ANNUAL REPORT 2017

GEN IV International Forum℠

Foreword from the Chair

2017 was a very fruitful year for the Generation IV International Forum (GIF) with progress in our collaborative R&D projects as well as several important international events.

The international workshop on "Advanced Reactor Systems (including Gen IV systems) and Future Energy Market Needs", organised by the Nuclear Energy Agency (NEA) as a side event of the GIF meetings in April 2017, was a great opportunity for our community to review the main expectations from the private sector, along with the new drivers and challenges in developing innovative reactor designs.

The challenges that the nuclear industry is facing were clearly identified and include safety concerns after the Fukushima accident, cost and regulatory uncertainties in a context where many electricity markets are dysfunctional as a result of cheap gas, subsidised renewables, and difficulties being faced by some ongoing nuclear new build projects in terms of construction costs and delays.

It was clearly recognised, however, that nuclear energy is a key technology in the fight against climate change and that many of the challenges facing the nuclear sector can be overcome through innovation and international co-operation so as to make the nuclear option more sustainable, even more proliferation resistant and more cost-effective, while maintaining the highest standards of safety.

In this context, we need three types of innovation: institutional, organisational and technological, all covered in some way through GIF activities.

Institutional innovation is important so as to share international safety standards with the objective of making progress towards stable and unified licensing processes. The work of a dedicated GIF Task Force to define safety design criteria (SDC) and guidelines (SDG) for the design of next-generation sodium-cooled fast reactors (SFR) represents an important step towards helping regulators become familiar with the technical characteristics of Gen IV systems and the associated safety research conducted within GIF. We also had the pleasure of having a panel at the international conference held in Russia on fast reactors (FR-17 in Yekaterinburg, Russia, June 2017) entirely dedicated to SDC and SDGs for SFRs. External review of this activity by the international regulatory community was also launched by the Ad Hoc Group on the Safety of Advanced Reactors (GSAR) jointly established under the aegis of the NEA Committee on Nuclear Regulatory Activities (CNRA) and the NEA Committee on the Safety of Nuclear Installations (CSNI).

This work is essential, and we hope for the extension of this effort to the other five GIF reactor systems. I would also like to mention the publication on the GIF website of other safety-related deliverables produced by the Risk and Safety Working Group that deals with the development of an Integrated Safety Assessment Methodology (ISAM) as a technology-neutral toolkit to evaluate the safety characteristics of all Gen IV systems, along with systems risk and safety assessment white papers.

Organisational innovation is also welcome in order to investigate new ideas and business models for nuclear reactors such as those developed by universities and start-up companies. The interest of private capital in advanced reactors (small modular reactor [SMR] and Gen IV) in some countries is a strong signal for the nuclear industry. It is an opportunity to better align our collaborative R&D with future market opportunities, and to attract and retain young skilled scientists and engineers. That's why the GIF launched a task force in 2015, providing a platform to enhance open education and training (E&T) and to facilitate networking of individuals and organisations involved in the development of Gen IV systems. A total of 12 webinars were organised in 2017 covering a wide range of topics with a high attendance record (totalling 3 387 views of live and archived webinars).

The Vice-chair mission on new market opportunities is also beginning to bear fruit with several contributions by the Senior International Advisory Panel and the Economic Modelling Working Group. Gen IV systems can address the need for dispatchable energy to meet the demand from the electricity and heat markets. Furthermore, Gen IV SMRs can substitute fossil fuel applications that generate CO_2 emissions and air pollution, and these SMRs can provide heat and power to remote locations.

We undoubtedly need technological innovation and international co-operation engaging in efforts both in relation to systems' research and cross-cutting activities.

Promising technologies, dealing for instance with modular construction, advanced concrete solutions, innovative fuels and materials (accident-tolerant fuels, ODS cladding), or 3D printing, are also growing in importance.

To better understand the impacts of these cross-cutting technologies on GIF R&D activities, the GIF Policy Group decided to launch a feasibility study on a possible new cross-cutting activity on advanced manufacturing and materials engineering.

In addition, high-performance computing and improved modelling capabilities are clearly opening the way for various applications in reactor physics and nuclear engineering, such as multi-criteria design optimisation, multi-scale and multi-physics calculation code systems, and the design of smart experiments for the qualification of innovative designs and components. To validate such tools and reduce uncertainties, we need proper R&D infrastructures, to be shared within the international community.

Under the supervision of the Vice-chair in charge of GIF external co-operation, it was therefore decided to launch a new task force on R&D infrastructures to identify existing key facilities, potential gaps and to enable access to such facilities.

Finally, I want to thank the involvement of all GIF systems and methodological working groups in paving the way for the major deliverables expected in 2018, namely the update of the 2009 R&D outlook and the preparation of the 4th GIF Symposium to be held in Paris (16-17 October 2018) and embedded in the 8th Atoms for the Future international conference jointly organised by GIF and the French Nuclear Energy Society Young Generation Network.

I look forward to seeing many of you for this important event for the GIF community and sharing with you the key conclusions of our outlook for the next decade.

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.

Among the signatories to the charter, 11 members (Australia, 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 agents to undertake the development of systems and the advancement of their underlying technologies. Argentina, Brazil and the United Kingdom[2] have signed the GIF Charter but did not accede to the FA; accordingly, within the GIF, they are designated as "non-active members". Australia, which signed the charter in June 2016, deposited its instrument of accession to the Framework Agreement in September 2017, nominating the Australian Nuclear Science and Technology Organisation (ANSTO) as its implementing agent, and this became effective 90 days later, on 13 December 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.

1. The European Atomic Energy Community (Euratom) is the implementing organisation for development of nuclear energy within the European Union.
2. The United Kingdom participates in GIF activities through Euratom.

Figure 1.1: GIF governance structure in 2017

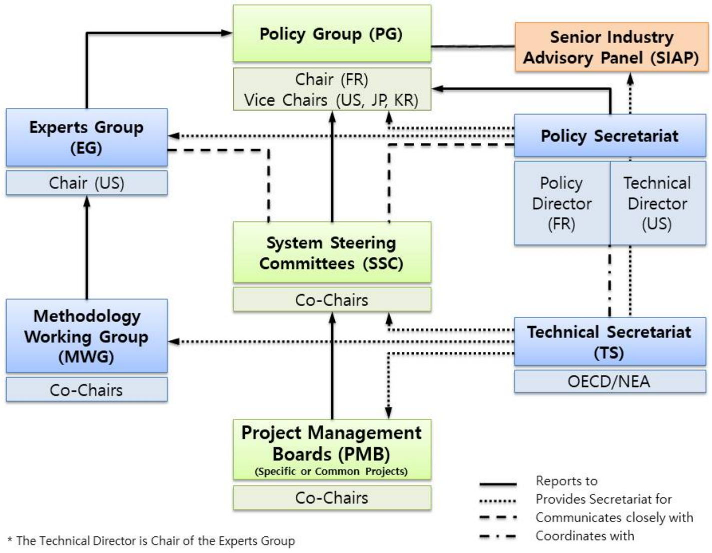


Table 1.1: Parties to GIF Framework Agreement, System Arrangements and Memoranda of Understanding as of 31 March 2018

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)	Australian Nuclear Science and Technology Organisation (ANSTO)	09/2017				12/2017		12/2017
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 and ICT (MIST) and Korea Nuclear International Cooperation Foundation (KONICOF)	08/2005 (02/2015)			04/2006 (02/2016)	11/2006 (03/2017)	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)	02/2018	01/2017

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 2017, the PG met in Paris in April, hosted by the Nuclear Energy Agency (NEA), and in Cape Town in October, hosted by the Republic of South Africa (Figure 1.2).

Figure 1.2: Policy Group in Cape Town, South Africa, October 2017

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. Project Arrangements have a ten-year duration, and each PA can simply be extended by written statement of all its signatories. In 2017, the SFR AF (Advanced Fuel) PA was not extended as the deadline for receiving such written statements was missed, but a new PA has been set up to continue the work. The SFR GACID PA on the other hand was voluntarily terminated. The SFR CDBOP (Component Design and Balance-of-Plant) project was extended for another ten years. Early 2018, both the VHTR HP (hydrogen production) PA and the FFC (Fuel and Fuel Cycle) PA were also extended for another ten years. The amendment of the VHTR MAT (Materials) PA to welcome China's INET as a member of that PMB was also successfully completed on 30 January 2018. Table 1.2 shows the list of signed arrangements and provisional co-operation within GIF as of 31 March 2018.

Table 1.2: Status of signed arrangements or MOU and provisional co-operation within GIF as of 31 May 2018

	Effective since	AUS	CA	EU	FR	JP	CN	KR	ZA	RU	CH	US
VHTR SA	Extended 30 Nov 2016	X		X	X	X	X	X			X	X
HP PA	19 March 2008 Extended		X	X	X	X	S	X			O	X
FFC PA	30 January 2008 Extended			X	X	X	X	X				X
MAT PA	30 April 2010			X	X	X	X	X			X	X
CMVB PA	Provisional			P		P	P	P			O	P
SFR SA	Extended 16 February 2016			X	X	X	X	X		X		X
AF PA	21 March 2007 Expired			X	X	X	X	X		X		X
AF PA (Phase II)	18 April 2018			X	X	X	X	X		X		X
GACID PA	27 Sept 2007 Expired				X	X						X
CDBOP PA	11 October 2007 Extended			O	X	X	O	X		O		X
SO PA	11 June 2009			X	X	X	X	X		X		X
SIA PA	22 October 2014			X	X	X	X	X		X		X
SCWR SA	Extended 30 Nov 2016		X	X		X	X			X		
M&C PA	6 Dec 2010		X	X		O	X			O		
TH&S PA	5 October 2009		X	X		O	X			O		
SIA PA	Provisional		P	P		P	P			P		
GFR SA	Extended 30 Nov 2016			X	X	X						
CD&S PA	17 Dec 2009			X	X							
FCM PA	Provisional			P	P	P						
LFR MOU				X		X	O	X		X		X
MSR MOU		X		X	X	O	O	O		X	X	X

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

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

CHAPTER 2

Highlights from the year and country reports

2.1. General overview

After becoming a member of the GIF in 2017 by signing its Charter, Australia accessed the Framework Agreement on 13 December 2017, with the Australian Nuclear Science and Technology Organisation as implementing agent.

GIF maintains a long-standing collaborative relationship with the International Atomic Energy Agency (IAEA). The 11th GIF-INPRO Interface Meeting was held in February 2017 in Vienna, Austria. While traditionally the collaboration's emphasis was on IAEA's International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) and covered co-operation on evaluation methodologies for economics, safety, physical protection, and proliferation resistance, its focus is gradually shifting to include generic advanced reactor technical information exchange, in addition to the GIF's SFR Safety Design Criteria (SDC) and Safety Design Guidelines (SDG) activities. It is expected that, in the future, the collaboration will be expanded to cover other areas of mutual interest, like special safety requirements for advanced reactors, future market conditions/requirements for advanced reactors (e.g. integration with renewable energy sources), non-electric applications of nuclear energy, and education and training.

At its 44th meeting, the GIF Policy Group approved the creation of a Research and Development Infrastructure Task Force (RDTF). The objectives of the RDTF are to:

- promote the utilisation of experimental facilities for collaborative R&D activities among GIF partners;
- facilitate GIF the partners' access to the various R&D facilities in the GIF member countries;
- identify essential large experimental infrastructure needed in support of Gen IV systems R&D activities in terms of their feasibility/performance, as well as demonstration/deployment;
- facilitate R&D collaboration across Gen IV systems.

The RDTF has drafted the terms of reference and is currently developing a work programme and list of deliverables.

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)

Based on the resolution documents published in response to the comments made by four external review bodies (*viz.* the US Nuclear Regulatory Commission (NRC), China's National Nuclear Safety Administration (NNSA), IAEA, and France's IRSN), the SFR SDC-TF has finalised the "GIF SFR System Design Criteria, Revision 2" report. The report outlines a set of criteria reflecting the GIF safety approach to achieve harmonised safety requirements for SFR systems. It is now being reviewed by the GIF Experts Group and will be published upon approval by both the GIF Experts and Policy Groups.

Education and Training Task Force (ETTF)

The ETTF's focus is on promoting education and training by identifying and advertising training courses; identifying and engaging collaboration with other international education and training organisations; delivering webinars dedicated to Gen IV systems and to related cross-cutting topics; and creating and maintaining a modern social medium platform to exchange information and ideas on Gen IV R&D topics, as well as on related GIF education and training activities.

In 2017, the ETTF has delivered 12 Webinars covering five of the six GIF systems as well as various cross-cutting issues addressing the economics of the nuclear fuel cycle, sustainability aspects of Gen IV systems, nuclear fuels and materials issues, the thorium fuel cycle, energy conversion systems, and the feedback from Phénix and Superphénix operation.

The ETTF is actively working towards engaging co-operation (e.g. exchange of information on activities of mutual interest such as the 2018 GIF Symposium) with other international organisations, specifically the European Nuclear Education Network (ENEN), and the Africa Network for Nuclear Education, Science and Technology (AFRA-NEST).

Advanced manufacturing and materials engineering

Tasked by the Policy Group to explore the potential for a GIF cross-cutting activity in the field of advanced materials engineering and manufacturing technologies, the Experts Group has submitted to the Policy Group a memorandum recommending the establishment of an ad hoc group to identify cross-cutting activities supporting advanced materials and manufacturing solutions to a high TRL[3]. It is recognised that such activities would have the potential to drive innovation in advanced manufacturing technologies and materials thus reducing impediments to Gen IV systems deployment. Depending on its results, the ad hoc group could evolve into a GIF Task Force (and possibly Working Group).

The group will involve the widest possible community and engage to gauge interest with both GIF countries research institutions and nuclear companies. For the latter, the approach taken will be both flexible and accessible, with clearly identified mechanisms for directly involving both prime, as well as small and medium-sized advanced nuclear reactor companies from GIF countries. The group will develop a priority list of R&D areas and initiatives, and deliver a white paper.

GIF comments to the IRSN[4] report

A revised draft report, summarising GIF SSCs and pSSCs comments, reviewed by an Experts Group member was completed and submitted to the RSWG for review. The RSWG comments were resolved with the GIF SSCs and pSSCs and a final draft was prepared and submitted to the Policy Group for approval.

2018 Update GIF R&D Outlook for Gen IV Nuclear Energy Systems Report, and Fourth GIF Symposium

The Experts Group adopted at its 38th meeting a revised structure for the 2018 Update of the GIF R&D Outlook Report. Drafting and reviewing of various SSCs, pSSCs, MWGs and

3. Technology readiness level.
4. https://www.irsn.fr/EN/newsroom/News/Documents/IRSN_Report-GenIV_04-2015.pdf

TFs contributions, as well as of the report's major messages was initiated. The revised draft will be prepared by end of January 2018, and a workshop to resolve all comments and finalise the report is planned for February 2018. The final report will be submitted to the Experts Group for review and approval at its 39th meeting in May 2018.

The 4th GIF Symposium will be held on 16-17 October 2018 in Paris. It will be embedded in the "Atoms for the Future" international conference, an event that is organised by the French Nuclear Society Young Generation Network (SFEN JG). The Symposium's International Scientific Programme Committee, after engaging with all the GIF stakeholders to identify the Symposium's major topical areas, has developed the Call for Papers and a first draft agenda comprising eight technical tracks and one panel on "Innovation and R&D in support of design, licensing, demonstration and deployment of Gen IV systems". The Symposium website is located at http://gifsymposium2018.gen-4.org.

SIAP design review of mature GIF systems

One element[5] of SIAP's three-year work programme consists in the offer to the GIF SSCs and pSSCs to perform design reviews and suggest R&D activities as well as milestones to be met in view of the deployment of their most "mature concepts". The criteria adopted for defining a "mature concept" are defined by the following timeline: pre-FOAK[6] by 2030-2035, FOAK by 2037-2040, and commercial by 2045. Consistently with the provisions of the GIF 2014 Roadmap Update, the pre-FOAK's objective is to demonstrate the technological, industrial and licensing feasibility, as well as elements of the economic viability of the respective design.

The SIAP developed a questionnaire to be completed by the SSCs/pSSCs wishing to submit their designs. In a first test phase, the questionnaire, reviewed by the Experts Group and approved by the Policy Group was returned by the (VHTR SSC, which submitted two GIF designs, specifically the HTR-PM under construction in China (start-up of the 1st module foreseen in 2018), and the SC HTGR, currently in the conceptual design and fuel qualification stage at Areva, USA (start-up of the 1st module foreseen after 2030). The responses to the questionnaire and the reviews performed by the SIAP during this test phase allowed fine tuning of the questionnaire. Based on this outcome, SIAP prepared a revised questionnaire and distributed it to all the six GIF SSCs and pSSCs.

Market issues

The Experts Group and the SIAP are supporting the vice-chair's for market issues two-year programme to elucidate Gen IV market issues. This programme aims at a survey of the key market drivers, opportunities and constrains. The survey will address key issues determining the political decision-making process as well as the industrial needs. In a next step, the study will analyse ways and means for GIF to respond to market drivers and maximise the valorisation of the attributes of the Gen IV designs. A SIAP paper on Gen IV Reactor Market Issues was discussed with Experts Group members and the SSCs/pSSCs. SIAP proposes to qualify its GIF IV systems priority attributes (economics, public acceptance, and integration in the future low-carbon energy mix) by developing associated challenges, and related R&D priorities, and to put these attributes in perspective and embed them into a wider concept of sustainability. First reflections on two aspects (concept flexibility and project finance risk management in view of cost reductions) were initiated.

5. In addition to the Panel's annual charges (for 2018 defined as "Among the recent and ongoing technological and organisational innovations for Gen III reactor designs and deployment, which topics and lessons learnt should in priority be capitalised for Gen IV reactors?"), and its activities in support of the GIF vice chair for market issues.
6. First of a kind.

GIF systems integration with renewables

The Experts Group has tasked the EMWG to analyse opportunities and challenges in view of the integration of Gen IV reactors into systems with an increasing share of variable renewable energy sources. The expected output of this analysis is a position paper summarising the issues, and recommendations from the Experts Group to the Policy Group.

This activity is performed in collaboration with the SIAP and is also in support of the GIF Vice-Chair's for market issues two-year programme, in terms of its pursuit of measures to enhance market drivers for Gen IV systems and the enhancement opportunities for Gen IV reactors to be integrated into systems relying on an increased share of renewable energy resources. Preliminary findings indicate that new-built reactors will have to be more flexible as compared to the current reactor generation for integration into power grids relying on a significant renewables share. There will be a need of clear policies aiming at an optimum mix of renewable, nuclear, other energy sources and energy storage capabilities. Hybrid systems could constitute a solution, provided that grid-scale energy storage, and flexible cogeneration applications of thermal and/or electrical energy are available.

2.3. Country reports

Australia

On 6 September 2017, following scrutiny by the Australian parliament and approval by the Federal Executive Council, Australia's Minister for Foreign Affairs, the Honourable Julie Bishop MP, signed Australia's Instrument of Accession to the Generation IV International Forum Framework Agreement.

Subsequently, on 14 September 2017, the Australian Ambassador to the OECD, His Excellency Mr Brian Pontifex, and ANSTO CEO, Dr Adi Paterson, deposited Australia's Instrument of Accession with the Secretary-General of the OECD. In depositing Australia's Instrument of Accession, Ambassador Pontifex thanked the GIF Policy Group, the GIF Technical Secretariat, and the OECD Office of Legal Counsel for their support throughout the membership and accession processes. The Framework Agreement entered into force for Australia on 13 December 2017, upon which Australia became a full and active member of the GIF.

In Australia, the responsibility for undertaking parliamentary scrutiny of new treaties falls to the Joint Standing Committee on Treaties. In recommending to the parliament that Australia should accede to the Framework Agreement, the Joint Standing Committee on Treaties found that:

> *Participation in the GIF is expected to help Australia maintain its national capacity as a leading edge nuclear technology developer in material sciences and fuel technologies. In particular: "Australian industry membership will provide participation for Australian scientists and engineers, with avenues for collaboration in the world-leading teams developing our next generation of nuclear and related technologies and with access to the technologies themselves."*

In carrying out its inquiries, the Joint Standing Committee on Treaties conducted a public hearing and called for public submissions. The overwhelming majority of public submissions welcomed Australia's participation in the GIF.

As foreshadowed in previous Policy Group meetings, in depositing our instrument of accession, the Australian government nominated the Australian Nuclear Science and Technology Organisation as Australia's implementing agent.

Although it might appear that Australia has progressed from nominee to full GIF member in a relatively short period of time, the Joint Standing Committee on Treaties noted in its report that a bid for membership of GIF had been under consideration in Australia since 2006. This has been a long-term ambition for those at ANSTO who have been involved in such deliberations for more than ten years. On 14 December 2017, ANSTO signed the Very High Temperature Reactor System Arrangement and the Molten Salt Reactor Memorandum of Understanding, and started engaging with the respective System Steering Committees to discuss its contribution to those systems.

The last of the processes that were ongoing as an outcome of the South Australian Nuclear Fuel Cycle Royal Commission came to a close in 2017, with the release of the report of the South Australian parliament's Joint Committee on Findings of the Nuclear Fuel Cycle Royal Commission. As was expected, the majority of the Committee recommended that the South Australian government should not commit further public funds investigating the proposal to establish an international high-level radioactive waste disposal facility. The members of the committee could not agree on any other recommendations, although there was a general view that the proposal for an international spent fuel repository did not pose insuperable safety or technical challenges. The South Australian government has said that it does not intend to pursue the proposal due to the withdrawal of bi-partisan support.

Separately, the Australian government is continuing its efforts to establish a National Radioactive Waste Management Facility, which will provide centralised, co-located facilities for low-level waste disposal and intermediate-level waste storage. Three volunteered sites, in two different communities, have been accepted by the Minister for Resources to progress to phase two of the site selection process, on the basis of broad support in both communities. The three sites are now undergoing more detailed community consultation and technical and heritage assessments, and the Department of Industry, Innovation and Science, with support from ANSTO, has established a permanent presence in both communities to ensure that they are fully engaged in the process. It is anticipated that one of the three sites will be selected as the preferred location for the national facility by the end of 2018.

Canada

Nuclear power in Canada

The government of Canada's position is that nuclear energy, as a nearly 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 remains 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.

In June 2017, the House of Commons Standing Committee on Natural Resources tabled a report entitled "The Nuclear Sector at a Crossroads: Fostering Innovation and Energy Security for Canada and the World", which can be found on their website at www.ourcommons.ca/DocumentViewer/en/42-1/RNNR/report-5. The report found that the nuclear sector in Canada is at a crossroads following several major changes over the past few years in the Canadian nuclear sector including the recent restructuring of Atomic Energy Canada Limited and made seven recommendations to advance the viability and competitiveness of Canada's nuclear industry with respect to regulatory and safety practices, research and innovation, leadership in nuclear power generation, and the development and commercialisation of next-generation nuclear technologies. In October 2017, the government of Canada responded to the report agreeing to all of the recommendations. The government response can be found on the Our Commons website at www.ourcommons.ca/DocumentViewer/en/42-1/RNNR/report-5/response-8512-421-241.

Nuclear energy developments

- Domestic

In the province of Ontario, a planned investment of CAD 26 billion (Canadian dollars) is ongoing to 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, is currently undergoing a 40-month refurbishment. 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.

Atomic Energy of Canada Limited (AECL), on behalf of the government of Canada, is investing more than CAD 1.2 billion to revitalise the Chalk River Laboratories and build a new world-class science facilities. Recent infrastructure investments have included: over CAD 55 million for a new hydrogen lab complex in 2015, a new materials science lab worth over CAD 100 million in 2016, CAD 40 million for a new tritium lab currently in final commissioning stage and another CAD 190 million has been directed to other major infrastructure projects that began in 2017.

Small modular reactors

The government of Canada has convened a process to develop a Canadian roadmap for the potential development and deployment of SMRs in Canada. The SMR Roadmap aims to produce its final report later in 2018.

The SMR Roadmap process began with provinces, territories and utilities, and has grown to include all essential enabling partners, including (but not necessarily limited to): national laboratories, the regulator, the waste management organisation, industry and academia. Demand-side stakeholders have also been engaged, including mining and oil sands industry stakeholders, as well as Indigenous and northern people.

The SMR Roadmap seeks to credibly and transparently demonstrate the market and stakeholder views.

Two additional initiatives underway in Canada will complement the SMR Roadmap. The first is the work of the Canadian Nuclear Safety Commission (CNSC) to ensure regulators readiness for SMRs in Canada.

The CNSC has been approached by a number of SMR vendors; 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. As of early 2018, ten SMR companies have started the VDR process with the likelihood that others will follow in the near term.

Additionally, in response to CNSC discussion paper DIS-16-04: "Small Modular Reactors: Regulatory Strategy, Approaches and Challenges", the CNSC published a "What We Heard Report" on 18 September 2017. This report summarises the results of the CNSC's consultation on DIS-16-04 and outlines some of the next steps the CNSC plans to undertake regarding the regulatory framework for SMRs. The CNSC report can be found on their website at www.nuclearsafety.gc.ca/eng/acts-and-regulations/consultation/completed/dis-16-04.cfm.

The second is being led by Canadian Nuclear Laboratories (CNL) to identify viable technologies and options for demonstration. In 2017, CNBL launched a Request For

Expressions of Interest (RFEOI) on small modular reactors (SMR) to gather feedback and initiate a conversation on the potential for an SMR industry in Canada, and the role CNL can play in bringing SMR technology to market. CNL issued a report summarising the findings, entitled "Perspectives on Canada's SMR Opportunity". The report can be found at www.cnl.ca/site/media/Parent/CNL_SmModularReactor_Report.pdf. The RFEOI yielded responses from 80 organisations representing a variety of interested stakeholders including international respondents. The report compiles the information but does not attempt to make recommendation or conclusion from the responses. CNL has identified SMRs as one of seven strategic initiatives the company intends to pursue as part of its long-term strategy, with the goal to demonstrate the commercial viability of the SMR by 2026.

International

In 2017, Canada was co-lead with the United States and Japan, in the preparation of a proposal for the "Nuclear Innovation: Clean Energy Future (NICE Future)", a new initiative under the Clean Energy Ministerial (CEM) to encourage formal discussion between member countries about nuclear energy options for both electric and non-electric applications. Canada will host the 10^{th} CEM Meeting in 2019, where progress on the NICE Future will be reported.

The Clean Energy Ministerial (CEM) is a high-level global forum to promote policies and programmes that advance clean energy technology, to share lessons learnt and best practices, and to encourage the transition to a global clean energy economy. Initiatives are based on areas of common interest among governments and other stakeholders of the 24 member countries and the European Commission.

Also in 2017, Canada has committed to join, as a founding member, a new initiative launched by the Nuclear Energy Agency (NEA), called the Nuclear Education, Skills, and Technology (NEST) framework. This is new multilateral joint undertaking that aims to help nuclear nations attract global top talent working in the nuclear sectors, and marshal geographically-distributed researchers under focused project statements to generate useful innovative breakthroughs to real-world energy problems. NEST will employ a project-based approach, with projects led by a managing institute and supported by fellows at universities, labs or private entities across several countries.

Activities within GIF

In 2017, Canada signed the amendments to the Project Arrangements for Thermal-Hydraulics and Safety as well as Materials and Chemistry of the GIF Supercritical Water-Cooled Reactor (SCWR) system as part of its continued active participation. Canada continued the participation in the three Cross-Cutting Working Groups to develop modelling tools or methodologies in support of the six Gen IV systems. In addition, Canada continues to be engaged in GIF initiatives such as compiling a list of infrastructures available and those for future needs to support the development of the Gen IV nuclear reactor systems.

Supercritical water-cooled reactor research and development

Canada has started the verification and validation phase of key components (such as mechanical components, thermal-hydraulics, materials, chemistry, fuel channel behaviours, fuel, reactor physics, economic modelling, etc.) to improve the confidence of the Canadian SCWR concept. Benchmarking exercises of analytical tools have been identified using experimental data obtained in previous phases of the project. These exercises have been scheduled with partners within the SCWR system. A second round robin corrosion test has been initiated within the SCWR Materials and Chemistry Project. Canada completed the corrosion testing and provided the results to other participants for comparison. A new manufacturing technique using the 3D metal printer has been explored to join two different materials. Canada plans to share the experience with other participants at the new Advanced Manufacturing Task Force. A strategy for developing a small SCWR concept has been established in Canada. It aims to generate power and produce process heat safely and economically for off-grid small remote communities, mining operations and oil-sand production. Canada is co-ordinating the R&D effort with other SCWR partners to develop small SCWR concepts. Canada is hosting the SCWR Information Exchange Meeting in 2018 and the 9th International Symposium on SCWRs in 2019 to provide the forum for researchers to disseminate the research findings, exchange ideas and establish collaborations. Researchers from GIF and non-GIF member states are invited to participate.

People's Republic of China

Nuclear energy legislation and regulation

China's parliament passed a new nuclear safety law on 1 September, aimed at improving regulation in the nuclear power sector as new projects are built across the country. The new law reflects China's rational, co-ordinated and balanced nuclear safety outlook, as well as its commitment to fulfilling obligations under international treaties.

China should carry out international exchanges and co-operation to prevent and deal with the threat of nuclear terrorism, the law proposed.

Under the new law, the government is required to set up an inter-agency co-ordination mechanism for nuclear safety and a national committee in charge of emergency response to nuclear accidents.

The law also introduced a set of protocols for nuclear facility operators based on their full responsibility for nuclear safety. The law will go into effect on 1 January 2018.

The nuclear power plants in operation and under construction in China mainland

- On 25 May, 15 days ahead of schedule, the hemispherical dome was installed on Fuqing unit 5, marking the completion of construction work on the pilot project and the beginning of the installation stage. Fuqing unit 5 is the first pilot project featuring HPR1000 technology, the third-generation reactor designed and developed by CNNC.
- On 21 July and 4 August, the first batch AP1000 projects in China, Sanmen unit 1 and Haiyang unit 1 respectively passed the comprehensive nuclear safety audit organised by NNSA, which is the most critical inspection before the initial fuel loading.
- Fuqing unit 4 began commercial operation on 17 September, marking the completion of the first phase of CNNC's project in Fuqing, with its four units now generating electricity.
- By the end of September, 37 nuclear power units in operation have kept a good record in safety and operation performance, and 19 units under construction are progressing as scheduled, only some demonstration projects are delayed.

International co-operation in nuclear energy

- China had completed a low-enriched uranium (LEU) renovation project in miniature neutron source reactor (MNSR) in Accra, Ghana on 10 August. The Ghana project undertaken by CNNC, under the guidance of CAEA, has proven the practicality and credibility of China's LEU technology and demonstrated a model for other countries in MNSR remoulding.

- The initiative to establish a fifth World Association of Nuclear Operators (WANO) centre in Shanghai received unanimous votes at a council conference held in Paris on 22 June, and will take effect after being approved at the WANO Plenary Session in October.
- During the Fourth MDEP Conference held by the NEA in London on 13 September Mr Liu Hua who is NNSA Administrator and also the China PG Member, his proposal of setting up HPR1000 Working Group in MDEP was agreed by all the members.

Gen IV Nuclear Energy Systems activities

- **SFR**: The R&D of demonstration SFR has been carried out, and it is expected that the FCD will be reached by the end of this year. At present, the review and evaluation of PSAR is in progress. CNNC and Terra Power will set up a new joint venture, to develop a 300 MWe demonstration Travelling Wave Reactor Program.
- **VHTR**: HTR-PM demonstration project progresses well. It will be connected to grid by the end of 2018 and be in full power operation in 2019 in accordance with the current plan. The installation is now in the final stage and commissioning test has started. The signature process of Project Arrangement to entrance into MAT-PMB and HP-PMB are undergoing. The preparation of CMVB Project Plan is nearly finished.
- **SCWR**: The R&D on SCWR and pre-conceptual design of the experimental reactor of CSR1000 have been proceeding continually in China. In terms of co-operation in SCWR, a new international benchmark exercise is almost finished based on the SCW 2X2 rod bundle tests from NPIC for assessing the computational fluid dynamics (CFD) models. On 29 May and 22 June, China respectively signed the Amendment to the Generation IV International Forum Project Arrangement on M&C and TH&S for the International Research and Development of the Supercritical Water-cooled Reactor Nuclear Energy System.
- **LFR**: China attended the 20th (21-22 March in Paris) and 21st (9 September in Seoul) GIF LFR pSSC meeting as observers and dedicated to finalise several technical documents such as SDC, SRP, SSA, IRSN report, etc.
- **MSR**: TMSR Research Centre in China recently completed the preliminary engineering design and site selection for the 2 million watts liquid-fuelled thorium molten salt experimental reactor (TMSR-LF1) and the conceptual design for a small modular thorium molten salt reactor. A TMSR simulator (TMSR-SF0) is under construction. It is expected that all the equipment would be built, installed and commissioned before June 2018.
- **Nominations**: China nominated Mr LYU Huaquan as the Chinese SIAP alternate member and planned to update the Chinese MWG members.

Euratom

- The current five-year Euratom research and training programme will expire at the end of 2018. Negotiations are ongoing for the extension of the Euratom Research training programme to 2019-2020 with commitment to include all activities linked to advanced reactors.
- The latest Euratom project proposal call of 2016-2017 was very successful in selecting, out of 72 proposals, 25 projects, of which 5 key projects supporting Gen

IV systems. These are: (1) ESFR-SMART: European Sodium Fast Reactor Safety Measures Assessment and Research Tools; (2) GENIORS: Gen IV Integrated Oxide fuels recycling strategies; (3) GEMINI Plus: Research and Development in support of the GEMINI Initiative (HTR and Cogeneration); (4) INSPYRE: Investigations Supporting MOX Fuel Licensing in ESNII Prototype Reactors (fast Reactors); (5) GEMMA: Generation IV Materials Maturity.

- A new call will be published at the end of October 2017 covering the year 2018 (EUR 60 million overall Euratom budget) but will be focused on the launch of a European Joint Programme in the field of waste.
- Then will follow a two-year call covering 2019-2020, (EUR 120 million overall Euratom budget) where there should be much larger opportunities for projects supporting Gen IV related research.

Systems

GFR: Consortium "V4G4 (Visegrad 4 Generation IV) Centre of Excellence", working on the design of the ALLEGRO reactor core performed feasibility studies on UOX fuel type (Slovak Republic). The Project SUSEN for Sustainable Energy (Czech Republic) included the experimental helium loop S-ALLEGRO.

MSR: The SAMOFAR (Safety Assessment of the Molten Salt Fast Reactor [MSFR]) summer school was held with over 90 participants from all over the world. The NUSTEM project in the United States was recently granted, which will lead to more co-operation between SAMOFAR and the United States in the field of education and information exchange on MSR. Two new experimental rigs have being constructed: DYNASTY facility at POLIMI (IT) for natural circulation and the SWATH facility at CNRS Grenoble for heath exchange with walls). SALIENT irradiations of salt samples in the Petten HFR have started in August 2017 and will investigate fission product stability/management)

LFR: A US-EU project started in March 2017 on techno-economic assessments for LFR small modular reactors with several key industrial and research organisation involved on both sides. A Facility for material testing in Lead environment successfully commissioned in Petten by JRC (the facility is a part of the JRC's Liquid Lead Laboratory – LILLA). Falcon consortium to support ALFRED construction in Romania under renewal and signature expected by end of October.

> *MYRRHA: compatibility testing performed on americium and neptunium bearing transmutation fuels in contact with the liquid metal coolant LBE under representative accident conditions. In addition, Knudsen Cell Effusion Measurements (KEMS) and SEM investigations on helium and fission gas release mechanisms*

SCWR: In Material research the corrosion resistance of different candidate materials was investigated with in-analytical methods (electrochemical measurements) have been developed. The in-pile material loop was assembled (in the out-of-pile conditions) and first tests were completed. In thermal-hydraulic and safety: heat transfer experimental studies and CFD modelling of flow in SCW were performed.

VHTR: Safety investigations on VHTR fuel in 2017 included a number of successful heating tests in the Cold Finger Apparatus (KüFA), simulating hypothetical accident scenarios on irradiated HTR fuel elements. Extensive post-irradiation examinations (PIE) have been performed on TRISO-coated particle fuel for HTR.

SFR: Fresh and irradiated FR MOX fuels with an initial Pu content of 24% were investigated within the H2020 ESNII+, in support of ASTRID. The fuels microstructures were characterised.

France

Situation of EDF nuclear fleet

In 2017, EDF nuclear power generation accounted for about 72% of the total electricity production, with 379 TWh generated. The nuclear electricity production was down by 1.3% compared to 2016. This is primarily explained by a series of outages as part of the inspections conducted regarding the certification of some components manufactured at "le Creusot" forge. The temporary shutdown of the Tricastin nuclear power plant in order to conduct engineering work to strengthen the nearby dikes also played a role.

Regarding the EPR reactor under construction in Flamanville, major construction steps were achieved for this new build project, in line with the completion schedule announced in 2015 towards commissioning by the end of 2018/beginning of 2019:

- most of the equipment of the nuclear section, such as the conventional island, has been delivered and installed on-site;
- completion of the main civil engineering work;
- first start-up of the turbine and the alternator;
- transfer of the control room to EDF teams that will operate the reactor.

In December 2017, EDF successfully completed the cold functional test phase. This stage is part of the system performance testing, which started in the first quarter of 2017, to check and test operation of all the EPR systems.

Update the French Energy Master Plan (PPE)

In 2017, the French government decided to launch the process to update the French Energy Master Plan for the period 2019 to 2029. This programmatic document will set the priorities for the evolution of the French energy mix, taking into account the objectives of the French Energy Transition Law and the 2017 climate plan.

Restructuring of the French nuclear industry

The key steps of the process have now been achieved, with the recapitalisation of Areva (now Orano) and the effective transfer of its power reactor business (now Framatome) to EDF as new major shareholder of the corresponding company.

Regarding Orano, the reorganisation of fuel cycle activities in a new entity was completed with an overall capital increase of EUR 3 billion. In January 2017, this project received a formal approval from the European Commission regarding compliance with EU competition rules. In March 2017, MHI and Japan Nuclear Fuel Limited (JNFL) agreed on an investment protocol and shareholders' agreement by which both companies will each take 5% of the capital of NewCo.

On the reactor side, Framatome was taken over by EDF and a joint company called Edvance in charge of nuclear island engineering studies for new build projects has been set up.

A EUR 4 billion capital increase of EDF was also approved by the French government, as the majority shareholder of the company.

Six EPR reactors are under construction worldwide and their status is as follows:

- Flamanville 3 in France and Olkiluoto 3 in Finland concluded their cold tests in 2017, towards fuel loading by the end of 2018/beginning of 2019;
- the fuel load for the first EPR in Taishan, China, should take place in 2018;

- the first concrete for the first EPR at Hinkley Point C in the United Kingdom is expected for spring 2018, following extensive ongoing site preparation.

The overall governance of the French nuclear sector has also been strengthened through an increasing role of the National Committee of the French Nuclear Sector "Comité National de la Filière Nucléaire" (CSFN) that is now chaired by EDF. In 2017, the CSFN identified four priorities for the French nuclear sector:

- maintaining, developing and valorising nuclear competencies;
- delivering EDF long-term operation programme "Grand Carénage";
- strengthening the co-ordination of French nuclear exports;
- promoting technological change through collaborative R&D efforts (including with small and medium-sized enterprises).

Progress of the ASTRID international project

The ASTRID demonstrator project is continuing its basic design phase that is to be completed by the end of 2019. International collaborations on this project have recently been strengthened, most notably with Japan.

In March 2017, France and Japan signed a new co-operation agreement paving the way for a joint development on the ASTRID project during a high-level bilateral meeting between Japanese Prime Minister Shinzo Abe and then French President François Hollande.

In parallel, activities for the detailed design of the ASTRID project are progressing. For instance, the first full-scale hydraulic test campaign for the qualification of hydraulic shock absorber system (dashpot) were completed by Areva and delivered to CEA. These tests support the design of the assembly for the first reactor core, as this innovative system will play a key role for the control of reactivity.

A number of technical deliverables have also been completed. For instance, the simulation of a gas power conversion system based on Brayton cycle has been greatly improved using the latest version of the Cathare 3 thermal-hydraulic modelling code that integrates the nitrogen real gas state. This design option would eliminate the risk of water-sodium interaction.

Progress of the RJH international project

In 2017, the construction of the Jules Horowitz reactor (RJH) in Cadarache successfully achieved its key milestones in line with the new construction schedule decided in 2015.

The first fuel load is expected to be charged between 2020 and 2022, with isotopes production taking place 18 months later.

In parallel, CEA is preparing with its partners from the JHR International Consortium the first joint research programmes on innovative fuels and materials. To gather scientific community around JHR, pre-JHR joint programmes with support of operating MTRs are under preparation applying for programmes both under the European Framework Program scheme (H2020) and under the NEA joint programmes scheme to be performed in the next few years with the objectives to be implemented in JHR starting from 2022.

Progress with decommissioning and waste management activities in France

CEA is currently dismantling more than 35 nuclear facilities. Due to his long history, both in civilian and in defence activities, CEA has gained a unique experience on its own decontamination, dismantling and legacy waste retrieval programmes.

In 2017, the decommissioning and waste management activities of CEA Nuclear Energy Division have been reorganised in a single directorate that integrates transversal programmes, R&D activities and dismantling units in charge of operations into nuclear facilities.

In support to these dismantling operations, CEA has been developing for many years a comprehensive R&D programme in this field.

Regarding ongoing projects, CEA successfully implemented a remote controlled laser cutting at UP1 and APM and developed new tools for laser cutting of the fuel debris at Fukushima Daiichi reactors. New developments were also launched in the field of treatment of waste from decommissioning (in can melting, plasma under water, new binders, tritium removal and B4C treatment) and in the field of characterisation (autoradiography, mobile tomography).

Moreover, Orano manages many operations of dismantling and waste retrieval, and expanded its current range of waste packages based on the use of:

- Incineration/fusion/vitrification process (PIVIC) intended for the treatment and conditioning of mixed (organic and metal) alpha-contaminated waste and contributed to the European THERAMIN project.
- Cementation, encapsulation, compaction and other technologies.

Finally, EDF conducts many programmes for decommissioning of plants in its own fleet.

In parallel, they defined in particular specific devoted automated/remote tools for cutting irradiated components such as vessel internals for Chooz A.

EDF created in 2016 Cyclelife, a new international platform in waste treatment, with activities in the United Kingdom, Sweden and France.

Progress with the Cigéo deep geological repository project

As part of the licensing process for the start of the project, a "safety options file" was submitted by the Agency for radioactive waste (Andra), to the French safety authority (ASN). The report sets out the chosen objectives, concepts and principles for ensuring the safety of the facility.

In July 2017, IRSN submitted its conclusions to ASN: IRSN highlighted that the project achieved overall a satisfactory technical maturity and underlined the substantial work undertaken by Andra with a view to demonstrate the safety of the installation.

ICERR Affiliates' agreement extended to three new countries

As part of the IAEA International Centre based in Research Reactors (ICEER) framework, the CEA has signed three more agreements with Jordan, Algeria and Indonesia in 2017, for a total of six Affiliates (Slovenia, Tunisia and Morocco being the first ones) that are now linked with CEA under the framework of this IAEA initiative, facilitating the access to CEA experimental facilities.

Selection of CEA's technology to demonstrate the feasibility of earth decontamination in the Fukushima region

A joint venture between Areva and Atox has been selected by the Japanese Environment Ministry in order to demonstrate the feasibility of earth decontamination in the Fukushima region based on a process patented by CEA in 2012. This process was subsequently developed at the industrial scale through a collaboration with Areva and Veolia that was financed by the French strategic investment fund (PIA).

Japan

Current status of nuclear policy

- Strategic energy plan: In August 2017, the government of Japan started the discussion on revising the Strategic Energy Plan, which serves as a basis for Japan's energy policy. The Strategic Policy Committee (for Natural Resources and Energy) of Advisory Committee for Natural Resources and Energy has been working on it, considering the plan, the current edition of which was established in 2014, is required to be revised at least every three years by law. The Round Table Studying the Energy Situation was launched to discuss desirable long-term energy future considering the Paris Agreement, and has been working on it in parallel.
- Fast reactor development: Following the new fast reactor development policy issued in December 2016, the government established the Strategic Working Group (WG), comprised of working-level members, under the Council on Fast Reactor Development chaired by the Minister of Economy, Trade and Industry in March 2017. Since then, the WG has been devoting to developing the Strategic Roadmap, which determines development tasks in the coming decade, towards its finalisation in 2018. To consult experts, the group has invited Mr William D. Magwood IV, Director-General of the Nuclear Energy Agency (NEA); Mr Yang, Reactor Engineering Department Director of China Institute of Atomic Energy (CIAE); domestic experts on the development strategy in Russia and India and on the social acceptability and on the safety of fast reactors.
- The prototype fast breeder reactor MONJU: In December 2016, the government of Japan finally decided to decommission MONJU. In June 2017, the MONJU decommissioning team, established under the government, formed its basic policy accordingly. Soon after, JAEA formulated the master plan for the decommissioning and started preparation for it.
- High-temperature gas-cooled reactors (HTGRs): A memorandum of co-operation in the field of HTGR technologies was concluded in May 2017 by JAEA and National Centre for Nuclear Research (NCBJ) in Poland, based on the Action Plan for the Implementation of the Strategic Partnership agreed by foreign ministers. Also, the international co-operation including joining the GEMINI+ project in EU has been proceeded. Following these activities, the industry-academic-government forum for HTGRs determined to establish a WG for overseas development strategy at their fifth meeting in June 2017. The WG held the first meeting and formulated the strategy for the HTGR technology development of Japan in the following August.
- The nuclear fuel cycle services: Nuclear Reprocessing Organization of Japan, authorised by the Minister of Economy, Trade and Industry, was established in October 2016. It aims at advancing steady and efficient reprocessing services in the midst of changing business environments for domestic utilities due to the electricity deregulation.
- The site selection for high-level radioactive waste: Although no significant progress had been made in siting for geological disposal of high-level radioactive waste for a long time, Agency for Natural Resources and Energy unveiled Nationwide Map of "Scientific Features" relevant for Geological Disposal in July 2017 as the first step on a long way towards completion of geological disposal. This is attributed to the Cabinet decision, clarifying that the government leads the

completion of geological disposal, based on the basic policy for the final disposal site revised in 2015.

Fukushima Daiichi nuclear power station (1F)

- Current status of the reactors: Each unit is under cold shutdown condition. Approaches to in-core inspection, to dealing with taking out the fuel debris, and to the decommissioning are being investigated based on the mid- and long-term roadmap which has been revised four times since its establishment in 2011.
- Technical Strategic Plan 2017: in August 2017, Nuclear Damage Compensation and Decommissioning Facilitation Corporation published Technical Strategic Plan 2017 for Decommissioning of the Fukushima Daiichi Nuclear Power Station of Tokyo Electric Power Company Holdings, Inc. It is to contribute to steadily implementing the mid- and long-term roadmap and to appropriately revising the roadmap by providing the technical justification.
- The radioactive water treatment: Preventive and multi-layered countermeasures against the contaminated water have been taken based on the three principles: a) isolating the polluted water from any other water sources; b) containing the contaminated water without any leak; and c) removing radioactive substances from the water. According to these principles, it has been pumping up the ground water and building the land-side impermeable wall or “frozen soil wall” for which refrigeration of the remaining openings started in August 2017 for a). The closure of the sea-side impermeable wall completed in October 2015 for b). It was confirmed that the quality of the water has been improved steadily due to the significant decrease of the radioactive material outflow to the sea. For c), the sub-committee was launched in September 2016 and comprehensive deliberation on long-term treatment of the clarified water by using Advanced-Liquid-Processing-System (ALPS) is being conducted from the technical and social views including harmful rumours.
- The fuel removal from the spent fuel pools: the removal work of the spent fuel of unit 4 completed in 2014. The operator is currently installing a cover dome for the removal at unit 3, aiming at starting the work in mid-2018. The preparation for rubble removal and decontamination for units 1 and 2 is in progress.
- The fuel debris removal: the in-core investigation is advancing step by step to the debris removal. In 2017, the inspection of the containment vessels of the units, from 1 to 3, was conducted with a dosimeter and camera. The project has taken a big step towards the decommissioning by successfully obtaining a lot of valuable data including pictures and radiation dose through this investigation.

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

- Safety review results: NRA has granted permission for 14 units of 7 sites to alter their reactor installation among 26 units of 16 sites, which had applied for NRA’s conformity assessment to the new regulatory requirement for restarting an NPS.
- Permittees: Sendai NPS units 1 and 2, Genkai NPS units 3 and 4 of Kyushu Electric Power Company; Ikata NPS unit 3 of Shikoku Electric Power Company; Takahama NPS units 1 to 4, Mihama NPS unit 3 and Ohi units 3 and 4 of Kansai Electric Power Company; Kashiwazaki-Kariwa NPS units 6 and 7 of Tokyo Electric Power Company.
- Out of these, five reactors are in operation: Sendai NPS units 1 and 2, Ikata NPS unit 3 and Takahama NPS units 3 and 4.

- Nuclear fuel cycle facilities: As a result of the conformity assessment to the new regulatory requirement, NRA has granted permission for Global Nuclear Fuel Japan's (GNF-J) fuel fabrication plant in April 2017, Japan Nuclear Fuel Limited's (JNFL) uranium enrichment plant in May 2017, Mitsubishi Nuclear Fuel's fuel fabrication plant in November 2017, and for Nuclear Fuel Industries Ltd.'s Tokai Plant in December 2017 to modify their operations.

Current situation of facilities of Japan Atomic Energy Agency (JAEA)

- Progress of Decommissioning of MONJU: JAEA has submitted the master plan mentioned earlier for the decommissioning to NRA, which mainly focuses on the removal of its fuel assemblies as the first step. It aims to complete the project in about 30 years after giving due consideration in characteristics of a sodium-cooled fast reactor including its low decay heat and radioactivity. JAEA will also refer to the technology and knowledge accumulated through previous decommissioning activities in Japan and overseas.
- The experimental fast reactor Joyo: Although JAEA had applied to NRA for alteration of its reactor installation to verify the conformity to the new regulatory requirement in March 2017, NRA required adding more information on preventive measures against core damage and containment vessel failure, and on emergency power supply systems in line with the requirement for power reactors. JAEA is preparing for reapplication accordingly.
- The high-temperature gas-cooled test reactor (HTTR): In November 2016, JAEA applied to NRA for alteration of its reactor installation to verify the conformity to the new regulatory requirement towards approval in 2018. NRA completed the substantive assessment in November 2017.

Korea

Nuclear power

A total of 24 nuclear power plants are operated in Korea, including Shin-Kori unit 3, an APR-1400 (1 400 MWe advanced power reactor), which started commercial operation in December 2016. In 2016, the nuclear power plants in Korea generated 161 995 GWh of electricity, which is responsible for 30.64% of the total electricity production. As of August 2017, the installed nuclear capacity from the 24 reactors accounts for 19.73% (22 529 MWe) of the total installed capacity. The capacity is slightly smaller than on April 2017 because Korea's first nuclear reactor, Kori-1, was permanently shut down on 19 June 2017. Three nuclear power reactors, Shin-Kori unit 4 and Shin-Hanul units 1 and 2, are under construction.

Nuclear energy policy

The new Korean government has stated that domestic electric generation would be gradually derived from renewables and gas power plants, and that the Korean nuclear energy policy will focus on reinforcing the safety of nuclear power plants, preparing for decommissioning, managing spent fuel, and encouraging the export of nuclear power plants and their relevant technologies. After this statement, on 24 July 2017, a public engagement committee was established to gather the public opinion and then to recommend to the government whether or not the construction of Shin-Kori units 5 and 6 will continue to be completed. The committee formed a citizens' participation group, which conducted deliberations on whether to keep the construction or not. Based on the voting results by the group on 20 October, the committee recommended the continued construction of Shin-Kori units 5 and 6, and the reduction of nuclear power share in

Korea's energy mix. The Korean government announced its "Roadmap of energy transition" on 24 October, which includes the construction of Shin-Kori 5 and 6 kept as planned, and the shares of renewables and natural gas applied to power generation increased.

Current status of construction project of NPPs in United Arab Emirates (UAE)

KEPCO (Korea Electric Power Corporation) has been constructing four units of the Barakah Nuclear Power Plant in the United Arab Emirates (UAE) since 2012. Barakah unit 1 has been waiting for fuel loading after hot functional tests. The unit 2 nuclear reactor has been installed, and structural construction is currently ongoing in units 3 and 4. The project will be completed by 2020. Furthermore, KEPCO and the UAE's Emirates Nuclear Energy Corporation (ENEC) signed a contract to jointly invest in a corporation, the NAWAH energy company, which will be in charge of the operation and management of the four units.

Current status of the Jordan Research and Test Reactor (JRTR) project

The construction project of the Jordan Research and Test Reactor (JRTR) was successfully completed through the holding of the inauguration ceremony on 7 December 2016 at the JRTR site of the Jordan University of Science and Technology. In addition, the JRTR was finally commissioned on 15 June 2017 and was delivered to the Jordanian Atomic Energy Committee (JAEC), which issued a Taking-Over Certificate (TOC) to the KAER-Daewoo Consortium (KDC). KAERI (Korea Atomic Energy Research Institute) and JAEC signed a separate contract to transfer operation technologies to JAEC. Moreover, both countries have planned for technical co-operation to promote research activities utilising the JRTR, such as the joint development of neutron beam devices.

SMR developments

The SMART (System-integrated Modular Advanced Reactor) Pre-Project Engineering (PPE) design has been underway since 1 December 2015 for building two first-of-a-kind (FOAK) SMART reactors in Saudi Arabia. Outstanding Korean nuclear industries, such as KEPCO E&C, KEPCO NF, and Doosan Heavy Industries & Construction are participating in the SMART PPE project. In parallel with the FOAK engineering design, 41 engineers from K.A.CARE (King Abdulah City for Atomic and Renewable Energy) of the Kingdom of Saudi Arabia have been trained to learn the SMART NSSS system technology since the middle of July 2016 through class room training (CRT), on-the-job training (OJT), and on-the-job participation (OJP). KAERI and K.A.CARE are jointly promoting SMART in energy-related international conferences and exhibitions.

Gen IV systems developments

Design activities of the Prototype Generation IV Sodium-cooled Fast Reactor (PGSFR) are being conducted during the basic design phase. The detailed design of a large integral test loop (STELLA-2) will be finished this year. Ten topical reports covering the PGSFR design, analysis methodologies, and related design codes will be prepared by the end of this year.

To develop and demonstrate key technologies for nuclear hydrogen production by 2030, a new project, "VHTR key technology performance improvement," was initiated for a three-year period in March 2017. The key technologies to be considered are the design analysis codes, thermo-fluid experiments, TRISO fuel (tri-structural isotropic), high-temperature material database and high-temperature heat applications.

Russian Federation

Vision of nuclear energy development

The State Corporation for Atomic Energy "Rosatom" is the only authority in Russia responsible for the use of nuclear technologies. It unites more than 350 enterprises and scientific organisations including all civil companies of the Russian nuclear industry, enterprises of nuclear weapons complex, research organisations and the world's only nuclear icebreaker fleet. Rosatom is the largest generating company in Russia, which provides 40% of electricity in the European part of the country.

Rosatom holds the leading position in the world market of nuclear technologies, occupying 1st place in the world for the number of nuclear power plants being simultaneously build abroad; 2nd rank in uranium reserves and 5th place in terms of its production; 4th place in the world in nuclear electricity generation, providing 40% of the world market of uranium enrichment services and 17% of the nuclear fuel supply market.

Rosatom offers an integrated product that covers NPP life cycle from NPP design and construction to service and fuel supply, and back-end solutions including radioactive waste and spent nuclear fuel management, as well as decommissioning.

Rosatom is diversifying its business through offering products for nuclear non-energy markets (nuclear medicine, isotope production, etc.) and products of non-nuclear technology (3D printers, HTS, etc.).

Combination of these two factors ensures stability of Rosatom's business, allowing it to respond flexibly to market changes.

For advancing stock the corporation is constantly working in three key areas:

- modernisation of existing technologies;
- creation of new technologies for energy markets;
- modernisation of existing and creation of new technologies for non-energy markets.

First direction encompasses development of new projects of NPP with thermal (VVER 3+ VVER-TOI) and fast neutron reactors (BN-1200), small and medium nuclear reactors, gas centrifuges of new generation, light water reactors (LWR) fuel for foreign nuclear power plants (TVS-Kvadrat). Technologies for nuclear power plants decommissioning are being developed.

Rosatom is successfully diversifying into non-nuclear business. Rosatom had made the decision to invest in wind energy. Another direction is nuclear medicine, including deliveries to many countries of a wide spectrum of isotope products, diagnostic equipment and devices for the treatment of various diseases. The Russian nuclear industry is also interested in practically of all the most "fashionable" technology directions – from energy accumulation to artificial intelligence and robotics. Rosatom generates business in the field of additive technologies, creating industrial 3D printers that print in metal, developing nuclear batteries that will be very compact, with a lifetime of tens, if not hundreds of years.

It is also important to say about one of the Rosatom's priorities, namely the closed nuclear fuel cycle (CNFC) on a two-component basis. It assumes that fast reactors (FR) shall be incorporated into an already existing system of thermal nuclear power units. FR in such system will not only generate electricity, but also will contribute to answer the challenges of nuclear energy: to destroy long-lived high activity radioactive wastes and to build-up materials for fuel production. Closed nuclear fuel cycle with fast reactors due to extensive use of ^{238}U fundamentally eliminates the problem of fuel resources exhaustion for nuclear energy, and allows to increase their fuel base by approximately 140 times and that brings nuclear energy on a leading position in energy resources balance.

An additional confirmation of the correctness of the chosen path of nuclear power industry development in Russia is the outcome of the International conference on fast reactors and the respective fuel cycles (FR-17), conducted by the International Atomic Energy Agency (IAEA) with the support of the Russian government and the State Atomic Energy Corporation "Rosatom" in Yekaterinburg in June 2017. The conference was attended by more than 650 participants from 29 countries and 6 international organisations. Panel and breakout sessions covered a wide range of scientific issues and were dedicated to the promising concepts of reactors, active zones, fuels and fuel cycles, operation and decommissioning, security, licensing, construction materials, industrial implementation. Technical tour to the Beloyarsk NPP with BN-600 and BN-800 fast reactor units was of great interest for all the participants of the conference.

On the opening day of the conference participants were shown video messages of the IAEA Director-General Yukiya Amano and Rosatom Director-General Alexey Likhachev, who emphasised that "the future of the world nuclear power is inextricably linked with the closure of the nuclear fuel cycle, integral part of which are 'fast' reactor technology." "This suggests that in the foreseeable future world nuclear power will be a truly renewable source of energy, based on radiation-equivalent usage of fissile materials."

To demonstrate the stable operation of a full range of facilities that ensure the closure of the nuclear fuel cycle a pilot energy complex is being created in the framework of the "Breakthrough" project which includes fast neutron reactor, fabricating – refabricating module of mixed nitride uranium-plutonium fuel and module of spent nuclear fuel processing.

Implementation of a closed nuclear fuel cycle based on FR has a number of advantages:

- exception of accidents requiring evacuation of population;
- resolution of the problem of handling long-lived high-level waste and spent nuclear fuel accumulation, which can remove the restriction associated with the public acceptability of nuclear energy;
- technological support of non-proliferation regime, which can remove the limitation related to the political acceptability of nuclear energy;
- long-term nuclear energy security (thousands of years) with raw fuel material resources, which will improve the competitiveness of nuclear energy.

To ensure further successful development of the technology of two-component nuclear energy system, Rosatom also aims its efforts at creating an International Research Center on the basis of a multipurpose fast neutron research reactor MBIR (IRC MBIR). The MBIR will be the world's most powerful high-flux multipurpose research fast neutron reactor with unique consumer properties. MBIR will allow not only to preserve but also to bring a new level of research capacity for advanced reactor RD&D.

Testing, the timely development and commercialisation of technologies of the closed nuclear fuel cycle will allow to:

- ensure large-scale development of nuclear energy of Russia to the end of the century;
- resolve all issues of nuclear energy in the field of SNF and HLW;
- gradually eliminate the use of nuclear energy in weapons technologies and materials;
- expand of nuclear technologies and services export.

If successful, nuclear energy could play a crucial role in solving today's key challenge for the country – transition of its economy to an innovative sustainable development path.

South Africa

Nuclear legislation, policy and energy planning

The government has reiterated that nuclear power will be procured at a "scale and pace that the country can afford". 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 Department of Energy gazetted a draft revised Integrated Resource Plan (IRP) and integrated energy plan for public consultation in November 2016. In December 2017, on the margins of the Energy Indaba conference, Minister of Energy pronounced the approval of the revised IRP by Cabinet.

The Department of Energy is further amending on several pieces of legislation viz, the Nuclear Energy Act 46 of 1999 and the National Nuclear Regulator Act 47 of 1999. New legislation that is in draft includes the Radioactive Waste Management Fund Bill.

Nuclear new build programme

Eskom (designated procurer and owner-operator for nuclear power plants) and Necsa (designated procurer and owner-operator for front-end fuel cycle facilities and multipurpose reactor) jointly issued an open Request for Information (RFI) in December 2016 with a closing date of 31 January 2017 for expression of interest and 28 April 2017 for the RFI. As at 31 January 2017, 27 companies including major nuclear vendors from China (SNPTC), France (EDF), Russia (Rosatom) and Korea (KEPCO) had responded. Plans were for a competitive procurement process to commence during mid-2017.

On 26 April 2017, the Western Cape High Court issued a judgement against the Department of Energy which impacted and delayed major milestones of the nuclear new build programme. The court set aside some inter-governmental agreements that South Africa entered into with potential vendor countries and the Section 34 Determination made under the Electricity Regulations Act. Since then, the Department of Energy has undertaken a process to review and standardise all the IGAs for further re-negotiation with these countries.

Despite this setback, the South African government remained committed to the roll-out of the nuclear new build programme since nuclear still forms part of the energy mix for the country.

Nuclear safety and licensing

Plant life extension (PLEX) continues for the Koeberg Nuclear Power Station (KNPS). The steam generator replacement as part of the PLEX strategy is intended to extend life of Koeberg to 60 years as well as a 10% thermal power uprate.

Following Eskom's submission of an application for a nuclear installation site licence for the Thyspunt and Duynefontein sites to the National Nuclear Regulator – the NNR completed a preliminary review of the Site Safety Report and a detailed review is ongoing.

Eskom recently achieved one of the major milestones on the siting process for future nuclear power plants. Following an extensive public consultation process under the National Environmental Management Act 102 of 1996, on 11 October 2017 – the

Department of Environmental Affairs issued a positive record of decision (i.e. an environmental authorisation) to Eskom for siting of future nuclear power plants on the Duynefontein brown field.

Following the IAEA Integrated Regulatory Review Service (IRRS) mission, the NNR as part of addressing the mission recommendations is in the process of developing regulations and guidance on long-term operation in anticipation of possible plant life extension application by Eskom for the Koeberg nuclear power plant. Eskom is also in the process of addressing recommendations of the IAEA Safety Aspects of Long Term Operation Mission.

On engineering, maintenance and operations, the Koeberg Nuclear Power Station currently has the best availability factor of above 93%, which is the best in Eskom's fleet of power stations. In 2017 the station achieved its second-best ever duration for a refuelling outage in its quest of achieving short duration outages. During the first quarter of 2017, the World Association of Nuclear Operators (WANO) conducted a peer review at Koeberg, and the station report was the best that it has ever achieved for a WANO Peer Review in all its years of operation.

In order to mitigate the Western Cape's shortage of fresh water as a result of the worst drought in the past 35 years, the station is working towards the construction of desalination plant on its site.

South Africa and the Generation IV International Forum

South Africa remains committed to the Generation IV International Forum (GIF). As you are all aware, the South African Pebble Bed Modular Reactor project which contributed to research and development activities at project level under the Very High Temperature Reactor System Arrangement of GIF was terminated in 2010. The PBMR intellectual property and its assets are currently under care and maintenance status.

South Africa's plan with regard to R&D for high-temperature reactors is subject to Cabinet lifting the care and maintenance status of the PBMR and actually giving a go-ahead on resuscitation of associated activities.

South Africa successfully hosted the GIF 38th Experts Group Meeting and 44th Policy Group Meeting from 16-20 October 2017 in Cape Town.

Nuclear skills development

Phase III of training is currently ongoing under the agreement entered into between South Africa and the South African Civil Nuclear Energy Training Program organised by State Nuclear Power Technology Corporation (SNPTC) of China. This training spans various areas including but not limited to engineering design, project management, commissioning and start-up, module manufacture and construction technology which contributes towards human capacity development with the South Africa nuclear sector.

Two of South Africa's universities, that is the North West University and Wits University completed the International Atomic Energy Agency's (IAEA) International Nuclear Management Academy peer review mission. The implementation of the recommendations will assist in the promotion and fostering of knowledge management in these identified universities and by extension in the entire country and the continent on a wide range of issues related to the peaceful use of nuclear technology.

Switzerland

General decision of Switzerland about nuclear power

In May 2017, the Swiss voters accepted the new Energy Strategy Law that stipulates a phase-out of the nuclear energy production in Switzerland. The new law forbids the construction of new nuclear power plants but allows the operation of the existing plants as long as they are safe. Operation until 60 years of lifetime and beyond is therefore formally possible if the operators make the needed investments. The regulator ENSI checks the safety of the plants.

Operation of the Swiss nuclear power plants

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

BKW Energie AG, the operator of the small BWR unit has confirmed the shutdown of the reactor for the end of 2019 and is preparing the post-operation phase and the dismantling of the facility.

Unit 1 of the Beznau NPP is still in shutdown and is planned to be restarted in spring 2018. The delay for the restart of the reactor results from an extra demand from the safety authorities in the documentation and assessment of the safety relevance of the small defects detected in the pressure vessel material.

In the Leibstadt power plant, the indication of local dry-out on the upper part of some fuel assemblies has resulted into an extended pool investigation during the summer and fall 2016 and a long outage of the plant. The reactor went online again in February 2017 with a maximum power limited to 90%. Further analyses are ongoing and three rods have been delivered to the Swiss hot laboratory at Paul Scherrer Institute for detailed material investigation.

The operators of the power plants are facing further economic difficulties due to the very low market price for the kWh.

The process to find the best site for a deep geological high-level waste repository is going on according to schedule. More detailed studies of three locations in Opalinus clay layers are ongoing. The national association for waste disposal (Nagra) is in charge of the search process in a publicly transparent manner. Nagra will make its provisional site selection around the year 2022, when it will be known where the repository is expected to be constructed. Definitive site selection and the decision of the Federal Council on the general licence are not expected before 2029.

Nuclear power related research in Switzerland

As already stated in the 2016 report, the new Energy Law with the associated phase-out of nuclear power electricity production, does not impact the nuclear research and education in Switzerland. As the centre of competence with a TSO function, the Nuclear Energy and Safety division (NES) of the Paul Scherrer Institute keeps its key mission to maintain nuclear competence in Switzerland. Also very important for NES is to keep the capacity to perform detailed scientific analyses of highly radioactive materials including spent nuclear fuel in the hot laboratory.

The focus of the division is the education of the next generation of nuclear engineers and scientists, the safety of light water reactors (LWR) and the scientific support for the safety of deep geological repositories.

NES continues its activities in the frame of the Gen IV International Forum with research on high-temperature materials for VHTR and GFR, safety studies of molten salt reactors (MSR) and collaboration on ASTRID reactor concept safety. In particular, since the beginning of 2017, NES is co-ordinating the European Sodium Fast Reactor Safety Measures Assessment and Research Tools ESFR-SMART project (H2020-EURATROM programme) stating the broad knowledge of the division in this field. NES has also stated its interest on behalf of the Swiss government to participate at the NEA Nuclear Education, Skills and Technology (NEST) Framework.

United Kingdom

Current UK areas of nuclear research and development

- UK nuclear innovation programme

At the beginning of 2014, the UK government convened the Nuclear Innovation and Research Advisory Board (NIRAB) to provide independent expert advice on the publicly funded research required to underpin government policy. In March 2016, the NIRAB published its recommendations in a report entitled "UK Nuclear Innovation and Research Programme Recommendations". These recommendations were subsequently prioritised in a report entitled "Prioritisation of UK Nuclear Innovation and Research Programme Recommendations". In formulating these recommendations, the NIRAB took into account the previously stated vision of government and industry for nuclear to continue to play a significant and increased role in the UK's energy mix by the middle of the century, noting that this may require the development and deployment of advanced reactor systems different to those currently being built around the world.

Given the long development time and high upfront investment required to commercialise new reactor systems and related fuel cycle infrastructure, the UK government has recognised it can play a role in supporting early stages of research. It announced in 2015 its intention to fund "an ambitious nuclear research and development programme intended to revive the UK's nuclear expertise and position the United Kingdom as a global leader in innovative nuclear technologies. The scope of this new programme has taken into account the recommendations of the NIRAB together with input from a range of other sources including international partners. The first, two-year phase of the UK's Nuclear Innovation Programme has been commissioned. The programme contains several main themes as follows:

Development of advanced nuclear fuels

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, 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 overall aim of a five-year programme is 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 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.

Advanced reactor technologies

In December 2017, the UK Department for Business, Energy and Industrial Strategy (BEIS) established a research initiative for advanced modular reactor (AMR) technologies, which are at an early stage of development and could offer novel functionality, such as high-grade heat, or a step-change in the delivery of low-cost electricity.

For the purposes of this specification the Department for Business, Energy and Industrial Strategy (BEIS) has defined AMRs as being a broad group of advanced nuclear reactors (including Generation IV designs), which differ from the technologies of conventional reactors that utilise pressurised or boiling water for primary cooling purposes. They maximise the use of off-site factory fabrication of modules and target applications that include:

- delivering low-cost electricity;
- increased flexibility (e.g. load-following) in delivering electricity to the grid;
- increased functionality (e.g. heat output for domestic and/or industrial use, facilitate the production of hydrogen);

- providing alternative applications that generate additional revenue or economic growth.

The AMR feasibility and development research initiative will progress through the following two phases:

- Phase 1: funding to undertake a series of feasibility studies for AMR designs.
- Phase 2: subject to UK government approval, up to GBP 40 million may be available for successful selected designs from Phase 1 to undertake applied R&D.

UK nuclear research facilities

2017 saw the remainder of the UK's recent suite of nuclear research facilities developed since 2014. These now include the following:

- High Temperature Research Facility

The High Temperature Research Facility (HTF) was established to investigate, develop and advance structural materials technology for future systems applications such as Generation IV nuclear fission, nuclear fusion, advanced gas turbine materials and other advanced energy concepts.

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.

- The U/Th/Beta-Gamma Active Process Chemistry R&D Laboratory (UTGARD)

The UTGARD Laboratory has been established for the study of chemical processes in support of spent nuclear fuel recycle and waste management. It is part of the National Nuclear Users Facility (NNUF) initiative and is an open access laboratory housed in dedicated facilities at Lancaster University.

The UTGARD Laboratory provides academic and industry users with a unique resource for the study of the chemistry and engineering of spent fuel recycle and waste management processes. With glove boxes for the study of aqueous and non-aqueous samples, it has the capacity to study fully nuclear hydrometallurgical separations processes. It is licensed for work on beta/gamma active fission products, uranium, thorium and low-level alpha tracers.

- The Nuclear Fuel Centre of Excellence (NFCE)

The NFCE is hosted by Manchester University's Dalton Nuclear Institute and the UK's National Nuclear Laboratory (NNL). NFCE's purpose is to create an advanced fuel R&D capability within existing facilities to further the UK's capability in fuel technology. A key focus is on growing UK talent specialising in advanced fuels.

Supported by funding from government to strengthen the existing fuel R&D facilities at NNL and The University of Manchester and create an integrated UK capability, the NFCE builds fuel fabrication and performance experience from decades of research and development on past, present and future fuel types. It will support the creation of improved fuel for current reactors, a new Generation III+ fleet, small modular reactors and, ultimately, fast reactor systems.

- The Materials Research Facility (MRF)

The MRF has been established to analyse material properties in support of both fission and fusion research. It is part of the National Nuclear Users Facility (NNUF) initiative to provide greater accessibility to world-leading research facilities, as a collaborative effort from four complimentary nuclear research hubs within the United Kingdom. The MRF provides academic and industry users with a unique resource for micro-characterisation of materials.

With hot cells for processing and micro-characterisation of neutron-irradiated samples, it has the capacity to cut, polish and encapsulate individual samples up to the tera-Becquerel level for analysis either on-site or back at the user's institute. The facility supports research in lifetime extension, nuclear new build, Generation IV reactor designs and fusion.

- Remote Applications in Challenging Environments (RACE) Research Centre

The RACE Research Centre based at the UKAEA's Culham site, home of the Culham Centre for Fusion Energy and the Joint European Torus. RACE explores many areas of remote operations including inspection, maintenance and decommissioning and is instrumental in developing new remote tools and techniques with academia and industry.

- Materials for Innovative Dispositions from Advanced Separations Laboratory (MIDAS)

The MIDAS laboratory has been established as a part of the National Nuclear User Facility (NNUF). It supports research in the management and disposal of radioactive wastes from the nuclear fuel cycle, providing characterisation and analysis capability for materials. The facility is operated as an open facility, available for use by academic, public and private sector organisations and is staffed by an experienced team of researchers who provide advice and guidance on utilising the equipment. The facility is capable of working with high active alpha and beta/gamma materials.

- The Pyrochemical Reprocessing Laboratory (PRL)

The PRL at the University of Edinburgh provides the facilities to develop and demonstrate integrated pyrochemical reprocessing of nuclear fuel using inactive, fuel-relevant compositional mixtures at laboratory scale, along with the required process monitoring.

The laboratory consists of a suite of interconnected integrated controlled atmosphere dry-boxes, equipped with the necessary furnaces, cell systems and electrochemical and spectroscopic characterisation equipment required for research into and development of each of the essential elements of pyrochemical reprocessing at the laboratory scale. The PRL is an open access laboratory and is affiliated to the NNUF.

- The Sir Henry Royce Institute for Advanced Materials

The Sir Henry Royce Institute is the UK national centre for research and innovation of advanced materials. Its founder members are The University of Manchester, Sheffield, Leeds, Liverpool, Cambridge, Oxford and Imperial College London, as well as the UK Atomic Energy Authority and the UK's National Nuclear Laboratory. The institute's hub is at Manchester University, with activities spread out across founder institutions. Nuclear materials research is one of the nine research areas that form part of the institute's programme and is oriented around on two core areas:

- nuclear fuels and waste streams in the nuclear fuel cycle;
- structural materials for fission and fusion energy.

The institute is developing capability for scientists and industry to prepare, test and analyse radioactive materials for fission and fusion applications and undertake programmes of work on irradiated nuclear materials.

United States

Nuclear energy continues to be a vital part of the United States' energy development strategy for an affordable, secure and reliable energy future. A number of nuclear energy initiatives pursued during 2017 are intended to advance reliable, economical, and emission-free nuclear energy in the electricity market and to encourage the development of advanced reactor designs. On 29 June 2017, at an Energy Week event held at the US Department of Energy (DOE) Headquarters in Washington, President Trump outlined a set of energy initiatives, including a call for a comprehensive nuclear energy policy review.

The DOE's Gateway for Accelerated Innovation in Nuclear (GAIN) continued its efforts to provide the nuclear community with access to technical, regulatory, and financial support necessary to move innovative nuclear energy technologies towards commercialisation while also ensuring the continued safe, reliable and economic operation of the existing nuclear fleet:

- In support of nuclear energy innovation and application of advanced nuclear technologies, DOE provides vouchers to assist applicants seeking access to the world-class expertise and capabilities available across the DOE complex. On 26 June 2017, GAIN announced that it would be providing to 14 businesses, vouchers worth approximately USD 4.2 million. This was a follow-on effort to the previous year in which vouchers totalling USD 2 million were awarded.
- On 13 July 2017, GAIN, in collaboration with Nuclear Science User Facilities (NSUF), conducted a Thermal Hydraulics Workshop at the Idaho National Laboratory (INL). The workshop's objective was to develop a ranked list of thermal-hydraulic research and development needs in the reactor technology areas of light water reactors, fast reactors, high-temperature gas reactors and molten salt reactors.

The DOE's Office of Advanced Reactor Technologies, continues 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. Activities related to molten salt reactor concepts have also increased with commercial interest. The Office of Advanced Reactor Technologies, in collaboration with DOE's National Laboratories, is developing technology roadmaps to identify research needed to reach commercial viability for each of these advanced reactor concepts.

In the area of light water reactors (LWRs), construction of two Westinghouse AP1000 pressurised water reactors at the Alvin W. Vogtle Electric Generating Plant in Georgia continues, with completion of construction expected by 2022. To support the construction of these reactors, Secretary of Energy Rick Perry announced conditional commitments for up to USD 3.7 billion in additional loan guarantees. Construction activities on two other AP1000 reactors in South Carolina were suspended citing numerous regulatory and budgetary challenges.

On 31 May 2017, the US Nuclear Regulatory Commission (NRC) issued a combined licence to Dominion Resources for a GE Economic Simplified Boiling Water Reactor (ESBWR) at Dominion's North Anna unit 3 site in Virginia. Development and licensing of the AP1000 and the ESBWR designs were supported through cost-share arrangements with the DOE's Nuclear Power 2010 programme.

The DOE LWR Sustainability (LWRS) programme focused on conducting research and development on advanced technologies that can improve reliability, sustain safety, and extend the operating life of the current LWR fleet. The LWRS programme has also helped the industry address current economic challenges by introducing new technologies to help gain efficiencies and improve safety. Dominion Resources and Exelon have announced their intention to seek an extension of the operating licences of the Surry plant in Virginia and the Peach Bottom plant in Pennsylvania, respectively, for an additional 20 years, which would mean a total of up to 80 years of operation for these reactors. A final decision by NRC on approving these subsequent licence renewals would likely be made by the early part of the next decade.

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, in April 2017, New York State enacted a law under its Clean Energy Standard which allowed the purchase of zero-emission credits by New York investor-owned utilities and other energy supplies, placing value on the intrinsic carbon-free emissions from nuclear power plants. This recent action by the New York state government prevented the premature shutdown of the Fitzpatrick plant. Additionally, Illinois approved the Future Energy Jobs Bill, which went into effect in June 2017, to provide subsidies to keep its Clinton and Quad Cities nuclear power plants open.

The DOE views small modular reactors (SMRs) as an innovative and emerging technology that can help meet the nation's growing energy demands, providing a safe, affordable option for the replacement of ageing fossil plants. To this end, the DOE SMR Licensing Technical Support (LTS) programme provided cost-shared financial support for the certification and licensing of innovative designs that improve SMR safety, operations and economics. Among SMR LTS programme participants, NuScale Power, LLC made significant progress towards its certification goals, meeting key project milestones such as completion of critical plant component testing and development of plant safety analyses. NuScale submitted its design certification application to the NRC on 12 January 2017. NRC has completed its acceptance review of the application and issued a 42-month review schedule. NuScale has also partnered with Utah Associated Municipal Power Systems (UAMPS) to deploy the first NuScale SMR, for which a preferred site was identified at the Idaho National Laboratory (INL). UAMPS is currently developing its business case to inform its decision to proceed to a combined licence application phase of the project. If favourable, a combined licence application will be developed and submitted to the NRC sometime in the 2019-2020 time frame with commercial operation projected for the mid- to late 2020s. In May 2016, the Tennessee Valley Authority (TVA) submitted to the NRC a technology-neutral early site permit (ESP) application for the development of an SMR project at its Clinch River site in Tennessee. The ESP application, which references a plant parameter envelope encompassing characteristics of all US light water-based SMR designs, was docketed by the NRC on 30 December 2016. TVA is currently focused on the NuScale design as its preferred SMR technology, and has recently begun to work with UAMPS in hopes of sharing efforts and cost towards the development of a reference combined license application.

In the area of advanced reactor technologies, several important actions are underway:

- DOE continues to work with X-Energy LLC and Southern Company through cost-shared awards, finalised in 2016, to support the further development of advanced reactor concepts. X-energy is pursuing a high-temperature gas reactor, and Southern Company is pursuing a molten chloride salt fast reactor.
- With regards to licensing efforts, NRC anticipates publishing the final version of "Guidance for Developing Principal Design Criteria for Non-Light Water Reactors" in early 2018. This regulatory guide (identified as DG-1330) explains how the NRC's "general design criteria" for traditional light water nuclear power plants could be applied to non-light water nuclear reactor design submissions, enabling applicants to develop principal design criteria as part of their regulatory filings.

Another important initiative within the DOE involves the development of accident-tolerant fuels, a next-generation nuclear fuel with higher performance and greater tolerance for off-normal events. These fuels would give operators additional time to respond to conditions, such as those experienced at the Fukushima Daiichi NPP. The congressionally directed programme is framed on a phased approach from feasibility to qualification and is executed through strong partnerships between 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 INL Advanced Test Reactor and the Halden Reactor in Norway. Many US nuclear utilities are interested in accelerating the development and use of accident-tolerant fuel concepts and have arranged with the industrial research teams to install lead test rods in commercial reactors as early as 2018, four years sooner than originally planned.

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 2017, DOE made 62 awards totalling USD 50 million for nuclear energy research and infrastructure enhancements through the Consolidated Innovative Nuclear Research Funding Opportunity Announcement.

As the DOE strives to meet the challenges of energy security in safe and economically viable ways, the United States will rely heavily upon nuclear energy as a key element in modernising the US energy portfolio.

CHAPTER 3

System reports

This chapter gives a detailed overview of the achievements made during 2017 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 capable of break-even core, 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. The core consists of an assembly of hexagonal fuel elements, each consisting of ceramic-clad, mixed-carbide-fuelled pins contained within a ceramic hex-tube. The favoured material at the moment for the pin clad and hextubes is silicon carbide fibre reinforced silicon carbide (SiCf/SiC).

The whole primary circuit with three loops is contained within a secondary pressure boundary, the guard containment. The produced heat will be converted into electricity in the indirect combined cycle with three gas turbines and one steam turbine. The cycle efficiency is approximately 48%. A heat exchanger transfers the heat from the primary helium coolant to a secondary gas cycle containing a helium-nitrogen mixture which, in turn drives a closed-cycle gas turbine. The waste heat from the gas turbine exhaust is used to raise steam in a steam generator which is then used to drive a steam turbine. Such a combined cycle is common practice in natural gas-fired power plants so represents an established technology, with the only difference in the GFR case being the use of a closed-cycle gas turbine.

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 MW_{th} 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 the Slovak Republic. The "V4G4 Centre of Excellence" is a legal body registered in the 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, etc.).

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

After legal establishment of V4G4 Centre of Excellence the EU VINCO project (Visegrad Initiative for Nuclear Cooperation) has been proposed to launch capacity building in nuclear technologies in Central European countries. The main objective of VINCO project (Horizon 2020) is to conduct a variety of capacity building activities aiming at strengthening the co-ordinating role of the V4G4 Centre of Excellence and supporting its member organisations.

One of the activities carried out in the frame of VINCO is covered by WP3 and devoted to learning exercises via mutual studies for gas-cooled reactors. The WP3 is aimed at sharing knowledge and mutual learning and exchange of scientific staff between the laboratories of the V4G4 Centre of Excellence and CEA (France).

The collaboration mechanism includes activities oriented on experienced employees of participating institutes and is focused on the establishment of co-operation rules and test case analyses related to the key problems of gas-cooled reactors and to better understanding the basis for the gas-cooling systems (GFR and VHTR) reactors.

The above goal is realised through VINCO benchmark learning exercise devoted to development of computer models as well as the efficient use of various calculation tools utilised by different users.

The main objectives of benchmark exercises is to assess if the present system codes are able to reproduce correctly the gas system dynamics as well as to support the development of consistent models of gas-cooled reactors in the different organisations.

For the benchmark exercise the ALLEGRO demonstrator concept developed in co-operation of CEA France and V4G4 has been selected as a reference GFR unit. The ALLEGRO Project Coordination Team (PCT) has been created to conduct technical meetings on regular basis as a platform for the knowledge and experience exchange among the V4G4 members and CEA.

Thermal-hydraulic benchmark activities

As a starting point the ALLEGRO demonstrator model for CATHARE2 code developed by CEA in 2009 has been used by all participants. CEA ALLEGRO CATHARE2 model had not been further developed since then and did not include all the modifications requested in the thermal-hydraulic benchmark specification (e.g. modified size of reactor pressure vessel, core radial heat transfer, core flow distribution, core power distribution, decay heat removal (DHR) secondary side and many others). Therefore all the participants using CATHARE2 code has been asked to perform modifications by their own generating four new CATHARE2 ALLEGRO national models.

In addition VUJE and ÚJV developed new ALLEGRO thermal-hydraulic models for RELAP5-3D and MELCOR codes.

In order to encourage participants to perform requested modifications the part of the qualification procedure has been proposed to harmonise the key characteristics of ALLEGRO demonstrator focusing on harmonisation of global volumes, passive and active heat exchange areas, mass of the structures in the core, pressure losses along the ALLEGRO system loops and key initial conditions of the steady state. Additional output of the activity was to identify the distortions among the models and use this information to explain the differences observed in transient calculations.

Having independent and harmonised ALLEGRO models for CATHARE2 code and new ALLEGRO models for RELAP5-3D and MELCOR code the further step was to perform the station blackout and loss-of-coolant accident (LOCA) transient analyses as specified in the benchmark definition. The results has been compared among the participants and evaluated from qualitative point of view.

The phenomena characterising each type of transient have been specified and the evaluation was focused on the prediction of such phenomena emphasising and explaining the observed differences among compared calculations.

The three-inch LOCA calculations have predicted all the expected phenomena typical for this type of accident. Discrepancies among the models were related mostly to different models of water to air heat exchanger. The VUJE RELAP5-3D model of water to air heat exchanger was far more effective than CATHARE2 models. This produced discrepancies in prediction of feed water temperature of MHX affecting core inlet (outlet) and fuel cladding temperature. This raised the question how to properly model water to air heat exchanger using aluminium finned tubes. Discrepancy in MHX feed water temperature was observed in MELCOR model related to initial temperature of the cooling air. The MELCOR and RELAP5-3D predictions with respect to MHX feed water temperature drop were comparable in the first phase. Another discrepancy was observed in the prediction of guard vessel temperature and pressure in RELAP5-3D model linked with the different heat transfer between gas mixture and internal guard vessel structures.

The station blackout calculations have predicted all physical phenomena typical for the total loss of electricity supply. The major difference was linked with the prediction of the core flow rate during the natural convection phase. The calculations clearly showed the sensitive dependence between the core flow and peak cladding temperature. The different core flow rates are determined by the different calculated pressure losses in the core, primary and secondary DHR circuit. In MTA-EK the negative water flowrate in DHR secondary system has been observed. This phenomenon is linked with the U tube design of the DHR heat exchanger and with not used circulation pump during nominal operation possibly establishing water temperature gradient in hot and cold leg of DHR secondary system. The temperatures in secondary system are unified in both hot and cold legs causing uncertainty in direction which natural circulation develops. The outcome can be reflected in the further design and operating measures of the DHR system.

The main outcome of the thermal-hydraulic benchmark exercise was the creation of the functional international team having members from Central European Visegrad countries able to conduct further safety studies by the dedicated thermal-hydraulic models for RELAP5-3D, CATHARE2 and MELCOR codes supporting further development and design activities related to GFR ALLEGRO demonstrator.

Neutronic benchmarks activities

The main objective of neutronic benchmark exercise was to assess applicability of existing neutronic code systems for static and dynamic characterisation of the gas-cooled reactor core. Up to now last part of GFR neutronic benchmarking – fuel assembly oriented benchmark was based on experience from previous benchmarks.

Reactor physics benchmark – detailed calculation of the whole ALLEGRO reactor (GFR demonstrator) core – was specified in the ESNII+ Project (FP7). The goal of the calculation exercise was to verify the reactor physics codes, namely to get information about the modelling uncertainties. The obtained deviations between the participants are characterising the user effects, the modelling uncertainties and the influence of the nuclear data differences altogether, without the possibility of their separation because of the complexity of the benchmark problem.

To enable identification of the deviation reasons simpler problem – an infinite regular lattice problem with burnup and with leakage represented by fixed buckling was defined and solved successfully as the first part of a neutronic benchmark in the VINCO project.

Second part of VINCO neutronic benchmark was oriented on burnup of 2D numerical models of ALLEGRO fuel assembly at infinite lattice without fixed buckling and without critical spectrum, with MOX and UOX fuel. More complex definition (assemblies) is much closer to usual utilisation of tested codes – preparation of libraries for macrocodes. Exclusion of critical spectrum calculations (both for burnup and for reactivity effects) enables more detailed comparison of deterministic and stochastic calculations. Assembly oriented benchmark includes progressive features – approximation of realistic temperature distribution and wide set of results: infinite multiplication factor, concentrations of key actinides, reactivity coefficients, kinetic parameters and transport cross section.

Nine solutions prepared at five participating organisations by four code complexes (ERANOS, SERPENT, HELIOS, SCALE) were included into comparison. Because of huge amount of results benchmark evaluation was divided into four categories as follows:

- solutions prepared with the ENDF library;
- solutions prepared with JEF library;
- library effect;
- buckling selection effect (zero or critical).

Comparison of ENDF and of JEF solutions shows, that differences at multiplication coefficient curves are significant. Not negligible are also differences at key actinides concentrations, prepared with JEF library for UOX fuel. Concerning reactivity coefficients (RC) differences are acceptable for Doppler RC, significant for pellet expansion RC and huge for He dilution RC. Differences are also very high for prompt-neutron lifetime (both libraries) and delayed neutron fraction (ENDF).

Evaluation of library effect was performed by comparison of two SERPENT calculations with identical inputs but different cross-section libraries – ENDF and JEF. Comparison shows, that library selection influences significantly infinite multiplication coefficient, concentration of a few actinides and delayed neutron fraction and very strongly He dilution reactivity coefficient.

Influence of buckling selection (zero or critical) was analysed by calculation by selected code and library with identical input but various buckling selections. This procedure was performed with ERANOS + JEF combination and with HELIOS + ENDF combination. In general, buckling selection has significant influence on infinite multiplication coefficient, some actinide concentrations and He dilution reactivity coefficient.

Obtained results of neutronic benchmark indicate possible fields of improvement at codes, methods and libraries. It should be confirmed by consequent analyses. Important benchmark result is also improved co-operation of participating teams and organisations.

UOX feasibility

Because of possible better availability of UOX fuel for first ALLEGRO cores feasibility study is going on with recent limits as follows:

- power density 50 W/cm^3;
- uranium enrichment not higher than 19.5%;
- burnup reserve 3% (k_{eff}^{BOC} ~1.03);
- reduction of burnup reserve at UOX core;
- EOC reactivity at UOX core the same as at MOX core (cycle length 660 FPD).

Usual core enlargement methods – addition of assemblies (rings), axial enlargement and fuel pellet diameter increase are taken into account. Two institutes – VUJE and ÚJV Řež – are involved in the activity. Partial conclusions state possible UOX utilisation at ALLEGRO at two different ways:

- addition of one ring of fuel assemblies, thicker fuel pin and axial enlargement; pin diameter enlargement have not negative influence on centreline temperature and heat transfer from the cladding at normal operation;

- addition of two rings of fuel assemblies, without change of radial geometry, with proportional axial enlargement; irradiation ability can be improved by changing the radial profile of enrichment.

3.2. Lead-cooled fast reactor (LFR)

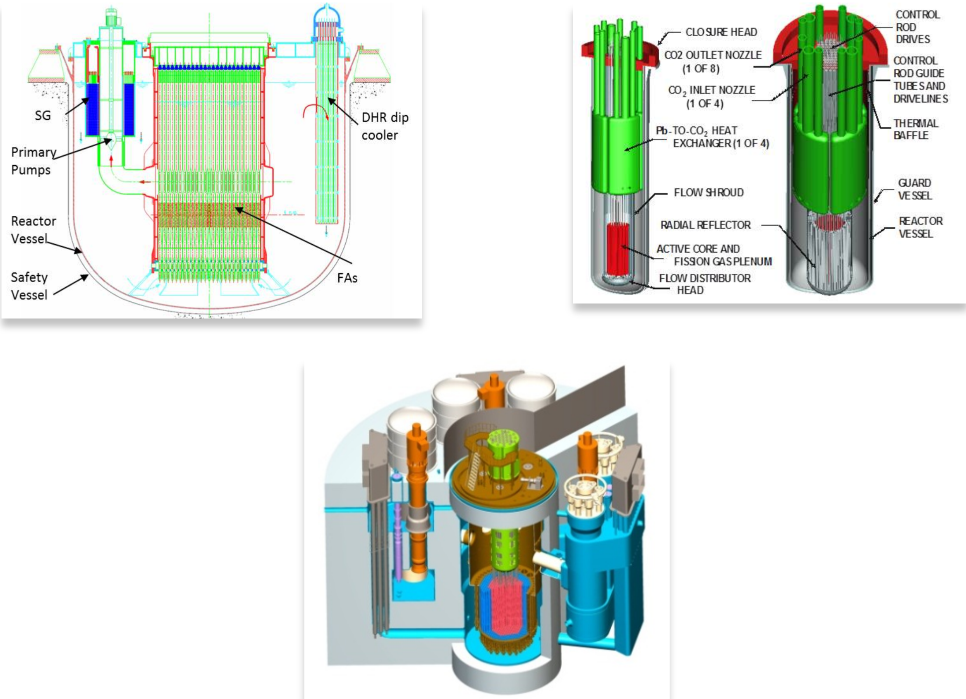
Main characteristics of the system

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 fissile burner/breeder, also with thorium matrices. An important feature of the LFR is the outstanding safety and design simplification that result from the fact that lead is a relatively inert coolant with a very high boiling point and the ability to operate at near atmospheric pressure. These systems have the potential to provide for the electricity needs of remote or isolated sites or to serve as large interconnected 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-OD-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.

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

Figure 3.1: Sketches of GIF LFR Reference Systems: ELFR, BREST-OD-300 and SSTAR



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 (LBE) 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 2025. Following the reformulation of the GIF LFR pSSC in 2012, the SRP was completely revised, and the issuance of the revised plan is expected early in 2018.

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.

Table 3.1: Key design parameters of the GIF LFR concepts

Parameters	ELFR	BREST-OD-300	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

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 a System Agreement is signed: system integration and assessment; lead technology and materials; system and components design and fuel development.

Main activities and outcomes

Two of the GIF LFR pSSC meetings took place in 2017. The first was held in Paris on 21-22 March. It was hosted by NEA in their new offices in Boulogne-Billancourt and the second was held in Seoul, Korea in conjunction with the first Global Symposium on Lead and Lead Alloy Cooled Nuclear Energy Science and Technology (GLANST) which was jointly organised and conducted by Seoul National University, the Korea Radioactive Waste Society and the GIF LFR pSSC on 7-8 September 2017.

The first meeting was characterised by the attendance of industrial representatives from LeadCold, Hydromine and Westinghouse, three companies actively developing new LFR concepts. The presentations by the representatives of these industrial organisations complemented the presentations of committee members who described their respective national programmes with detailed presentations and discussions of industrial development related to their specific LFR concepts. A very active discussion followed these presentations.

The second meeting in Seoul was characterised by the presentations on the status of activities in MOU signatories and observer countries. These presentations were embedded in the GLANST conference programme. The presentations were made available to the public as part of the conference programme and were included in the published proceedings. A closed session of the committee was then held on 9 September to discuss additional internal business of the LFR pSSC. The GLANST conference successfully demonstrated the rapidly growing worldwide interest in LFR technology by virtue of participation by a large number of delegates from countries throughout the world. The next GLANST conference has been announced to be held in Europe in 2021.

In 2017 the internal activities of the LFR pSSC have been centred on top level reports for GIF. After the issuance of the LFR White Paper on Safety in collaboration with the GIF RSWG, the pSSC has been 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 realised that the IAEA SSR-2/1 (on which the SFR SDC was based) did not require many of the features identified for the SFR to be adopted for the LFR due to fundamental differences between the two LMFR technologies (note additionally that IAEA SSR-2/1 refers primarily 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 was then updated taking into account such inputs and was completely revised to comply with the new version of IAEA SSR-2/1 issued at the end of 2016. The report is presently under internal review and a revised version is expected to be provided to and discussed with RSWG at the beginning of 2018.
- ***LFR System Safety Assessment***: In 2014, the RSWG asked the SSC chairs to develop a report on their systems to analyse them systematically, assess their safety level and identify further safety-related R&D needs. The initial LFR assessment report was prepared by the LFR pSSC and a revision of the report addressing comments from the RSWG is now in preparation. Detailed discussions are expected to take place at the beginning of 2018, with the objective of bringing the report to a final and agreed form.
- ***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 will be used as a basis for the development of the corresponding LFR-SDG report. On the other hand the LFR pSSC decided to postpone the compilation of the report after the issuance of LFR SDC.

- ***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 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.

- ***Co-operation Agreement Euratom-Rosatom***: Following the signature in May 2014 of a Cooperation Agreement (CooA) between the BREST and LEADER projects, by NIKIET (on behalf of Rosatom) and Ansaldo (on behalf of the LEADER consortium), two dedicated meetings were organised and conducted. Presently the two organisations (Nikiet and Ansaldo) are discussing the possibility for a renewal of the Cooperation Agreement.

- ***US/Euratom new LFR INERI project***: In conjunction with the regular LFR pSSC meeting, a new INERI project was started in March 2017. The title of the project is: "Small Modular Lead-cooled Fast Reactors in regional energy markets: safety, security, and economic assessments". The project envisages collaboration between a US DOE-sponsored organisation, in this case the Naval Postgraduate School, Monterey, CA, and Euratom R&D and industrial organisations, led by the JRC. Other key organisations involved are: Argonne National Laboratory (ANL), Lawrence Livermore National Laboratory (LLNL), Ansaldo Nucleare, ENEA, RATEN-ICN, SCK•CEN, Hydromine, Westinghouse. This joint US/Euratom project is investigating the feasibility and assessing the potential deployment of small modular lead-cooled fast reactors in regional energy markets and for insular applications. An INERI programme review was held in Oak Ridge, Tennessee, in July 2017, and an INERI Joint Project Plan was completed in September 2017.

- ***IAEA FR-17 Conference, Ekaterinburg (RU)***: The LFR community was strongly represented at this very important event for fast reactors. Several papers were presented by our Russian Colleagues with topics ranging from fuel cycle, experimental facilities and results as well activities related to LFR design and licensing. Additionally, many researchers participated from Europe, Korea, China and the United States and presented their work showing the increasing worldwide interest in HLM technology.

- ***GLANST, Seoul (KR)***: The GIF LFR pSSC supported the organisation of the first Global Symposium on Lead and Lead Alloy Cooled Nuclear Energy Science and Technology (GLANST) in September 2017 in Seoul. More information on the event can be found in the dedicated section on the status of activities in Korea.

Main activities in Russia

An innovative fast reactor BREST-OD-300 with inherent safety is being developed as a pilot and demonstration prototype for the basic commercial reactor facilities of future nuclear power with a closed nuclear fuel cycle. The main goals of the system are:

- elimination of nuclear accidents requiring evacuation, i.e. elimination of emergency planning zone with no resettlement of public;

- closure of the nuclear fuel cycle (NFC) for the full use of the energy potential of uranium raw material;

- reduction of the produced waste to radiation-equivalent (relative to natural raw materials) RW disposal;

- technological strengthening of non-proliferation regime (i.e. no uranium enrichment for nuclear power, elimination of core blanket weapon-grade plutonium production or extraction during SNF processing, and substantial reduction of nuclear material transportation volumes);
- ensuring competitiveness in comparison with other energy generation types.

The lead coolant properties make it possible to implement in fast reactors the following:

- in combination with application of (U-Pu)N fuel, complete breeding of fissile materials in the reactor core while maintaining a constant small reactivity margin to prevent the potentially disastrous effects of an uncontrolled power increase as a result of unintended introduction of the reactivity margin due to equipment failures or personnel errors;
- avoidance of the void reactivity effect due to the very high boiling point and the high density of lead;
- prevention of 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;
- reduction of the possibility of fuel damage as a result of the high heat capacity of the coolant circuit;
- flattening the FA power distribution and reduction of the peak fuel pin temperatures respectively, with corresponding safety improvement by taking into account the high density of lead and its albedo properties;
- greater time lags of the transient processes in the circuit, which make it possible to reduce 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 confirmation of the foundations on which these approaches are based to ensure inherent safety.

Special attention in the reactor development is paid to justification 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 from two previously withdrawn FAs are being subjected to post-irradiation studies. Seven FAs with nitride fuel elements are currently being irradiated in the BOR-60 research reactor.

In designing the reactor core components, 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 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 FAs for 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. Mechanical tests included transverse bending, torsion, axial tension and compression. Vibration tests were conducted using running and stationary water. Vibration tests were also performed in air. Hydraulic tests of FA mock-ups were conducted using lead coolant.

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

A mock-up of a 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 and 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.

The absence of data from physical experiments with nitride fuel led to the necessity of carrying out additional experimentation using the BFS critical facility (Figure 3.3). In the associated simulations 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 neutronics calculations. The results of the calculations carried out using the validated software tools sow 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.

Figure 3.2: Full-scale FA mock-up and FA mock-up with a retort for testing

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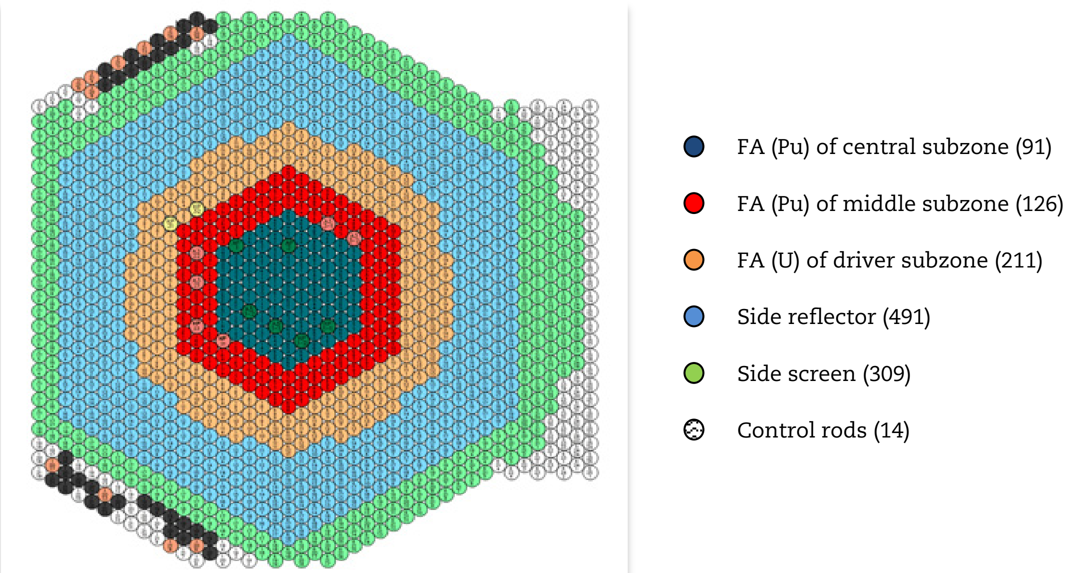
achieve a small reactivity margin during the reactor operation and provision of a practically stable power density field during the duration of the fuel lifetime.

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.

A wide range of calculations and experimental studies were required to confirm the serviceability of such a vessel type (Figure 3.4), 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 3.5), heating-up modes have been optimised, and the gas emission parameters have been determined. 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.

Figure 3.3: Map of BFS critical assembly with BREST-type fuel composition



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.

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 3.6), it was demonstrated that it is impossible for a single SG tube rupture to develop into a multiple tube rupture (dependent rupture exclusion).

Figure 3.4: Distribution of first primary stresses σ_1 in concrete filler of reactor vessel by the end of heating-up

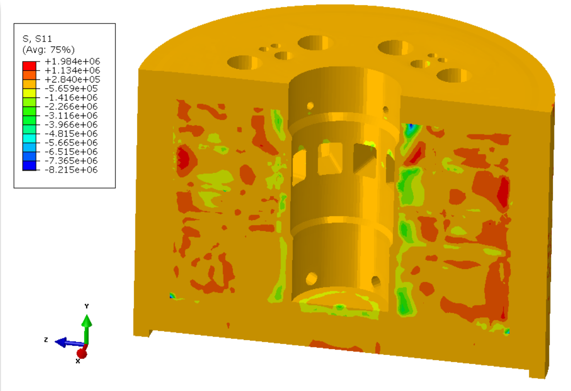


Figure 3.5: Full-scale mock-up of reactor vessel's central part

Figure 3.6: Tube rupture experiment

The reactor main coolant pump (MCP) 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 MCP mock-up have been developed;
- the flow characteristics of the lead coolant flow path have been obtained for levels up to 80% of the required flow (less than 100% due to 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 (Figure 3.7), 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}, ^{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 $8.65{\cdot}10^{-9}$ 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 is being carried forward. The next stages include completion of planned R&D, construction and operation of the BREST-OD-300 power unit as a part of the pilot and demonstration of the full energy complex.

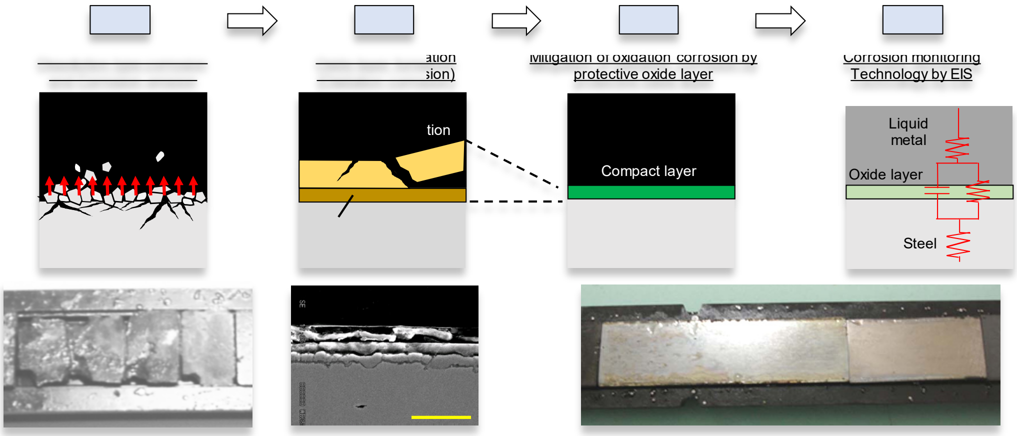
Figure 3.7: Ex-reactor loop facility with lead coolant and gas circuit

Main activities in Japan

Fundamental experimental and theoretical studies for the LFR have been carried out by the Tokyo Institute of Technology.

In the material studies, material compatibility investigations for the LFR has been pursued. The corrosion characteristics of 13 kinds of steels (e.g. 316-type austenitic steel, 9Cr martensitic steels, 12Cr martensitic steels, Si-rich martensitic steel, Al-rich ferritic steels and 18Cr ferritic steel) have been investigated by means of corrosion tests with a non-isothermal type forced convection loop. The compatibility studies could be summarised into four stages as shown in Figure 3.8. First, it was found that the occurrence of severe corrosion-erosion on the steels was induced by the destruction of their corroded surfaces in flowing Pb-Bi at low oxygen concentration. However, this severe corrosion-erosion could be suppressed by the formation of protective oxide layers on the steel surfaces in the flowing Pb-Bi and then the corrosion losses were greatly mitigated, if the oxygen concentration was adequately controlled in the flowing Pb-Bi. The formation of Si or Al-rich oxide layers, which had excellent stability, was effective in protecting the steel surfaces in the flowing Pb-Bi for long-term duration.

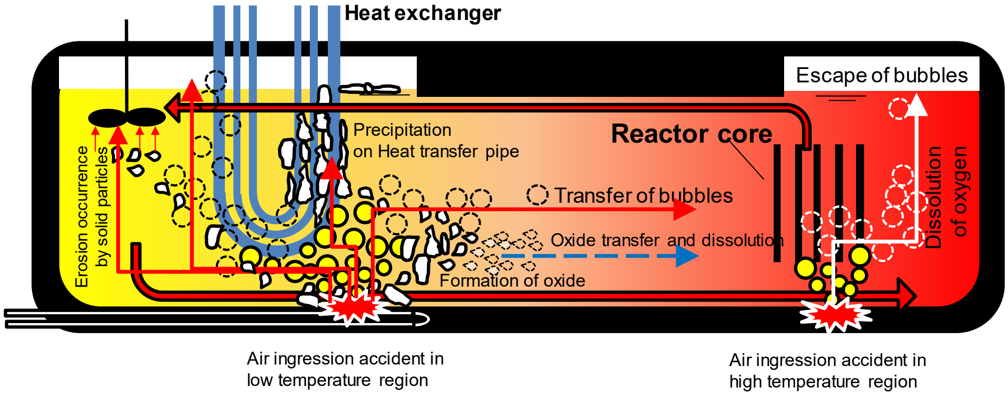
Figure 3.8: Staged approach of material compatibility studies at the Tokyo Institute of Technology



The oxygen sensor is one of the essential technologies for corrosion mitigation. The performance of solid electrolyte oxygen sensors was improved as a result of refinement of the sensor structure. The in situ corrosion monitor is also one of the key technologies. An in situ corrosion monitor was developed based on electrochemical impedance spectroscopy (EIS). The properties and the effectiveness of the oxide layers were analysed in situ in static Pb. The EIS signals indicated that there were changes in the layer thickness related to the growth of the oxide layers in the liquid metal. Crack initiation and propagation in the protective oxide layers were also detected by the change of electrical resistance and capacitance in the EIS signals.

Accident tolerance is one of the important features of Pb-based coolants. The Pb-Bi coolant does not rapidly react with air in the case of an air ingress accident. However, the coolant can be oxidised by air ingress and the resulting chemical characteristics can be changed as a result of this oxidation. The possibility of coolant oxidation and various other coolant behaviours in the reactor system was investigated, and coolant oxidation only in the low temperature region was indicated as shown in Figure 3.9. Oxidation tests were also performed to investigate the mechanism of oxidation in Pb-based coolants. The test results indicated that PbO was preferentially formed in Pb-Bi alloys, and that Pb was depleted from the alloys by the oxidation. The ternary oxide of Pb-Bi-O and Bi_2O_3 were formed only after enrichment of Bi in the alloys due to the Pb depletion.

Figure 3.9: Chemical and physical behaviours of LBE coolant after air ingress accident



In the theoretical study, innovative LFR concepts have been studied. The use of Lead alloy as a coolant can make the neutron economy good in fast reactors. By using this characteristic, Breed and Burn reactor concepts and CANDLE burning reactor concepts have been studied. Those reactors need only natural uranium or depleted uranium for the fuel once they come into an equilibrium condition. It is also possible to achieve high burnup of fuel, up to 40%. Studies have been performed to solve the problem of fuel integrity in high burnup, the design of initial cores to start-up the reactor, and the design of reactor cores with new concepts.

Main activities in Korea

The new Government of the Republic of Korea started in May 2017 with a programme of renewable energy expansion from the present level of about 3% to 20% by 2030. The energy transition policy of the new government, however, has met with strong criticism primarily from academia. Gradually, the public has joined expert opinion with respect to nuclear energy in urging the government to revisit the policy through public debate. New reactor builds have been restarted after the overwhelming approval from the public. Nuclear fuel cycle R&D is receiving scrutiny in the public debate. Meanwhile, the emphasis of government R&D has moved towards the decommissioning and safety reinforcement of operating plants, while the non-electricity applications of nuclear technology has been greatly expanded.

Under this situation, LFR R&D has been redirected towards marine propulsion and space power development, by taking advantage of the excellent safety, very long refuelling intervals and economy of LFR. LFR R&D progress has been made mainly within university programmes during the past 20 years, since the first Korean study begun in 1996 at Seoul National University. LFR R&D has expanded into the Ulsan National Institute of Science and Technology (UNIST), the Korea Advanced Institute of Science and Technology (KAIST), Pohang Institute of Science and Technology (PosTech) as well as Sungkyunkwan University (SKKU).

The Korean LFR Program has presently three main objectives:

- Micro-modular reactors for marine propulsion, including ice-breakers for transporting natural gas from arctic Russian production centres. It is envisaged that such propulsion application can be expanded to container ships and other remote station applications.
- A technology development requirement for sustainable power generation using energy produced during nuclear waste transmutation has been reformulated towards increased safety.

- 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 safety margin against worst case scenarios such that an emergency planning zone size can be as small as 100 metres.

To meet the first goal, a compact micro-modular reactor called HARMONIUM has been designed based on URANUS as the reference. HARMONIUM has innovative features including compact core with the help of pony pumps and the use of supercritical CO_2 cycle on the secondary side while keeping the reactor core life of 30 years covering the entire life cycle of ice-breakers and container ships without refuelling.

To meet the second goal, the Korean first LFR-based burner PEACER (Proliferation-resistant Environment-friendly Accident-tolerant Continual-energy Economical Reactor) 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 TORIA (Thorium Optimized Radioisotope Incineration Arena) with the leadership of the Nuclear Transmutation Energy Research Centre of Korea (NUTRECK) at Seoul National University.

As a part of the second goal Korea has also started to develop PASCAR (Proliferation-resistant, Accident-tolerant, Self-supported, Capsular and Assured Reactor) for 20-year operation without on-site refuelling. Recently the Korean government-funded international collaborative R&D has been completed and resulted in design improvements and materials development for URANUS (Ubiquitous, Rugged, Accident-forgiving, Non-proliferating and Ultra-lasting Sustainer).

PEACER (Proliferation-resistant Environment-friendly Accident-tolerant Continual-energy Economical Reactor) is a Pb-Bi cooled fast reactor being developed at NUTRECK at 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 (MCP), and using the Rankine cycle for power generation. As with other similar Pb-Bi cooled fast reactor concepts, the operating coolant temperature of PEACER spans over 300-400°C to assure corrosion-resistant conditions over the entire reactor lifetime.

PEACER family 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 by taking advantage of chemically inert coolant. The steam at the turbine inlet is superheated to 360°C at 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 including the innovative PyroGreen technology 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 final residual wastes extracted from the last cycle of SFR operation can be transmuted in one unit of

TORIA that has less than 100 MW of nuclear power. All the 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). To meet the third goal for distributed power stations, URANUS has been developed. Based on the PEACER design, a small proliferation-resistant transportable power capsule designated as PASCAR has been developed at NUTRECK by capitalising on outstanding natural circulation and chemical 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 pumps, while enriched uranium dioxide fuel can be used for near-term applications. Recently the core design has been changed to use fresh enriched UO_2 fuel rods in a hexagonal geometry. Like the PASCAR design, URANUS is targeted for 30 years of operation without on-site refuelling at an electric power up to 100 MW and 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 1) three-dimensional neutronic and thermal-hydraulic analysis code validation, 2) corrosion-resistant (FGC) functionally graded composite) materials production, and 3) 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.

FGC materials. As part of the material development, a group of researchers designed a FGC tube pilgering process using three-dimensional finite element analysis (FEA). In this study, it was shown that the curvature and plastic strain are developed on the rolled product with the same roll speed and the 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.

Large Scale Thermal hydraulic Test Systems. The first large-scale LFR test facility in Korea, HELIOS, has been moved from SNU to the Ulsan National Institute of Science and Technology (UNIST) where a new LFR development programme has been started with the government support. At SNU, a new mock-up, designated as PILLAR (Pool-type Integral Leading test facility for Lead-Alloy-cooled small modular Reactor), has been designed, built and operated since 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.

Launch of GLANST at SNU. The new Global Symposium on Lead and Lead Alloy Cooled Nuclear Energy Science and Technology (GLANST) was launched in September 2017 with a five-year interval in order to provide an additional forum in parallel with the already successful HLMC, with the sponsorship of the Generation IV International Forum (GIF) provisional System Steering Committee for the Lead Fast Reactor (LFR), the Korean Radioactive Waste Society and the Korea National Research Foundation. On this basis, the Scientific Committee of GLANST, including key members of the GIF LFR pSSC as well as key members of HLMC, organised and convened the GLANST-2017 conference. The inaugural conference held at SNU was well received with the participation of about 50 members from GIF member states with keynote speakers invited from the United States, Russia and Korea.

Main activities in Euratom

Following the signature of the FALCON (Fostering ALfred CONstruction) Consortium Agreement in December 2013 by Ansaldo, ENEA (Italy) and ICN (Romania) a new text of the consortium agreement has been discussed during 2017 and the final formal signature of the new FALCON consortium is expected to take place at the beginning of 2018. The main motivation for this new formulation is the opening of possible participation to new partners spread not only within the European Community but also internationally.

In 2017, 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) configurations; (iii) evaluation of different options for primary pumps; (iv) integration of a new decay heat removal (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. Design activities for a test facility of the DHR anti-freezing system (SIRIO) have been started benefiting from a grant of the Italian government and the construction of the facility is expected to start at the beginning of 2018.

During 2017 a new facility has been commissioned for conducting pre-normative, separate effect tests of candidate structural materials for lead-cooled fast reactors (LFRs) inside realistic environmental conditions in temperatures up to 650°C. The facility is a part of the JRC's Liquid Lead Laboratory (LILLA).

Concerning the Steam Generator Tube Rupture Event (SGTR), in the frame of MAXSIMA Project (Methodology, Analysis and Experiments for the "Safety In MYRRHA Assessment" – Euratom H2020) an experimental campaign of four runs, investigating heavy liquid metal-water interaction, in a large configuration, was carried out at the CIRCE facility in 2017. Experimental runs provided new verifications that no propagation of the rupture to the surrounding tubes occurred (no domino effect) during the tests. Post-test analysis was able to predict pressure and temperature time trends in agreement with experimental data, providing a contribution to code validation for water-HLM interaction scenarios in a large pool facility. The analyses performed provided the evidence that a suitable design of a depressurisation system (e.g. rupture discs) could allow for the mitigation of the postulated SGTR event in heavy liquid metal nuclear systems with confidence and safety. Figures 3.10 and 3.11 illustrate the facilities and the results of the SGTR experimental campaign.

Figure 3.10: SGTR Test Section in CIRCE facility (MAXSIMA Project, H2020)

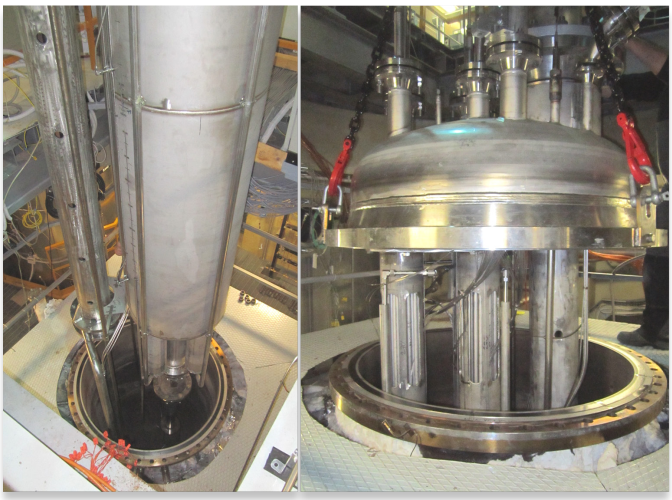
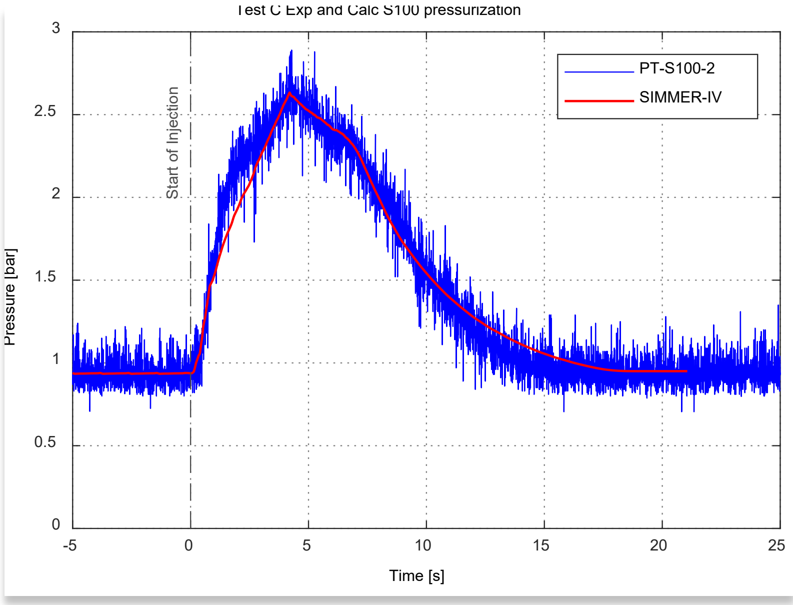


Figure 3.11: SGTR Experiment in CIRCE facility (MAXSIMA Project, H2020). Calculated and experimental pressure time trends in the main vessel cover gas



As regards the projects co-funded by the Euratom H2020 programme, SESAME (Thermal-hydraulics simulations and experiments for the safety assessment of metal-cooled reactors) and MYRTE (MYRRHA research and transmutation endeavour) continue their respective R&D activities with activities co-ordinated through the conduct of joint meetings to discuss and report progress. In 2017, three new collaborative projects of interest for Generation IV and LFR technology have been funded and already launched:

- GEMMA: materials for Gen IV LFRs, with a total budget of EUR 6.6 M and co-ordinated by ENEA-Italy, started in June 2017;
- INSPYRE: fuel for FRs, with a total budget of 9.4 M€ and co-ordinated by CEA France, started in September 2017;
- M4F: materials for Gen IV and fusion, with a total budget of EUR 6.5 M and co-ordinated by CIEMAT-Spain, started in September 2017.

A new Euratom H2020 call for project proposals has been published at the end of October 2017. In the call, project proposals related to safety and severe accident simulations of Gen IV reactors as well as projects related to innovation aspect of nuclear safety are sought. The call will end in September 2018 and the first projects are expected to start in the beginning of 2019.

Main activities in China (observer)

In China, the Chinese government has provided continuous national support to develop lead-based reactors technology since 1986, by the Chinese Academy of Sciences (CAS), the Minister of Science and Technology, the NSF, etc. Following the last 30 years of research on lead-based reactors, the China Lead-based Reactor (CLEAR) was selected as the reference reactor for both ADS and fast reactor systems, and the program is being carried out by the Institute of Nuclear Energy Safety Technology (INEST/FDS Team), CAS. The activities on CLEAR reactor design, reactor safety assessment, design and analysis software development, lead-bismuth experiment loop, key technologies and components R&D activities are being carried out.

Several "13th Five-Year" plans by government related to lead-based reactor were published. CLEAR-M project aim at construction of small module energy supply system has been lunched as one of these plans. The engineering design for the first step as prototype mini-reactor CLEAR-M10a with 10 MWth were carried out.

For ADS system, several concepts and related technologies are under feasibility assessed. For example, the detailed conceptual design of CLEAR-I with the final goal of minor actinide (MA) transmutation, which has operation capability of subcritical and critical dual-mode has been finished. An innovative ADS concept system as advanced external neutron source driven travelling-wave reactor CLEAR-A for energy production was proposed.

In order to support the China Lead-based Reactor projects, as well as validate and test the key components and integrated operating technology of lead-based reactor, three integrated test facilities have been built and start commissioning since 2017, including the lead alloy-cooled engineering validation reactor CLEAR-S, the lead-based zero power critical/subcritical reactor CLEAR-0 coupled with HINEG neutron generator, the lead-based virtual reactor CLEAR-V.

Main activities in the United States (observer)

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

With regard to design concepts, of particular relevance is the past development of the Small, Secure Transportable Autonomous Reactor (SSTAR), carried out by Argonne National Laboratory (ANL), Lawrence Livermore National Laboratory (LLNL) and other organisations over an extended period of time. SSTAR is 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.

University-related design activities include past work at the University of California on the Encapsulated Nuclear Heat Source (ENHS) and more recent efforts at the University of Alaska and Texas A&M University to design a Passively Operated Lead Arctic Reactor (POLAR). Other efforts in the current year included university research on methods for in-service inspection (ISI) in LFRs as well as ongoing research associated with the EU-US INERI project "Small Modular Lead-cooled Fast Reactors in Regional Energy Markets: Safety, Security, and Economic Assessments".

In the US industrial sector, current 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 an ongoing 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

The number of possible salts and material combinations leads to a broad range of potential MSR concepts (see Table 3.2). However, for a basic representation, MSRs can be classified into the following groups:

- **salt-fuelled reactors**, in which a flowing fuelled salt contains fissile material that fissions when in the core and flows throughout the primary system serving as fuel and coolant;
- **salt-cooled reactors**, in which a solid fuel undergoes fission and is cooled by a separate, non-fuelled primary salt.

Both salt-fuelled and salt-cooled concepts can be fast, epithermal or thermal spectrum reactors. Salt-fuelled thermal spectrum reactors typically use fluoride salts with fixed moderating material within the core. Fast spectrum salt-fuelled concepts can use fluoride or chloride salts and do not require in-core solid moderating material.

Salt-fuelled and salt-cooled MSR concepts have many common technology needs. Examples include the need for materials development and qualification, affordable fabrication and construction methods, large-scale salt production, large-scale pumping and heat exchange, source term definition and behaviour characterisation, and modelling and simulation tools to evaluate performance and facilitate licensing. Depending on the country, MSR development efforts may have government, industry or private elements.

Within the GIF MSR pSSC (provisional system steering committee), research is **performed on both subclasses**, under an MOU signed by Euratom, France, Russia (from year 2013), Switzerland (from year 2015), the United States and Australia (from year 2017), with Canada, China, Japan, and Korea as observers. The mission of the MSR pSSC is to support development of future nuclear energy concepts that have the potential to provide significant safety and economic improvements over existing reactor concepts.

Fast spectrum molten salt reactor concepts

In the beginning, MSRs were typically thermal-neutron-spectrum graphite-moderated design concepts. Since 2005, liquid-fuelled MSR R&D has focused on fast spectrum MSR options combining the generic advantages of fast neutron reactors (extended resource utilisation, waste minimisation) with those related to molten salt fluorides as both fluid fuel and coolant (low pressure, high boiling temperature and, optical transparency without fission products). Recent MSR developments in Russia on the 1 000 MWe molten salt actinide recycler and transmuter (MOSART) and in France, Euratom and Switzerland on the 1 400 MWe non-moderated thorium molten salt reactor (MSFR) address the concept of large power units with a fast neutron spectrum in the core. The fast neutron spectrum MSRs 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).

More recently, a third concept has been under development by TerraPower Inc: the "molten chloride fast spectrum reactor" (MCFR). It represents the first US government funding for a liquid-fuelled MSR in 40 years. Southern Company Services is the lead for this programme, and TerraPower, Oak Ridge National Laboratory (ORNL), EPRI, and Vanderbilt University are supporting institutions. The MCFR is intended to have a very hard neutron spectrum to avoid requiring fissile material input after its initial core load or separation of fissile materials from the remainder of the fuel salt.

Table 3.2: List of the different MSR systems

Name	Developer	Power, MWt	Fuel/carrier/moderator
Thermal Spectrum Liquid Fuel MSRs			
Thorium Molten Salt Reactor, Liquid Fuel (TMSR-LF)	Shanghai Institute of Applied Physics (SINAP), China	395	ThF_4-$^{233}UF_4$/^7LiF-BeF_2/graphite
Integral Molten Salt Reactor (IMSR)	Terrestrial Energy, Canada and the United States	400	UF_4/fluorides/graphite
ThorCon Reactor	ThorCon International, Singapore	557x2	UF_4/NaF-BeF_2/graphite
Liquid-Fluoride Thorium Reactor (LFTR)	Flibe Energy, United States	600	ThF_4-$^{233}UF_4$/^7LiF-BeF_2/graphite
FUJI-U3	Japan	450	ThF_4-$^{233}UF_4$/^7LiF-BeF_2/graphite
Advanced Molten-salt Break-even Inherently-safe Dual-mission Experimental and Test Reactor (AMBIDEXTER)	Ajou University, Korea	250	$^{233}UF_4$-ThF_4/^7LiF-BeF_2
Transatomic Power MSR (TAP)	Transatomic Power, United States	1 250	UF_4/FLiNaK/SiC clad $ZrH_{1.6}$
Compact Used fuel BurnEr (CUBE)	Seaborg Technologies, Denmark	250	SNF/fluorides/graphite
Process Heat Reactor	Thorenco, United States	50	UF_4/NaF-BeF_2,/Be rods
Stable Salt Thermal Reactor (SSR-U)	Moltex Energy, United Kingdom	300-2 500	UF_4/fluorides/graphite
Fast/Epithermal Spectrum Liquid Fuel MSRs			
Molten Salt Fast Reactor (MSFR)	SAMOFAR, France – EU – Switzerland	3 000	ThF_4-UF_4/^7LiF-
Molten Salt Actinide Recycler and Transformer (MOSART)	Kurchatov Institute, Russia	2 400	$TRUF_3$ or ThF_4-UF_4/^7LiF-BeF_2 or NaF-^7LiF-BeF_2
U-Pu Fast Molten Salt Reactor (U-Pu FMSR)	VNIINM, Russia	3 200	UF_4-PuF_3/^7LiF-NaF-KF
Indian Molten Salt Breeder Reactor (IMSBR)	BARC, India	1 900	ThF_4-UF_4/LiF-
Stable Salt Fast Reactor (SSR-W)	Moltex Energy, United Kingdom	750-2 500	PuF_3/Fluorides
Molten Chloride Fast spectrum Reactor (MCFR)	TerraPower, United States		U-Pu/Chlorides
Molten Chloride Salt Fast Reactor (MCSFR)	Elysium Industries, United States and Canada	100-5 000	U-Pu/Chlorides
Dual Fluid Reactor (DFR)	Dual Fluid Reactor, Germany	3 000	U-Pu/Chlorides

Table 3.2: List of the different MSR systems *(Cont.)*

Name	Developer	Power, MWt	Fuel/carrier/moderator
Solid Fuel MSRs (all thermal spectrum)			
Molten-Salt Reactor with Micro-Particle Fuel (MARS)	Kurchatov Institute, Russia	16	TRISO-coated LEU/FLiBe/Graphite pebble bed
Advanced High Temperature Reactor (AHTR)	ORNL, United States	3 400	Coated U particles in blocks or plates/FLiBe/Graphite
Small Advanced High Temperature Reactors (SmAHTR)	ORNL, United States	125	Coated U particles in blocks or plates/FLiBe/Graphite
Pebble Bed – Fluoride Salt-Cooled High Temperature Reactors (PB-FHR)	UC Berkeley, MIT and UW, United States	242	TRISO-coated LEU/FLiBe/Graphite pebble bed
Thorium Molten Salt Reactor, Solid Fuel (TMSR-SF)	SINAP, China	395	TRISO-coated U-Th/FLiBe/Graphite pebble bed
Indian High Temperature Reactor (IHTR)	BARC, India	600	TRISO-coated U-Th/FLiBe/Graphite pebble bed

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

Thermal spectrum molten salt reactor concepts

Canada, China, Japan and South Korea are focused on the development of the small and medium power liquid fuel units with thermal spectrum graphite-moderated cores. In China, the Thorium Molten Salt Reactor (TMSR) programme was initiated by the Chinese Academy of Sciences (CAS) in 2011, which involves a closed U-Th fuel cycle for MSR. The new candidate site for the liquid-fuelled 2 MWt TMSR-LF1 test reactor was selected in 2017. It will be located in Wuwei, Gansu Province, about 2 000 Km from Shanghai.

Fluoride salt-cooled high-temperature reactor (FHR)

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. 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. China and the United States are currently developing such reactors.

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 reprocessing unit and wastes conditioning. The mastering of technically challenging MSR technologies 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 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 MSFR is relevant, additional developments are required before FHRs can be considered for deployment.

- continuous fibre ceramic composites;
- FHR specific fuel elements and assemblies;
- tritium release prevention technologies.

Main activities and outcomes

MSR pSSC activity

Two partners signed the MOU in 2017:

- the United States signed on 5 January 2017;
- Australia signed on 14 December.

List of MSR related meetings actively supported by pSSC members in 2017 is given below:

- 23th MSR pSSC Meeting 23-24 January 2017, PSI, Villigen, Switzerland;
- 24th MSR pSSC Meeting, 28-29 September 2017, IAEA, Vienna, Austria;
- Webinar Series 8: Fluoride-Cooled High-Temperature Reactors, 27 April 2017, Prof. Per Peterson, UC Berkeley, United States;
- Webinar Series 9: Molten Salt Reactors, 23 May 2017, Prof. Elsa Merle, CNRS, France;
- 11th GIF-INPRO/IAEA Interface Meeting, 20-21 February 2017, IAEA Vienna, Austria;
- First IAEA Workshop on the Challenges for coolants in fast spectrum system: Chemistry and materials, 5-7 July 2017, IAEA, Vienna, Austria;

- IAEA CM on MSR Technology, TECDOC, 25-27 September 2017, IAEA, Vienna, Austria;
- MSR Workshop, 24 January 2017, PSI, Villigen, Switzerland;
- MSR Summer school, 2-4 July, 2017, SAMOFAR, POLIMI, Lecco, Italy;
- MSR Workshop 2017: RD&D Priorities and Regulatory Issues, 3–4 October, 2017, ORNL, United States.

MSFR concept development (France and Euratom)

- Basic plant simulator

The safety and optimisation studies performed in previous projects and in SAMOFAR have led to the initial design of the liquid fuel circuit and the emergency drain system of the MSFR (see Figure 3.12). These systems are now being optimised in terms of safety in the technical work packages of SAMOFAR.

This initial design of the fuel circuit and emergency drain system has been evaluated and was approved by international experts during the SAMOFAR progress meeting in June 2016 and is continuously being improved to take into account the results obtained in the SAMOFAR project.

In August 2017, the first version of the basic plant simulator was released. This system code has been developed jointly by CNRS (primary fuel circuit) and POLIMI (intermediate and conversion circuits) and is undergoing validation. This will be done by comparing the results to those of well-known system codes. Afterwards the code will be used to define the operation procedures of the MSFR including the identification of safety issues. Preliminary calculations with the LiCore code developed at CNRS (primary fuel circuit) show excellent behaviour of the MSFR in response to load-following variations (Figure 3.13). Doubling the power from 1.5 to 3 GW leads to a fuel salt temperature change of only a few tens of degrees. The thermal behaviour of the heat exchangers during transients in the intermediate circuit has been evaluated by POLIMI using the Modelica code.

More information on the safety approach and methodology applied in SAMOFAR can be found below, in the safety assessment section.

Figure 3.12: Overview of the MSFR system (left) for two possible designs, including the fuel circuit and the emergency draining system

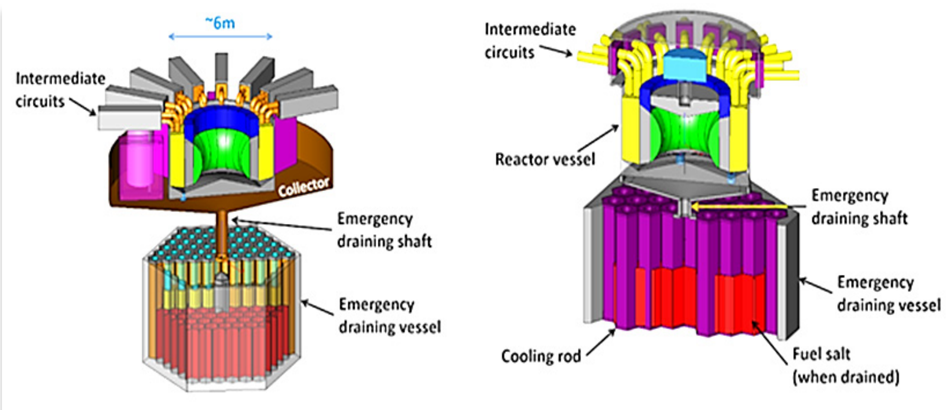
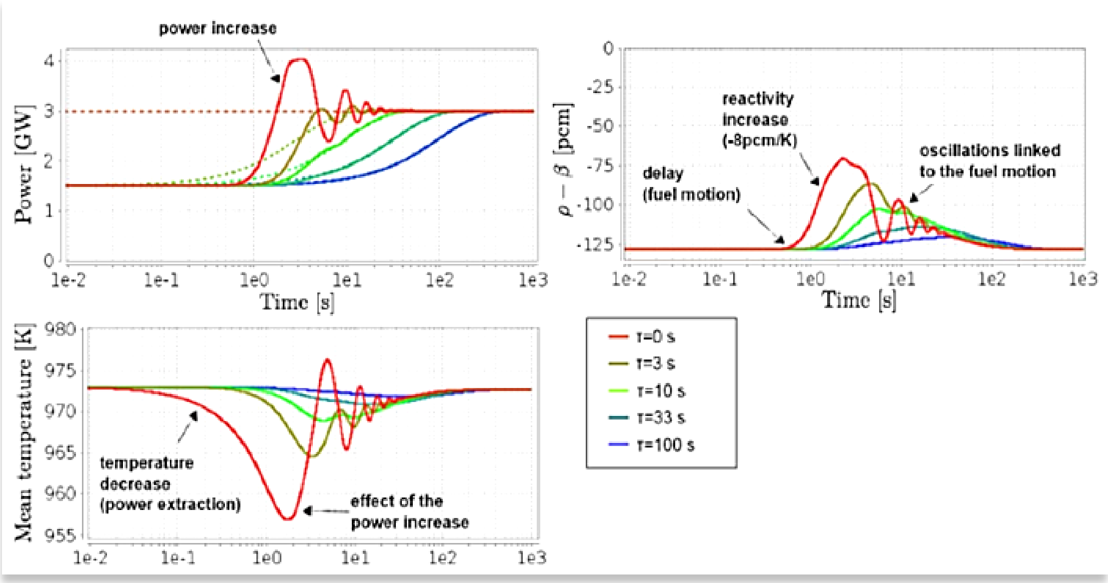


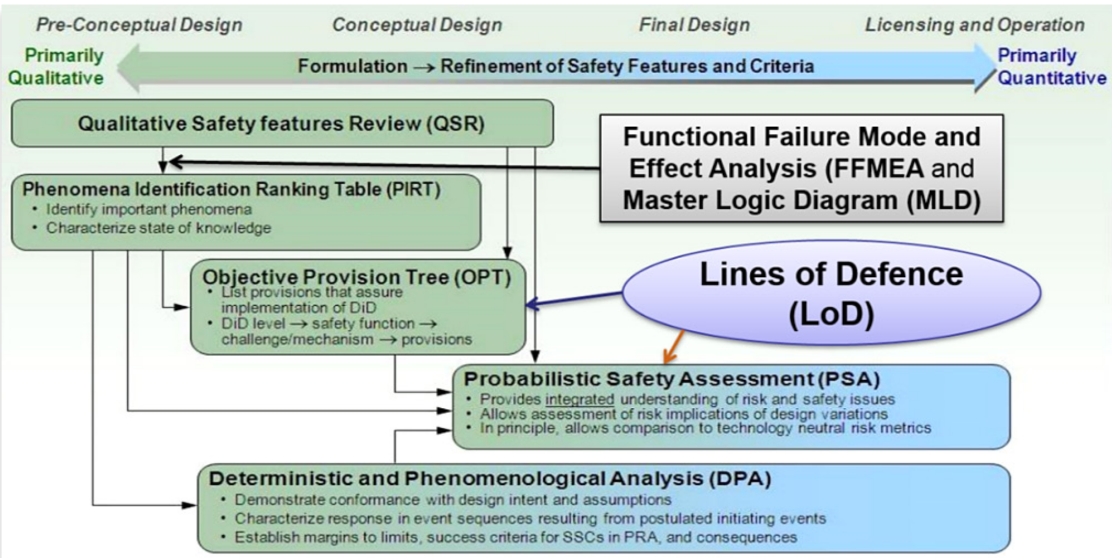
Figure 3.13: Load-following transients from 1.5 to 3 GW simulated with the LiCore code by varying the power extracted in the heat exchangers



▪ Safety assessment of the MSFR concept

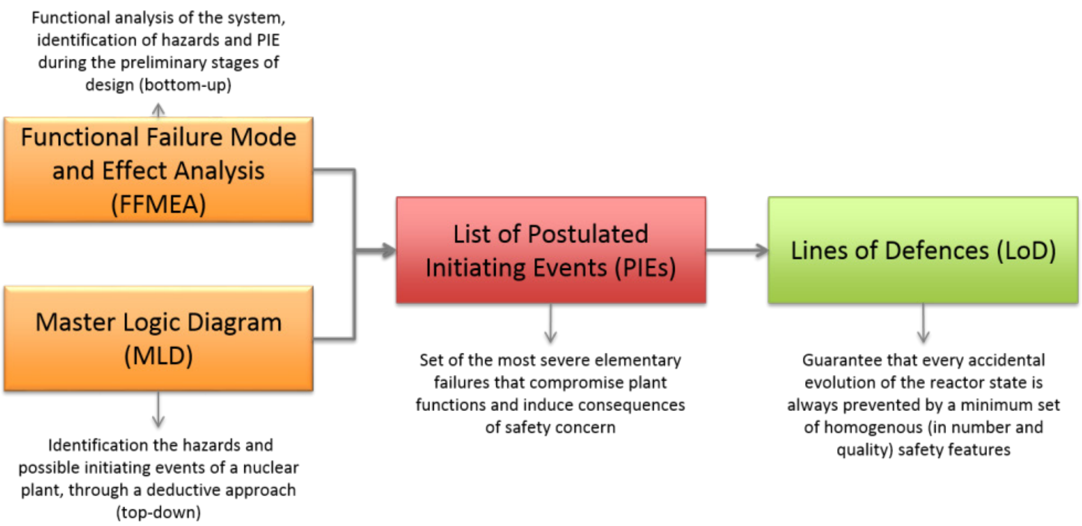
As described in the 2016 GIF annual report, the analysis of safety and optimisation studies has led to the proposal of an initial design of the MSFR systems, including both the fuel circuit and the emergency draining system.

Figure 3.14: ISAM methodology with some analysis tools identified for the MSFR application



Driven by IRSN, the French TSO, a safety methodology dedicated to liquid fuel fast reactors was developed in 2017 in the frame of the H2020 SAMOFAR project. One of the main purposes of this methodology is for it to be used jointly with the design studies and to give useful feedbacks and guidance to the designer in order to have a safety "built-in" rather than "added-on". Starting from the ISAM (Integrated Safety Assessment Methodology) approach of GIF (see Figure 3.14) and taking into account other safety methodologies and guidelines, the application procedure and required tools to apply it to the MSFR have been identified. These safety assessment tools are being applied to the MSFR, mainly the Functional Failure Mode and Effects Analysis (FFMEA) by CIRTEN/POLITO and CNRS/LPSC, the Master Logic Diagram (MLD), and the Lines of Defences (LoD) approach by CNRS/LPSC and Areva (see Figure 3.15).

Figure 3.15: Safety assessment tools applied to the MSFR



The FFMEA and the MLD have been used in order to identify the broad set of hazards from which the more relevant will then be selected as postulated initiating events Both methods have been applied to fulfil this task because of their complementarity – the first one being a bottom-up, and the second one a top-down approach – and have been applied in parallel to be as exhaustive as possible in the identification of the initiating events.

On one hand, the FFMEA is an inductive method based on a functional approach where the failure modes are obtained by the negation of a function rather than the malfunction of a component. It is therefore suitable to define possible accident initiators of the MSFR despite the lack of design details to allow an evaluation at the component level. This methodology includes the list of the systems and main components in the plant breakdown structure (PBS), the definition of the main functions (process functions, safety functions, investment protection functions, etc.) of the system through the functional breakdown structure (FBS) and finally the filling of the FFMEA table (see example in Table 3.3), the final objective being to provide a list of potential initiating events (IEs). The method has been applied on the MSFR in normal operation, and more specifically for power production, with a focus on the fuel circuit and the systems in direct interaction with it (fertile blanket system, intermediate circuit, gas processing unit, etc.).

Table 3.3: Extract from the MSFR FFMEA table

Process function	PBS elements	Op. Md.	Failure mode	Cause	Consequences	PIE
P1.1.1.1 To keep and preserve the integrity and leak-tightness of the core cavity	Core vessel	NOp-P	Loss of containment leak-tightness	Rupture in the core vessel	The fuel flows outside the core cavity; The chain reaction shuts down; The fuel is drained in the EDS and cooled down in order to remove residual head; Etc.	Loss Of Liquid Fuel

On the other hand, the MLD is a deductive method allowing one to identify the initiating events through a structured approach, particularly well adapted for the MSFR early design stage as the identification of hazards is not linked to detailed design assumptions. The main steps of the method are to identify the top event (which is the undesired event to be prevented), to decompose the top event into detailed sub-events and to deduce all possible causes likely to lead to the failure. The diagram is usually presented in the form of a qualitative fault tree beginning with the top event and where the lower levels of the tree show the elementary failures (see Figure 3.16). The MLD has been applied to the MSFR for the power production mode, similarly to the FFMEA application, and more specifically for the hazard "degradation/leakage of the core vessel".

The combined use of the MLD and the FFMEA has allowed us to produce a list of hazards for the normal operation of the reactor during power production and a list of PIEs is currently under definition. For example, an extract of this preliminary list is available in Table 3.4.

This study has also helped to establish a list of design key points that are relevant for safety and should be further documented, such as the type of pumps used for the fuel circulation, the definition of the decay heat removal system or the components of the fission product removal systems (see Figure 3.17). It has also highlighted the need to further define the operation and accidental procedures. For instance, the cases in which the emergency draining system, the routine draining system or in-core shutdown are used should be defined.

Figure 3.16: Extract from the MSFR MLD

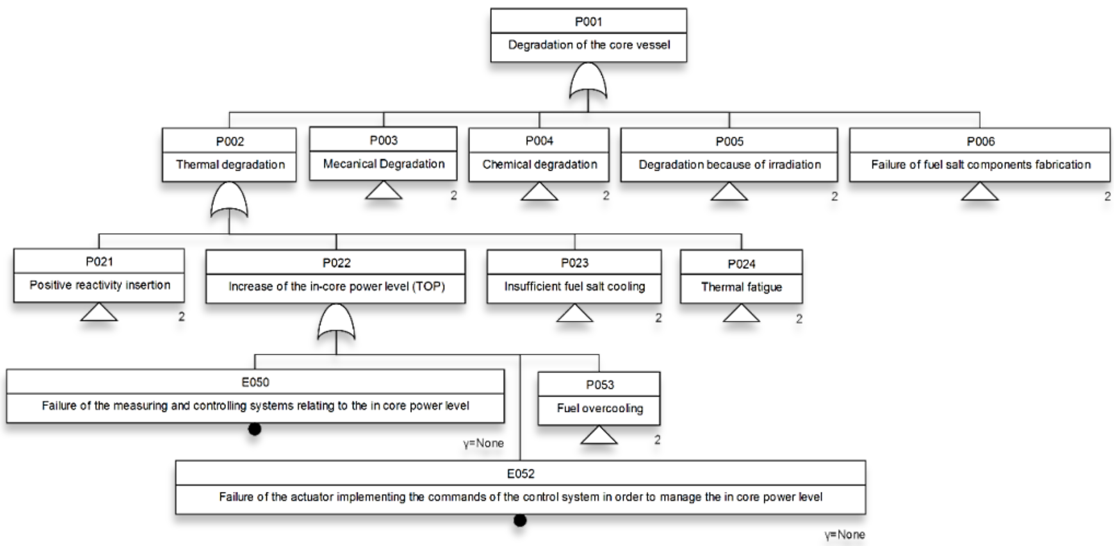
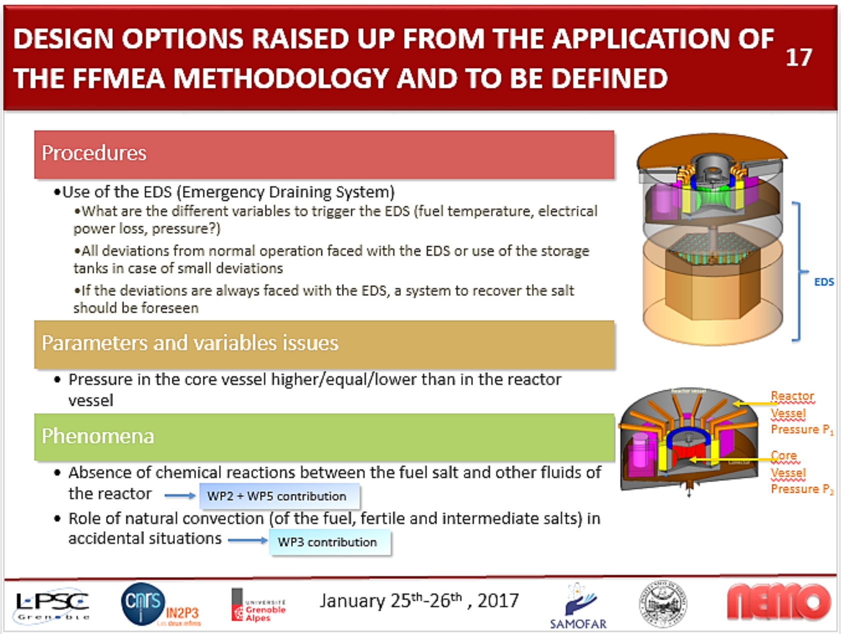


Table 3.4: Extract from the preliminary list of PIEs

Family	Postulated initiating event (PIE)
Loss of Liquid fuel	Breach in the upper reflector (with/without rupture of a radial fuel outlet pipe of the expansion vessel system and/or damages to the structure cooling system)
	Breach in the lower reflector (with rupture of the structure cooling system)
Loss of integrity of the core cavity	Complete rupture of the pressurised sampling device
	Rupture of the blanket tank wall between fuel and fertile salt with rupture of the cooling circuit for internal structures
	Breach of a heat exchanger plate/channel
Reactivity insertion accident	Accidental insertion of fuel
Loss of fuel flow	Complete rupture of the fuel circuit pump
Loss of heat extraction	Leak of intermediate salt
	Rupture of one or several intermediate pump
Overcooling	Over-working of the fuel circuit pump
	Over-working of the intermediate circuit pump
Loss of pressure/volume control in the core cavity	Obstruction of the vertical inlet pipe for the fuel from the core to the expansion vessel
	Rupture of the connection between the free surface of the fuel storage tank and the free surface of the core for the gas in the part between the core cavity and the valve
Loss of critical geometry	Collapse of the welded joints taking the recirculation sectors in the correct position
Loss of chemistry control	Rupture of the gas separation chamber
	Rupture of horizontal bubble injector for salt cleaning
Loss of support function	Total loss of electric power

In parallel to this work, the safety provisions of the plant are under definition, and the present conclusions and results are the following: the application of defence-in-depth principles has helped to define several proposals for the confinement barriers of the MSFR. These analyses will then have to be applied to the other systems of the plant (e.g. processing unit, energy conversion circuit) and to the other operation procedures of the MSFR such as start-up, shutdown and load-following.

Figure 3.17: Examples of questions on the design options raised by the preliminary safety assessment of the MSFR

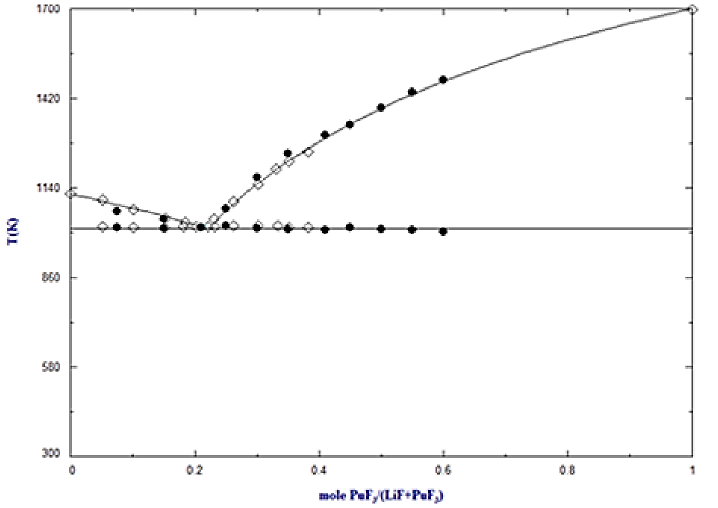


▪ Molten salt chemistry and behaviour

Recently at JRC Karlsruhe, the synthesis of pure PuF_3 has been achieved, and the first experimental results on systems containing PuF_3 have been obtained, extending the knowledge of the LiF-PuF_3 phase diagram as shown in Figure 3.18.

Another highlight was the experimental demonstration of the retention capacity of caesium in the MSFR fuel solvent using a Knudsen effusion mass spectrometry, a unique device to measure volatility of nuclear materials. The results of this campaign are summarised in Figure 3.19, indicating reduction of CsF volatility of nominal concentration of 1 mol% by more than 2 orders of magnitude due to the fact that it is dissolved in the fuel. This effect reduces the source term in case of an accidental release of fuel salt. Using calorimetric facilities, the melting temperature of the uranium-based MSFR fuel salt has been determined, and the influence of caesium and iodine on the fuel salt melting point has been investigated showing no major effect with respect to reactor operation. To complement the study of the caesium behaviour in the MSFR fuel, the full thermodynamic assessment of the ternary CsF-ThF_4-LiF system has been made requiring the assessment of the CsF-ThF_4 subsystem (LiF-ThF_4 and LiF-CsF have already been implemented in the JRC molten salt database). The CsF-ThF_4 subsystem was studied using various techniques, including differential scanning calorimetry for the determination of equilibrium data, Knudsen effusion mass spectrometry for the determination of activity coefficients, and X-ray diffraction measurements to reveal the structure and stability of various intermediate compounds. Using this novel information, the CsF-ThF_4 system has been assessed and is shown, together with the measured equilibrium points, in Figure 3.20. To understand the fuel behaviour under accidental scenarios, the vaporisation behaviour of the uranium-based fuel salt has been investigated at elevated temperatures, providing fundamental thermodynamic data on partial vapour pressures of gaseous species, which are in equilibrium with the molten fuel salt. These data were used for the extrapolation of the vaporisation behaviour up to the boiling point of the fuel salt.

Figure 3.18: Assessed LiF-PuF_3 phase diagram; solid symbols: data measures at JRC Karlsruhe; open symbols: data measured by ORNL



In order to study the methods for the MSR fuel salt clean-up, a facility has been designed and put into operation for the synthesis of pure fluoride actinides and for electrochemical measurements of actinides in molten fluorides. The syntheses of pure UF_4, ThF_4 and PuF_3 have been established, and the purity of the products has been verified experimentally. The electrochemical studies of selected actinides in molten fluoride salt media are still ongoing.

Figure 3.19: Vapour pressure of 1 mol% CsF in a eutectic mixture of LiF-ThF_4

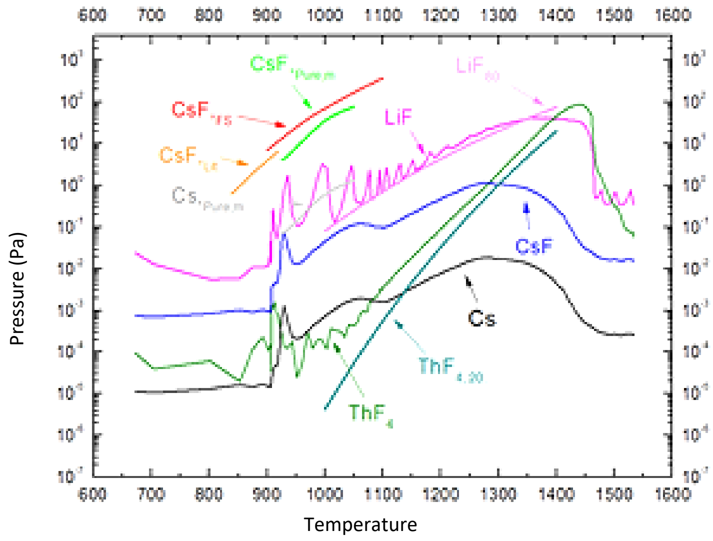
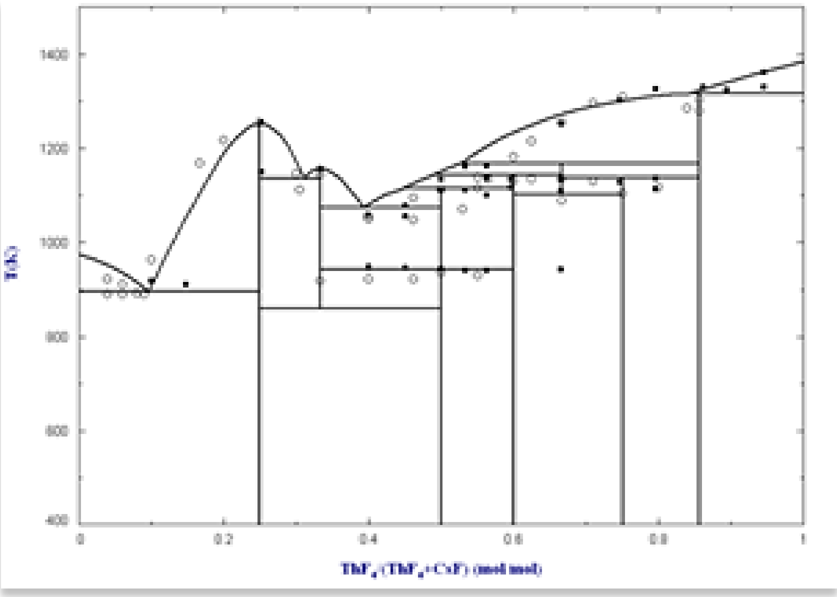


Figure 3.20: Assessed CsF-ThF$_4$ system; solid symbols: data measures at JRC Karlsruhe; open symbols: data measured by ORNL



▪ Inactive testing loops

At POLIMI, the DYNASTY loop (see Figure 3.21) and the data acquisition system have been developed. Some preliminary experiments have been carried out with water, and the molten salt that will be used in the facility at the next stage has been characterised. The facility will be used to study the dynamics behaviour of natural circulation systems subject to distributed heating. In DYNASTY experimental campaigns have already been started to support the validation of analytical and numerical simulation tools developed by PoliMi, EDF and TU Delft. The experimental results show the impact of the thermal inertia on the behaviour of natural circulation systems. The dynamic instabilities in the form of periodic oscillations and pulsed flow behaviour will be studied soon. The extension to DYNASTY for the experimental simulation of the passive decay heat removal (DHR) system has been designed and commissioned. This extension is aimed at investigating the coupled dynamics of the primary loop and the passive DHR system.

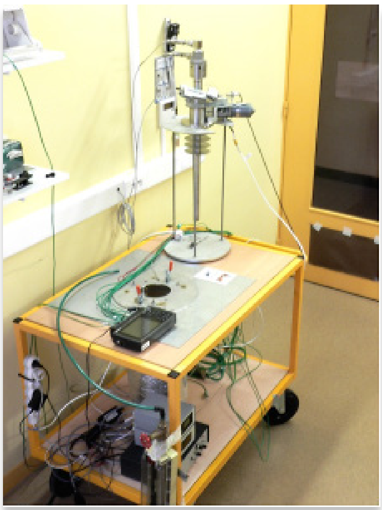
Two facilities have been built at CNRS to investigate heat transfer and solidification phenomena of molten salts:

- SWATH-W using water as working fluid: to study the accuracy of the CFD models predictions regarding the flow velocity field with particle image velocimetry (PIV) experiment implementation.
- SWATH-S using FLiNaK salt: to investigate salt heat transfer and phase change phenomena.

One of the main objectives of the SWATH experiments (LPSC, France, SAMOFAR European project 2015-2019) is to improve molten salt numerical models used for design and safety studies, and more specifically during the fuel salt draining. Complex thermal heat transfer involving conduction, convection (the later reinforced by turbulence mixing) and radiative heat between various phases (solid, liquid and gas) will exist in some of the MSFR components. The relatively high Prandtl values characterising molten salts imply that the thermal development lengths will be in general larger than the hydraulic development lengths. Therefore, accurate prediction of the heat exchange will require a precise prediction of the flow field.

Figure 3.21: DYNASTY facility at PoliMi labs

Figure 3.22: SWATH-S solidification experiment at CNRS



The operation of both SWATH facilities 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 3.23 presents a sketch of the SWATH-S facility, which is composed by two salt storage tanks, pipes, and a glove-box where test sections are placed under an argon atmosphere. The pressure control system is designed to maintain a stable flow during the operation of the loop by regulating the argon cover gas pressure of the tanks. Figure 3.24 displays a global view of the setup.

Figure 3.23: Sketch of the SWATH-S facility

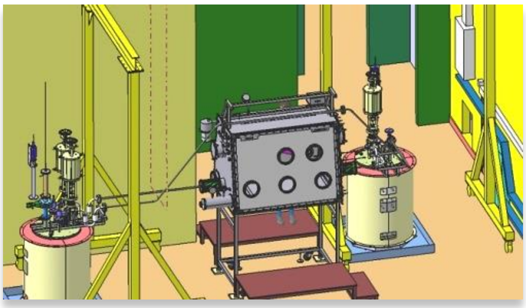
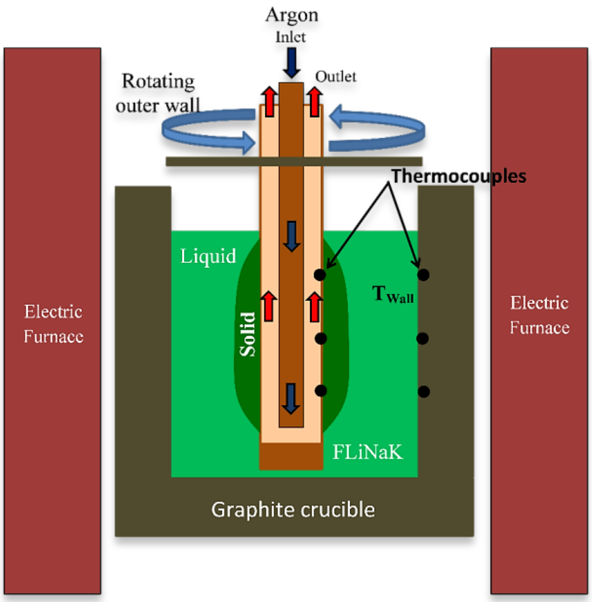


Figure 3.24: Global view of the SWATH-S setup facility

Experience related to solidification processes are also in progress in SWATH project.

Since it is expected that the presence of flow convection in the fluid phase has a significant effect on the shape of the solidification front, two different boundary conditions will be investigated: (i) Natural convection and (ii) forced convection. In order to decrease the uncertainties associated with the numerical modelling of flow velocity field conditions, a relatively simple geometry (and flow field) has been adopted for the experiment. As can be seen in Figure 3.25, the solidification experiment employs a rotating tube inside an annular cavity filled with molten FLiNaK. The rotating tube contains an inner tube that allows for the circulation of a gas coolant (argon) to decrease the temperature of the external wall of the outer tube below the FLiNaK melting point and thus to initiate the solidification process.

Figure 3.25: Setup for solidification experiment



Other experiments related to solidification and fusion processes are conducted under SWATH project to develop an efficient cold plug design. These studies are based on results obtained from a horizontal cold plug device already being used as a safety device in the Forced Fluoride Flow for Experimental Research (FFFER) facility, which was developed at the LPSC (CNRS Grenoble) prior to the SAMOFAR project for helium bubbling studies (Figure 3.26). The working principle of the cold plug relies on the control of the heat transfer balance inside the device, which determines whether the salt inside the cold plug solidifies or melts. When cooling of the assembly is stopped (e.g. due to a loss of electrical power), the thermal energy stored in the mass is quickly transferred by conduction to the solidified salt region, causing it to melt.

Tests of helium bubbling and liquid-gas separation have been done in the FFFER facility (LPSC-CNRS Grenoble) in 2017, showing satisfying results in the configuration used (about 1% vol. gas, 1.9 salt litre/s). The design of the liquid-gas separator can be improved for running higher flow.

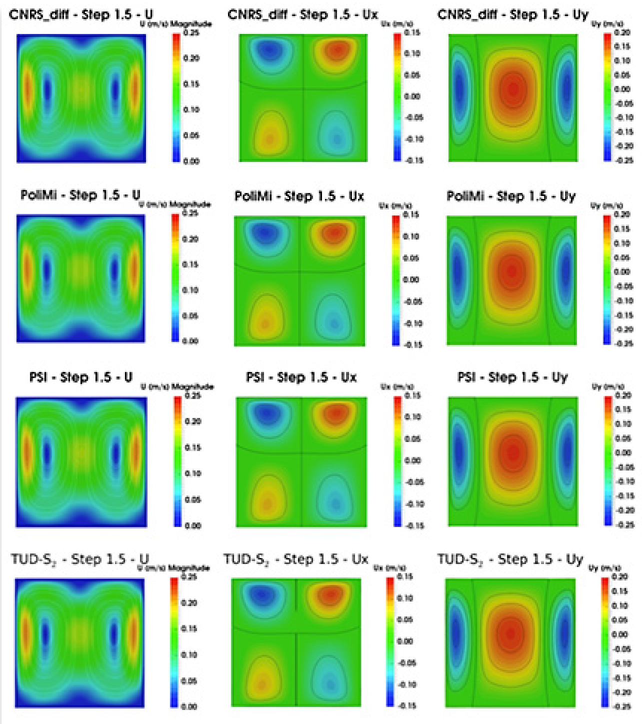
Figure 3.26: Forced Fluoride Flow for Experimental Research (FFFER) facility

Reactor physics

During the first two years of the project, the code systems of the SAMOFAR partners have been extended to include the unique aspects of the molten salt reactor that prevent the possibility to employ standard reactor physics code packages for its simulation. These simulation packages include OpenFoam tools at POLIMI and PSI based on neutronics diffusion theory. The CNRS code package constitutes a coupling between OpenFoam and the Monte Carlo code SERPENT, using the transient fission matrix approach. KIT and EDF have further extended the SIMMER code to include the correct thermodynamics properties of the salt. At TU Delft, the code coupling consists of a discrete ordinates code coupled to a new discontinuous Galerkin finite element flow code.

The correctness of these codes has been assessed through a benchmark study between the partners. This benchmark has been defined by CNRS, is specifically devised for the multi-physics processes in the molten salt reactor, and gradually increases the complexity of the physics to be modelled, making the identification of possible errors easier. The results showed very good agreement between partners. At the same time, the benchmark was effective in initially highlighting code problems that were solved along the way. An example result of the benchmark is shown in Figure 3.27. The code systems are now in good shape to proceed with multi-physics transient analyses.

Figure 3.27: Results of different partners for steady-state coupling between neutronics and CFD

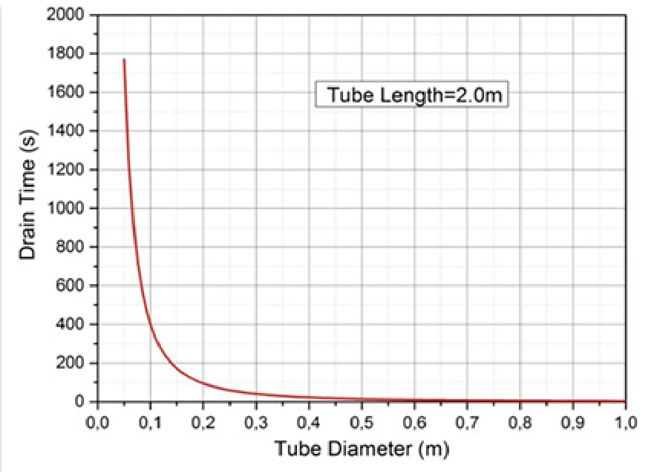


Note: (Left) velocity magnitude, (middle) x-component of velocity with isolines, (right) y component with isolines.

A set of transients has been selected that are considered most important to study the safety of the MSFR. Examples of the selected set include Unprotected Loss of Heat Sink (ULOHS) transient, Unprotected Loss of Fuel Flow (ULOFF), and blockage of fuel salt in the draining system. The list of transients has been augmented with modelling suggestions to unify approaches taken.

Some of the partners have already started modelling transient scenarios. Notably, KIT and EDF have investigated by various approaches the speed of draining of the salt from the core during an emergency. As an example, Figure 3.28 shows the dependency of the draining time on the draining tube diameter. Fundamental studies of plug melting are also being performed that precede the draining process.

Figure 3.28: Core draining time obtained by KIT as function of the draining tube diameter



Material corrosion

Recently, CINVESTAV, the Mexican partner in SAMOFAR, has delivered the first Yttria Stabilized Zirconia (YSZ) samples for the corrosion studies at CNRS in LiF-ThF_4 (77-23%mol) molten salt. YSZ samples with different compositions have been prepared in the form of pellets (8, 11.3, 17 and 20%mol Y_2O_3) and Hastelloy coatings (17%mol Y_2O_3 and 17%mol Y_2O_3 + 10%mol C). All pellets have been sintered at 1 300°C in air.

At CNRS, a preliminary study of the chemical and electrochemical behaviour of zirconium (introduced in the molten salt as ZrF_4) has been performed in LiF-NaF-KF (46.5-11.5-42%mol) molten salt at 550°C. A complex electrochemical response of zirconium was observed on the tungsten and molybdenum electrodes. Several reduction and oxidation processes have been identified. The role of oxide ions has been studied as well. Complementary studies are necessary for a better knowledge of the zirconium behaviour and its interaction with oxide ions. Later, the same system will be evaluated in LiF-ThF_4 molten salt.

An extensive study of the chemical behaviour of iodide has been accomplished at the CNRS. This study was executed in two different fluoride molten salts, LiF-NaF-KF (500°C) and LiF-ThF_4 (650°C). Through voltammetry techniques, three oxidation processes have been identified in the two molten salts: oxygen evolution, iodine evolution and metallic gold oxidation. A redox potential inversion has clearly been observed among the I_2(g)/I– and O_2(g)/O_2- redox systems present in LiF-NaF-KF and LiF-ThF_4. The redox potential shift of O_2(g)/O_2- redox system to more anodic potentials shows a high stabilisation of the oxide ions in LiF-ThF_4 molten salt, which was attributed to the formation of a stable and soluble thorium oxifluoride specie ($ThOF_2$) in the salt. In LiF-ThF_4 salt, the presence of a more oxidant redox system than I_2(g)/I– lead to a spontaneous oxidation of iodide ions. This chemical reaction is related to the presence of oxygen (2 ppm) in the inert gas. The fluorination extraction was electrochemically simulated in LiF-ThF_4 molten salt. A yield extraction of iodide higher to 95% was obtained.

MOSART development in Russia

MSR activities in Russia now focus mainly on liquid fuel fluoride-based systems. These activities continue to be directed through the Rosatom Offices of Innovations and Fuel Cycle. Recent Rosatom developments concerning the MOSART concept address the advanced large power unit with the main design objective being to close nuclear fuel cycle for all actinides, including Np, Pu, Am and Cm (see Figure 3.29).

Figure 3.29: Closed nuclear fuel cycle with MOSART

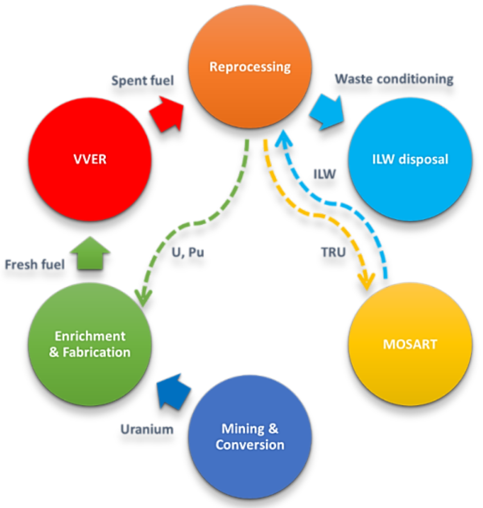
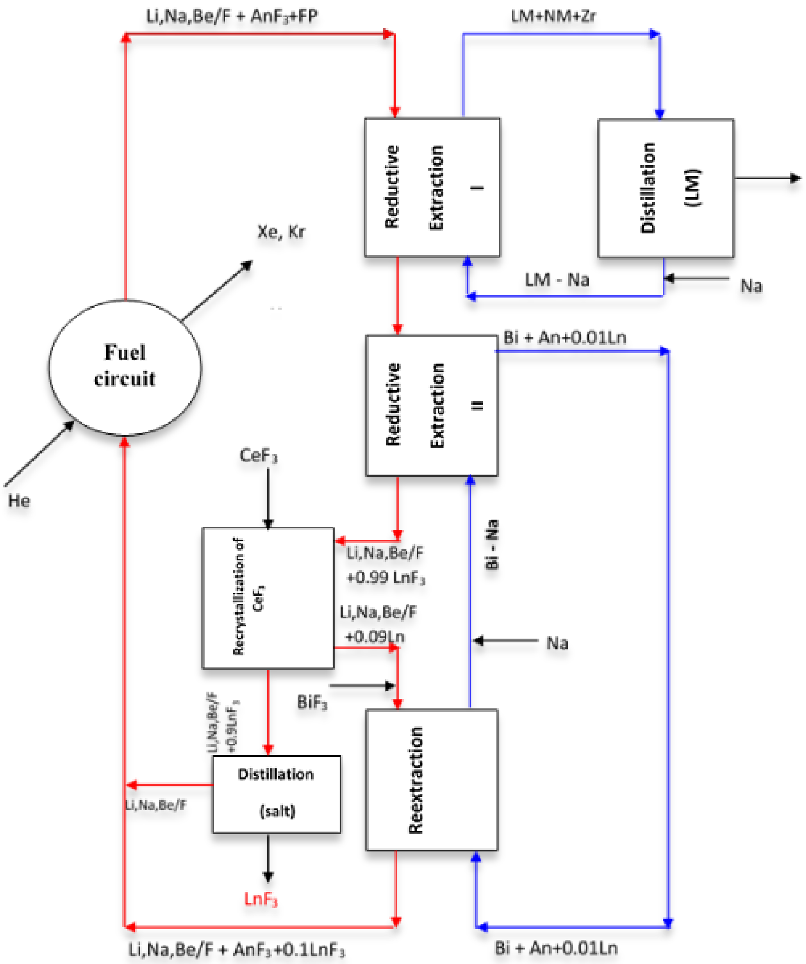


Table 3.5: Methods and cycle times for fission product removal and TRU recycling

Component	Removal time	Removal operation
Kr, Xe	50 sec	Sparging with He
Zn, Ga, Ge, As, Se, Nb, Mo, Ru, Rh, Pd, Ag, Tc, Cd, In, Sn, Sb, Te	2.4 hr	Plating out on surfaces+ To off gas system
Zr	1-3 years	Reductive extraction
Ni, Fe, Cr		
Np, Pu, Am, Cm		
Y, La, Ce, Pr, Nd, Pm, Gd, Tb, Dy, Ho, Er, Sm, Eu		
Sr, Ba, Rb, Cs	>30 years	
Li, Be		Salt discard

The optimum design for Li,Be/F MOSART is fast spectrum of homogeneous core without graphite moderator. The effective flux of such a system is near $1x10^{15}$ n cm^{-2} s^{-1}. The main attractive features of the MOSART system include the use of (1) a simple configuration of the homogeneous core (no solid moderator or construction materials under high-flux irradiation); (2) proliferation-resistant multiple recycling of actinides (separation coefficients between TRU and lanthanide groups are high, but within the TRU group are very low); (3) the proven container materials (high nickel alloys) and system components (pump, heat exchanger etc.) operating in the fuel circuit at temperatures below 1023K, (4) inherent safety of the core due to large negative temperature reactivity coefficient (-3.7 pcm/K), and (5) the long period for soluble fission product removal (see Table 3.5). The fuel salt clean-up flowsheet for the Li,Be/F MOSART system, based on reductive extraction in to liquid bismuth, is given on Figure 3.30.

Figure 3.30: Conceptual scheme of Li,Na,Be/F MOSART fuel salt clean-up



The Mining and Chemical Combine (MCC) is being considered as a possible site for the construction of the Li,Be/F MOSART reactor plant. The unique technical and technological capabilities of the MCC site provide the opportunity to place an experimental 100 MWt MOSART unit with a fast neutron spectrum in close proximity to the VVER used fuel reprocessing facilities, linking it to the Experimental Demonstration Centre infrastructure. Following Rosatom's request, MCC and NRC "Kurchatov Institute" are developing a programme plan for development of the Demo MOSART.

In 2017, Rosatom sponsored an MSR workshop at Bochvar VNIINM (Moscow) that included representation of MSR developers from Rosatom and RAS Institutions, as well as NRC "Kurchatov Institute". The main focus of this workshop was on Li,Be/F MOSART and Li,Na,K/F MSFR designs.

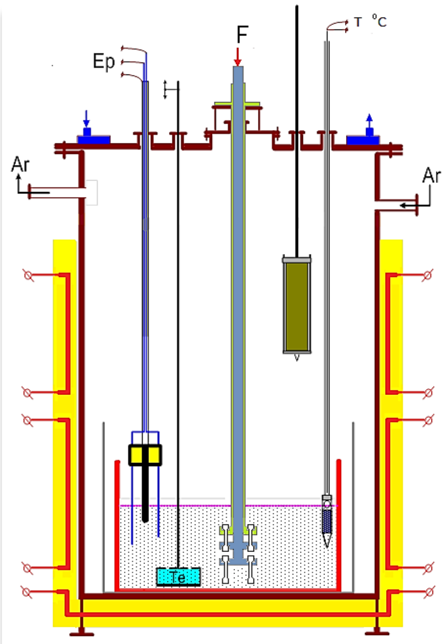
^{99}Mo production in a very small power MSR with $LiF–BeF_2–UF_4$ fuel salt is also of interest for Rosatom. It was examined at NRC "Kurchatov Institute". The proposed method of ^{99}Mo production relies on the behaviour of gaseous and noble fission products in the fuel salt. Molybdenum, together with some other noble- and semi-noble metals, does not form stable compounds in $LiF–BeF_2–UF_4$. At least 50% of the molybdenum in a MSR will be in a gas-aerosol phase above the fuel salt surface. Neutronic, thermal-hydraulic and mass transfer evaluations were performed for the very small power MSR operating in a natural convection mode.

In 2017, NRC "Kurchatov Institute" and the Shanghai Institute of Applied Physics (SINAP) signed a bilateral agreement to co-operate on the development of MSRs. The most significant NRC "Kurchatov Institute" development resulting from this collaboration in 2017 was the completion of a Li,Be,U/F corrosion facility, the layout of which is shown as Figure 3.31. This corrosion test included high nickel alloys (see Table 3.6) developed in Russia (HN80MTY), China (GH 3535) and the United States (Hastelloy-N). At this facility the electrochemical behaviour of UF_n (n=3.4) in a molten 71LiF-27BeF_2-2UF_4 (in mole%) salt mixture containing metallic Te was already studied by cyclic voltammetry. Formal analysis of the obtained dependencies showed that, in our experimental conditions, the recharge U(IV) to U(III) is qualitatively consistent with voltammetric criteria and can be classified as quasi-reversible for reaction $U^{4+}+e^{-}\rightarrow U^{3+}$. Molybdenum was used as the material for both the reference and working electrodes. The [U(IV)]/[U(III)] ratio for the molten 71LiF-27BeF_2-2UF_4 (in mole%) salt mixture containing this redox buffer couple in the presence of metallic Te was measured in these tests accurately and reliably by a voltammetric analysis at temperatures up to 1 073 K. Parallel chemical probes of the melt samples confirmed this conclusion.

Table 3.6: High nickel alloys under study at NRC "Kurchatov Institute"

Element	in mass%				
	GH3535	GH3535	Hastelloy-N	UNS10003	kHN80MTY
Ni	base	base	base	base	base
Cr	6.8	7.01	7.1	7.2	6.81
Mo	16.4	16.72	15.9	16.2	13.2
Al	0.12	Al +Ti 0.22	0.23	0.13	1.12
Ti	<0.005		<0.005	<0.005	0.93
Fe	3.2	4.04	3.1	3.3	0.15
Mn	0.46	0.57	0.50	0.50	0.013
Nb	0.013	-	0.05	<0.009	0.01
Si	0.24	0.31	0.3	0.15	0.04
W	<0.005	-	0.014	<0.005	0.072
Cu	0.01	0.01	0.033	0.015	0.02
Co	0.031	0.01	0.1	0.027	0.003
V	<0.03	0.01	<0.03	<0.03	0.003
B	-	0.001	-	-	0.003
S	-	0.001	-	-	0.001
P	-	0.002	-	-	0.002
C	-	0.57	-	-	0.025

Figure 3.31: Layout of the Li,Be,U/F corrosion facility at NRC "Kurchatov Institute"



In 2017, the NRC "Kurchatov Institute" has been also co-operating with the EU SAMOFAR Project and contributing to the IAEA Report "Status of Molten Salt Reactor Technology".

MSR activities in the Czech Republic

The experimental development of molten salt technologies devoted to molten salt reactor (MSR) and fluoride salt-cooled high-temperature reactor (FHR) systems continued in the Czech Republic. A new four-year project was launched in January 2017 that focuses on the research and development of selected areas of MSR and FHR reactor technology. The project is a follow-up and broadening of existing Czech activities in MSR. The aim of the project is to contribute to the development of MSR and FHR reactor technology in the area of reactor physics, nuclear-chemical engineering and material research. One of the main objectives of the project is the experimental determination of main neutronic properties and characteristics of MSR and FHR reactors cooled by ^{7}LiF-BeF_2 salt (FLiBe salt). The other objectives of the project are focused on the MSR fuel cycle technology and MSR reactor core chemistry, further development of MSR/FHR structural material-Ni-based alloys and subsequent design and manufacture of selected components of the MSR/FHR technology. The project also creates a platform for running Czech-US co-operation in MSR/FHR development.

The project is conducted by a consortium of Czech research institutions and industrial companies led by the Research Centre Řež. The other members of the consortium are ÚJV Řež – Nuclear Research Institute, COMTES FHT, MICo Ltd and ŠKODA JS – Nuclear Machinery.

The main work packages of the project are:

- theoretical and experimental physics of MSR/FHR system;
- chemistry and chemical technology of MSR;
- structural materials and components of MSR/FHR technology.

These main work packages are complemented by system studies covering also the non-proliferation and physical protection issues of the thorium – uranium fuel cycle and an MSR mock-up design.

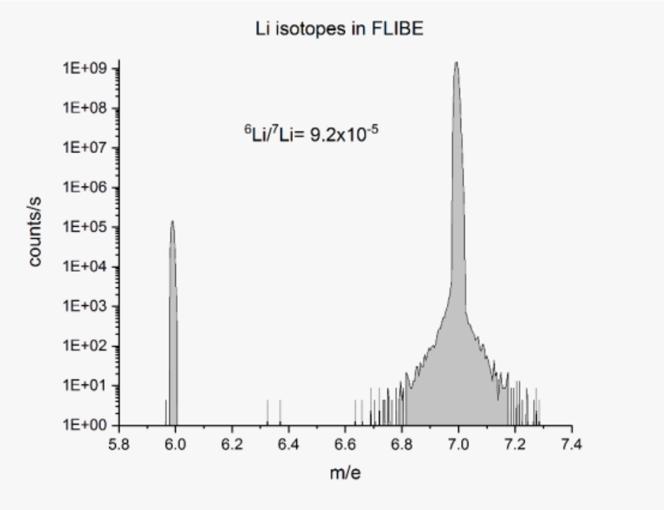
- Theoretical and experimental physics of MSR/FHR system

The effort, which is a follow-up to previous activities, is focused mainly to the interconnection of theoretical and experimental studies of thermal spectrum MSR reactor physics and MSR/FHR neutronic studies. The main part of experimental work concerning the pure FLiBe salt neutronics and FLiBe with thorium and uranium fluorides neutronics has been carried out at the LR-0 experimental reactor of Research Centre Řež. The LR-0 core consists of six pin-type fuel assemblies (VVER-1000 design) with nominal enrichment of 3.3% and an empty experimental channel, forming a driven zone in the core centre. Material insertions are put into the driven zone, occupying one position in the lattice.[5] The tests with FLiBe were performed with real MSR/FHR reactor LiF-BeF_2 (66-34 mol%) coolant salt containing Li-7 isotope (99.994 mol%), which was provided by ORNL, and were aimed at studying the neutron spectrum shape to confirm previous results obtained with LiF-NaF salt. The salt used in these experiments originally comes from the Molten Salt Reactor Experiment and was supplied to the Czech Republic through the 2012 Czech-US collaborative agreement on MSR/FHR R&D. Material specimens were put into the driven zone of the LR-0 reactor, occupying one of the lattice positions. The neutron spectrum behind the layer of salt was measured by recoiled protons in different energy ranges. The independent measurements were taken by a set of hydrogen proportional detectors (HPD) for energies 0.1-1.3 MeV and by an organic scintillator (Stilbene) detector for energy ranges 0.8-10 MeV. The inserted zone with the FLiBe salt is shown in Figure 3.32. The analysis of isotopic composition of Li in FLiBe was determined by SIMS method, and the result of the Li-6 and Li-7 isotope rate is evident from Figure 3.33.

Measurements with FLiBe were taken at room temperature, while neutronic tests planned within the new project will be performed in a special heated insert zone in the LR-0 at the temperature range of 500-750°C. The main objective of the tests will be determination of reactivity coefficients. These tests mark the continuation of a close collaboration between Research Centre Řež and Oak Ridge National Laboratory in this area.

Figure 3.32: Loading of FLiBe zone into LR-0 reactor

Figure 3.33: Evaluation of Li isotopes by SIMS method



- Chemistry and chemical technology of MSR

Existing research and development studies in chemistry and chemical technology were focused on the verification of liquid MSR fuel processing – experimental production of UF_4 and ThF_4, basic electrochemical studies of actinide/fission product separation from fluoride molten salt media and the flowsheet studies of the single-fluid and double-fluid online pyrochemical reprocessing of MSR thorium-breeder. The present effort and future directions cover also the development and experimental verification of a fused salt volatilisation technique proposed for the extraction of uranium (in the chemical form of UF_6) from the MSR fuel salt. The previous programme in electrochemistry, realised by ÚJV Řež, was focused on the development of an experimental setup for molten fluoride salt media – including the development of reference electrode based on the Ni^0/Ni^{2+} redox couple and the evaluation of redox potentials for uranium, thorium and selected fission products in individual selected molten fluoride salts (LiF-NaF-KF, FLiNaK, LiF-BeF_2, FLiBe and LiF-CaF_2). The present effort is focused on the development and verification of quantitative electrochemical extraction of uranium and thorium and on removal of main neutron poisons (fission products) from the MSR carrier salt (FLiBe). Special attention will be paid to the electrochemical studies of protactinium. These studies are planned to be realised in collaboration between the Research Centre Řež, ÚJV Řež and the European JRC, Institute for Transuranium Elements Karlsruhe.

- Structural materials and components – the molten salt loop programme

Material research for molten fluoride technologies played an important role in existing R&D activities focused on MSR development. The most important was the development of nickel-based superalloy MONICR. MONICR was designed and developed by the COMTES FHT company as the Czech structural material for MSR and FHR technology. The basic corrosion and irradiation tests of MONICR were completed in previous projects, whereas further development of the semi-pilot production and further tests of high-temperature microstructure stability, high-temperature mechanical stability and radiation embrittlement are studied in the new project. Another study concerning MSR/FHR component development includes the continuation of special graphite gasket seals development and of the design and development of pumps (impellers) and valves for fluoride salt media. These activities are conducted by MICo Ltd and by ŠKODA JS Company. Regarding the development of materials and components, a molten fluoride salt loop program was initiated. The out-of-pile loop programme will contribute to the preparation of the MSR mock-up design, which should be a final stage of the new project.

A new forced FLiBe loop was built and put in operation in the first half of 2017. The loop is intended for material research and testing of components of the MSR and FHR technologies. The loop is electrically heated and thermally insulated and consists of an impeller, two experimental channels for samples, a freeze valve and a storage tank. The main structural materials of the loop are Inconel 718 and MONICR. The working temperature range is from 550°C to 750°C. The loop programme covers material corrosion tests, development and verification of special graphite gasket seals and further development of pumps and valves for fluoride salt media. A picture of the FLiBe loop is provided in Figure 3.34.

US MSR activities

In 2017, the US government signed the GIF MSR memorandum of understanding. US MSR efforts are led by industry with an emphasis on deployment, while government efforts are more broadly defined with a focus towards advancing fundamental science and technology and developing the next generation of the nuclear workforce. Support for US MSR efforts has diversified significantly, and many important activities are more broadly classified as support for advanced non-LWRs. Both liquid and solid (a.k.a. FHRs)-fuelled MSRs are included within the scope of US activities.

Figure 3.34: FLiBe loop in the Řež Research Centre

Industry

The Nuclear Energy Institute, which represents the US nuclear industry, is co-ordinating a number of activities supporting MSR development and deployment under its Advanced Reactor Working Group. NEI's technical working group (TWG) on MSRs is seeking to co-ordinate the common elements of MSR industry interests with the DOE Office of Nuclear Energy (DOE-NE). The NEI MSR TWG provides a forum to identify and collaborate on technology-specific issues, recommend course corrections, and rapidly transfer progress into designs. NEI's Licensing Modernization Project (LMP), which is being cost-shared with the US Department of Energy (DOE), has a primary objective to develop technology-inclusive, risk-informed, and performance-based regulatory guidance for licensing non-LWRs for the US Nuclear Regulatory Commission (NRC) to review and potentially approve. NEI sponsored an advanced reactor modelling and simulation workshop in early 2017 that included an overview of current and planned US MSR modelling and simulation tools. Presentations from the workshop are available on the Gateway for Accelerated Innovation in Nuclear's (GAIN) website.

NEI issued an advanced reactor security white paper (ML17026A474) that proposes new physical security requirements for non-LWRs with enhanced engineered safety and security features. Under NEI's proposal, MSR plants would be required to maintain the capabilities to detect and assess threats and to promptly summon local law enforcement assistance with the interdiction and/or neutralisation of the threat would be performed by local law enforcement officers rather than plant staff.

Government funding (2017)

Funding to support MSR activities included support for Gateway for Accelerated Innovations in Nuclear (GAIN), Nuclear Energy University Programs (NEUP), an MSR Integrated Research Project (IRP), and an Industry Award to Southern Company.

GAIN awarded seven vouchers (USD 2.1 million) to MSR companies, which increased access to the R&D capabilities within DOE national laboratories:

- Synthesis of Molten Chloride Salt Fast Reactor Fuel Salt from Spent Nuclear Fuel;
- NEAMS [Nuclear Energy Advanced Modeling and Simulation] Thermal-Fluids Test Stand for Fluoride Salt-Cooled, High-Temperature Reactor Development;

- Development of the Micro-Scale Nuclear Battery Reactor System;
- Conversion of Light Water Reactor Spent Nuclear fuel to Fluoride Salt Fuel;
- Evaluation of Power Fluidic Pumping Technology for Molten Salt Reactor Applications;
- IMSR® [Integral Molten Salt Reactor] Fuel Salt Property Confirmation: Thermal Conductivity and Viscosity;
- Fuel Salt Characterization.

The DOE's NEUP programme awarded one MSR Integrated Research Project (IRP) and four MSR NEUP grants in 2017. The IRP NuSTEM: Nuclear Science, Technology and Education for Molten Salt Reactors (IRP-17-14541; USD 3 000 000) is led by Texas A&M University with team members from the University of California at Berkeley, the University of Wisconsin, and an international partnership with the EU SAMOFAR program. The IRP project will contribute to the molten salt fast reactor concept while educating new workforce in molten salt systems and will focus on the following five technical areas:

- material and corrosion science;
- optical/chemical sensor development;
- modelling, multi-physics simulation, and uncertainty quantification;
- thermal-hydraulic science;
- $^{35}Cl(n,p)$ cross-section measurements.

The MSR DOE NEUP projects are:

- Methods to Predict Thermal Radiation and to Design Scaled Separate and Integral Effects Testing For Molten Salt Reactors, University of California Berkeley (CFA-17-12664), USD 800 000.
- Design of a Commercial-Scale, Fluoride Salt-Cooled, High-Temperature Reactor with Novel Refueling and Decay Heat Removal Capabilities, University of Massachusetts Lowell (CFA-17-12972), USD 400 000.
- Radiative Heat Transport and Optical Characterization of High Temperature Molten Salts, University of Wisconsin-Madison (CFA-17-13232), USD 800 000.
- Bimetallic Composite (Incoloy 800H/Ni-201) Development and Compatibility in Flowing FLiBe as a Molten Salt Reactor (MSR) Structural Material, University of New Mexico (CFA-17-13020), USD 800 000.

The DOE-NE also continued funding the Southern Company Services led (partnering with TerraPower, Oak Ridge National Laboratory, Electric Power Research Institute, and Vanderbilt University) project to perform integrated effects tests (IET) and materials suitability studies to support development of TerraPower's Molten Chloride Fast Reactor with a total government cost share of up to USD 40 000 000 over five years. TerraPower's announced plans are for the IET to lead to test reactor operations in 2025 and a prototype in 2030.

Government funding (2018 and beyond)

Solicitations for advanced reactor funding were announced:

- additional NEUPs to support advanced reactor R&D (DE-FOA-0001772);
- advanced Research Projects Agency-Energy (ARPA-E) programme: Modeling-Enhanced Innovations Trailblazing Nuclear Energy Reinvigoration (MEITNER);

- DOE Industry Funding Opportunity Announcement (DE-FOA-0001817);
- inclusion of MSRs within DOE's Small Business Innovative Research (SBIR) initiative.

The 2018 NEUP MSR focus areas are:

- down-selection of cladding materials for structural components in liquid-fuelled molten salt reactors;
- innovative new alloys for molten salt reactor structural applications;
- development of molten salt reactor fuel salt irradiation capabilities;
- advanced in-reactor instrumentation;
- fluoride salt-cooled high-temperature reactors – flow loop testing and reactor core and plant modelling capabilities;
- predicting the chemical speciation, structure, and dynamics of salts solutions for molten salt reactors;
- understanding the structure and speciation of molten salt at the atomic and molecular scale.

The MEITNER ARPA-E call (USD 20 000 000 total funding) seeks to identify and develop innovative technologies to enable the advanced nuclear reactor design community to mature their designs for future commercial deployment. MEITNER Awardees will perform key enabling technology development for nuclear reactor systems, components, and structures, moving those technologies towards commercialisation. The MEITNER programme will require a system-level approach in describing and quantifying how new and innovative enabling technologies fit into a plant design to make the plant "walkaway" safe, quickly deployable, safeguardable, cost-competitive and commercially viable.

The DOE Industry Funding Opportunity Announcement (approximately USD 400 million, contingent upon congressional appropriations over five years) is to support innovation and competitiveness of the US nuclear industry through cost-shared, cross-cutting basic/fundamental, applied R&D, and demonstration/commercial application R&D activities for all aspects of existing and advanced reactor development. These activities may include development of technologies that improve the capability of the existing fleet, methods to improve the timelines for advanced reactor deployments, the cost and schedule for delivery of nuclear products, services, and capabilities supporting these nuclear technologies, design and engineering processes, and resolution of regulatory/certification issues potentially impeding the introduction of these technologies into the marketplace. The solicitation is organised into three tiers of proposals with different objectives, requirements and funding levels.

- first-of-a-kind nuclear demonstration readiness projects;
- advanced reactor development projects;
- regulatory assistance grant and technology development opportunities.

MSRs are also included in DOE's Office of Basic Energy Sciences SBIR call. The MSR topics for the call are Bimetallic structures for liquid-cooled, high-temperature reactor systems and molten salt and material interactions.

- NRC activities

The NRC activities underway supporting advanced reactor licensing are described on its advanced reactor web page. The NRC is supporting activities related to the NEI co-ordinated, DOE cost-shared LMP. The NRC has recently issued a Draft Final Regulatory Guide 1.232 (ML18011A659) which provides Guidance For Developing Principal Design

Criteria for non-light-water reactors. The draft regulatory guide provides safety-equivalent design criteria for non-LWRs to the criteria provided for LWRs in the Code of Federal Regulations (specifically 10CFR50 Appendix A). The NRC has also begun developing MSR specific guidelines for reviewing non-power reactor applications (MSR version of Regulatory Guide 1537) that would be used for licensing test MSRs in the United States.

In response to the NRC's regulatory information survey, multiple prospective MSR vendors have informed the NRC that they intend to submit documents for NRC review in the next few years. For example, Terrestrial Energy USA (ML16336A508) and TerraPower (ML17172A187) have submitted regulatory issue summary responses indicating that they intend to submit MSR licensing documents to the NRC by the end of 2019. Transatomic Power also submitted non-public information (ML16298A026) to the NRC about their future licence submittal activities.

NRC recently issued a draft white paper on functional containment performance criteria for non-LWRs (ML18010A516) that is intended to be presented to the Advisory Committee on Reactor Safety and subsequently to the commission in 2018. The purpose of the white paper is to seek commission approval of the staff's recommendation to adopt a technology-inclusive, risk-informed, performance-based approach to establishing performance criteria for structures, systems, and components and corresponding programmes to limit the release of radioactive materials from non-LWR designs. If approved, this approach would have profound implications for MSR licensing.

The NRC commissioned ORNL to develop a 12-module training seminar to introduce its staff to MSR technology and assess regulatory infrastructure needs and readiness. Over 100 NRC staff attended the training. Similar training materials were also presented to Canadian Nuclear Safety Commission staff. The seminar presentations are available under package ML17331B100. The specific modules are listed in Table 3.7

Table 3.7: MSR training modules

MSR training module title	NRC accession number
History, Background and Current MSR Developments	ML17331B113
Overview of MSR Technology and Concepts	ML17331B114
Overview of Fuel and Coolant Salt Chemistry and Thermal Hydraulics	ML17331B115
MSR Neutronics	ML17331B116
Materials	ML17331B117
Systems and Components	ML17331B118
Overview of MSR Instrumentation	ML17331B120
Fuel Cycle and Safeguards	ML17331B121
Operating Experience	ML17331B123
Safety Analysis and Design Requirements	ML17331B125
Regulatory Issues and Challenges	ML17331B126
MSR Development and R&D Issues	ML17331B128

NRC has begun to consider the requirements for qualifying molten salt fuel. An overview of a potential pathway for liquid fuel qualification was presented to the NRC at the August 2017 public stakeholder meeting (ML17220A315). Extensive effort has been undertaken by industry and NRC on assuring that the behaviour of reactor fuel is well understood under all potential operational conditions including accidents.

▪ Standards activities

DOE and NRC jointly sponsored a standards forum meeting in September 2017 to identify codes and standards that need to be developed that are not currently being developed in a timely manner by standards development organisations. MSR standards which required updating were identified by the NEI TWG Chair (ML17272A069).

American Nuclear Society (ANS) standards under development that directly pertain to MSRs include:

- ANS-20.1, "Nuclear Safety Design Criteria for Fluoride Salt-Cooled High-Temperature Reactor Nuclear Power Plants";
- ANS-20.2, "Nuclear Safety Design Criteria and Functional Performance Requirements for Liquid-Fuel Molten Salt Reactor Nuclear Power Plants";
- ANS-30.1, "Integrating Risk and Performance Objectives into New Reactor Nuclear Safety Designs";
- ANS-30.2, "Categorization and Classification of Structures, Systems, and Components for New Nuclear Power Plants".

Additionally, the ASME Boiler and Pressure Vessel Code committee is investigating the addition of clad structures, brazing as a joining technique, as well as the possibility of adding alloy N or a near-derivative to the high-temperature nuclear portion of the code.

▪ DOE-sponsored meetings

DOE sponsored three workshops with significant emphasis on MSR technology during 2017. In August, DOE's Office of Basic Energy Science sponsored a workshop on the Basic Research Needs for Future Nuclear Energy. The first identified priority research topic identified was to "Enable design of revolutionary molten salt coolants and liquid fuels".

The DOE Office of Nuclear Energy organised a workshop on Molten Salt Chemistry Workshop at Oak Ridge National Laboratory on 10-12 April 2017, for the purpose of identifying innovative science-based, technology driven approaches to accelerate MSR development and deployment. The workshop's five recommended future research directions were:

- Understanding, predicting and optimising the physical properties of molten salts highlights the need to apply modern measurement techniques and modelling and simulation tools to accelerate the design, discovery and characterisation of salts optimised for various types of MSRs.
- Understanding the structure, dynamics, and chemical properties of molten salts highlights how modern X-ray and neutron scattering and spectroscopic tools (which were not available during the studies in the 1960s and 1970s) and electrochemical methods can be coupled with advanced modelling capabilities to provide new insights into the structure, dynamics, and properties of salt species on the length and time scales needed for phenomenological understanding.
- Understanding fission and activation product chemistry and radiation chemistry highlights the need to understand the rapid decay and chemical transmutation of fission and activation products and the unique phenomena related to radiation-induced chemistry.

- Understanding materials compatibility and interfacial phenomena addresses the need to fundamentally understand material degradation (such as corrosion) and interfacial reactions (including the combined chemical and radiation effect) to develop new, more stable materials for MSR.
- Guiding next generation materials for molten salt reactors addresses the need to develop the next generation of materials that will enable MSR developers to reach the reactor performance targets. Structural materials proposed for MSR must endure extreme environments, including high fluences of neutrons, high operating temperatures and corrosive environments.
- Creating a virtual reactor simulation focuses on developing the modelling and simulation tools necessary to understand the behaviour of the reactor throughout its lifetime and provide dynamic reactor and irradiation capsule emulation for chemical and isotopic source terms.

ORNL hosted the third annual MSR workshop the first week of October 2017. The workshop objective was to provide a forum for sharing information and status of MSR R&D programmes, technology developments, international collaborations, and state of maturity of evolving MSR technologies; and for 2017 put a special emphasis on safety and licensing topics to initiate discussions on these key areas to identify needed R&D and tool development to support the ultimate licensing of MSRs. The workshop had an attendance of ~250 individuals representing utilities, reactor developers, component suppliers, DOE, IAEA, NRC, national laboratories and universities. The workshop featured the premiere of a video on remote maintenance of molten salt reactors (link available from the workshop homepage) that had been made at ORNL in 1959 but lost for more than 50 years.

DOE-NE technical campaign

DOE-NE elevated the status of its technical campaign in 2017 making it a peer to the other advanced reactors that DOE is pursuing (HTGRs and SFRs). The DOE-NE technical campaign has four major topic areas primarily carried out through its national laboratories.

- Identifying, characterising and qualifying successful salt and materials combinations for use in MSRs.
- Developing an integrated reactor performance modelling capability that captures the appropriate physics needed to evaluate plant performance over all appropriate timescales and licence MSR designs.
- Establishing a national salt reactor infrastructure and economy that includes affordable and practical systems for the production, processing, transportation and storage of radioactive salt constituents for use throughout the lifetime of molten salt reactor fleets.
- Licensing and safeguards framework development to guide research, development and demonstration.

MSR development in China

The Chinese Academy of Sciences (CAS) has reached an agreement with the Gansu provincial government on building the 2 MW thermal power molten salt test reactor in the Province of Gansu. The candidate reactor site is in the Minqin County, which is one of the 58 counties of the Gansu Province and is part of the Wuwei prefecture. Analysis reports on candidate reactor site safety and environmental impact have been prepared and are ready to be submitted to the National Nuclear Safety Administration to apply for the reactor site permit. The designer of the test reactor, Shanghai Institute of Applied Physics (SINAP), is in the process of completing the preliminary engineering design.

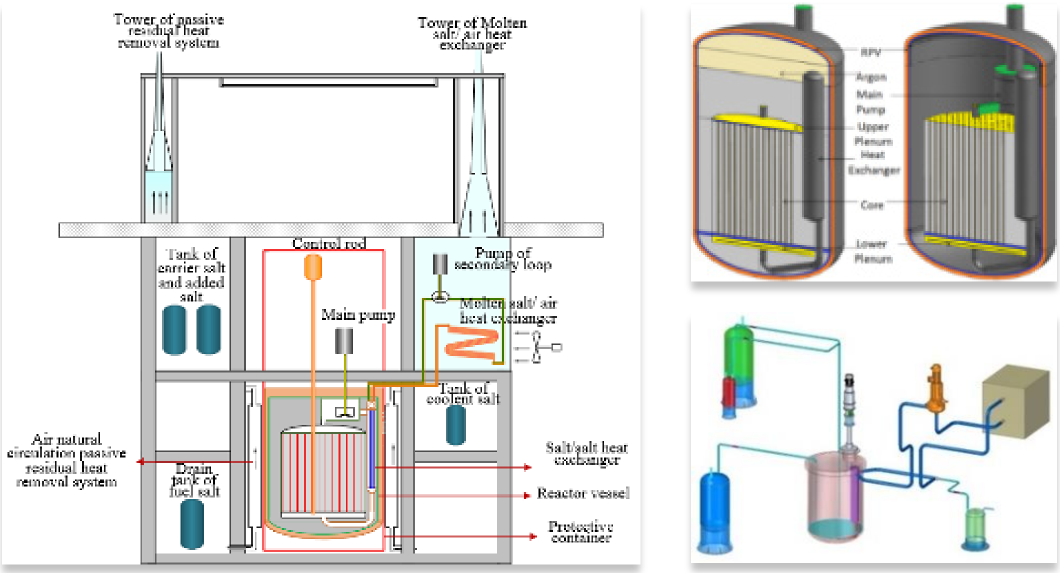
SINAP is also negotiating contracts with domestic and foreign suppliers to procure nuclear fuel, alloy and graphite components, and fluoride salts that are to be used to build the test reactor. The construction of the test reactor is expected to be completed by the end of 2020. SINAP, currently an observer of the GIF MSR pSSC, has briefed the Chinese GIF Liaison Office and recommended that China sign the MSR MOU and eventually an MSR System Arrangement (SA) agreement.

A reprocessing flowsheet for TMSR fuel cycle based on pyroprocessing techniques has been established. Uranium, carrier salt (^{7}LiF-BeF_2) and thorium can be separated from the fuel of TMSR following this flowsheet, and the different steps have been validated both by bibliographic study and some experimental determinations:

- Kilogram-scale experimental equipment for validating the feasibility of fluorination and distillation processes have developed, and processing parameters have been optimised to guarantee high purity and high recovery of uranium and carrier salt.
- For fluorination process, online infrared spectrum analysis, sorption and desorption process, and gradient condensation have been developed for process monitoring, product purification and UF_6 collection respectively. Frozen wall technique is under developing to reduce to corrosion during fluorination process, and the experiments shows that the corrosion rate of the metal material under the protection of the frozen wall can be reduced by 90%.
- Distillation mode has been selected. In the case of this mode, temperature gradient is main driving force for salt vaporisation and condensation. This working mode has substantially enhanced recovery efficiency of vaporised salt, and the recovery rate reaches to 99% with DF of FPs more than 10^2.
- A system integrating fluorination and distillation processes has been developed, and the quantitative transportation technology of liquid salt based on pressure difference is used to integrate two processes. Technology optimisation of both fluorination and distillation is in progress.
- Electrowinning is used to recycle thorium from the molten salt after fluoride volatility and vacuum distillation. Thorium, which is in the form of oxide or fluoride in the residuals, is electrochemically reduced to metal in chloride eutectic salt. The crude metallic thorium contains some salts, which can be further removed by vacuum distillation. High purity thorium is obtained after these procedures and the chloride salts can be used again.

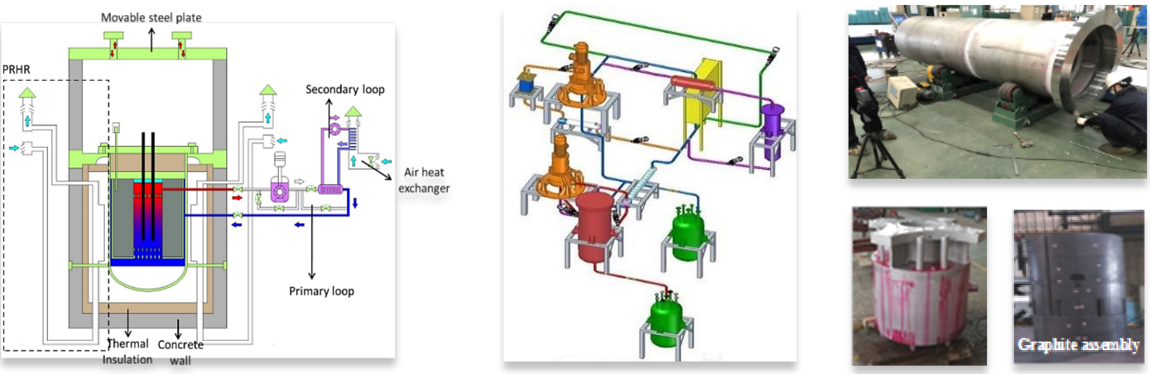
The conceptual design of the 2 MW_{th} TMSR-LF1 has been completed and the preliminary design is about to start. The construction of the reactor in Gansu, China will start in 2018, and the reactor will reach criticality and full power by 2020. Integrated design replaced the complex loop, that is, the core, the main pump, and the salt/salt heat exchanger are contained into the reactor vessel. The fuel salt (LiF-BeF_2-ZrF_4-(ThF_4)-$UF_4$17wt%^{235}U, 99.95at%^{7}Li) inclusion, reactor vessel and protective container form three radioactive containment barriers. Negative temperature feedback and passive residual heat removal system ensure reactor's safety. The main system has been designed including the reactor vessel, the salt loops, instrument control system, nuclear auxiliary system and the power plant layout. The key equipment R&D, such as the pumps, heat exchangers and passive residual heat removal system has been carried out. Series of important tests are being carried out. The preparation of fuel salt with nuclear purity has reached tons level. High density and low impregnation graphite and nickel-based alloy N10003 with independent intellectual property rights has being developed and prepared by domestic manufacturers.

Figure 3.35: The TMSR-LF1 system



The scaled experimental device TMSR-SF0 is built to solve the uncertainty of the thermal-hydraulic design of TMSR-SF1 and relevant experiments verification. Physical scheme design and preliminary design of TMSR-SF0 have been finished in 2016. All equipment are being installed after commissioning and experiment. The whole project will be completed in June 2018. Main systems of TMSR-SF0 include core heating system, Reactor vessel, Primary and secondary loops (include pumps, heat exchanger, radiator, and salt storage tank), instrumentation and control, Molten salt sampling, purification and treatment system, Auxiliary systems, engineering prototypes and public facilities. The foundation and steel frame construction, as well as the public facilities including water, electricity and gas and ventilation have been completed in 2017. Domestic nuclear power manufacturer conducted the main equipment fabrication of TMSR-SF0. At present, fabrication of main equipment including the reactor vessel, metallic structure, heat exchangers, frozen valves and salt pumps is about to finish. The core graphite, heaters and power, passive residual heat removal, cover gas over-pressure protection, tritium control and series of engineering prototypes have been manufactured and tested.

Figure 3.36: The TMSR-SF0 system



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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 China, and a pressure tube concept proposed by Canada. Other than 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 these reactor types are common, which 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, 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. Canada, China and Euratom signed the extension of Project Arrangements for Thermal-Hydraulics and Safety as well as for Materials and Chemistry in 2017. Table 3.8 lists the members and shows the status of these PMBs.

The fuel qualification testing (provisional) PMB has been consolidated into the system integration and assessment (provisional) PMB. Prior to the consolidation, Canada and Euratom were collaborating informally to pursue in-reactor irradiation of SCWR fuels at supercritical pressures in the Řež research reactor in Czech Republic. China was also interested to participate in future testing.

Table 3.8: 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 China Russia	November 2006 (renewed 2016 by Canada and Japan) July 2014 (renewed 2016) July 2011 (renewed 2016)
Thermal-Hydraulics and Safety Project Arrangement	Canada China Euratom	July 2017 (extension signed) June 2017 (extension signed) July 2017 (extension signed)
Material and Chemistry Project Arrangement	Canada China Euratom	July 2017 (extension signed) June 2017 (extension signed) July 2017 (extension signed)
System Integration and Assessment Provisional Project Arrangement	Managed by the System Steering Committee	-

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 will be sought to minimise materials degradation and corrosion product transport and will be 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 both the nuclear and non-nuclear industries.

System integration and assessment

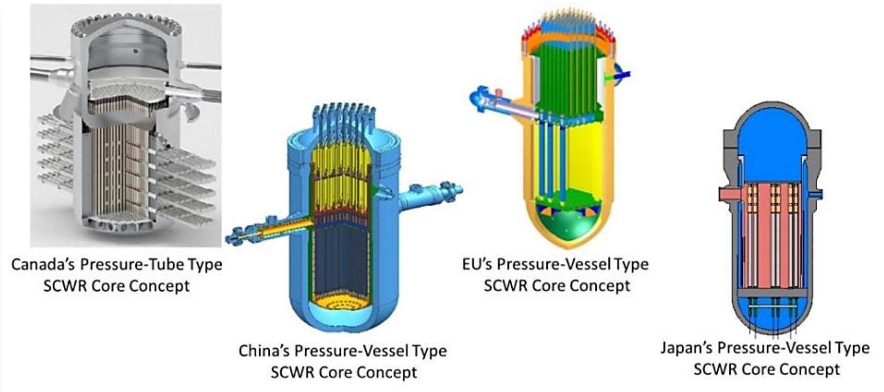
The system integration and assessment provisional PMB covers three main activities:

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

Four SCWR core concepts with thermal spectrum have been proposed, as shown in Figure 3.37. Canada, EU and Japan have completed their concept development. China is continuing the development of core and plant concepts for their pressure vessel type thermal spectrum SCWR. Their plan to host a review meeting with international peers has been deferred to 2019.

A collaborative project has been proposed in developing small SCWR concepts ranging from 10 to 300 MW in electric power. Canada has developed a preliminary small pressure tube type SCWR concept. Work on finalising this concept is ongoing. China is focusing on completing their reference SCWR concept but has also an interest to pursue the development of a small pressure vessel type SCWR concept. EU is also interested in R&D for the small SCWR concept.

Figure 3.37: SCWR Thermal Spectrum Core Concepts



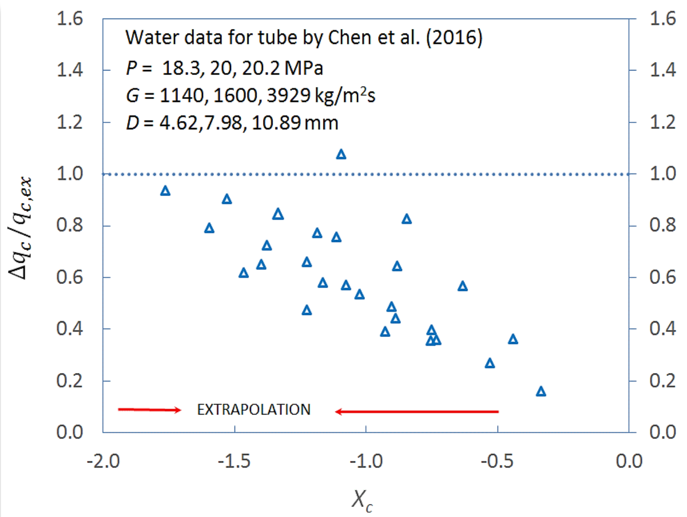
The construction of a supercritical water test loop has been completed at the Řež Research Centre in Czech Republic. It is being commissioned out-reactor for material testing. Figure 3.38 shows the test loop and the test section during out-reactor testing. Licensing effort is continued for approval to install the loop into the LVR-15 reactor for in-reactor testing. It may continue for fuel testing depending on the outcome. Fuel testing may be feasible in 2025.

Figure 3.38: Supercritical Water Test Loop and Test Section at Řež Research Centre in Czech Republic

Thermal-hydraulics and safety

Predictions of critical heat flux (CHF) are required in establishing the start-up and shutdown processes and in analysing postulated large-break loss-of-coolant accidents. The CHF look-up table has been widely used in the prediction. However, the experimental database for developing the table was limited at pressures near the critical point. This has led to increase in prediction uncertainty. Recently, a number of experimental studies have been performed to obtain CHF data at pressures near the critical point in support of the SCWR development. The CHF data was compiled for assessing the prediction capability of the CHF look-up table. The CHF look-up table tends to over-predict the experimental values on average by 27% within the applicable range. However, the discrepancy increases to 64% outside of the applicable range of the table. As illustrated in Figure 3.39, the deviation increases as the conditions deviate further away from the applicable range.

Figure 3.39: Comparison of Predicted CHF Values of the CHF Look-Up Table and Experimental Values of Chen et al.



The multi-fluid trans-critical look-up table (MTC-LUT) was developed for predicting heat transfer coefficient over a wide range of flow conditions at subcritical and supercritical pressures. It has been incorporated into the CATHENA code in support of the safety analyses of the Canadian SCWR concept. A verification exercise was performed to confirm the implementation. Figure 3.40 compares experimental heat transfer coefficients and predictions of the CATHENA code using the MTC-LUT. Good agreement has been observed confirming the success of the implementation. Deviations between predicted and experimental heat transfer coefficients were attributed to the normalisation process of a large number of experimental values within the tabulated ranges.

A three-dimensional CFD study of the fluid flow and heat transfer at supercritical pressures was performed against the wire-wrapped bundle experiments at Xi'an Jiaotong University (XJTU). The SST k–ω turbulent model was applied in the calculation. It has been shown that the CFD calculations are sensitive to the turbulent Prandtl number. The calculated surface temperature distributions along the rods and wires are presented in Figure 3.41. An increase in surface temperature is illustrated from inlet to the outlet along the rods. The results showed over-predictions of the surface temperature by up to 65°C using the CFD tools and a deviation in peak temperature location observed in the experiment.

Figure 3.40: Comparisons of experimental and predicted heat transfer coefficient (HTC) using the CATHENA Code with the MTC-LUT

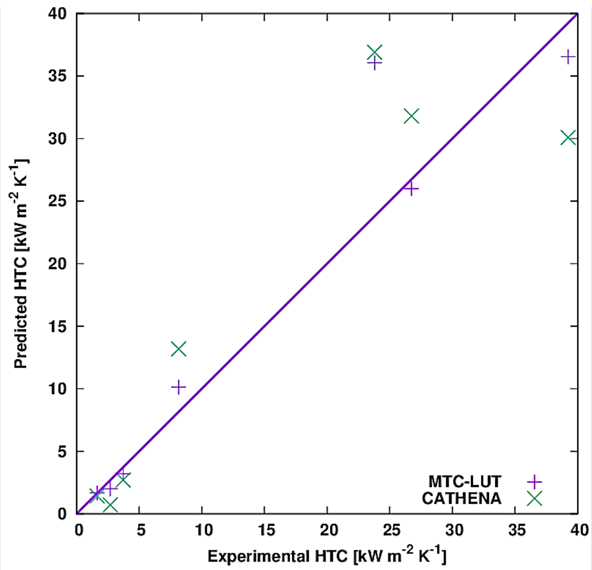
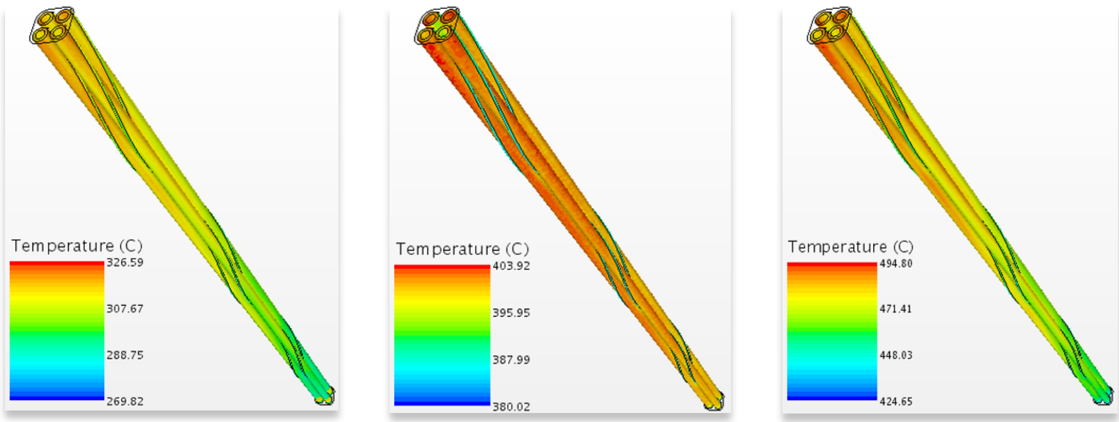


Figure 3.41: Overall predicted temperature distribution in the fluid domain along the heated length of the 2 × 2 wire-wrapped fuel bundle assembly

(a) subcritical test condition (b) pseudocritical test condition (c) supercritical test condition



For rod#1 at an axial location of 0.5 m along the heated length, the experiment reported a bimodal peak and a valley in the circumferential distribution of temperature on the fuel rod (Figure 3.42). The dominant temperature peak is seen in the narrow-gap region (180°) and another smaller peak at the 270° location, whereas a small dip in temperature (valley) is generally seen at an angular location of 225°. The CFD predictions were able to capture the order of temperature reported in the experiments as seen from Figure 3.42. However, the location of the peak temperature was predicted away from the narrow-gap region (180°) for all three cases. The experiment reported the peak temperature in the narrow-gap region, 180 °, whereas the CFD predicted the peak temperatures further downstream (circumferentially) at 225-245°. Based on the results presented in Figure 3.42, it can be inferred that the k–ω turbulence model was not able to resolve the turbulence correctly in both the corner and rod-to-rod gap regions.

Several conservative assumptions were implemented in thermal-hydraulic analyses of the fuel assembly in the Canadian SCWR fuel channel. A two-step approach has been applied to assess the effect of spacing devices (i.e. wire wraps) on heat transfer characteristics. The first step consisted of selecting and evaluating wire wrap subchannel models from a literature review, while the second step is to perform simulations with selected models to verify the assumption. The literature review indicated a complex phenomenon of the wrapped-wire effect on heat transfer. Three separate components in the model were reviewed for improving the prediction accuracy: 1) flow-resistance calculation to account for the increased wetted perimeter and enhanced turbulence induced by the wire, 2) mixing calculations to account for the swirl flow induced by the wire, and 3) heat transfer calculation to account for the extended surface of the wire that increases the heated perimeter and the fin effect. Figure 3.43 shows predicted cross-flow and pressure profiles in the fuel assembly of the Canadian SCWR fuel channel.

Figure 3.42: Assessment of the CFD predictions with measurements on rod#1 at 0.5 m using the k–ω turbulence model at sub-, pseudo- and supercritical test conditions

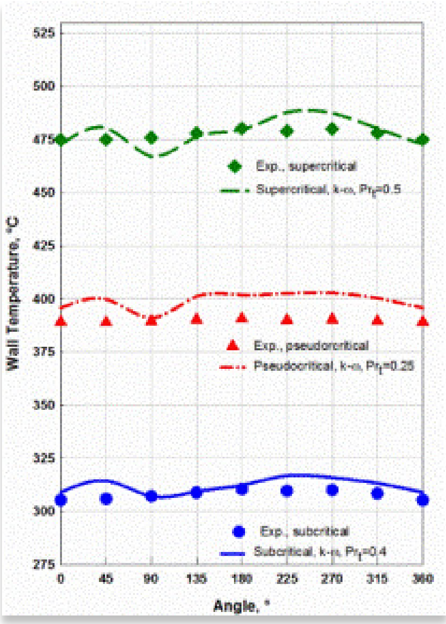
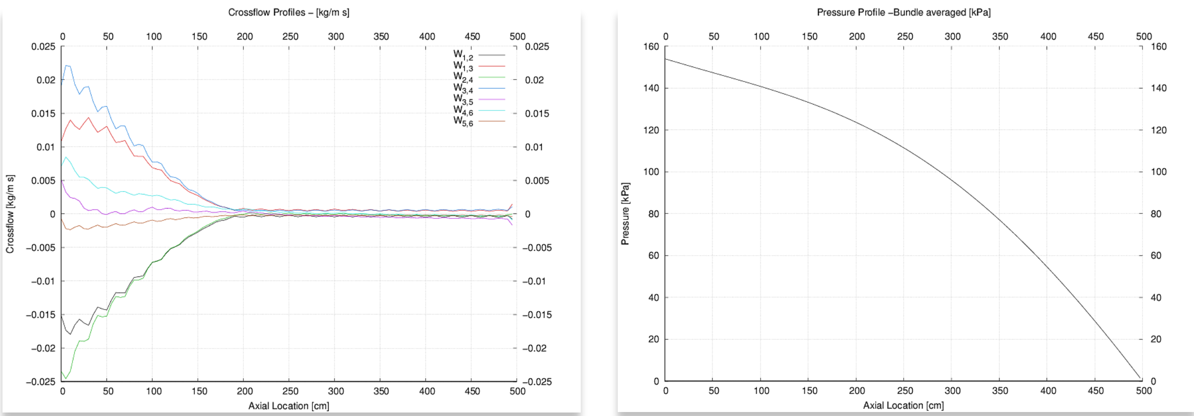
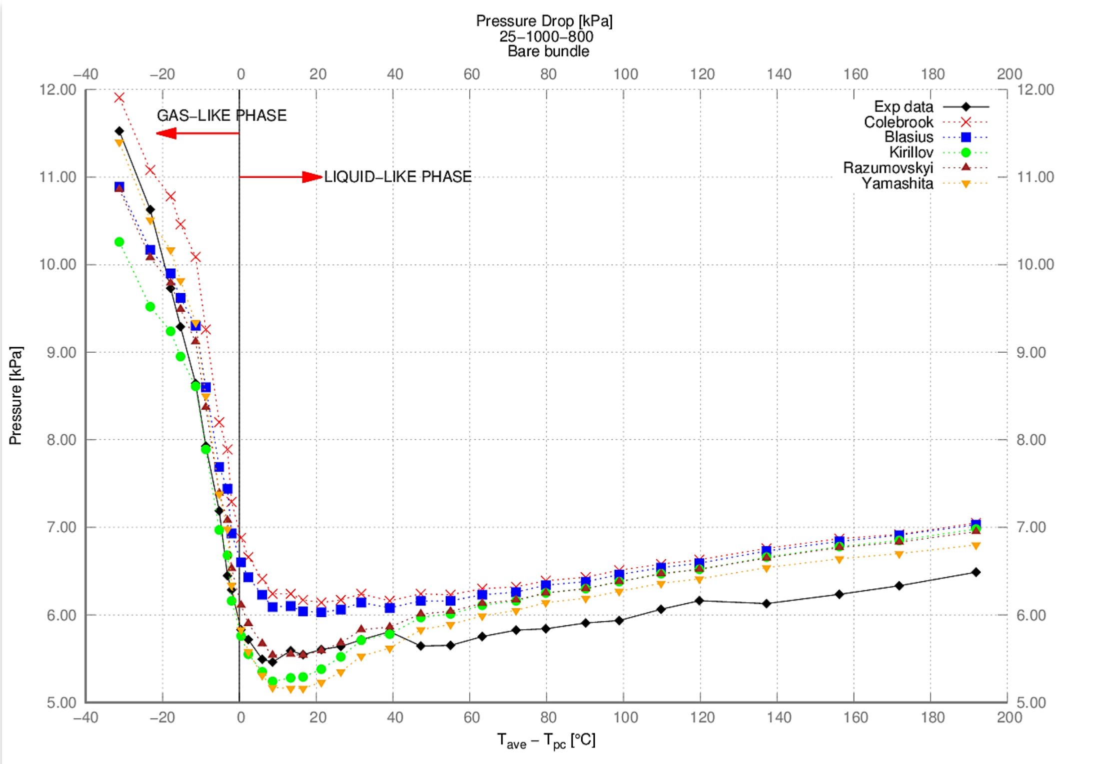


Figure 3.43: ASSERT-PV predictions of the cross-flow and pressure profiles for the Canadian SCWR fuel channel



An update of the three identified components have been implemented in the subchannel code ASSERT-PV V3R1m2 using the following models: the Cheng and Todreas inter-subchannel mixing model to take into account the swirl flow, the Cheng and Todreas hydraulic resistance model for wire-wrapped bundles, and a modification of the supercritical heat transfer correlation to take into account the wire wrap (fin effect) and the geometry effects. The updated model was assessed against experimental data performed with water flow through three-rod and seven-rod bundles at the National Technical University of Ukraine and with water flow through a four-rod bundle at the Xi'an Jiatong University. Preliminary results showed that isothermal friction factor correlations tend to over-predict the pressure drop for supercritical flows. Three supercritical friction factor correlation were assessed: Kirillov, Razumovskiy and Yamashita. Prediction results are consistent with the experimental trends. Figure 3.44 compares calculated friction factors using several friction factor correlations against experimental values of the 4-rod bundle test at XJTU. The use of a wire wrap model, which is a multiplier, resulted in overprediction of the pressure drop.

Figure 3.44: Pressure drop comparison between ASSERT-PV V3R1m[2] predictions and experimental data



The analytical model for China CSR1000 was established based on SCTRAN code which includes complete steam loop and feed water loop (Figure 3.45). Four start-up processes with control systems were put forward. The calculation results show that the thermal parameters of the circulation loop and once-through direct cycle meet the expectation, and the maximum cladding surface temperature does not exceed the limit temperature 650℃ (Figure 3.46). The feasibility of the start-up scheme and the security of the start-up process have been verified.

Figure 3.45: SCTRAN model of CSR1000 system

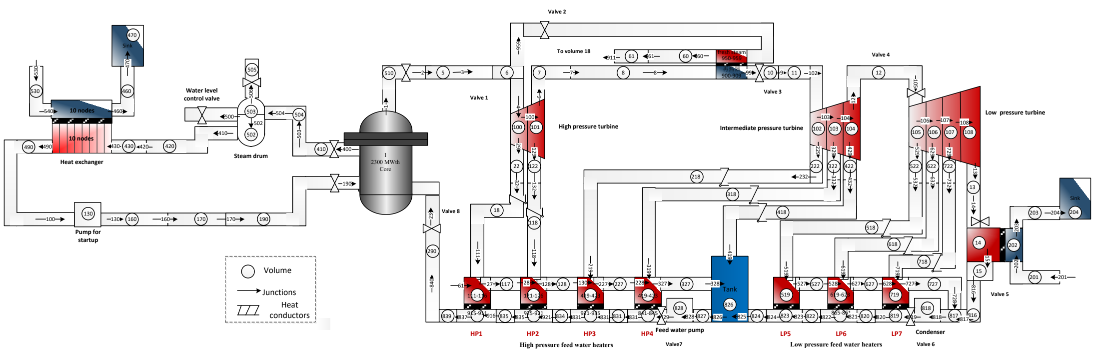
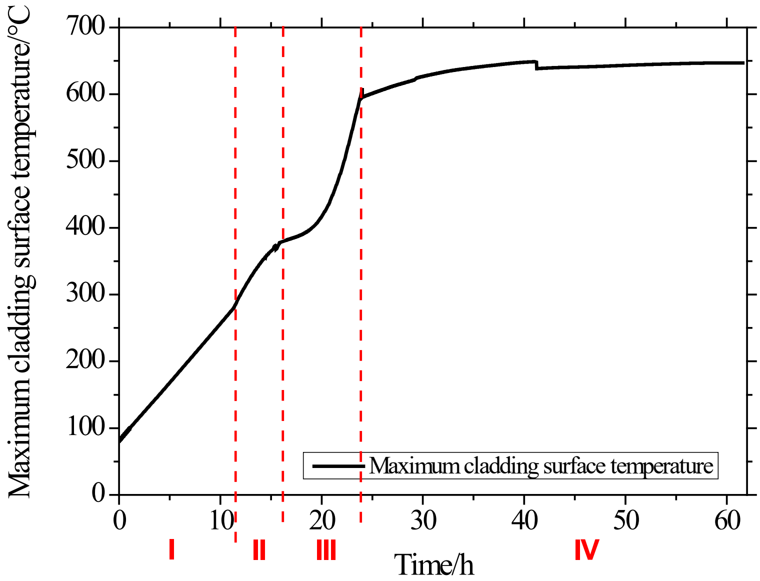


Figure 3.46: Maximum cladding surface temperature of start-up procedure



Three typical postulated accidents were analysed for CSR1000 during the start-up process (Figure 3.47): the loss-of-flow accident (LOFA), the loss-of-coolant accident (LOCA) and the reactivity insertion accident.

Figure 3.47: A schematic diagram of the SCWR safety system

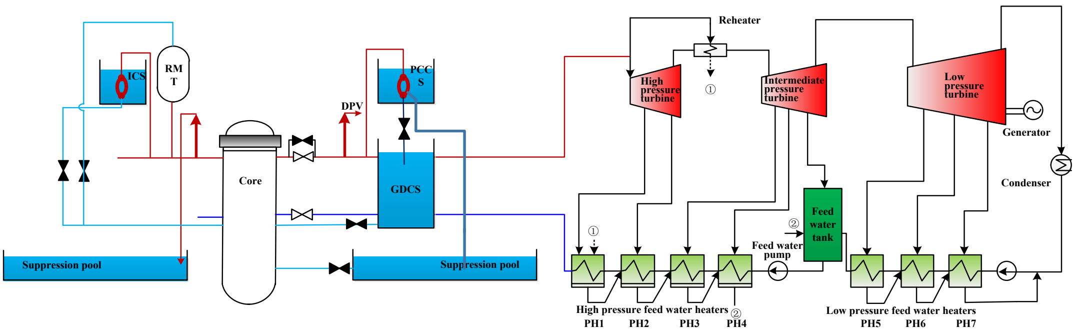


Figure 3.48 illustrates the responses of maximum cladding surface temperature with flow conditions and power for an LOFA that was initiated by the RCPs trip. Recirculation loop was always connected in the accident. Although the pump speed was 0 rpm after 5.0 s, there was still heat generated in the core. The recirculation loop flow did not drop to 0 and a lower natural circulation rate existed. After about 4.0 s (4.6 s-5.6 s) of shutdown, the RMT valve was triggered to open. After 200 s of the accident, the fill tank level became low, and ICS valve was triggered to open. The core remained single-phase throughout the start-up procedure avoiding any DNBR issue.

Figure 3.49 illustrates the responses of maximum cladding surface temperature with flow conditions and power of a reactivity insertion accident, which introduced a positive reactivity. Reactor power gradually increases and the maximum cladding surface temperature increases. In the low-pressure region (below 23 MPa), when the power reaches 120% of the set power, the "power high" signal triggers the reactor shutdown signal. The first and second-stage release valves of the steam pipe are triggered, when the core pressure drops. The opening of the relief valve causes fluctuations of coolant flow in the recirculation loop.

Figure 3.48: Results of LOFA analysis during the end of phase III

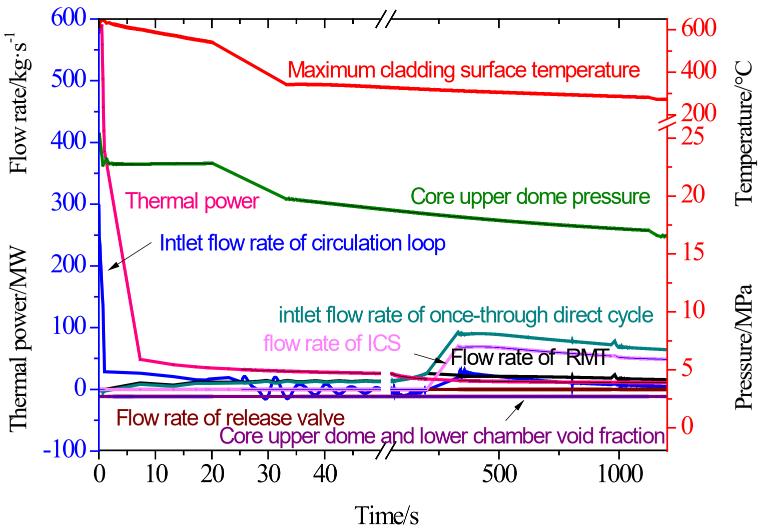
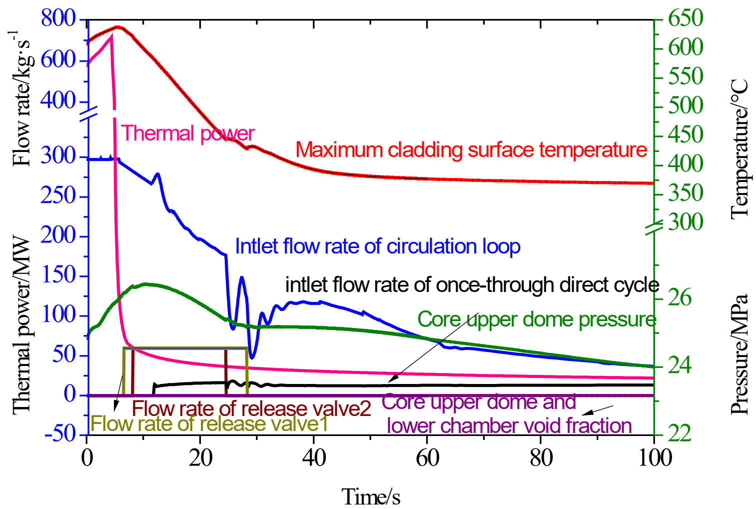


Figure 3.49: Results of reactivity insertion accident analysis during the end of phase III



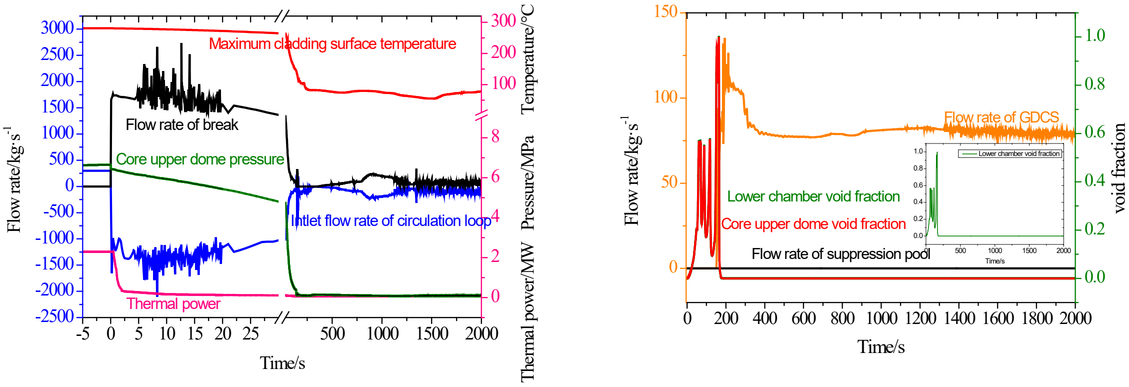
The LOCA accident analysis is based on the large break in the cold section of the recirculation loop (Figure 3.50). The accident occurred at 0 s, simultaneously, the system coolant quickly lost and the core coolant flow immediately dropped. Then, a counter-current occurred while the system pressure dropped rapidly. After the occurrence of a large break in the cold pipe section, a critical flow occurs at the breach. As the GDCS system was put into work, new coolant was injected into the system and the coolant flow at the breach was constantly fluctuating. When the coolant in the GDCS tank is not enough to cool the core, the coolant in the suppression pool begins to inject into the core, providing long-term cooling.

The analysis results show that the triggering signal can ensure the effective and timely operation of the safety system and ensure the safety of the reactor during the start-up process. The maximum cladding temperature of the reactor occurs at the end of the fourth stage of start-up procedure with LOCA. The temperature value is 850℃, which stays below the safety criterion by a large margin.

Figure 3.50: Results of LOCA accident analysis during the end of phase III

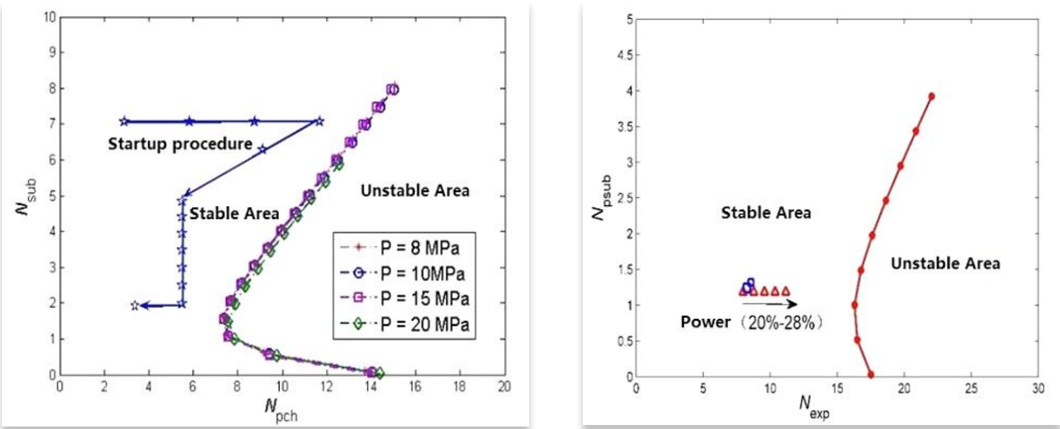
(a) The changes of core power, system pressure, maximum cladding surface temperature, break and loop flow rate

(b) The changes of safety system flow rate and void fraction



The density wave instability boundary of CSR1000 in the process of start-up and rated conditions are obtained based on the nuclear thermal coupling, frequency domain method (Figure 3.51). The sliding pressure start-up procedure has been shown to be in a stable region under the subcritical and the supercritical pressures.

Figure 3.51: Instability analysis of subcritical pressure and supercritical pressure



Pressure transient is a significant phenomenon related to the nuclear reactor safety, occurred under the process of the SCWR's start-up, shutdown and some accidents. Experiments on fluid heat transfer characteristics during pressure transients under supercritical pressures have been performed with R134a on SUFTEL (Supercritical Freon Test Loop). The experiment parameters were as follows, the inner diameter of the test tube was 10 mm. The mass flux was 800 kg/m^2·s and the heat flux varied from 30-60 kW/m^2. The pressure varied from 3.8 to 4.5 MPa in pressure increasing transients, while varied from 4.5 to 3.8 MPa in pressure decreasing transients. In the present experiments, the outer wall temperature varies rapidly during both the pressure increasing and decreasing transients. Figure 3.52 shows the variations of the wall temperature and outlet fluid temperature during pressure increasing transients, while Figure 3.53 is obtained during pressure decreasing transients at the corresponding same heat and mass flux with Figure 3.52. The beginning of the wall temperature jumping occurs when the pressure approaches to the critical pressure, therefore, a slow pressure changing should be taken when starting or shutting down the SCWRs to reduce the changing rate of wall temperature, especially close to the critical pressure.

Figure 3.52: Pressure increasing transients

(a) (b)

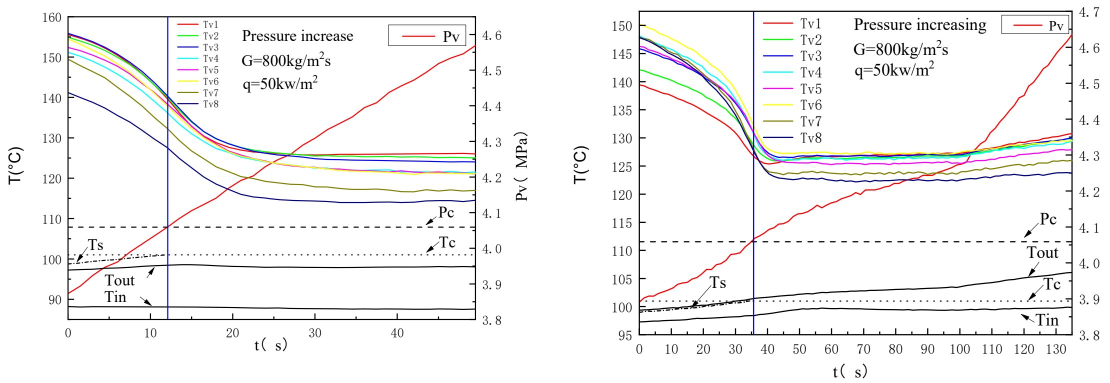
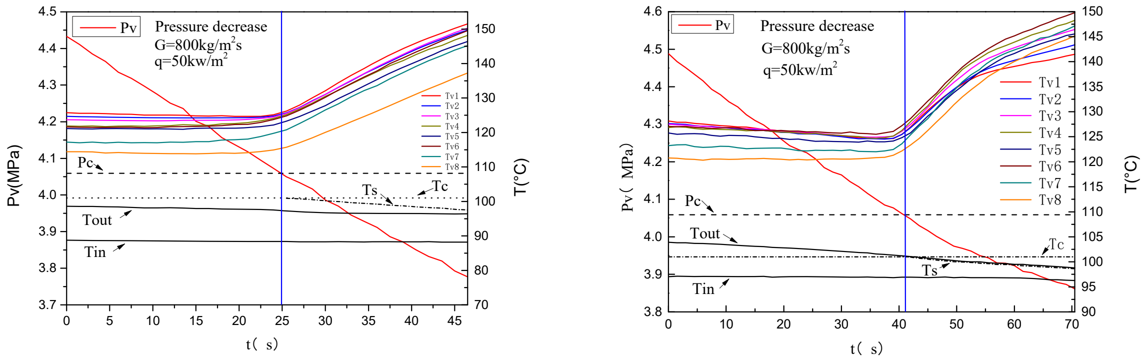


Figure 3.53: Pressure decreasing transients

(a) (b)



An experiment has recently been completed to obtain the wall temperature and heat transfer coefficient of water at subcritical pressures in a SCWR subchannel. The test section was wire-electrode cut to simulate the central subchannel of a 2×2 rod bundle (Figure 3.54). Experimental parameters covered the pressures of 11-19 MPa, mass fluxes of 700-1 300 kg/m²s and heat fluxes of 200-600 kW/m². Heat transfer characteristics in single-phase and two-phase regions were analysed with respect to the variations of heat flux, system pressure and mass flux. For a given pressure, it was found that the wall temperature increases with increasing heat flux or decreasing mass flux in the steam-water two-phase region.

Figure 3.54: Structure of the subchannel test section

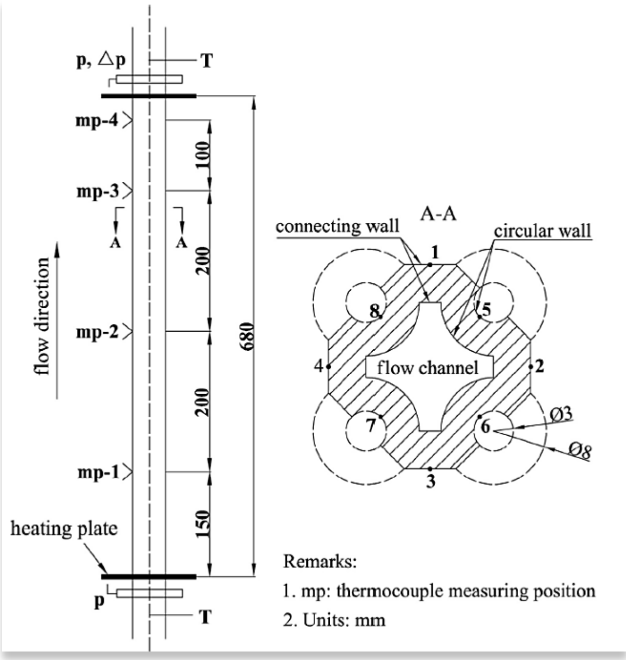
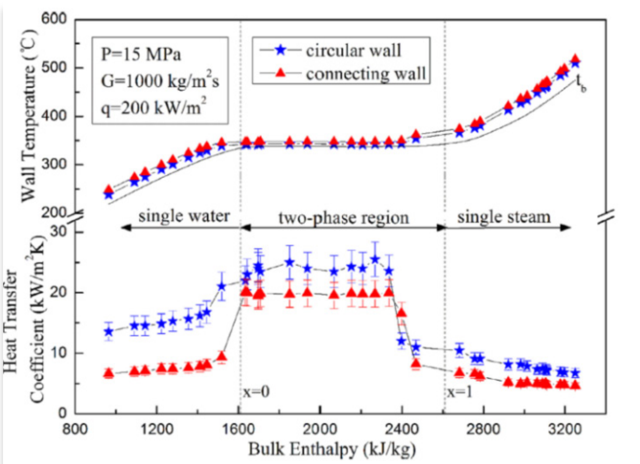


Figure 3.55 expresses the variations of inner wall temperature and heat transfer coefficient with bulk enthalpy at the connecting wall (measuring points 1, 2, 3 and 4 in Figure 3.54) and the circular wall (measuring points 5, 6, 7 and 8 in Figure 3.54). It is seen that the wall temperature of the circular wall is lower than that of the connecting wall within the entire bulk enthalpy region. As a consequence, the corresponding heat transfer coefficient of the circular wall is relatively high.

Figure 3.55: Heat transfer difference along the circumference of the test section.



The variations of wall temperature and heat transfer coefficient with bulk enthalpy and system pressure are shown in Figure 3.56. It is seen in Figure 3.56a that the wall temperatures overlap with each other in low-enthalpy single-phase region in which the bulk temperatures are insensitive to pressure change. With the increase of bulk enthalpy, the bulk flow enters into two-phase region at 11 MPa first, followed by 15 MPa and 19 MPa.

In this region, the wall temperature keeps nearly constant, but the temperature level is promoted with increasing pressure as the saturated bulk temperature is increased with pressure. Deteriorated heat transfer occurs at the pressures of 15 MPa and 19 MPa with a slight rise in wall temperature near the bulk enthalpy of 2 000 kJ/kg. In high steam quality region, the wall temperatures increase with bulk enthalpy accordingly. The higher the pressure is, the higher the wall temperature will be. From the distributions of heat transfer coefficient plotted in Figure 3.56b, it is concluded that the effect of pressure on heat transfer coefficient is weak in the subcooled-water region and superheated steam region.

Figure 3.56: Effects of pressure on heat transfer characteristics: (a) wall temperature; (b) heat transfer coefficient

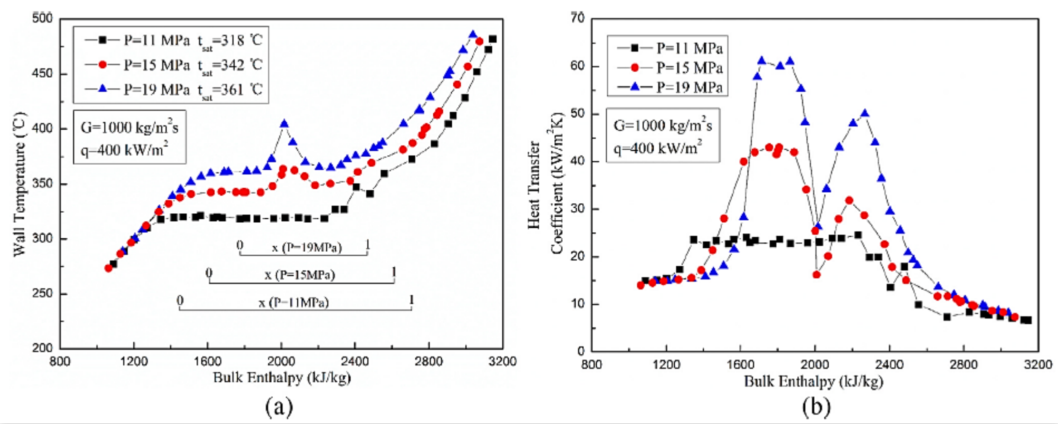


Figure 3.57 expresses the profiles of wall temperature and heat transfer coefficient plotted against bulk enthalpy and mass flux at the pressure of 15 MPa and heat flux of 400 kW/m^2. At a high mass flux of 1 300 kg/m^{2}s, the wall temperature increases gradually with bulk enthalpy in single-phase subcooled region. As the bulk flow approaches the saturated temperature, the wall temperature remains steady in the two-phase region until the steam quality reaches about 0.85, and finally increases with bulk enthalpy in the superheated region. When the mass flux is decreased to 1 000 kg/m^{2}s, the wall temperature profile varies similarly except that a mild heat transfer deterioration appears at a steam quality of about 0.28. A further decrease in mass flux leads to a higher peak of the wall temperature.

Critical heat flux (CHF) experiment with uniform heating was performed in a tube of 8.2 mm in inner diameter and 2.4 m in heated length. The water flowed upward through the test section. The pressure covered the range from 8.6 to 20.8 MPa, mass flux 1 157 to 3 776 kg/m^{2}s, inlet quality -2.79 to -0.08 (subcooling 19-337℃), and local quality -0.97 to 0.53. For the pressure close to the near-critical point, the CHF decreased substantially with the pressure increasing. For the subcooling larger than a certain value, the CHF was related to the local condition. But for low subcooling and saturated condition, the CHF was related to the total power. The present results were in agreement with the previous experiment for the same local subcooled condition. Based on the present experimental results with subcooled and saturated conditions, an empirical relation of the CHF was presented. The comparison of the experimental data with the present correlation is shown in Figure 3.58. For more than 95% data points, the deviations are less than 15%, and the average error and the root mean-square error are 0.03% and 7.3%, respectively.

Figure 3.57: Effects of mass flux on heat transfer characteristics: (a) wall temperature; (b) heat transfer coefficient

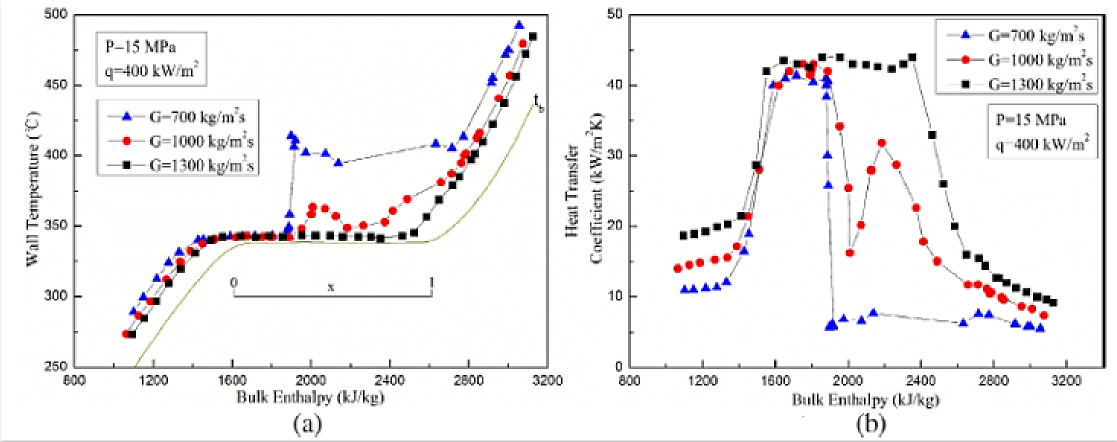
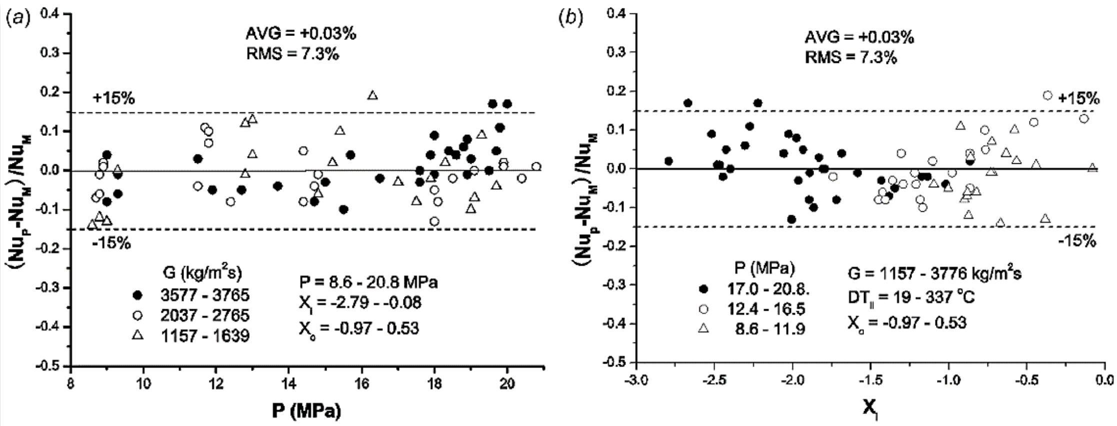
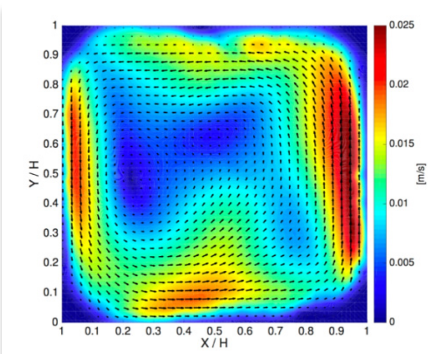


Figure 3.58: The comparison of the present experimental results with the empirical correlation



Flow field in a Rayleigh Bénard cell was investigated both by experimental and numerical method. Figure 3.59 shows the results measured with particle image velocimetry (PIV). The colour scale indicates the magnitude of the mean velocity, while the arrows show its direction.

Figure 3.59: Time averaged velocity field of the PIV experiment



An experimental data bank for heat transfer of supercritical water in circular tubes was established with data from 24 different sources, a wide range of parameters and more than 20 000 data points. Heat transfer data from the databank were partly selected to develop heat transfer correlations in circular tubes at supercritical conditions. The selection criteria consist of data reproducibility, consistence in energy balance, consistence in other parameters and comparability with each other. A methodology has been also used to assess the intrinsic consistency of the experimental information contained in the databank. Contributions of the neighbouring (experimental) nodes were considered (whenever possible) for each experimental node. The contributions are just scaled values of the corresponding experimental values. According to the above mentioned criteria, more than 14 000 data points were selected and considered as reliable data, which can be used for further purposes, e.g. development of correlations.The experimental data in supercritical water were compared with several correlations for supercritical fluid. As shown in Table 3.9, most of the correlations overestimate the heat transfer coefficient (HTC). At G=500-1 500 kg/m^{2}s, with lower q/G ratio, correlations deliver better prediction of the experimental data. With higher q/G ratio, the phenomenon of heat transfer deterioration (HTD) is more obvious and leads to bigger deviation between experimental data and correlations.

Table 3.9: Comparison of test data in circular tube with various correlations

Test data From	N		Correlations					
			D-B	Bishop	Swenson	Jackson	Cheng	Watt
G=500-1500 q/G=0-0.4	5518	μ	0.489	0.378	0.112	0.214	0.101	0.176
		σ	0.524	0.295	0.194	0.379	0.278	0.258
G=500-1500 q/G=0.4-0.8	11638	μ	0.501	0.587	0.489	0.465	0.215	0.25
		σ	0.562	0.687	0.515	0.389	0.198	0.301
G=500-1500 q/G=0.8-1.2	6255	μ	0.521	0.457	0.415	0.687	0.274	0.487

In order to evaluate dimensionless numbers for developing heat transfer coefficient (HTC) of SCW in circular tubes, 14 dimensionless numbers were selected, as indicated in Table 3.10.

Table 3.10: Dimensionless numbers for developing heat transfer correlations

ID No.	1	2	3	4	5	6	7
Dimensionless Numbers	P/P_c	$\frac{T_B - T_{PC}}{T_{PC} - T_c}$	$\frac{h_B - h_{PC}}{h_{PC} - h_C}$	$\frac{\rho_W}{\rho_B}$	$\frac{C_{P,A}}{C_{P,B}}$	$\frac{\mu_W}{\mu_B}$	$\frac{\lambda_W}{\lambda_B}$
ID No.	8	9	10	11	12	13	14
Dimensionless Numbers	Re_B	Pr_B	$\frac{q \cdot \beta_B D}{\lambda_B}$	$\frac{q \cdot \beta_B}{C_{P,B} G}$	Gr_B	$Gr_{q,B}$	Bu

Three correlation methods were used to evaluate the effect of various dimensionless parameters on heat transfer, i.e.:

- distance correlation;
- Pearson product momentum correlation;
- Spearman’s rank correlation.

Figure 3.60 shows the levels of the correlation factor of the HTC deviation factor (with respect to the D-B correlation) for the 14 dimensionless parameters with the three correlation methods above. It may be seen from Figure 3.60 that the dimensionless numbers with the IDs 4, 7 and 10 (to be identified in Table 3.9) have rather higher levels of the correlation factor and can be further considered for the development of heat transfer correlations.

CFD simulation was carried out using LES method. One of main objectives is to investigate the effect of the conjugated heat transfer. Figure 3.61 shows the results of the transient wall temperature behaviour with and without conjugated heat transfer.

Figure 3.60: Correlation factors for 14 dimensionless numbers

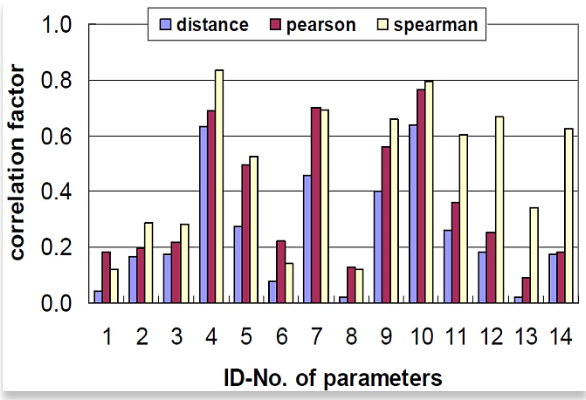
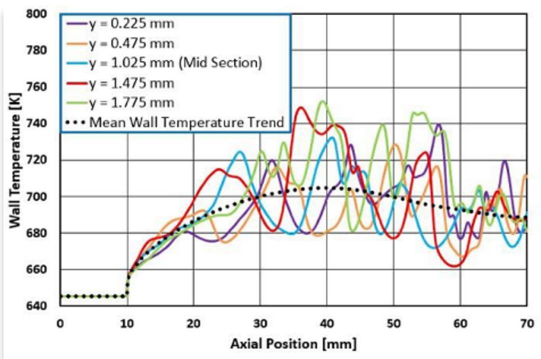
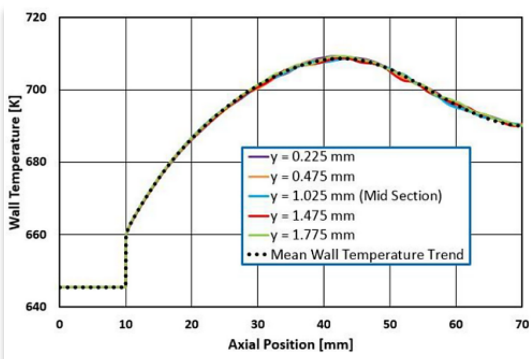


Figure 3.61: Effect of conjugated heat transfer on wall temperature

a) imposed heat flux at the internal wall surface



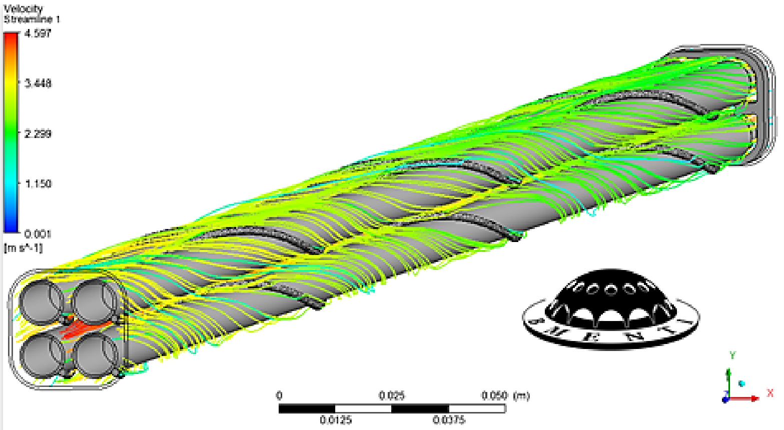
b) conjugate heat transfer



As it can be noted by the screenshots of wall temperature distribution reported in Figure 3.61a and Figure 3.61b, when the heat flux is imposed at the wall surface (2nd kind boundary condition) the wall surface temperature is allowed to oscillate, owing to the local and instantaneous changes in heat transfer efficiency, affected by turbulence; when a conjugate heat transfer approach (3rd kind boundary condition) with wall dynamics is adopted, instead, a considerable damping of the oscillations is observed. In fluids with strong dependence of properties on temperature (as in supercritical pressure fluids close to the pseudocritical conditions), this results in changing the velocity pattern close to the wall, i.e. turbulence characteristics. Details on these effects are reported in Pucciarelli and Ambrosini.

The so-called wrapped wire (or wire wrap) spacer explicates more mayor effects on the flow field of SC water in the investigated geometry. The most evident one of these effects is the change in the main direction of the SC water flow. It can be well visualised by three-dimensional streamlines (Figure 3.62). Three-dimensional twisting like flow (which is caused by the so-called guiding effect of the wire geometry) can be recognised by the curved streamlines which strictly follows the curvature of the wires (Figure 3.62).

Figure 3.62: Flow behaviour in a 2x2 rod bundle



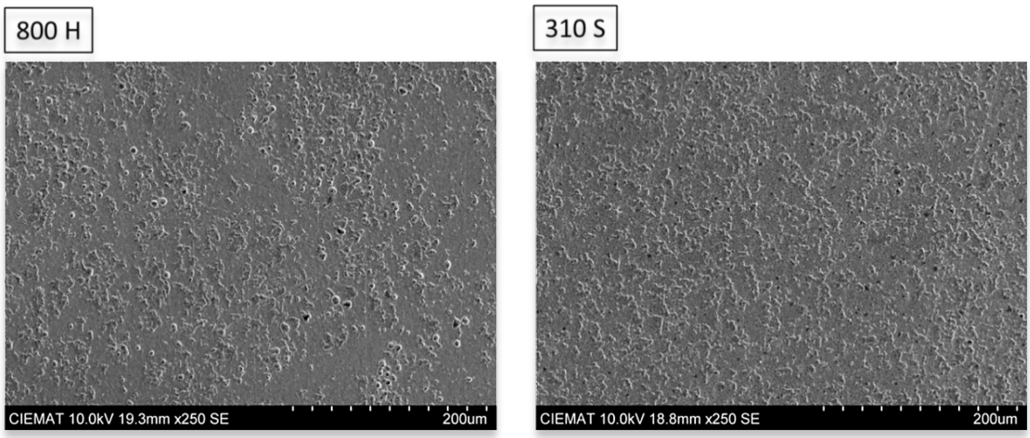
Materials and chemistry

In 2017, 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. Also during 2017, collaborations have been established to examine: stress corrosion cracking (SCC) under SCW conditions on candidate alloys; the effects of coatings on oxidation behaviour; and irradiation effects on mechanical properties and microstructure evolution. The ultimate goal has been to promote activities towards in-pile tests both in Europe and China.

European laboratories along with Canadian and Chinese research centres are involved in the 2nd international round robin testing on the oxidation behaviour of candidate materials (Alloys 310 and 800) in SCW. The aim of this co-operative effort is to study the oxidation behaviour of candidate materials for SCWRs and, at the same time, identify the discrepancies observed in the 1st round robin testing. Rectangular coupon specimens of alloy 800H and 310S stainless steel were prepared by JRC IET and distributed to participants. Coupons were exposed for 1 000 h to deoxygenated supercritical water at 550°C and 25 MPa. At CNL, the coupons were exposed in a 500 mL static autoclave constructed of alloy 625. Both alloys showed remarkably similar corrosion behaviour, at approximately 10 mg/dm^2 weight gain and 25 mg/dm^2 descaled weight loss (0.5 µm). Results from Ciemat have shown differences in the oxidation behaviour of alloy 800 H and 310 S in deaerated SCW at 550ºC:

- Weight gain of alloy 800 H samples tested in SCW at 550ºC are greater than the weight gain of 310 S samples.
- It was found from SEM analysis that both alloys were covered by oxide particles. Nevertheless, the density of these particles was higher in the 310 S samples (Figure 3.63).
- According to the Auger elemental composition profiles Cr is incorporated to the outer layer in the 800 H samples but not in the 310 S.

Figure 3.63: Morphology of the oxides found in the 800 H and 310 S tested at 550°C in deaerated SCW



The results from the second round robin testing will be published in 2018. During 2017, the first step has been taken to prepare a third international round robin testing for the SCC behaviour of candidate materials. This Round Robin Testing will start in 2018.

In 2017, Canada focused on construction and planning of new testing facilities. The high temperature Supercritical Water Chemistry and Materials Test Loop was constructed and partly commissioned (see Figure 3.64). This loop is designed to allow 1 000 h corrosion tests at 25 MPa and 795°C, and 850°C and 19 MPa, using chemistry controlled and purified water. This loop has also been equipped with thermocouples to monitor the degradation of heat transfer as oxide films grow.

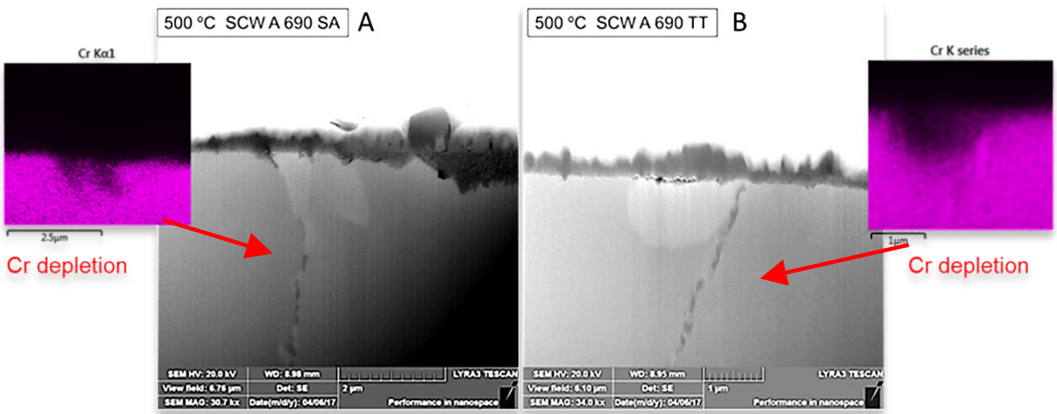
Figure 3.64: Supercritical Water Chemistry and Materials Test Loop

Ciemat activities in 2017 were mainly focused on the study of the influence of intergranular carbides in the corrosion behaviour of nickel-based alloy 690, which has been used as a replacement of alloy 600 in PWR for many years. Alloy 690 has shown an optimum behaviour to stress corrosion cracking since it was first installed in the late 1980s. For this reason, it was selected as a candidate material for the SCWR. However, recent results have pointed out that, contrary to the expectation, the SCC crack growth rate of alloy 690 is lower without intergranular carbides in its microstructure, in PWR primary water conditions. Considering these results, the Structural Materials Division of Ciemat started two years ago a line of research on the role of intergranular chromium carbides in the oxidation and SCC behaviour of nickel-based alloys in supercritical water.

As part of this work, Alloy 690 TT and SA (with and without intergranular carbides, respectively) oxidation samples and tensile specimens machined from a Control Rod Drive Mechanism (CRDM) were tested in deaerated (<10 ppb O_2) SCW at 400°C and 500°C and 25 MPa. After these tests, samples were weighed analysed by Scanning Electron Microscope equipped with a focused ion beam (SEM/FIB). Moreover the elemental composition profiles of the oxide layers were obtained by Auger spectroscopy. Results from this work have shown that the alloy 690 has an optimum oxidation behaviour in SCW. In all cases thin oxide layers were observed after 500 h of testing.

The absence of carbides in the grain boundaries of alloy 690 seems to promote the oxidation processes in the material. A possible explanation is that the absence of carbides promotes the diffusion of elements along the grain boundaries. Some observations showed Cr depletion in the grain boundaries of Alloy 690 TT and SA samples 500°C in SCW. This result was confirmed studying the cross sections of selected samples obtained by FIB (Figure 3.65). It is suspected that the Cr depleted zones observed in the surface of the samples may be produced by the carbides dissolution in SCW. In addition to this, the Cr depletion underneath the surface of the material may occur as a consequence of the migration of grain boundaries. However, more work is needed to elucidate these points. Differences in the Fe composition in the oxide layer of the specimens tested in SCW at 400°C and 500°C support the idea stated by other authors of a possible change in the corrosion mechanism in SCW between 400°C and 500°C.

Figure 3.65: Cr depletion zones observed in the cross sections of A 690 SA and A690 TT. The cross sections were obtained by FIB



From the study of the fracture surfaces of the tensile specimens of A 690 TT and SA tested in deaerated SCW at 500°C many questions about the role of creep mechanisms in the corrosion behaviour of alloy 690 in SCW rose up. Nevertheless, more work is planned to be done in order to examine this subject in depth.

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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 fuel recycling be developed and qualified for use. Important safety features of the SFR 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, nitrogen gas, and supercritical carbon dioxide are considered as working fluids for the power conversion system to achieve high performance in terms of thermal efficiency, safety and reliability. With innovations to reduce capital cost, the SFR is aimed to be economically competitive in future electricity markets. In addition, the fast neutron spectrum greatly extends the uranium resources compared to thermal reactors. The SFR is considered to be the nearest-term deployable system for actinide management.

Much of the basic technology for the SFR has been established in former fast reactor programmes including recently the Phénix end-of-life tests, and will be continued with the ASTRID project in France, the restart of Joyo in Japan, the lifetime extension of the BN-600, the operation of the BN-800 and development of the BN-1200 project 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 reprocessing at a central location serving a number of reactors as shown in Figure 3.66.
- An intermediate-to-large size (300 to 1 500 MWe) pool-type reactor with oxide or metal or nitride fuel and potentially minor actinides as shown in Figures 3.67 to 3.69.
- 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 3.70.

The two primary fuel recycle technology options are (1) advanced aqueous and (2) pyrometallurgical reprocessing. 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 reprocessing. Mixed nitride fuel potentially can be recycled by both advanced aqueous and pyrometallurgical reprocessing methods.

Figure 3.66: Japanese sodium-cooled fast reactor (loop-configuration SFR)

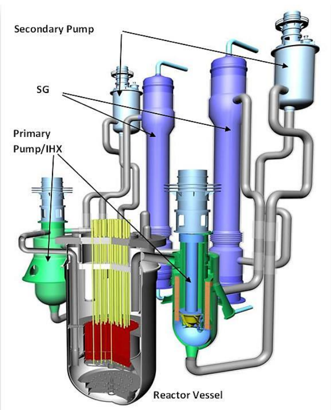


Figure 3.67: Korea advanced liquid metal reactor (pool-configuration SFR)

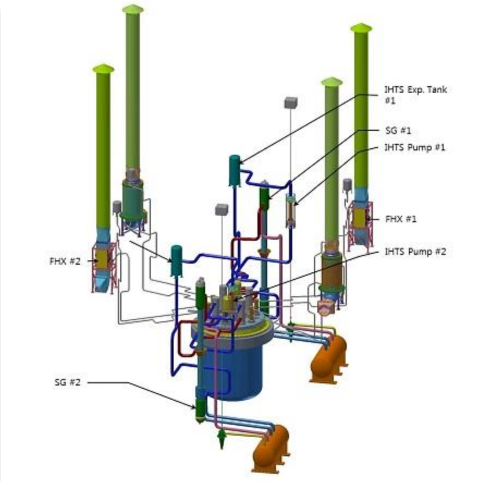


Figure 3.68: ESFR (pool-configuration SFR)

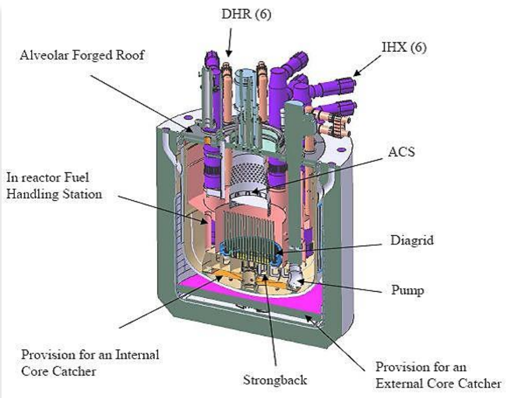


Figure 3.69: BN-1200 (pool-configuration SFR)

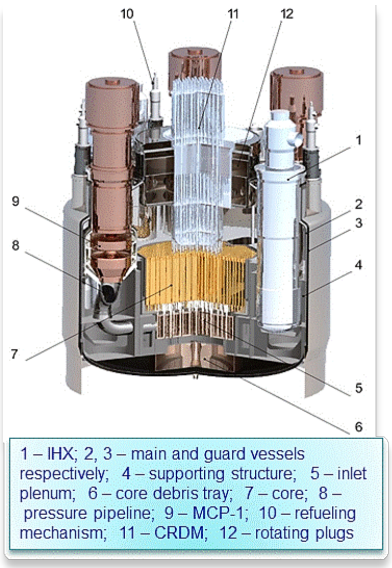
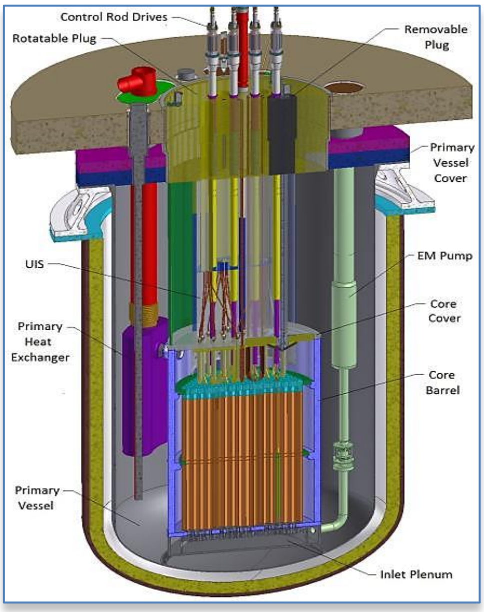


Figure 3.70: AFR-100 (small modular SFR configuration)



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;
- Ministry of Science and Technology, China.
- State Atomic Energy Corporation Rosatom, Russia.

Three Project Arrangements (PAs) were signed in 2007: Advanced Fuel (AF), Component Design and Balance-of-Plant (CD&BOP), and Global Actinide Cycle International Demonstration (GACID), and the PA for Safety and Operation (SO) was signed in 2009, the PA for System Integration and Arrangement (SIA) was signed in 2014.

The PA for AF and the PA for GACID expired in 2017 and the PA for CD&BOP extended for another ten years in 2017. It is started process of signing a new PA (Phase II) for AF for next ten years.

R&D objectives

The SFR development approach is based 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 four projects. The scope and objectives of the R&D to be carried out in these 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 are 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, Joyo, Phénix, BN-600, BN-800 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 inherent safety features and passive safety systems.

Advanced fuel project (AF: presently expired and phase II project is under preparation)

The Advanced Fuel Project aims at developing minor actinide-bearing (MA-bearing) high burnup fuel for SFRs 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-MA-bearing 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 (BOP) cover 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, sensor development and data treatment, for example, to detect defects using sensors immersed beneath the sodium surface. Some other important topics have been dealt with key technologies such as leak-before-break (LBB) assessment, steam generators and development of alternative energy conversion systems, e.g. using a Brayton cycle. 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: (1) development of advanced, high reliability steam generators and related instrumentation; and (2) the development of advanced energy conversion systems based on a Brayton cycle with supercritical carbon dioxide and nitrogen as the working fluid. In addition, the significance of the experience that has been gained from SFR operation and upgrading is recognised.

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 Proliferation Resistance and Physical Protection (PRPP) methodology.
  - **2014:** Solicit safety assessment using RSWG methodology.

SO project:

- Methods, models and codes.
  - **2008-2011:** Research collaboration on methods, models and codes for safety technology and evaluation among four countries of France, Japan, Korea and United States.
  - **From 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, Joyo, Phénix, BN-600 and CEFR) between France, Japan, Korea and United States. (Collaboration with Korea started in 2009).
  - **From 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.
  - **From 2012:** Research collaboration between Euratom, China, France, Japan, Korea, Russia 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:

- **2007-2021:** Viability study of proposed concepts.
- **2009-2027:** Performance tests for detailed design specification.
- **2014-2027:** Demonstration of system performance.

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 work packages (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 are 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 five design tracks are: JSFR (JAEA, loop), KALIMER (KAERI, pool), ESFR (Euratom, pool), BN-1200 (Rosatom, pool), and AFR-100 (DOE, modular). These tracks cover a broad range of SFR design characteristics. The China CFR-1200 may be proposed as design track in future.

The list of R&D needs has been updated by the SIA PMB members at every PMB meeting and has been 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 has shown to be 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 2017, the following trade studies were contributed within WP 1.1.2:

- CFR1200 design requirements study (Task 1.1.2.CH1);
- R&D needs for low void worth core safety (Task 1.1.2.EU1);

- Operation procedures to comply with grid regulation and plant lifetime constraints (Task 1.1.2.FR1);
- Metal and Oxide Core Trade Study (Task 1.1.2.US+JP1);
- Study of PGSFR design issues (Task 1.1.2.KR1);
- Safety Self-Assessment of JSFR Track (Task 1.3.2.JP1) was contributed within WP 1.3.2.

Safety and operation project

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

- WP SO 1: Methods, models and codes

CIAE (China) would like to analyse the heat transfer of fuel assembly during transportation process for CEFR. The basic function of the refuelling system is to transfer new fuel assembly to the core and transport spent fuel assembly to the cooling pool outside the reactor. During the transport process, the fuel assembly is exposed to different environments, which may result in a decrease in heat transfer capacity. If an accident occurs, the spent fuel assembly will be in the environment for a long time, and the adverse heat transfer condition will lead to the increase of the cladding temperature and even the damage. Therefore, it is necessary to obtain the maximum temperature of the cladding, by analysing fuel assembly heat transfer conditions during transportation process.

This research includes the below fields and conditions.

- screening representative transport conditions;
- build the test facility;
- the experiments include natural convection heat transfer experiments and forced convection heat transfer experiments;
- development of the code for fuel assembly heat transfer calculation during transportation process.

The research programme consists of four steps as below, and will be finished by the end of 2019.

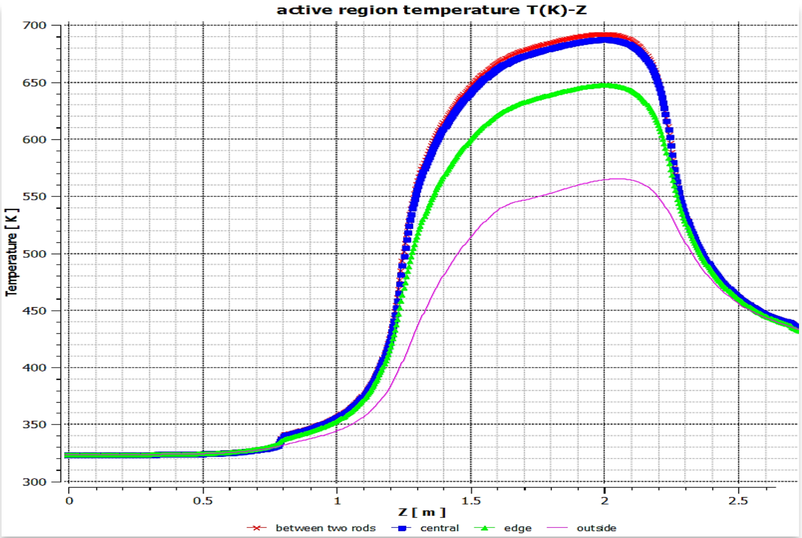
- research programmes will be determined;
- test facility will be build;
- test research work will be completed;
- the code development will be completed.

The temperature along the test section based on the preliminary analysis is shown in the Figure 3.71. Although there are some differences in the main gas temperature, the temperature of the gas is rising and then falling in the box or outside the box, and the maximum value appears at the top of the active zone. Gas temperature in the lower transfer zone began to rise slowly. Then the fuel rod power surge around the active zone, and its temperature increased exponentially. And then the temperature increases slowly because of the increase in the local gas flow rate and at a higher level. Heat can be transmitted to natural gas by heat conduction. The highest temperature is close to the top of the active section. After the inflection point is the upper conversion area, where the power is reduced and the temperature decreases rapidly. It can be seen that the gas temperature inside the box is closely related to the power of the fuel rods. There are two reasons. On the one hand, the gas flow in the box is weak. On the other hand, the natural convection heat transfer ability outside the box is stronger, and the heat can be rapidly transferred to environment along the radial direction, thereby reducing the temperature accumulation in the axial direction.

These analyses led to four conclusions as below.

- Flow in the component box is very weak. It is laminar flow.
- Flow outside the component is strong. It can form turbulent heat transfer.
- The gas temperature in the axial direction is closely related to the fuel rod power.
- Need to consider radiant heat transfer, especially in the box.

Figure 3.71: The predicted temperature along the test section based on the preliminary analysis



Euratom modelled ASTRID-like reactor building using three-dimensional finite elements, and performed dynamic analyses (Figure 3.72).

The objectives of this study were:

- feasibility study of the finite element model of the ASTRID-like reactor structure with the modelling of seismic isolators;
- the seismic response of the structure for various configurations of seismic devices;
- determination of the nuclear island displacements, accelerations, floor spectrum and the isolators displacements, shear and axial forces;
- assessment of the island response and the bearing capacity with the minimum and maximal isolator stiffness;
- study the floor spectrum for different points of interest of the isolated island response;
- verification of the design criteria for the isolators specified by the codes.

The characters of structure modelling were as follows:

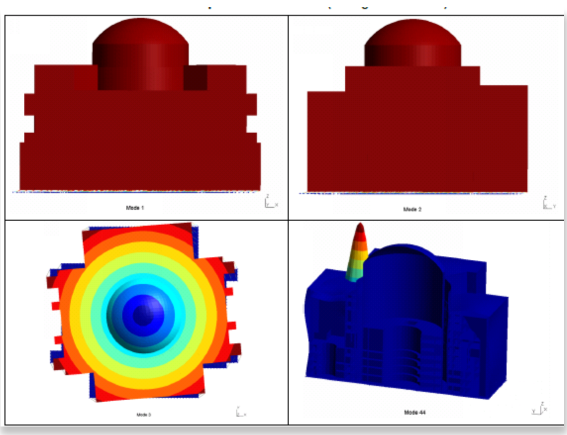
- the entire nuclear island is modelled by three-dimensional finite elements;

- slabs, walls, basemats are represented by shell elements;
- beams and columns are modelled using beam elements;
- soil structure interaction is represented using spring elements beneath the basemat;
- the behaviour of these elements is linear elastic;
- base isolators are represented by special finite elements connecting the nodes between the upper and lower basemats, using a linear behaviour of both vertical and horizontal stiffness;
- the properties of each element are adjusted in order to represent the mechanical behaviour of the number of real isolation devices.

From these analyses the design criteria for earthquake situation, bearing modelling were verified and the acceleration floor response spectra were calculated as below.

- various configurations of seismic devices are considered accounting for the variability of the shear properties of the bearings;
- verification of the design criteria for earthquake situation for all the configurations including: isolator repartition; modal analysis; displacements and accelerations (spectral method); bearing solicitations;
- bearings verification;
- acceleration floor response spectra;
- proposal for alternative design solution.

Figure 3.72: Principal mode shapes



JAEA performed the kinetic study of sodium-concrete reaction (SCR) for safety assessment from the view point of chemical reaction.

The objectives of this study is as below:

- confirm the reaction behaviour of NaOH-SiO2 by thermal analysis;
- consider SCR kinetic feature in comparison with the other reactions.

Therefore JAEA performed NaOH-SiO_2 reaction experiment.

- exothermic peaks were identified just after the melting point of NaOH (around 584 K);

- no peak shift when increasing heating rate (for the other reactions, peak shifts were identified);
  - reaction speed is too fast to set the condition of heating rate;
  - faster than reaction rate of the other possible reactions.
- sample eruption was observed just after the melting point of NaOH.
  - reaction product (water/vapour) by rapid heating is driving force.

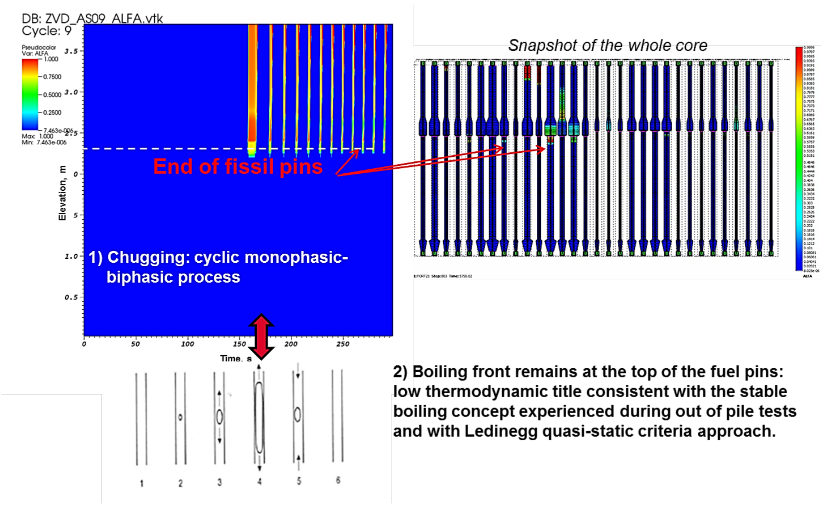
The conclusions in this study are shown below:

- Thermal behaviour of NaOH-SiO_2 reaction was investigated.
- Reaction onset was observed just after the melting point of NaOH.
- Eruption behaviour was observed by in situ measurement.
- Reaction behaviour of siliceous concrete with NaOH is similar to that of reagent based NaOH-SiO_2 reaction. Similar reaction behaviour of NaOH with Al_2O_3 as minor concrete composition was observed as well.
- Reactivity of NaOH-SiO_2 was more significant than the other possible reaction such as Na-SiO_2 reaction and Na_2O-SiO_2 reaction in the early stage of secondary mode of SCR.

CEA investigated sodium boiling phenomena with the CATHARE 2 thermal-hydraulic system code during a postulated ULOF transient. This study focuses on a stabilised boiling case, allowing to avoid a fast temperature excursion in the fuel channel above the Na boiling temperature. In stabilised regime, the inlet cooling flow rate can be sustained under natural circulation within the subassemblies even if they have reached the saturation temperature. The two-phase flow quality remains low (typically below 1%). In case of unstabilised boiling, a flow redistribution would lead to the downwards progression of the boiling front within the core: the subsequent rise of the quality would then induce fuel pins dry-out.

By the investigation with the CATHARE 2 code, 1D axial void fraction profile along the hottest core subassembly was obtained as shown in Figure 3.73. Boiling front remains at the top of the fuel pins: low thermodynamic title consistent with the stable boiling concept experienced during out-of-pile tests and with Ledinegg quasi-static criteria approach.

Figure 3.73: 1D axial Void fraction profile along the hottest core subassembly



The US DOE carried out work in two areas: 1) the development of mechanistic source term (MST) analysis capabilities, and 2) the development of an advanced, reduced-order three-dimensional modelling capability to represent thermal flow phenomena such as thermal stratification.

A trial mechanistic source term calculation was performed for a metal fuel, pool-type sodium fast reactor. This project consisted of two efforts. First, a mechanistic calculation was performed utilising best-estimate models and data to identify potential gaps in the current knowledge base. Figure 3.74 graphically depicts the analysis steps of the mechanistic source term calculation, which included the reactor response to the transient scenario, radionuclide release from damaged fuel, radionuclide transport through the reactor system, and off-site consequences. In parallel, a simplified sensitivity analysis was conducted with the goal of determining the importance of particular radionuclides and phenomena on off-site dose. The findings of both efforts were then combined to outline future research needs and develop a potential path forward. The results of the analysis predicted small off-site doses and demonstrated that a mechanistic source term calculation is possible utilising current modelling tools and data. However, gaps in available data and tools result in uncertainties or the use of conservative assumptions that could make it difficult for future SFR vendors to reduce site boundaries and emergency planning zones. The gaps were prioritised based on the findings of the sensitivity analysis and recommendations for future research were provided, shown in Figure 3.75. Of these gaps, the modelling of radionuclide transport within noble gas bubbles in the sodium pool and radionuclide migration within the fuel pins during irradiation were highlighted as the highest priority.

Figure 3.74: Mechanistic source term calculation analysis steps

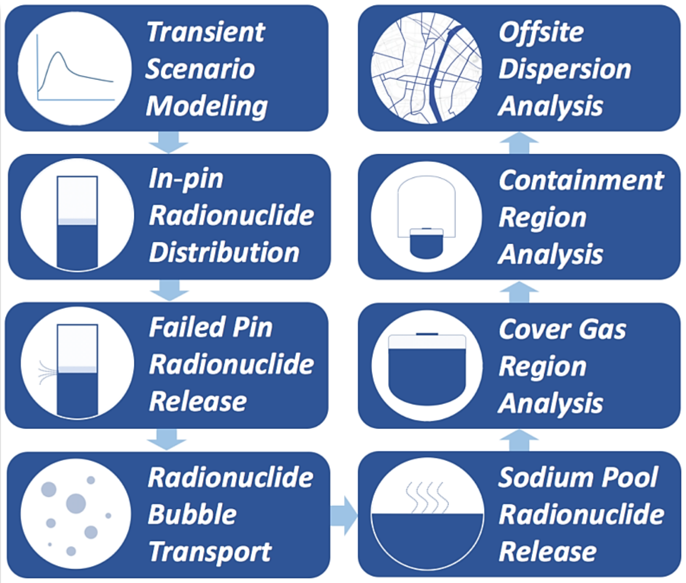
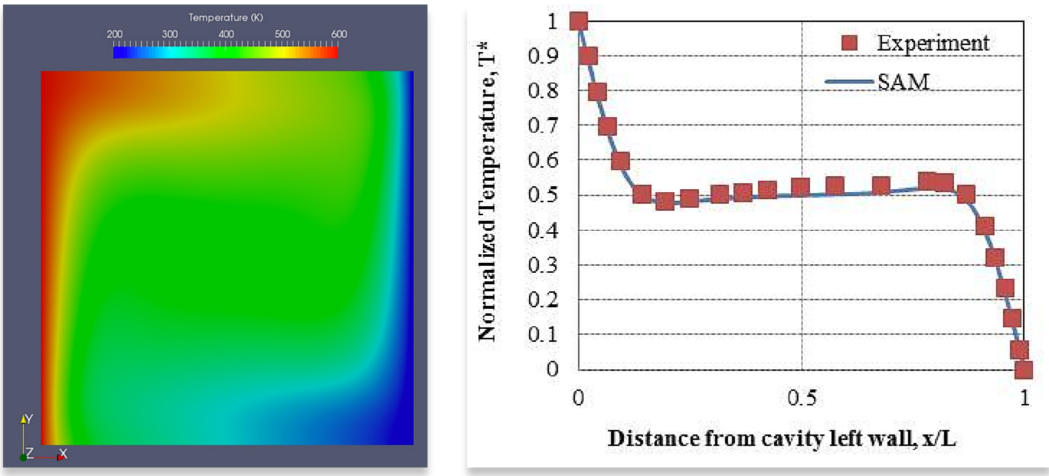


Figure 3.75: Mechanistic Source Term Trial Calculation – Recommendations for Future Research

Group	*Recommendations*
Bubble Transport	• *Formal completion of the IFR bubble code, including development of documentation and code licensing pathway.* • *Experimentation regarding failed fuel pin blowdown and entrainment of released radionuclides in bubble (cross-cutting research with reactivity effects of channel voiding and structural impacts of blowdown on neighboring fuel pins).* • *Experimentation regarding removal of radionuclides from bubble traveling through sodium pool.*
In-Pin Migration and Release	• *Continued metal fuel PIE to determine radionuclide migration within fuel pin during irradiation.* • *Experimentation regarding radionuclide release from failed high burnup fuel pins at high temperatures (above fuel melting point) in liquid sodium.*
Aerosol Behavior	• *Continued development of SFR version of MELCOR.* • *Assessment of available data regarding deposition/condensation/chemical interactions within cover gas region and containment.*
Hold-up/Leakage	• *Response of reactor head seals during transient conditions.*
Vaporization	• *Investigation of non-homogeneous mixing of radionuclides in liquid sodium.*
Dispersion	• *Assessment of applicability of dose conversion factors and deposition assumptions to chemical and physical radionuclide forms likely in SFR radionuclide release.*

Mixing, thermal stratification, and mass transport phenomena in large pools or enclosures play major roles for the safety of reactor systems. Depending on the fidelity requirement and computational resources, various modelling methods, from the 0-D perfect mixing model to 3D computational fluid dynamics (CFD) models, are available. Each is associated with its own advantages and shortcomings. It is very desirable to develop an advanced and efficient thermal mixing and stratification modelling capability embedded in a modern system analysis code to improve the accuracy of reactor safety analyses and to reduce modelling uncertainties. An advanced system analysis tool, SAM, is being developed at Argonne National Laboratory for advanced non-LWR reactor safety analysis. While SAM is being developed as a system-level modelling and simulation tool, a reduced-order three-dimensional module is under development to model the multi-dimensional flow and thermal mixing and stratification in large enclosures of reactor systems. The framework of a 3D finite element flow model has been developed and implemented in SAM. To prevent the potential numerical instability issues, the Streamline Upwind Petrov-Galerkin (SUPG) and Pressure-Stabilizing Petrov-Galerkin (PSPG) formulations have been implemented. Several verification and validation tests were performed, including lid-driven cavity flow, natural convection inside a cavity, and laminar flow in a channel of parallel plates. Based on the comparisons with the analytical solutions and experimental results, it is demonstrated that the developed 3D fluid model can perform very well for a range of laminar flow problems. This 3D flow model is based on solving the primitive variables in the conservative form of the governing equations for incompressible but thermally expandable flows. Combined with the use of the Jacobian-free Newton–Krylov (JFNK) solution method and high-order discretisation schemes, this flow model has great potentials for both efficient and accurate multi-dimensional flow simulations. The results from a SAM simulation of a natural convection test problem in a square cavity were compared with the available experiment results, as shown in Figure 3.76. The normalised temperature distributions at the centre-horizontal line agreed very well with the available experimental data.

Figure 3.76: Temperature distributions and comparison between experiment and SAM predictions of a square cavity test problem



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 Sodium Fast Reactors (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.

Rosatom analysed phenomenon of a local natural convection of sodium coolant occurring in different sections of the pipelines of SFR heat removal loops and its influence on a general coolant natural circulation in a closed circuit. In particular, this research is dedicated to evaluation of a scale of the local natural circulation phenomenon depending on characteristics of the circuit (layout of the circuit, diameter of the pipelines, heat losses level, etc.). It is specially analysed the nature and value of influence of the local natural circulation on the magnitude and stability of the general coolant natural circulation in the closed circuit.

▪ WP SO 2: Experimental programmes and operational experiences

JAEA has investigated the reactor vessel coolability of sodium-cooled fast reactor under severe accident condition with 1/10 scaled water experiment (PHEASANT).

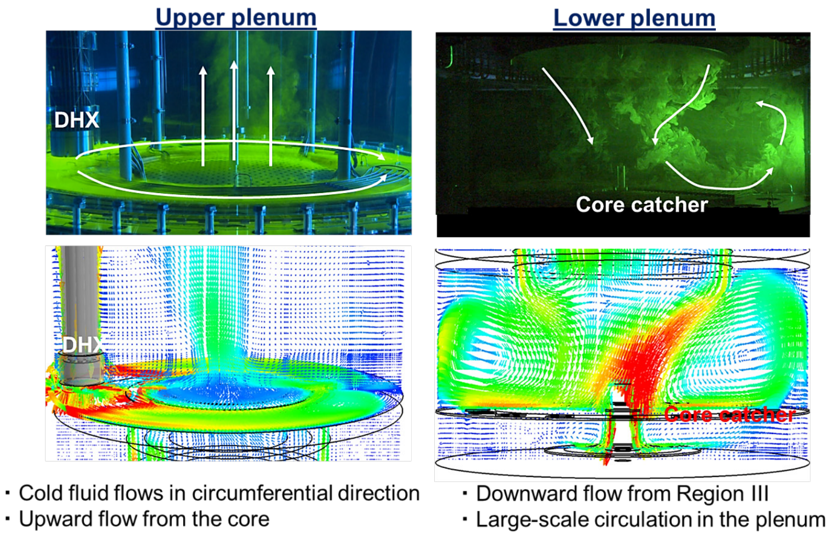
Flow path clarified from flow visualisation of water experiments is as below.

- cold fluid flowed on the bottom plate of upper plenum in the circumferential direction;
- some amount of cold fluid flowed towards the top surface of the core, and the rest penetrated into Region III;

- dyed fluid on the top surface of the core ascended upward direction;
- cold fluid penetrating through Region III reached the debris on the core catcher.

Fluid calculation by FLUENT 14.5 could well reproduce the thermal-hydraulic phenomena in PHEASANT as shown in Figure 3.77.

Figure 3.77: Comparison of velocity field



KAERI has developed Sodium thermal-hydraulic Experiment Loop for Finned-tube sodium-to-Air heat exchanger (SELFA) to evaluate heat transfer performance of the finned-tube sodium-to-air heat exchanger (FHX) and to validate FHX thermal-sizing code (FHXSA) as shown in Figure 3.78. SELFA is a separate effect test facility using liquid sodium with preservation of length scale ratio (1/1) and reduced power scale ratio (1/8).

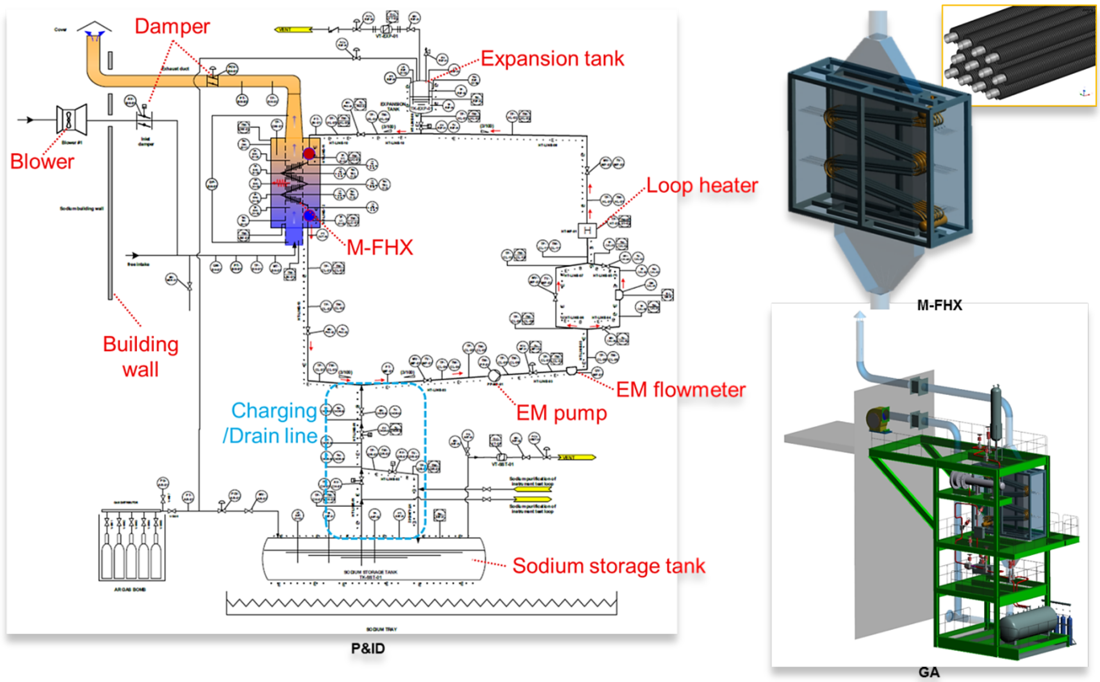
The cold and hot shakedown tests were finished after construction of the SELFA facility. The heat exchanger performance test procedure was determined with several operating tests. Total 41 FHX performance tests were performed until now, and the test results have been analysed including uncertainty information. Performance tests for the finned-tube sodium-to-air heat exchanger are scheduled to be completed by the end of 2017. Test database obtained from the SELFA facility will be used to validate thermal design and safety analysis codes.

WP SO 3: Studies of innovative design and safety systems

CEA surveyed innovative control rod system to manage reactivity. SFR's (ASTRID's) control rod system (CRS) has to fulfil the following functions:

- reach criticality from shutdown state and allow power rise up to nominal conditions (~1 600 pcm);
- compensate reactivity swing due to fuel burnup during one irradiation cycle (~1 600 pcm);
- adjust power level to electrical grid demand (~120 pcm);
- fine tuning of core power map and hot spots (~200 pcm);
- ensure reactor shutdown in normal and incidental conditions;
- keep reactor subcritical during refuelling and maintenance states.

Figure 3.78: SELFA overview



The reference material used for neutron absorption is boron carbide. This choice comes from its high efficiency in fast neutron spectrum and the large French experience on this material. Two axial enrichment zones exist in RBC and RBD:

- natural B_4C in the lower part, inserted in high flux in order to optimise residence time;
- 48% 10B enrichment in the upper part to improve worth when rod is completely inserted.

The design margins of ASTRID and the past European Fast Reactor design (EFR) are compared as shown in Figure 3.79.

Rosatom analysed a new decay heat removal system (DHRS) option, where the external surface of pipelines and equipment of main heat removal loops of the SFR secondary circuit (DHRS-2C) is used to remove decay heat to the outside air. Such a design solution significantly extends capability of DHRS comparing with a similar DHRS through the walls of the reactor vessels to the outside air that is considered in the PRISM reactor design and can be used only in designs of SFR with small power size. The optimisation of the characteristics of the proposed DHRS-2C is performed in relation of its application in the BN-800 secondary loops. In addition, for evaluation of the allowable power range of application of the DHRS-2C, the computational analysis of decay heat removal modes by this system for large power size SFR, namely for the SFR with electric power 1 600 MWe, so-called BN-1600, is carried out. The degree of influence of the specific DHRS-2C characteristics on its efficiency is studied.

Figure 3.79: EFR – ASTRID Comparison (Subcriticality margins)

Subcriticality margins

Subcriticality reserve		Margins (pcm) ASTRID	Margins (pcm) EFR
Fuel handling operation	Maximal incidental reactivity insertion	>1117	>1309
	Handling error: fuel insertion instead of control rod	518	584
	Handling error: 2 extracted control rods	940	1134
Start-up	Criticality approach procedure	1120	-549

Large margins for both architectures

Large margin for ASTRID between critical state and all inserted CR state

Criterion not met for EFR because RBS are withdrawn before criticality approach

Maximal use of the main SFR equipment is one of the advantages of this DHRS concept that allows abandoning the special heat exchange equipment for emergency decay heat removal and, thus, to reduce significantly capital costs for NPP construction. In addition, refusal of special "sodium-air" heat exchangers (AHX) permits to exclude danger of sodium freezing in the AHX heat exchange tubes both in transients related to putting DHRS into operation and in its standby modes. It should be noted that this DHRS performs additionally localising functions in SFR that allows greatly improving its safety against sodium leaks.

Component design and balance-of-plant project

The viability of designing appropriate sodium-cooled fast reactor (SFR) components and balance-of-plant (BOP) has been demonstrated with the design, construction and operation of previous sodium-cooled reactors. The main objective of this research and development (R&D) project is related to system performance, either through the design of advanced components and technologies to enhance the economic competitiveness or safety performance of the plant, or by research and development on the use of advanced energy conversion systems in the BOP that could allow further cost improvements. This R&D project is dedicated to essential efforts to support the design at component level.

Activities within this project will address experimental and analytical evaluation of advanced in-service inspection, instrumentation and repair technologies (ISI&R), leak-before-break assessment, development of advanced energy conversion systems (AECS) with Brayton cycles including the supercritical carbon dioxide (sCO_2) Brayton cycle and the nitrogen gas Brayton cycle, advanced steam generator technologies, detection of steam generator failure, sodium-water reactions, and include other relevant activities related with components and BOP system designs. The project will ultimately include assessment of the feasibility of the technology for desired utilisation. Project activities will be based in part on the extensive historical R&D experience with component design and balance-of-plant for sodium-cooled fast reactors. Details of each study are stated as follows:

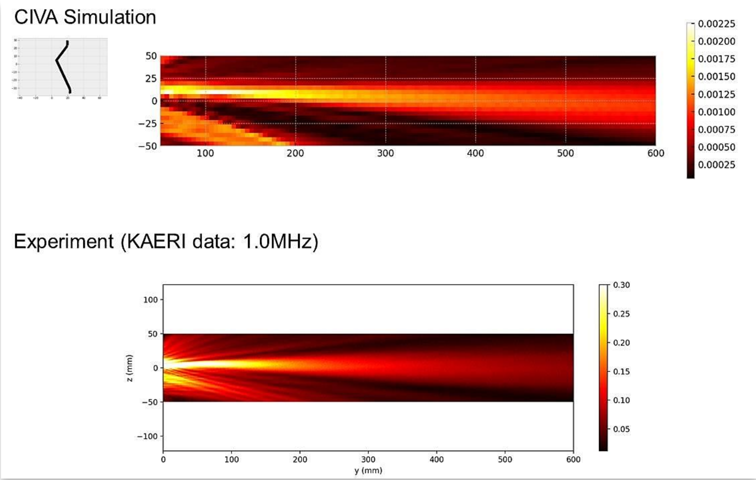
ISI&R technologies

This topic has largely been studied during 2017 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 sensor development and data treatment (image 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 must be able to provide under sodium movement of these sensors. Preliminary under sodium viewing were achieved in 2017.

Another approach to use ultrasonic waves under sodium surface is to use waveguide sensors. In 2017, CEA participated in the evaluation of the waveguide sensor technology by using a modelling tool (CIVA) in the framework of the collaborative work with KAERI. Comparison between experiments done by KAERI and simulation was conducted to confirm the accuracy of the one-way coupling approach. Evaluation of different design options in view of sensor performance improvement was realised.

Figure 3.80: Comparison between KAERI experiments and CEA CIVA simulation



AECS technologies

Concerning sodium gas heat exchangers (a key component in case of use of a gas as tertiary fluid/working fluid), CEA started to study the option of a compact design. Some generic studies such as optimisation of channel design and development of modelling tools were already started. The analysis of the performance of a first sodium gas heat exchanger mock-up was presented based on tests realised on DIADEMO sodium tests rig. Another study on the optimisation of the design of headers was initiated.

The testing in DIADEMO tests rig was designed as the first sodium gas heat exchanger with 530°C sodium inlet and 345°C sodium outlet temperature condition and 310°C nitrogen inlet and 515°C outlet temperature condition.

The qualification programme of sodium gas heat exchanger began at different scales as shown in Figure 3.81.

At the channel scale, models and calculations were validated.

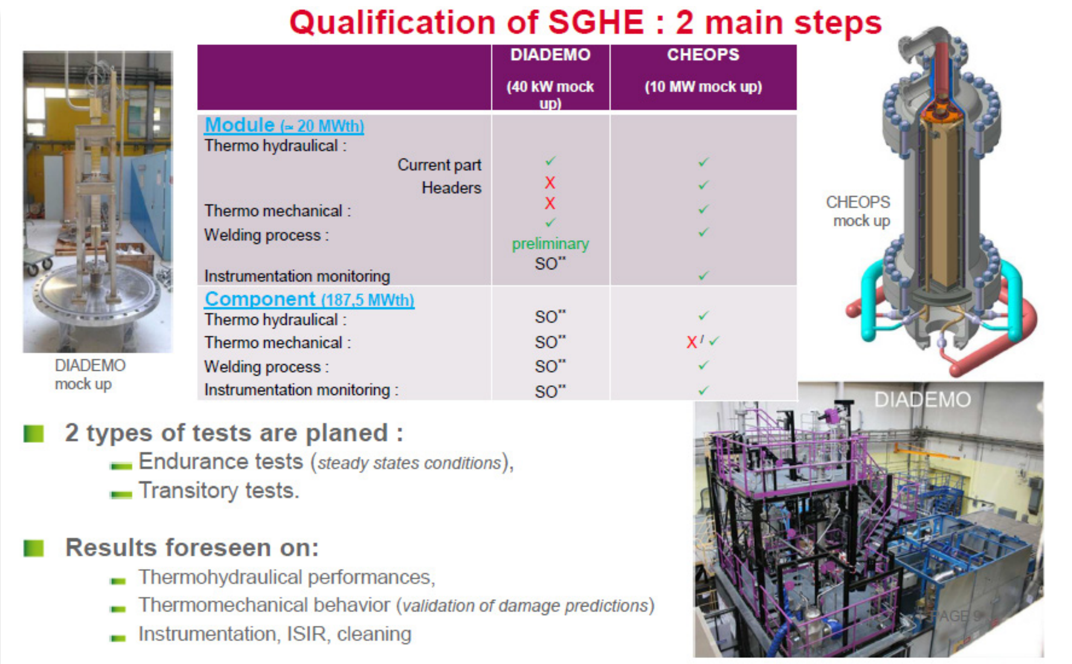
- several channel designs have been evaluated;
- thermal-hydraulic simulation tools are qualified at the channel scale.

At the component scale, low power functional mock-ups tested in the DIADEMO facility provided first results that fully confirmed the predicted thermal-hydraulic performance:

- improvement of the manufacturing process was achieved during fabrication of the mock-ups;
- experimental results confirm the choices made for the design;
- simulation tools well predict the performance: temperature map, pressure drop.

Following improvement of the heat exchanger manufacturing process with development of fabrication examination and improvement of the headers to avoid maldistribution, CEA will start a 10 MW mock-up test programme in the CHEOPS sodium facility.

Figure 3.81: Qualification programme of DIADEMO and CHEOPS



DOE/ANL analyses for the supercritical carbon dioxide Brayton cycle focused upon modelling of SFR nuclear power plants utilising dry air cooling by which heat is rejected directly to the atmospheric heat sink as well as optimisation of the control and operation of such power plants as the ambient air conditions change to maximise the benefits of dry air cooling.

Continuing application of the Plant Dynamics Code a small-scale sCO_2 Brayton cycle demonstration in the Integrated System Test (IST) facility provided for testing and validation of the code. The PDC simulation was carried out in three stages:

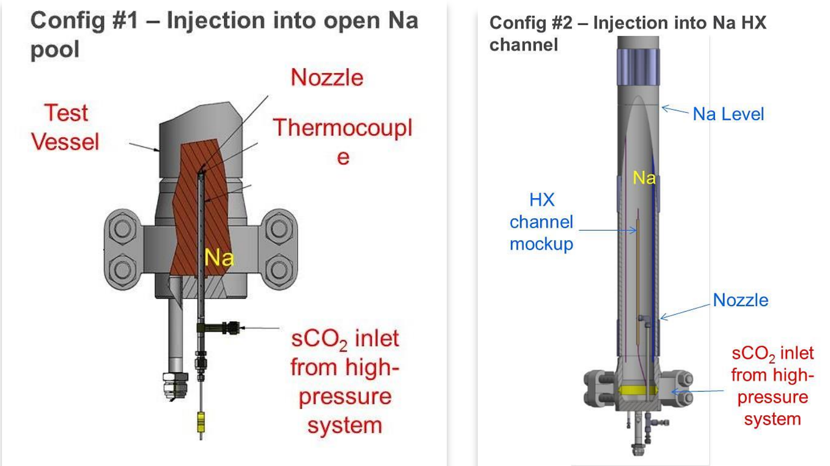
- Steady state: Good agreement, except for the power turbine performance.
- Transient with a given turbo-compressor shaft speed and control valve position: Same as in previous work. New information helped to identify an error in the assumed shell-and-tube heat exchanger tube mass.
- Full transient with active control: This case validates the control setup and shaft speed equations. Results for two tests show good agreement with the data. The turbo-compressor shaft power balance is affected by the approximation of windage losses.

Sodium-CO_2 interaction tests were also continued at ANL in the SNAKE (S-CO_2 Na Kinetics Experiment) facility. Experiments simulating CO_2 injection into a single semicircular sodium channel of a compact diffusion-bonded sodium-to-CO_2 heat exchanger were conducted in the test section shown in Figure 3.82.

- Sodium leakage and consequences

In this field, crack growth assessment procedures that will be employed in the JSME's LBB standard were developed. As shown in Figure 3.83, the penetrated crack length obtained by the crack growth assessment is used in the LBB assessment.

Figure 3.82: Two configurations of sodium-CO_2 interaction experiments



Based on the sensitivity analyses:

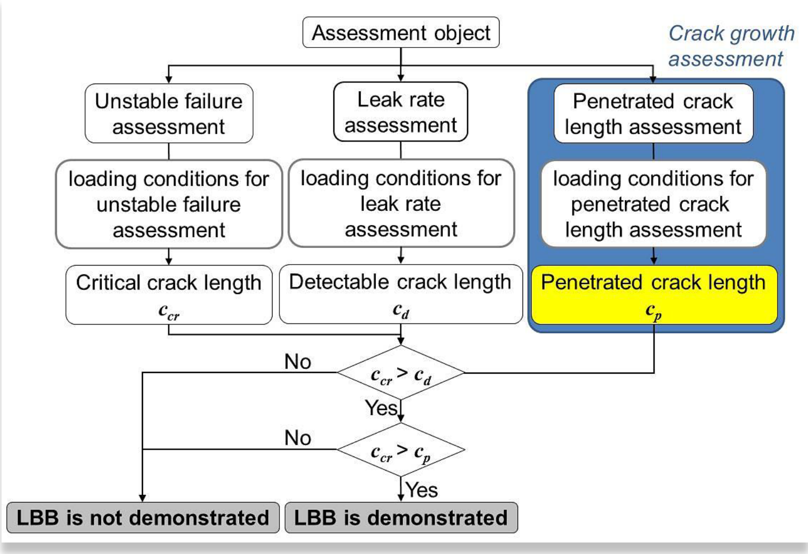
- The penetrated crack length in a pipe is larger than that in a plate.
- A longer penetrated crack length is predicted in the thicker pipe.
- The penetrated crack length strongly depends upon the ratio of the membrane stress to total stress. Pure bending stress results in the largest penetrated crack length.
- There was little influence of the initial crack size on the penetrated crack length, where the bending stress is predominant.
- A longer crack length was predicted without creep crack growth.
- In addition, the penetrated crack length depends upon the exponent, "m," in Paris' crack growth rule. A larger "m" results in a larger crack length.

Master curves as a function of the following parameters are proposed by polynomial approximation of the fatigue crack growth analysis results, both for axial and circumferential cracks:

- ratio of membrane/total stress;
- ratio of pipe radius/thickness, R/t, for circumferential cracks;
- exponent, m, of Paris' crack growth rule;
- contribution of creep crack growth is not taken into account for conservatism.

By using the master curves, the penetrated crack length can be estimated without fracture mechanics knowledge.

Figure 3.83: LBB assessment flow chart for SFR pipes



▪ Steam generators

JAEA continued to study tube failure propagation in a SFR steam generator. JAEA carried out development of a more comprehensive understanding of the sodium-water chemical reaction process. In this study, the primary and secondary surface reaction (the reaction between liquid sodium and water vapour) mechanisms in sodium-water reactions was investigated by an ab initio molecular modelling approach and by the differential thermal analysis (DTA) experiment technique.

Based on this investigation, the below items became clear.

- Reaction paths on NaOH generation and Na_2O generation in SWR were identified.
- The overall reactions considered are:
- Primary reaction: $Na+H_2O \rightarrow NaOH+1/2H_2$.
- Secondary reaction: $NaOH+Na \rightarrow Na_2O+1/2H_2$
- The primary reaction is much more rapid than secondary reaction.
- Melting points and transition temperatures of the reactants (Na&NaOH) were identified in the pure material test and reaction test.
- The decomposition temperature of NaH was identified in the reaction test.
- Na_2O as secondary reaction product was identified from XRD analysis.
- Na_2O generation by the secondary reaction, although slower than NaOH generation, is large enough that it should be considered in sodium-water reaction modelling.

KAERI improved the performance of their steam generator tube inspection system. The detectability of the combined inspection sensor was improved by modification of the sensor structure and the signal analysis software was also be enhanced by newly loading a phase analysis function. Furthermore, a new sodium test facility was designed and constructed to simulate the actual conditions of steam generator tubes with sodium deposits. Further investigation of the influence of sodium deposits on the measured signals of the combined inspection sensor was carried out.

Recent progress on development of the SG tube inspection technique is provided below.

Performance improvement of the combined inspection sensor system:

- Modification of sensor structure:

The material for the sensor protection casing was changed to improve the filling factor and signal sensitivity.

- Design and fabrication of the phase analysis circuit:

The phase analysis circuit was designed and fabricated to obtain phase information of the RFECT signals

- Loading the phase analysis function on the signal analysis software:

Phase analysis and display functions were newly added in signal analysis software is being upgraded to load the phase analysis function.

Further investigation of the influence of sodium deposits:

- Design and construction of a sodium test facility: A sodium test facility which can be operated over 500˚C to simulate sodium deposits was designed and constructed.
- Construction of test specimen: Test specimens having several kinds of defects were constructed for investigation of the influence of sodium deposits.

Sodium operation technology and new sodium testing facilities

For the MECANA sodium facility, mechanical manufacturing and main component construction started and were fully achieved. In parallel, the main specific sodium instrumentation has been manufactured and is now available. Instrumentation, control system and electric supply studies and realisation are now ongoing.

For the STELLA-2 integral sodium facility, design of the main system and auxiliary systems has been completed, and fabrication of each component or subsystem is in progress. Fabrication of key components and subsystems should be finished in 2017 and on-site assembly and installation is planned in 2018.

3.6. Very-high-temperature reactor (VHTR)

Main characteristics of the system

High- or very-high-temperature reactors were developed and operated between the 1960s-1990s, two are currently operational (HTR-10 is running, HTTR awaits regulator approval to restart) and two reactor units are under construction (HTR-PM). They are characterised by fully ceramic coated particle fuel, the use of graphite as neutron moderator and helium as coolant. All modern designs feature passive decay heat removal capability resulting in inherent safety. They are generally designed as modular SMRs and particularly suitable for highly efficient cogeneration of heat and power. Several such reactors have operated routinely in the reactor outlet temperature range 700-850°C, while operational experience was gained already in two reactors for longer periods of time up to 950°C which is considered a limit for current structural alloys. Beyond this temperature, new structural materials would be required.

The initial driver for the VHTR in GIF was the desire of several signatories to develop a reactor capable of powering a CO_2-free bulk hydrogen production facility, possibly using the thermochemical sulphur-iodine cycle. This process consumes heat at a temperature of 850°C thus, taking due account of heat transfer cascades, it would require a reactor outlet temperature of approx. 1 000°C. This remains the long-term target of the GIF VHTR system. At the same time, efforts are continuing to reduce the temperature requirements of H_2 production (by using catalysts or different processes) and to apply innovative heat transfer technology such that the reactor itself can operate at the lowest possible temperature.

However, more recent market research in several of the signatory countries has shown that process heat, mostly in the form of steam <600°C (plug-in market, see Figure 3.84), represents indeed a very significant existing market in all industrialised countries, globally several hundred GWth which today is almost entirely fossil-fuelled. For such an application, a rather conventional reactor outlet temperature of approx. 750°C would already be sufficient.

Figure 3.84: Market for HTGR process heat

Region	Plug-in market	Total market	GDP 2011 (approx.)
Europe	~ 800 TWh/y *(EUROPAIRS)*	~ 3,000 TWh/y *(EUROPAIRS)*	17,000 bn€ / 25% of world
USA	~ 1,100 TWh/y *(MPR Associates)*	~ 3,600 TWh/y *(MPR Associates)*	15,000 bn€ / 22% of world
Japan		*1,000 – 1,400 TWh/y (est.)*	5,900 bn€ / 8% of world
China		*1,200 – 1,700 TWh/y (est.)*	7,000 bn€ / 10% of world
India		*300 – 500 TWh/y (est.)*	2,000 bn€ / 3% of world
Russia		*300 – 500 TWh/y (est.)*	2,000 bn€ / 3% of world
World total	***3,000 – 5,000 TWh/y ~ 370 – 630 GWth***	***11,000 – 16,000 TWh/y***	**69,000 bn€**

Source: Bredimas

The technology basis for the VHTR had 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. Initially, low-enriched uranium fuel at very high burnup will be used in a once-through mode, while plutonium- or thorium-based fuels are longer-term options. Several solutions are being investigated to adequately manage the back-end of the fuel cycle and the potential for a closed fuel cycle. Although various fuel designs are considered in the VHTR systems, all exhibit similarities allowing for a coherent R&D approach with the TRISO-coated particle fuel form as the common denominator. This fuel form is composed of small kernels of fissile ceramic material (typically UO_2 or UCO), surrounded by a porous carbon buffer, and coated with three layers: pyrocarbon/silicon carbide/pyrocarbon. As demonstrated in many experimental and operational performance tests, this coating represents a very efficient barrier against fission product release under normal and accident conditions.

In the past, AVR and HTTR already demonstrated operation up to 950°C for longer periods of time. A VHTR could currently be designed to deliver heat and electricity over a range of core outlet temperatures between 700 and 950°C, and possibly up to 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). For higher temperatures, the development of innovative materials such as new super alloys, ceramics and compounds will be necessary.

In current or near-term projects of VHTR design and construction, the reactor delivers heat to steam generators which feed an indirect Rankine cycle for power conversion using the latest available technology from conventional power plants. However, direct helium gas turbine or indirect (gas mixture turbine) Brayton-type cycles can also be considered in the near future. The experimental reactors HTTR (Japan, 30 MWth, awaiting regulator approval for restart) and HTR-10 (China, 10 MWth, operating) support the advanced reactor concept development for the VHTR. They provide important information for the demonstration and analysis of safety and operational features. This allows improving the analytical tools for 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 start-ups, plant vendors and national laboratories respectively in China, the United States, Korea and Japan. The construction of the HTR-PM demonstration plant in China (two pebble-bed reactor modules with 250 MWth each delivering steam to a single superheated steam turbine generating 200 MWe) started on 9 December 2012. The reactor outlet temperature will be 750°C, which is well within the limits of the current state-of-the-art for materials and components, yet suitable for the generation of high-quality steam of 566°C. The HTR-PM demonstration plant is planned to be synchronised to the grid by the end of 2018, representing a major step towards the deployment of Generation IV technology.

Status of co-operation

The **Fuel and Fuel Cycle** (FFC) Project Arrangement (PA) became effective on 30 January 2008, with signatories from Euratom, France, Japan, Korea and the United States. The PA was extended to include China as a signatory and was amended in 2013. It went into effect in January 2014. The PA was extended by another ten years in early 2018.

The **Materials** (Mat) PA, which addresses graphite, metals, ceramics and composites, was signed by signatories from Canada, France, Japan, Korea, South Africa, Switzerland, the United States and Euratom in 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 amendment to include China's INET as a new signatory of the PA was finally signed by all parties in early 2018.

The **hydrogen production** (HP) PA became effective on 19 March 2008 with signatories from Canada, France, Japan, Korea, the United States and Euratom. The PA was extended for another ten years in early 2018. A new Project Plan to include China as an additional signatory is under preparation.

The **Computational Methods, Validation and Benchmarks** (CMVB) PA made significant progress in its preparation but remained provisional throughout 2017. In discussions during the CMVB PMB meetings in 2017, the PA, Project Plan (PP) and work plan for the first year were finalised. The CMVB PA is expected to be signed in 2018.

Two other projects on components and high-performance turbo machinery and on system integration and assessment were discussed by the VHTR SSC, but the associated research plans and Project Arrangements have not yet been developed.

R&D objectives

While VHTR development is driven by high-efficiency cogeneration of heat and power, originally with focus on bulk hydrogen production, current R&D targets are demonstration of inherent safety features, high fuel performance and coupling with process heat applications. The VHTR SRP describes the R&D programme to establish the basic technology of the VHTR system. As such, it covers the needs of the viability and performance phases of the development plan described in the Generation IV Technology Roadmap. The VHTR system currently runs three projects and one is in a very advanced preparation phase as discussed hereunder:

- The **Fuel and Fuel Cycle** (FFC) project is executing collaborative research focusing on the performance of TRISO-coated particles, the basic fuel form for the VHTR. The R&D underway is successfully increasing the understanding of the standard design (UO_2 kernels with buffer and PyC/SiC/PyC coatings) and examining the potential of uranium oxycarbide (UCO) kernels. It also investigates the possible use of ZrC coatings instead of SiC for enhanced burnup capability, reduced fission product permeation and even further increased resistance to core heat-up accidents (above 1 600°C). The R&D in this project 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 deep burn of plutonium and minor actinides (MA) in view of a closed fuel cycle.
- The **Materials** (Mat) project is strongly contributing to the development and qualification of structural and functional materials, design codes and standards, as well as manufacturing techniques which are all essential for VHTR system development, demonstration and deployment. 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 possibly 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, a significant amount of work is being invested in metal performance analysis for 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 database has been developed and is being further populated and used. It efficiently stores and manages test data, facilitates international R&D co-ordination and supports modelling to predict damage and lifetime assessment.
- The **hydrogen production** (HP) project is investigating technologies that make use of nuclear energy (heat and/or electricity) to power large-scale bulk hydrogen production plants. After screening many such processes in the 1980s, two were identified by the HP project as the most promising for VHTR applications, namely the sulphur-iodine thermochemical cycle and high-temperature steam electrolysis. As alternatives, two additional cycles could be identified: the hybrid copper-chloride process and the hybrid sulphur cycle. Such cycle processes typically consist of three elemental processes which are then combined to a cycle where the overall input consists only of water and energy (heat and/or electricity) and the output of hydrogen and oxygen. Ongoing R&D efforts in this project cover feasibility, optimisation, efficiency of the elemental processes, integration to a stable process cycle, economics analysis from laboratory to demonstration scale, and also encompasses component development such as advanced process heat exchangers. Process engineering options are also being investigated that aim at lowering temperature requirements so as to make hydrogen production

compatible with other Generation IV nuclear reactor systems. Coupling technologies with the nuclear reactor are also being investigated accompanied by safety analysis to minimise potential hazards between nuclear and chemical systems.

- The **Computational Methods, Validation and Benchmarks** (CMVB) project is in an advanced preparation phase and ready for signature in 2018. Work by the signatories had already begun while the project was still provisional. It is focusing on licensing-relevant subjects for the assessment of 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 will 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.

The VHTR SRP had identified in principle the need for two additional projects, one on **component development and qualification** and another one on **system integration and assessment**. However, the signatories did so far not formulate full projects in these areas.

A project on VHTR Components will need to address the development of components for reactor subsystems (core structures, absorber rods, core barrel, pressure vessel, etc.) and for power conversion or coupling processes (such as steam generators, heat exchangers, hot gas duct, valves, instrumentation and turbo machinery). Some components, especially when approaching 1 000°C, 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 and qualified in dedicated large-scale helium test loops, capable of simulating normal and off-normal events. A project on components could address development needs that are in part common to those of the GFR.

System Integration and Assessment is necessary to guide the R&D on different VHTR baseline concepts and new applications such as cogeneration or hydrogen production. Near- and medium-term projects should provide information on their designs to identify potential for further technology and economic improvements. At the moment, this topic is directly being addressed by the System Steering Committee.

Main activities of the System Steering Committee

In 2017, the GIF VHTR System Steering Committee (SSC) has provided guidance and monitored progress in the ongoing projects, and could help fix a number of difficulties, mainly related to the availability of signatory representatives and communication. The VHTR SSC has discussed and approved the participation of Australia in the VHTR Materials project and their initiative to create a Task Force on Cross-cutting Advanced Manufacturing technologies. The VHTR has welcomed Canada back in the SSC as observer reflecting growing interest for VHTR technology in this country.

The VHTR SSC has also maintained lively interactions with the Methodology Working Groups, in particular with the Risk and Safety Working Group (RSWG) for which a white paper and a safety self-assessment were prepared. A VHTR SSC representative has attended a meeting of the Proliferation Resistance and Physical Protection (PRPP) working group where future interaction could benefit from more design details. Based on interaction specifically with the Economic Modelling Working Group (EMWG), it was found that the interaction should be reinforced, in particular in view of reducing cost uncertainties and to identify avenues for cost reduction R&D (e.g. graphite, components).

In January 2017 and in collaboration with the GIF Education and Training Task Force, a VHTR SSC representative has given a webinar on the VHTR system. The VHTR SSC is also participating in a related LinkedIn group.

In September 2017, the VHTR SSC has facilitated the delivery of three questionnaires to the GIF Senior Industry Advisory Panel (SIAP) with detailed information on specific VHTR designs and has participated in one of their meetings.

Twice annually, the VHTR SSC is reporting progress in the system to the GIF Experts Group and Policy Group meetings, where it has delivered in particular a presentation on market needs for cogeneration. It has provided activity updates to the IAEA and a presentation on its safety and licensing-relevant activities to the NEA Group on the Safety of Advanced Reactors (with regulators and TSOs).

Finally, several of the VHTR SSC members are actively involved in the preparation of the HTR 2018 conference which will be held in October 2018 in Warsaw, Poland.

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. The 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 3.85.

Status of ongoing FFC activities

2017 was the last year of the current Project Plan 2012-2017. Significant achievements were made in the areas of irradiation and PIE, characterisation, safety testing and fuel cycle back-end issues. Also, special workshops on SiC materials, QA methods (leach-burn-leach), benchmarking of models predicting TRISO fuel performance under accident conditions were organised to further refine the understanding of TRISO performance. Apart from its project monitoring and co-ordination tasks, the FFC PMB prepared a new ambitious five-year Project Plan for signature in early 2018.

Irradiation and PIE

In the United States, PIE of the AGR-2 and AGR-3/4 experiments is still in progress both at Idaho National Laboratory (INL) and Oak Ridge National Laboratory (ORNL). Ongoing AGR-2 PIE consists of destructive compact examinations, including 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. UCO particle morphologies and microstructures generally have appeared similar to what was observed with the earlier AGR-1 irradiation experiment. Optical microscopy of a number of particles from different compacts indicated that the higher irradiation temperatures achieved in AGR-2 Capsule 2 resulted in less buffer fracture, presumably due to thermal creep allowing more stress relaxation than at lower temperatures. Figure 3.86 shows representative morphologies of particles from Compact 2-2-3 (time-average, volume-average irradiation temperature of 1 261°C) and Compact 5-2-3 (1 108°C). Note the increased occurrence of buffer fracture in Compact 5-2-3. In addition, detailed microanalysis of irradiated particles at relatively low length-scales using e-beam instruments is being performed to examine the migration of fission products in the coating layers.

Figure 3.85: Task structure

WP1. Irradiations and PIE
Task 1.1: Irradiation Design and Operation
Task 1.2: Hosted Joint Irradiations
Task 1.3: PIE Protocol and Procedures
Task 1.4: Irradiation and PIE results
WP2. Fuel Attributes and Material Properties
Task 2.1: Measurements of Critical Material Properties
Task 2.2: Fuel Material Property Database
Task 2.3: Characterisation Techniques
Task 2.4: Fuel Performance Modelling
WP3. Safety
Task 3.1: Pulse Irradiation Testing
Task 3.2: Heating test Capabilities
Task 3.3: Heating Tests
Task 3.4: Source Term Experiments
WP4. Enhanced and Advanced Fuel
Task 4.1: Process Development
WP5 Waste Management
Task 5.1: Head-end Process
Task 5.2: Graphite Management
Task 5.3: Disposal Behaviour and Waste Package
WP6. Other Fuel Cycle Options
Task 6.1: Transmutation
Task 6.2: Thorium Cycle

Figure 3.87 shows an example of analysis performed to identify fission product precipitates within the SiC layer microstructure and correlate these with the nature of the grain boundaries.

The AGR-3/4 PIE currently in progress includes analysis of fission products on the capsule components to help quantify total fission product release from the fuel, destructive examination of fuel compacts to examine the state of the particles and the distribution of fission products within the fuel, and heating tests to evaluate fission product transport at elevated temperatures. Fuel compact destructive examination begins with deconsolidation-leach-burn-leach analysis. This is more complex for the AGR-3/4 compacts than for standard cylindrical fuel compacts (such as those from the AGR-1 and AGR-2 experiments), because of the presence of "designed-to-fail" (DTF) particles in the AGR-3/4 compacts. Compacts must be deconsolidated such that the DTF particles are avoided, as they would be dissolved and overwhelm the solution activity, making measurement of fission product inventory in the compact matrix impossible. To achieve this, a method of radially deconsolidating the compacts has been developed and deployed in the hot cell at INL. This approach removes sequential, thin regions around the compact circumference, allowing intact TRISO particles to be collected and fission product inventory in the matrix to be measured while avoiding the DTF particles along the compact centre line. Three compacts have been deconsolidated to date.

Figure 3.86: Particle ensembles (left) and representative particles (right) from two AGR-2 compacts irradiated at time-average, volume-average temperatures of 1 261°C (Compact 2-2-3, top) and 1 108°C (Compact 5-2-3, bottom).

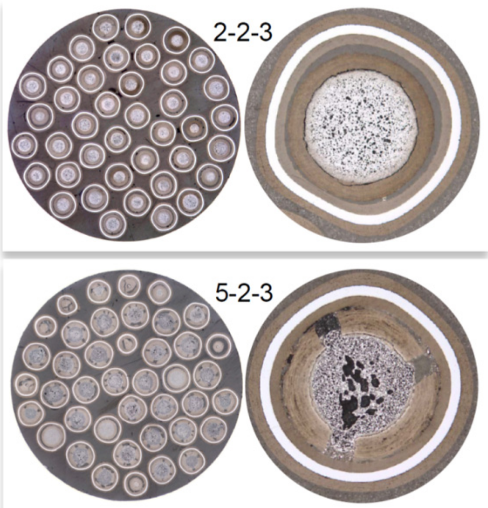
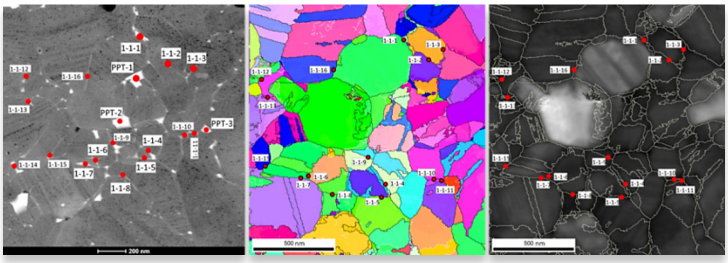


Figure 3.87: TEM micrograph (left) of the SiC layer of an irradiated AGR-2 particle, showing fission product inclusions within the microstructure. Images in the middle and right show analysis of grain boundary orientation



AGR-3/4 heating tests will include fuel compacts, fuel bodies (i.e. intact capsule internals consisting of fuel compacts surrounded by matrix and graphite rings), and individual matrix/graphite rings. The objective in all cases is to better understand fission product transport in the fuel and in matrix and graphite at elevated temperatures. Two AGR-3/4 compact heating tests in pure helium have been completed (one at an isothermal temperature of 1 400°C, the other involving isothermal holds at 1 600 and 1 700°C). Additional tests are planned in the next several years.

The final fuel qualification irradiation for the AGR programme is AGR-5/6/7. This experiment will consist of 194 fuel compacts and a total of approximately 575 000 particles. The AGR-5/6 portion of the test will irradiate the fuel over a broad range of burnup (approximately 6-18% FIMA), fast neutron fluence (1.5 to 7.5×10^{25} n/m^2, E > 0.18 MeV), and temperature (approximately 600-1 350°C), to approximate the range of values that would be experienced by the fuel in an HTGR core. The AGR-7 portion of the experiment constitutes a fuel performance margin test, which will involve temperatures far in excess of those expected in a gas-cooled reactor during normal operation. Time-average peak fuel temperature in this capsule will reach approximately 1 500°C, with burnups of approximately 18% FIMA. The AGR-5/6/7 fuel compacts were fabricated in 2017, and fabrication of the irradiation test train (five individual capsules in total) is completed. The experiment is awaiting insertion into the Advanced Test Reactor at INL in early 2018.

In the EU, the PIE of the HFR-EU1 irradiation test performed in the High Flux Reactor at Petten containing Chinese and German fuel irradiated at typical pebble bed conditions is also completed.

For China, two Chinese HTR-10 pebbles irradiated in HFR-EU1 were transported from JRC Petten to JRC Karlsruhe in 2016, and one of them was tested there at the simulated accident temperature, after the PIE of the irradiated pebbles. One high-temperature test was completed of an HTR-10 pebble, and two deconsolidations and coated particle examinations will be performed in 2018.

In Korea, the irradiation test in the High-flux Advanced Neutron Application Reactor (HANARO) had been completed in December 2013 at a maximum burnup of 37 344 MWd/MtU over five cycles. Different fuel forms were irradiated: kernel, coated particles, fuel compacts and graphite. Five irradiation cycles in HANARO were completed in March 2014 and the data analysis of the irradiation conditions is now completed. Non-destructive experiments (NDE) on irradiated rods (measurement of the rod diameters, gamma-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 discs) were performed. Destructive experiments were carried out on TRISO fuel particles (optical inspection, EPMA). Post-irradiation examinations on IG-110 and A3-3 graphite were performed in 2017 (thermal conductivity, hardness and Young's modulus).

Fuel attributes and material properties

In the EU, the pyrocarbon irradiation for creep and swelling/shrinkage of objects PYCASSO-I and PYCASSO-II were irradiations of surrogate particles from France, Japan and Korea. X-ray tomography and nano-indentation of PYCASSO-I samples from France are complete and will be delivered shortly. Plans have been established to analyse Korean surrogate particles from the same irradiation tests and are awaiting a funding decision.

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

In Japan, two main R&D projects are currently ongoing: (1) on Security-Enhanced Safety Fuel for Clean Burn HTGR to establish Pu-burn fuel technologies and, (2) on Sleeveless Oxidation-Resistant Fuel to develop advanced HTGR fuel with improved heat removal compared to the HTTR in order to decrease the maximum fuel temperature during normal operation.

In Korea, research was conducted on fuel fabrication technology. A packing fraction of 30% of TRISO fuel particles in matrix graphite was the target. An automatic over-coater was developed to control the thickness of the coating layers on TRISO particle fuels in view of maximising the packing fraction.

Based on work performed in 2016, the leach-burn-leach round robin test was continued to benchmark the leach-burn-leach (LBL) process as a Quality Assurance method for fuel. ORNL in the United States completed fabrication and characterisation of particles with simulated pre-burn or post-burn LBL defects that will be used in the round robin activity. These defects were seeded according to the experimental plan into seven round robin samples for each participant, along with aliquots of a certified impurity standard. ORNL characterisation results of the United States set of seven specimens compared favourably to the expected defect populations and impurity content. Samples have been shipped to KAERI and shipment to INET is pending. China completed fabrication of on a second set of round robin samples containing representative TRISO coatings on depleted-uranium UO_2 kernels, which will also be shipped to each participant when authorisations are acquired in early 2018.

Korea has made progress of the development of the test apparatus for LBL analysis. Measurement of the seven round robin samples containing simulated defects received from ORNL in January are planned over the next six to nine months. China has completed the characterisation of the China set of round robin samples containing representative TRISO coatings on depleted-uranium UO_2 kernels. The test samples from ORNL and INET, were delivered to both sites (INET and ORNL) at the end of 2017.

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

The advancement of the TRISO fuel performance models under accident conditions was discussed at the PMB meeting in June 2017. Fuel post-irradiation accident performance was modelled by the different participants (INL, JAEA and KAERI) and the final calculations from all three participants were completed. The results of these calculations were compared to each other and to available experimental data. It was noticed that the different codes compare well to each other but all over-predict the experimental data. After presentation of the results and discussion with the PMB members, the path forward was established as follows: INL will draft a final report that includes a description of the benchmark and of the codes, methodologies, and results from all participating members, as well as comparison to experimental data (one data set still awaited to complete the task). The final report is expected to be issued in early 2018, assuming no delay in the missing experimental data. In Japan, The code FORNAX-A was developed at JAEA to study fission product release behaviour under accident conditions. In addition to the modelling results of heating test of HFR-EU1bis (EU) pebbles delivered in 2016, the calculation results on heating tests of AGR-2 (United States) were delivered in 2017.

The 4th International Workshop on SiC Material Properties was held in June 2017 in conjunction with the FFC PMB meeting in Baotou, China. The workshop covered a multitude of topics including SiC performance results from post-irradiation examination, fundamental studies of SiC interaction with fission products, and TRISO behaviour during accident conditions. The primary topic of discussion was silver release from intact TRISO particles and the variables influencing this release.

Safety testing

In Korea and 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 has been constructed for a simulated heating test in a laboratory fundamental data are expected to be provided for the construction of actual heat-up test equipment for use in a hot cell. Specimens of Cs, Ag, Pd in a graphite container were tested at a maximum temperature of 2 000°C under Ar atmosphere.

AGR-2 fuel compact safety tests are in progress, with eight UCO and three UO_2 tests completed at various temperatures. In general, the results continue to demonstrate excellent performance of the fuel types. The UCO fuel, in particular, exhibits very low incidence of coating failure at temperatures as high as 1 800°C, and is similar in performance and behaviour to the previously tested AGR-1 fuel. The AGR-2 UO_2 fuel demonstrated excellent in-pile behaviour to a burnup as high as 10.7% FIMA, but also exhibits notable degradation of the SiC layer due to CO attack at elevated temperatures (1 600-1 700°C) that is characteristic of the UO_2 fuel type.

The Neutron Radiography (NRAD) reactor in the INL Hot Fuel Examination Facility is being used to re-irradiate AGR-2 fuel to generate short-lived fission products (^{131}I, ^{133}Xe) to evaluate release behaviour in heating tests. Two tests have been performed to date involving the re-irradiation of loose kernels, followed by heating in the FACS furnace in pure helium to observe iodine and xenon release. In addition, a method has been developed to mechanically crack loose, irradiated particles, such that these cracked particles can be re-irradiated in NRAD and heated in FACS to examine iodine and xenon release. Finally, the capability for re-irradiating whole AGR fuel compacts in the NRAD reactor is being developed, with similar tests on whole compacts expected to start next year. This will include the AGR-3/4 fuel compacts, which each contain 20 designed-to-fail particles with exposed kernels, providing a source of fission products that will be released during the test and measured.

The United States is also developing a furnace system that will be used to perform high-temperature tests of irradiated fuel specimens in oxidising conditions. The system will allow irradiated specimens to be heated to temperatures as high as 1 600°C in gas mixtures containing air or moisture, while measuring fission products released from the fuel. The system will be installed in a hot cell at the Materials and Fuels Complex at INL.

In Japan, oxidation tests with SiC-TRISO are being carried out. The oxidation testing furnace was built in 2015. Oxidation tests are progressing using dummy SiC-TRISO particles with/without OPyC layer at ~1 600°C with 20 ppm – 20% of O_2 atmosphere. The work was transferred to the next five years plan and will be delivered by December 2018.

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 the United States.

China is interested in developing UCO kernels with ZrC-TRISO coatings. The first stage on UCO kernel production was completed. Two different carbon blacks were used to study the influence on the performance of UCO microspheres. The first stage on Zr coating fabrication is also completed.

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. C-ADU Gel is prepared by external gelation method and then treated thermally. Thermal treatment of C-ADU gel equipment has been built. C-ADU gel particle manufacture conditions were described. SEM imaging showed that the globularity is not uniform. In addition, surface cracking was observed. This is due to a high viscosity.

Waste management and other fuel cycle options

The area covers three subjects, but at this stage, collaboration is still limited with results mainly from EU projects:

- spent VHTR fuel management (direct disposal vs. waste minimisation vs. reprocessing);
- irradiated graphite management (decontamination and possible recycling);
- transmutation using a VHTR (deep burn, Pu and minor actinide incineration).

The results from the European projects ARCHER, PUMA, CARBOWASTE and CAST are of interest to the other parties and related documents have been uploaded onto the GIF website.

For the time being the FFC Project had no new collaborative activity yet on the Th-U fuel cycle.

Materials

Although the term of the original Materials Project Plan (PP) was completed in 2012, the Materials Project Arrangement (PA) continued through 2017 while simultaneously pursuing an initial extension of the PP through 2015 and an additional extension through 2020. 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 (the European Union, France, Japan, Korea, Switzerland, and the 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 has been somewhat delayed by several factors including the need to reapprove the GIF Framework Agreement, but is expected early in 2018.

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 2015 were completed. Additionally, by the end of 2017, over 390 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 2017, an additional extension of the PA through 2020 was initiated which is expected to also add Australia (ANSTO) as a new member of the PMB.

In 2017, research activities continued 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 is the oxidation resistance of graphite as functions of temperature and exposure conditions. A summary figure illustrating the strength of graphite following different types of oxidation is shown in Figure 3.88.

Additionally, multiple signatories (Japan, Korea and the United States) continue to examine complementary approaches for improving the overall oxidation resistance of graphite by applying SiC, boron and B_4C coatings to the graphite. Data to support graphite model development was generated in the areas of microstructural evolution, irradiation damage mechanisms and creep. Support was provided for both ASTM and ASME development of the codes and standards required for use of nuclear graphite, 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. China was particularly active in testing and analysing multi-block, large-scale-models of graphite core support structures (Figure 3.89).

Figure 3.88: Residual strength of graphite as a function of oxidation mass loss and temperature of exposure

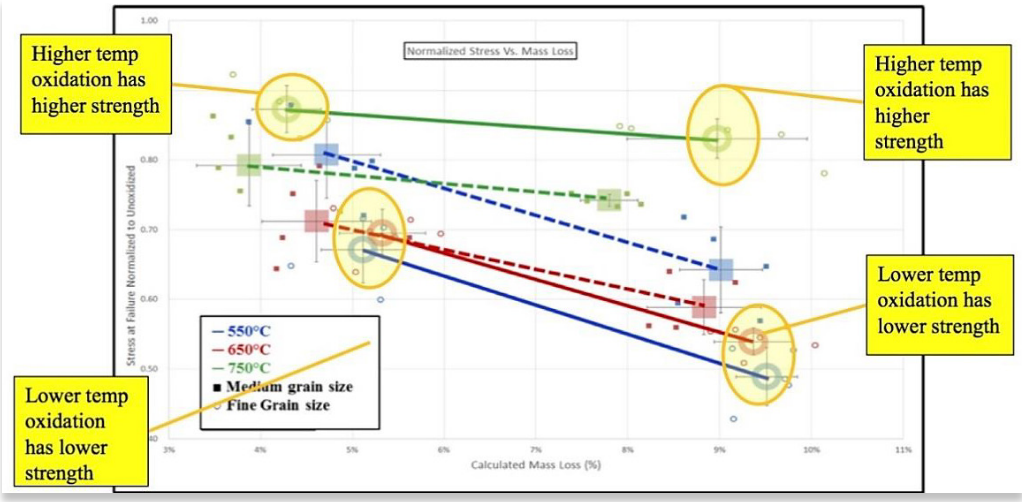
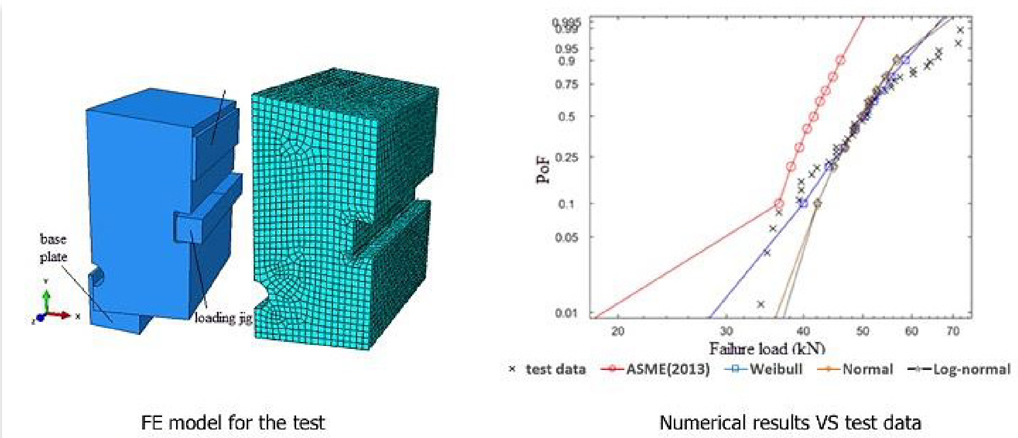


Figure 3.89: INET model and test results for large graphite components



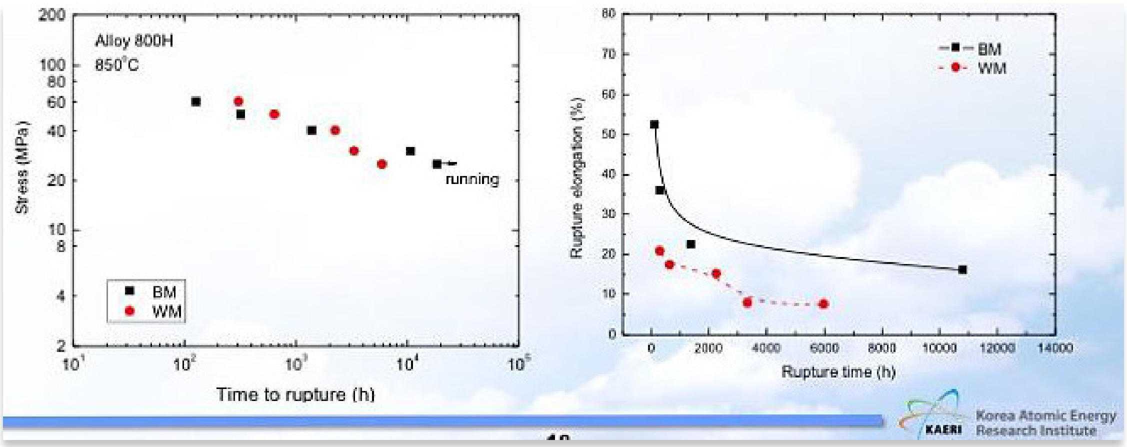
Examination of high-temperature alloys (800H and 617) provided very useful information for their use in heat exchanger and steam generator applications. These studies included an evaluation of the existing data base and an extension of it through creep, creep-fatigue and creep crack growth rate testing 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 h. Data for the Code Case was contributed from the United States, Korea and France. The lower temperature portion of the Code Case, allowing use of alloy 617 at temperatures up to 371°C was approved by ASME and the high temperature portion is expected to be approved in 2018.

An example of work on creep testing of alloy 800H performed by KAERI comparing properties of base versus weldment illustrates that the strength of the weld metal is slightly higher than the base metal, but that its ductility is slightly lower (Figure 3.90).

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.

In the near/medium term, metallic alloys are considered as the main option for control rods in VHTR projects, which target temperatures below about 950°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 continues to examine the thermomechanical properties of SiC and SiC-SiC composites and oxidation in C-C composites. Studies of fabrication, architecture, and processing on the properties and fracture mechanisms of the composites is being investigated. The results of this work is being actively incorporated into developing testing standards and design codes for composite materials, and to examine irradiation effects on ceramic composites for these types of applications.

Figure 3.90: KAERI Creep tests for Base Metals (BM) and Weld Metals (WM) of Alloy 800H at 850°C



Hydrogen production

Active participation of the signatories to the Hydrogen Production Project Arrangement continued during 2017. China remained a candidate and a concerted effort is now in place to complete their accession to the PMB as soon as possible.

The signatories have been focusing mainly on four hydrogen production processes, namely sulphur iodine (SI), high-temperature steam electrolysis (HTSE), copper-chlorine (Cu-Cl) and hybrid-sulphur (HyS) processes.

In Japan, JAEA has carried out demonstration tests of hydrogen production using the SI process. The goal has been to verify the integrity of process components and stability of the hydrogen production process. Following a 31-hour test in 2016 at a production rate of 20 L/h, the focus has been on quantitative evaluation of engineering issues in reactor sections. Prevention methods for leakage of HI-containing solution are also being considered. A longer-term stable operation of hydrogen production process is expected through implementation of improvements achieved from these efforts.

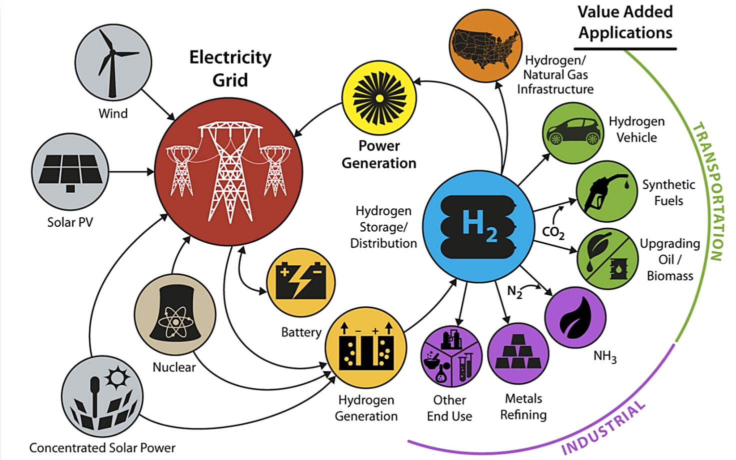
In China, at INET, R&D activities on nuclear hydrogen production have been progressing well. In 2016, an evaluation of the progress and prospect of the two main technologies, the SI and HTSE processes, was conducted, and the SI process was selected for future scale-up and potential coupling to HTR-10. Consideration was given to the difficulties of scale-up, coupling technology and application scenarios of nuclear hydrogen production with HTR-PM600. INET has carried out fundamental studies of improving the efficiency of the SI process (Figure 3.91), including H_2 separation by membrane technology, novel design of the Electro-ElectroDialysis (EED) process for HI concentration, kinetics of the Bunsen reaction, etc. In parallel, development studies of the key prototype reactors have been carried out. In addition, INET has developed a two-step R&D plan for piloting the SI process over the next ten years; the first step is on fully understanding the safety issues of nuclear hydrogen related to the key technologies for pilot scale SI process by 2020, and the second step is the development of HTR-SI coupling technology by 2025.

In Korea, a new three-year project "VHTR key technology performance improvement" was initiated in March 2017. Its purpose is to improve the level of key technologies to support VHTR development and demonstration in the future. The key technologies to be considered are the design analysis codes, thermo-fluid experiments, TRISO fuel, high-temperature materials database and high-temperature heat applications. As a part of this project, a high-temperature heat utilisation technology has been investigated. In 2017, material and heat balance analyses have been performed for high-temperature heat utilisation systems such as SI, HTSE, and Steam Methane Reforming processes in terms of hydrogen production efficiency, energy demand and thermal utilisation. This effort will provide basic information for an evaluation of the performance of coupled systems and optimisation of the VHTR and high-temperature heat utilisation system.

Figure 3.91: INET's sulphur-iodine process for nuclear hydrogen production

The US Department of Energy Office of Energy Efficiency and Renewable Energy (EERE) H2@Scale is funding Idaho National Laboratory to develop and build a 25 kW HTSE demonstration facility to study efficient H_2 production and energy integration (Figure 3.92). This facility will support system integration studies within the Dynamic Energy Transport and Integration Laboratory (DETAIL) at INL. It will demonstrate thermal integration with co-located systems for steam production and will support transient and reversible operation for grid stabilisation studies via a microgrid and Real-Time Data Simulation (RTDS) systems. This facility will be operational in summer of 2018.

Figure 3.92: Hydrogen generation and applications in an integrated energy system



The Sol2Hy2 project, funded by the European Fuel-Cell and Hydrogen Joint Undertaking, has focused on the bottle-necks solving materials research and development challenges, and demonstration of the relevant key components of the solar-powered, CO_2-free hybrid water splitting cycles, complemented by their advanced modelling and processes simulation with added conditions-specific technical-economical assessment, optimisation, quantification and evaluation. The main focus was on the hybrid sulphur cycles (HyS). The main results and achievements of the project are reported in the analysis, development and validation of new process flowsheets to include solar power input for key units of the plant, targeted at selected locations (specified by user) and allowing a flexible combination of different sources, inclusive of the new Outotec Open Cycle, where sulphuric acid is directed to be a commercial by-product to hydrogen rather than to be cracked and returned to the HyS cycle. This enables an increase of the renewable sources share, improves waste heat utilisation and ensures 24/7 plant operation, eliminating solar input instability, combined with reasonable capital costs and balance of the products streams. In the electrolyser unit – the core of H_2 generation – most of the challenges were solved – elimination of platinum-group metals catalysts, control of parasitic reactions, lowering the capital costs (e.g. ~3 times vs. existing analogues). The operation of the developed sulphuric acid cracking unit was successfully demonstrated on-sun at the solar tower Jülich (Germany). An extra development of the software for plant design and optimisation was also carried out allowing the user to analyse and optimise the hydrogen production process in an interactive and guided way, through the use of user-friendly graphical user interfaces. This enables any user to evaluate technical and economic performance of a hydrogen production plant in any feasible location well before field studies.

Significant technical progress has been achieved in the development of the steps of the Cu-Cl hybrid thermochemical cycle in Canada over the last year. Combined with the moderate temperature (<530°C) heat requirement of the process, the recent technical progress has provided the impetus for an accelerated development project to demonstrate an integrated process coupled to an industrial heat source. Process technical viability demonstration plans over a three-year period include a lab-scale system of 50 L/h, already in progress during this year, to elucidate any difficulties in the integration of the electrochemical, hydrolysis, thermolysis and separation/drying steps involved in the process. In parallel, a 1 ton/day capacity unit will be designed for demonstration in collaboration with industrial and academic partners.

Computational methods, validation and benchmarks

Since the Computational Methods, Validation and Benchmark (CMVB) Provisional Project Management Board (pPMB) was restarted in Weihai, China, just before the HTR-2014 conference, six CMVB pPMB meetings have been held in turn during the following three years by different provisional member countries. At these meetings, the main focus was the development of the work packages (WP) of the draft Project Plan (PP). This involved identifying the tasks with the most interest among all of the members in order to maximise participation. All the efforts were made to promote the signing procedure of the Project Arrangement (PA) and to commence the related research projects as early as possible.

The 16th provisional CMVB PMB meeting was held in June 2017 at the HTTR site in Japan. The progress of CMVB activities in different member countries were introduced and the current information was exchanged. All work packages and tasks of the draft PP were reviewed and updated to reflect the discussions made at the meeting.

The five work packages are listed in the following table:

WP No	WP Title	Lead
1	Phenomena identification and ranking table (PIRT) methodology	DOE (United States)
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)

It was agreed that the United States will lead WP1 instead of the EU, and some tasks were moved. Modifications were also made to the contribution sheets and the annual Work Plan (WP). As a large part of the PP is devoted to the validation and verification (V&V) of computational fluid dynamics (CFD) and system models, guidelines and best practices in V&V were discussed in detail.

INET hosted the 17th CMVB pPMB meeting in Beijing. The meeting mainly focused on the review and approval of the final version of PP, PA and First Year Work Plan. After review of the PP and each WP, all participants agreed to approve the CMVB PP. The Chair informed the VHTR SSC and EU that the CMVB pPMB approved the PP and it was delivered to the SSC members for their review and approval. The EU has nominated in the end of 2017 a new representative in the pPMB who is tasked to include new EU input to the project. Comments from the new EU representative on the PA and PP will be incorporated before final approval followed by final approval of the Project Arrangement in 2018. The next CMVB pPMB meeting will be held by KAERI in Seoul on 28-29 March 2018.

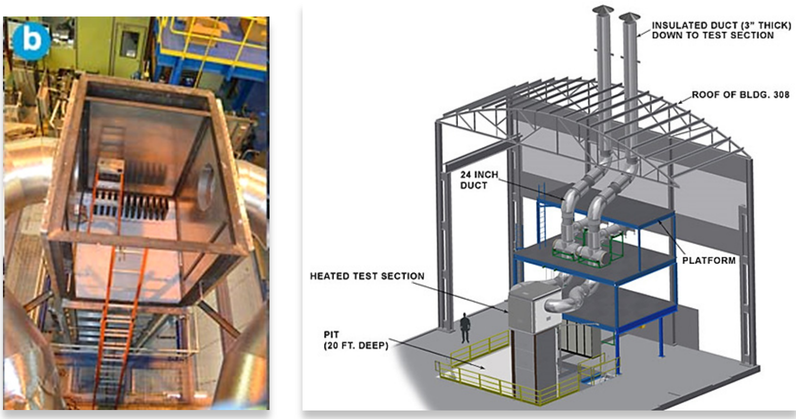
After in-depth discussion in several pPMB meetings, the past, current, and new test facilities and projects have been proposed as potential resources to carry out the experiment model development and benchmarking activities. In China, experiments using 16 separate engineering test facilities, including a helium circulator test facility (Figure 3.93) have been completed in support of HTR-PM development. The component installation of the HTR-PM is proceeding towards start-up testing, perhaps before the end of 2018. Critical tests and start-up physics testing will also contribute valuable data to the code development and validation. China's HTR-10 was restarted to test the major components and system operation. It was operated at power to conduct a melt-wire experiment to measure in-core temperatures. A cold shuffling stage was started in order to discharge the measurement elements out of the core for later inspection.

Figure 3.93: HTR-PM Main Helium Circulator Test Facility

The European ARCHER and GEMINI+ project focus on demonstration-oriented technology R&D with ARCHER completed in January 2015 and GEMINI+ running until 2020. Results are available to this project. Korea has focused its R&D on improvement and validation of VHTR passive safety features such as the hybrid air-cooled RCCS with water jacket. In the United States, the Department of Energy's Advanced Reactor Technologies programme supported the development and validation of core analysis tools, most notably with the construction and operation of thermal fluid test facilities (HTTF, NSTF, MIR, etc.). Data from Natural Circulation Shutdown Heat Removal Facility (NSTF) experiments is available for validation of air-cooled and water-cooled RCCS models.

The HTTF at Oregon State University was shut down after experiencing heater failures during start-up. The re-designed heat elements will be installed in 2018 prior to start-up testing. All these research activities carried out in test facilities and reactors play an important role for V&V of computer codes and calculation methods, which will benefit the CMVB work.

Figure 3.94: Natural Circulation Shutdown Heat Removal System at Argonne National Laboratory



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 (2002) and its update (2014) in terms of economics, proliferation resistance and physical protection, and safety.

4.1. Economics Modelling Working Group

The mandate of the Economic Methodology Working (Group) is to provide a methodology for the assessment of the Generation IV systems against the two economic goals mentioned in the Generation IV Technology Roadmap:

- to have life cycle cost advantage over other energy sources (i.e. to have a lower levelised unit cost of energy);
- to have a level of financial risk comparable to other energy projects (i.e. to have similar total investment cost at the time of commercial operation).

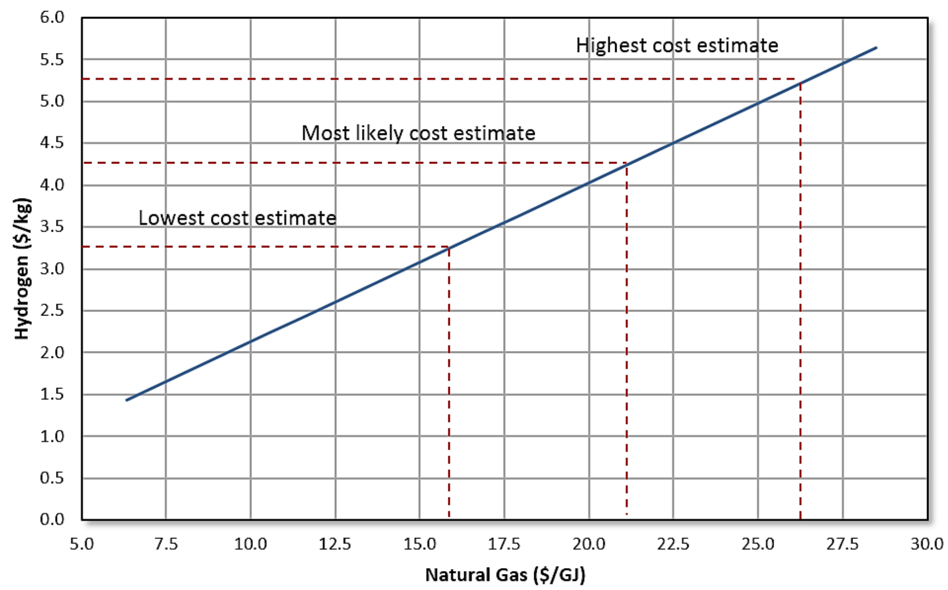
EMWG published its cost estimation guidelines in 2007 along with an Excel-based software, G4ECONS v2.0, for economic assessment of Generation IV systems. These guidelines and economic model have been used for the assessment of Generation IV concepts and several studies have been published to date. Highlights of the EMWG activities in 2017 are briefly described below.

EMWG member delivered a webinar on "Estimating Costs of Generation IV System" as part of the webinar series organised by the GIF Education and Training Task Force on 25 October 2017. The webinar discussed GIF cost estimating guidelines, and use of G4ECONS v2.0 to calculate two figures of merit; levelised unit electricity cost (LUEC) and total capital investment cost (TCIC) and also discussed the benchmarking of G4ECONS with IAEA's NEST economic tool for once-through and closed fuel cycles.

G4ECONS v2.0 was benchmarked against IAEA's Hydrogen Economic Evaluation Programme (HEEP) for economics of hydrogen production using Generation IV reactors, in collaboration with IAEA's Nuclear Energy Division. In this benchmarking analysis the economics of large-scale hydrogen production using a High-Temperature Steam Electrolysis (HTSE) plant connected to a supercritical water-cooled reactor (SCWR) is assessed using the two economic models. G4ECONS v2.0 calculates a levelised unit energy cost (LUEC) for the nuclear energy system and includes a module for calculating a levelised unit product cost (LUPC) for non-electricity applications such as hydrogen production. In addition to hydrogen production costs, HEEP also calculates the cost of hydrogen storage and transportation. Hydrogen LUPC were calculated over a range of capital costs, operating costs and discount rates. The two models predicted hydrogen LUPC that were within 2.6% of each other. Differences could be traced to the different ways of calculating interest on capital investment during the construction period; G4ECONS calculates interest quarterly while HEEP calculates interest yearly.

The cost of nuclear hydrogen production was further compared to that of conventional hydrogen production using natural gas reforming using US DOE's H2A model over a range of natural gas prices. Figure 4.1 shows the range of break-even natural gas prices to be competitive with nuclear hydrogen production over the range of uncertainties in the cost of nuclear hydrogen. The natural gas price in the Far East tends to be significantly higher than in North America. Also, the cost of construction of nuclear plants and associated hydrogen plant would vary with the region. Therefore, the economics of nuclear hydrogen production should be considered in the regional context. The results of this benchmarking study has been jointly documented with IAEA's Department of Nuclear Energy And submitted to a refereed journal for publication.

Figure 4.1: Comparison of nuclear hydrogen costs (red dotted lines) with conventional steam-methane reformed hydrogen (blue line) over a range of natural gas prices



In 2017, EMWG focused its efforts on studying the issues associated with integration of nuclear with renewable resources on the grid, in support of GIF Vice-Chair's initiative on market issues for deployment of Generation IV reactors. This study was based on review and analysis of the published information in close co-operation with the Senior Industry Advisory Panel. It was found that the increasing share of the renewable resources on the grid, driven by favourable policies, are causing unfavourable conditions for nuclear generation because of the requirement of operation under load-following mode. Although some of the current nuclear power plants in Europe already operate in load-following mode either because of excess nuclear generation capacity (e.g. France) or excess renewable capacity (e.g. Germany); flexible operation of nuclear plants leads to excessive maintenance costs, overall lower capacity factors leading to unfavourable economics and possibly early retirements. It has been suggested that small modular reactors could be more suitable for flexible operation for integration with renewable resources. Utility requirements for flexible operation of new nuclear reactors, in terms of ramp rates and depths and frequency response are already known in Europe and North America. It is recognised that the Generation IV systems will required to be designed for flexible operation taking into consideration the utility requirements. Therefore, the system developers should consider flexible operation requirements and include in their research and development stage the aspects, including but not limited to, fuel optimisation for extended period of low power operation, reactivity control for required ramp rates, materials to withstand thermal cycling and fatigue, rapid heat rejection etc.

Hybrid systems are proposed to optimise the profitability of electricity production and co-generation while providing dispatchable electricity to meet flexible demand from the grid. Such hybrid systems would require large-scale energy storage and flexible co-generation application which could be specific to a geographical location. Viability of new nuclear will also largely depend on policies that value the reliability provided by nuclear generation while promoting carbon-free renewable resources. EMWG will produce a position paper based on this study and will also reach out to the System Steering Committee to raise awareness of flexibility requirements for new reactors.

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

The Proliferation Resistance and Physical Protection Working Group (PRPPWG) was created to establish a framework for assessing Generation IV nuclear systems against the proliferation resistance and physical protection (PR&PP) goals of GIF. The PR&PP evaluation methodology developed by the group is described and documented in a publicly available document posted on the GIF public 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, available also in the form of a leaflet). 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) and at the PRPP International Workshop held in Jeju, Korea, October 2016 are also available on from the GIF website. The group maintains a bibliography providing 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/upload/docs/application/pdf/2017-11/gif_prppwg_bibliography_rev05b_2017-11-21_15-41-1_118.pdf).

In 2017, the PRPPWG continued 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 increased effort began in 2016 with the preparation by PRPPWG of a questionnaire addressed to all the GIF SSCs and is a follow-on effort to the joint study by the PRPPWG and the SSCs between 2008 and 2011. The previous effort involved several joint meetings and workshops, and the PRPPWG and SSCs jointly produced white papers on the PR&PP state-of-play, at that time, of each of the six design concepts. The joint study included discussions on cross-cutting issues common to the six concepts.

As one of the measures identified by means of the preparatory questionnaire, in April 2017, the PRPPWG held a joint workshop with representatives of the six systems to provide an overview of the purpose and principles of PR&PP and to discuss developments and design changes that have occurred since 2011. Hosted by the NEA in Paris, the workshop took place over two days. On the first day, the SSCs and the PRPPWG presented the current status of the six GIF system concepts and of the PR&PP Evaluation Methodology and of its application to get a better understanding of the SSC needs and to convey the existing methodology. In most cases, it was clear that the design variants under consideration had changed since the issue of the PR&PP Compendium Report of Gen IV systems in 2011. On the second day, the International Atomic Energy Agency (IAEA) and the GIF Senior Industry Advisory Panel (SIAP) presented some of the activities related to their mandate and relevant to the PRPPWG areas of interest. PR&PP-related activities were also presented by the PRPPWG members. The SSCs and the PRPPWG then discussed the next steps to develop a regular and sustained interaction between the groups. The workshop paved the way 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. The participants agreed to update the Systems PR&PP white papers from 2011 to be consistent with the current scope of the SSCs, and the PRPPWG offered to engage with SSCs as much as possible where there is interest in more substantial interaction.

The PRPPWG held its annual meeting at the European Commission Joint Research Centre in Ispra, Italy on 7-8 November 2017. On this occasion, the group had also the possibility to visit the JRC Advanced Safeguards Measurement, Modelling and Monitoring Laboratory (AS3ML). As the focus of the PRPPWG's near-term efforts centres on engagement with the SSCs, the attendance by members from two of the systems, LFR and MSR, was very useful in defining and refining the path forward with the SSCs. As a result, the PRPPWG and the SSCs have begun a process to update the existing white papers.

The PRPPWG has enjoyed positive and increased interaction with the Senior Industry Advisory Panel (SIAP) during 2017. SIAP representatives participated in both the April workshop and the November PRPPWG meeting, and these meetings provided valuable feedback to the PRPPWG on SIAP's considerations of PR&PP in evaluating the maturity of GIF system designs. This specifically resulted in an improved understanding of the content and purpose of a questionnaire that SIAP had prepared for the GIF systems and the opportunity to iterate on PR&PP-related questions to better achieve their desired result.

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 has focused its activities in recent years 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 – has been pursued actively. The group was represented in the two EG/PG meetings held in 2017 in Paris, France and in Cape Town, South Africa.

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 PRPPWG continues to provide status reports for the RSWG's meetings in order to remain well-connected as the groups plot a collaborative course.

The activity of outreach towards the scientific and technical community and dissemination of results continued in 2017. To this purpose a paper was prepared by the PRPPWG for the IAEA "International Conference on Fast Reactors and Related Fuel Cycles: Next Generation Nuclear Systems for Sustainable Development (FR-17)", held in Yekaterinburg, Russia on 26-29 June 2017. The paper summarises the status of the PR&PP methodology, the ESFR case study, and highlights challenges facing the group to strengthen its visibility and promote further uses of the approach by different stakeholders.

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 28th 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. A PRPPWG contribution to the GIF webinars is scheduled for 23 May 2018.

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, e.g. through participation of members of the group in IAEA meetings, consultancies and conferences. IAEA has been regularly attending and contributing to PRPPWG activities and, in 2017, participated in the April workshop as well as the PRPPWG annual meeting in November. This collaboration is expected to continue and even extend as the GIF designs further mature.

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 February 2017 at the IAEA Headquarters in Vienna, Austria, where fruitful discussions were conducted on opportunities for future collaboration. In October 2017, PRPPWG members also attended an INPRO Technical Meeting on Proliferation Resistance and an Overview of the INPRO Methodology. The attendance was based on national membership to INPRO. In December 2017, the PRPPWG provided training on the PR&PP methodology to a group of eight engineers at the request of the Chinese GIF Liaison Office. This training was very well received and an important example of PRPPWG's continued outreach efforts.

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

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4.3. Risk and safety assessment methodology

The Risk and Safety Working Group (RSWG) is a methodology working group established to provide a harmonised approach to safety assessment of all six Gen IV systems. Since its creation in 2005, the RSWG has proposed safety principles, objectives, and attributes based on Gen IV safety and reliability goals to inform R&D plans. The RSWG has also developed a Basis for Safety Approach (BSA) and an Integrated Safety Assessment Methodology (ISAM). The main goals of the BSA and ISAM are to ensure a consistent approach to safety applied to all six systems, to provide tools for entire cycle from concept development to basic design and then to licensing, and to improve understanding of safety-related design vulnerabilities and the contributions to risk.

Based on the Gen IV safety and reliability goals and the use of ISAM, the RSWG has launched several activities in close collaboration with the System Steering Committees (SSCs).

The Risk and Safety White Papers on pilot application of ISAM aim at demonstrating its applicability as a self-assessment for each of the six Gen IV systems and providing guidance on improving their safety architecture.

The Safety Assessment reports for six Gen IV systems have been launched by the RSWG upon request of the GIF Experts and Policy Group. Their development is led primarily by the respective SSCs with the RSWG involved as a technology-neutral reviewer. The Safety Assessment reports are intended to provide a snapshot of high-level safety design attributes, challenges and remaining R&D needs for the six Gen IV systems.

In 2017, the white papers related to the supercritical water reactor (SCWR) and very-high-temperature reactor (VHTR), and the Safety Assessment Report related to the sodium fast reactor (SFR) have been approved by the GIF Experts Group are now under open access on the GIF website.

The RSWG also actively contributes to the development of safety design criteria (SDC) for the GIF systems. Through its representatives in the GIF SFR Safety Design Criteria Task Force (SDC-TF), the RSWG participated, in 2017, to the second phase of the SDC-TF activity dealing with the development of the Safety Design Guidelines (SDG) on Key Structures, Systems and Components. Based on the SFR SDC and referring to IAEA NSG series, the SDG will focus on reactor core, reactor coolant system and containment system. With a total of 14 SFR-specific focal points, SDG report is expected to be completed in 2018.

The RSWG has also completed the review of a draft LFR Safety Design Criteria focusing on the consistency of LFR system's safety approach as the Gen IV reactor system based on the Basis for Safety Approach and ISAM reports.

The SDC development for the VHTR is expected to be conducted in close synergy with the ongoing work under an IAEA Coordinated Research Project (CRP) on HTGR Safety Design Criteria. In June 2017, the GIF RSWG and VHTR SSC representatives have attended the third Research Coordination Meeting held as "Cooperation between GIF and IAEA in Association with the CRP on Modular HTGR Safety Design".

A draft version of SDC for the GFR is being prepared and it is expected to be shared with the RSWG in the first trimester of 2018. Like for the SFR SDC, the document is based on the review of the IAEA SSR 2/1 Rev. 1 from the GFR design perspective.

The RSWG continues to advice the PG and EG on interactions with the nuclear safety regulatory community, international organisations and stakeholders relevant to Gen IV nuclear systems. In 2017, the RSWG was invited to attend the interface meeting of the Ad hoc joint NEA CNRA/CSNI Group on the Safety of Advanced Reactor (GSAR) responsible for dealing with regulatory and research activities in the primary area of advanced reactors and associated installations and has also participated to the GIF-INPRO/IAEA interface meeting to exchange information on the progress, status and future plans of activities related to R&D and technology innovations of Nuclear Energy Systems, including Gen IV reactors.

In the area of education and training, the RSWG and the other methodology working groups were invited at the end of 2017 to provide a training, organised by the NEA and supported by GIF liaison office of China, to a group of industrial engineers on the use of the different cross-cutting methodologies. The outcome of the training was extremely positive and it is likely to be repeated in the future. The GIF Education and Training Task Force has also requested the RSWG to provide a webinar that is planned for October 2018.

References

GIF RSWG, "Sodium-Cooled Fast Reactor (SFR) System Safety Assessment" (2017).

GIF RSWG, "Super-Critical Water Reactor (SCWR) Risk and Safety Assessment White Paper" (2017).

GIF RSWG, "Very High Temperature Reactor (VHTR) Risk and Safety Assessment White Paper" (2015).

CHAPTER 5

Task Force reports

5.1. Education and Training Task Force

The average age of the nuclear workforce has been gradually rising for the past several years, a phenomenon commonly referred to as the "ageing workforce". This nuclear workforce includes scientists, engineers, technicians and other specialists who have worked in the nuclear industry since its inception and carry with them a vast amount of knowledge and experience. The Gen IV International Forum (GIF) Education and Training Task Force was created to respond to the challenge of not only retaining qualified Gen IV workforce but also forming, training a new generation of engineers, and/or educating and informing a more general public, policy makers on topics related to Gen IV reactor systems and cross-cutting subjects.

GIF webinars development

The task force serves as a platform to enhance open education and training as well as communication and networking in support of GIF, and its objectives are to sustain the nuclear education needed for the promotion and development of Gen IV reactor systems, to increase the knowledge of new advanced concepts, and to serve as a knowledge repository to avoid the loss of the know-how and competences that could seriously and adversely affect the future of nuclear energy.

While many countries are either ramping up or developing nuclear power production as an important step towards economic development and environmental protection, a decrease or uncertainty of the fiscal year budgets have left organisations and agencies looking for new avenues for training and educating a qualified workforce. This has led to an increase in those looking for readily available education and training resources.

Using modern internet technologies, the GIF Education and Training Task Force has launched a webinar series on Gen IV systems in September 2016, which is accessible to a broad audience and is educating and strengthening the knowledge of participants in applications to advanced reactors. This achievement is the direct result of partnering with university professors and subject matter experts who conduct live webinars on a monthly basis. The live webinars are recorded and archived as an online educational resource to the public from the GIF website (www.gen-4.org). In addition, the webinars offer unprecedented opportunities for interdisciplinary crosslinking and collaboration in education and research. The GIF webinars presented in Table 5.1, targets a large spectrum of those that do not know but are desiring to learn about the many aspects of advanced reactor systems.

By exploiting modern internet technologies, the GIF ETTF will continue reaching out to a broad audience and raising the interest and strengthening the knowledge of participants in topics related to advanced reactor systems and advanced nuclear fuel cycles. Besides opening the classroom to everyone in the world, the webinars offer earlier opportunities for interdisciplinary networking and educational and research collaboration.

Table 5.1: GIF webinar series (September 2016 to December 2017)

N	Presenter	Title	Date
1	Dr John Kelly DOE, United States	Atoms for Peace – The Next Generation	29 September 2016
2	Prof. Myung Seung Yang Youngsan University, Korea	Closing the Fuel Cycle	19 October 2016
3	Dr Claude Renault CEA, France	Introduction to Nuclear Reactor Design	22 November 2016
4	Dr Bob Hill ANL, United States	Sodium-Cooled Fast Reactors (SFR)	15 December 2016
5	Dr Carl Sink DOE, United States	Very-High-Temperature Reactors (VHTR)	25 January 2017
6	Dr Alfredo Vasile CEA, France	Gas-Cooled Fast Reactor (GFR)	22 February 2017
7	Dr Laurence Leung CNL, Canada	Supercritical Water Reactors (SCWR)	28 March 2017
8	Prof. Per Peterson UC Berkeley, United States	Fluoride-Cooled High-Temperature Reactors (FHR)	27 April 2017
9	Dr Elsa Merle CNRS, France	Molten Salt Reactors (MSR)	23 May 2017
10	Prof. Craig Smith US Naval Graduate School, United States	Lead Fast Reactor (LFR)	12 June 2017
11	Dr Franco Michel-Sendis NEA	Thorium Fuel Cycle	12 July 2017
12	Dr Steven Hayes INL, United States	Metallic Fuels for SFRs	22 August 2017
13	Dr Richard Stainsby NNL, United Kingdom	Energy Conversion	21 September 2017
14	Dr Geoffrey Rothwell NEA	Estimating Costs of Gen IV Systems	25 October 2017
15	Mr Joel Guidez CEA, France	Phénix and SuperPhenisx Feedback Experience	29 November 2017
16	Dr Christophe Poinssot CEA, France	The Sustainability of Relevant Framework for addressing Gen IV Nuclear Fuel Systems	14 December 2017

In connection with this activity, flyers are developed to advertise the webinars on the Gen IV website and a strong emphasis has been dedicated to the creation and maintenance of a modern social medium platform (such as LinkedIn www.linkedin.com/groups/8416234) as well as a GIF ETTF webpage (www.gen-4.org/gif/jcms/c_97306/education-and-training) to exchange information and ideas on Gen IV R&D topics as well as related GIF education and training activities.

Brochures to advertise the GIF ETTF webinars activities are being developed, and participation at several national and international conferences have occurred. Since the first webinar presented by Dr John Kelly in September 2016, the GIF ETTF has co-ordinated 16 free, live, interactive webinars. As of December 2017, attendance during the live webcasts totals 1 142 and the number of viewings of recorded webinars in the online archive is 1 650 for a total of webinar viewing of 2 792 (Figure 5.1).

The participants in the GIF webinars include representatives from multiple organisations including federal agencies, national laboratories, various state agencies, universities, international organisations, contractors and commercial organisations. As shown in Figure 5.2, 30% of webinar participants are from international organisations. Representatives from state agencies comprise the largest single organisation type.

Figure 5.1: Webinar attendance and number of archived viewings

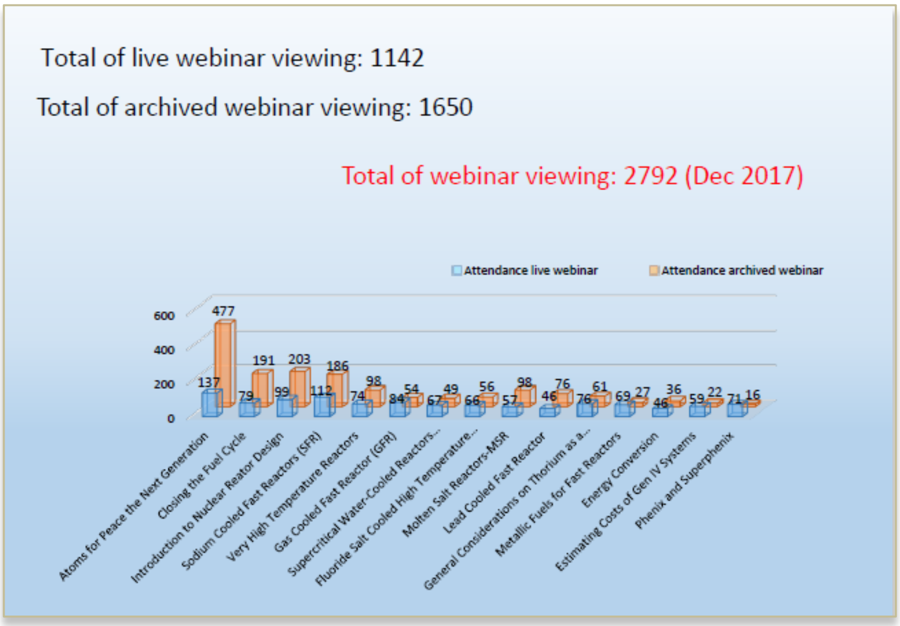
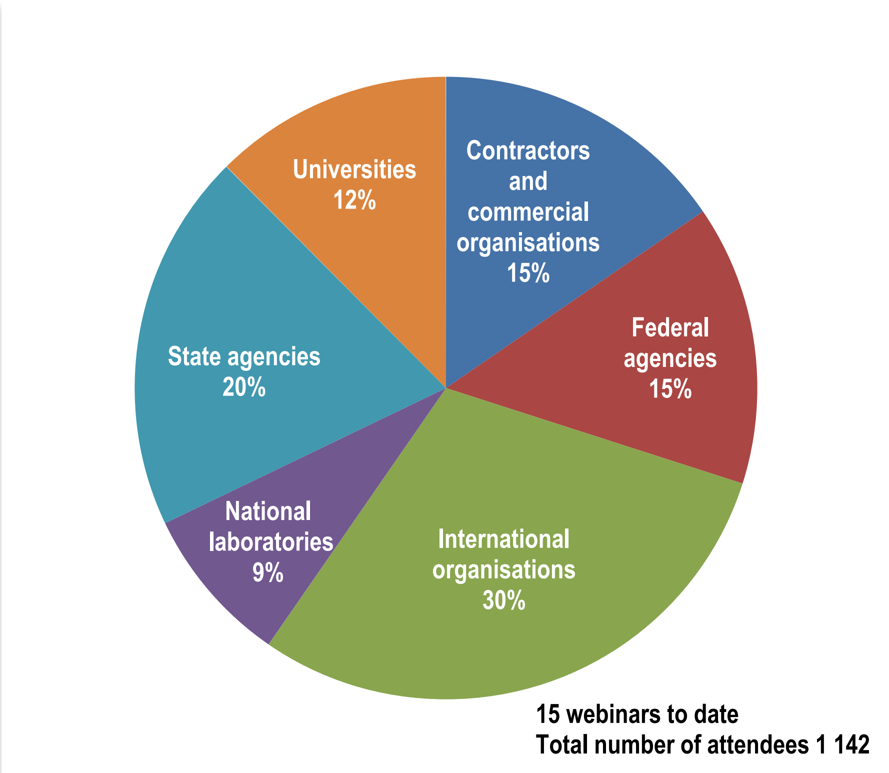


Figure 5.2: Participants by organisation type



There are no fees associated with these webinars, which make the webinars very attractive. The success of these webinars relies on the presenters who are internationally recognised experts.

The attendees thus far have been extremely positive about the quality and content of these webinars as reflected by the following statements:

"I thought it was very interesting. The material is not often presented in other than a graduate school setting so many of us don't have access to it; other than from books. Thank you for making it possible."

Collaborative synergy is being developed with other international education and training organisations such as the European Nuclear Education Network (ENEN), and the African Network for Education in Science and Technology (AFRA-NEST), to promote information, exchange on various training courses, academic education, opportunities for job and scholarship, as well as educational materials, to discuss common concerns, issues and challenges related to the institutional co-ordination for nuclear education.

"Excellent introduction. I look forward to the ongoing program."

"These webinars will benefit a vast audience, keep up the great work!!"

"Very good format. Great outreach. Please continue."

"Excellent, clear and well organized presentation that covered central issues on the topic."

"The technical content of the slides for this webinar were EXCELLENT."

"I like the link to the GIF webinars on the Gen IV webpage. This makes it very convenient to watch the webcasts and/or download the presentations."

Conclusions

To bolster interest and increase awareness in Gen IV reactor systems, the GIF ETTF is offering short (60 to 90 minutes) webinar presentations on specific advanced reactors topics which have been developed and are offered as interactive online conferences. The webinars are recorded and archived to become a library or collection of seminars for online access from the Gen IV website (www.gen-4.org). The GIF webinars have successfully reached a broad audience and continue to gain interest. The momentum and overwhelmingly positive feedback from participants affirm the benefits in these unique educational opportunities and validate the need for additional resources if the United States is to maintain its level of expertise in Gen IV systems. GIF webinars will continue to be a useful education resource for current and future workforce.

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5.2. Task Force on Safety Design Criteria

In 2017, GIF SFR Safety Design Criteria Task Force (SDC-TF) completed the first SFR Safety Design Guidelines report titled "Guidelines on Safety Approach and Design Conditions of Generation IV SFR Systems" and updated the previously published SFR Safety Design Criteria based on the external feedback from the national regulators and IAEA. These accomplishments were highlighted as part of a dedicated panel discussion at the International Conference on Fast Reactors and Related Fuel Cycles (i.e. FR-17) in Yekaterinburg, Russia. The "safety approach" guidelines report was also distributed to NEA Ad hoc Group on the Safety of Advanced Reactors (GSAR) and the IAEA. At the same time, the most significant effort of the SDC-TF in 2017 focused on continued development of design guidelines for Gen IV SFR design tracks. The second safety design guidelines report "Guidelines on Structures, Systems and Components for Generation IV Sodium-cooled Fast Reactor Systems" (SSC SDG) has been under preparation throughout 2017 with its anticipated completion in 2018.

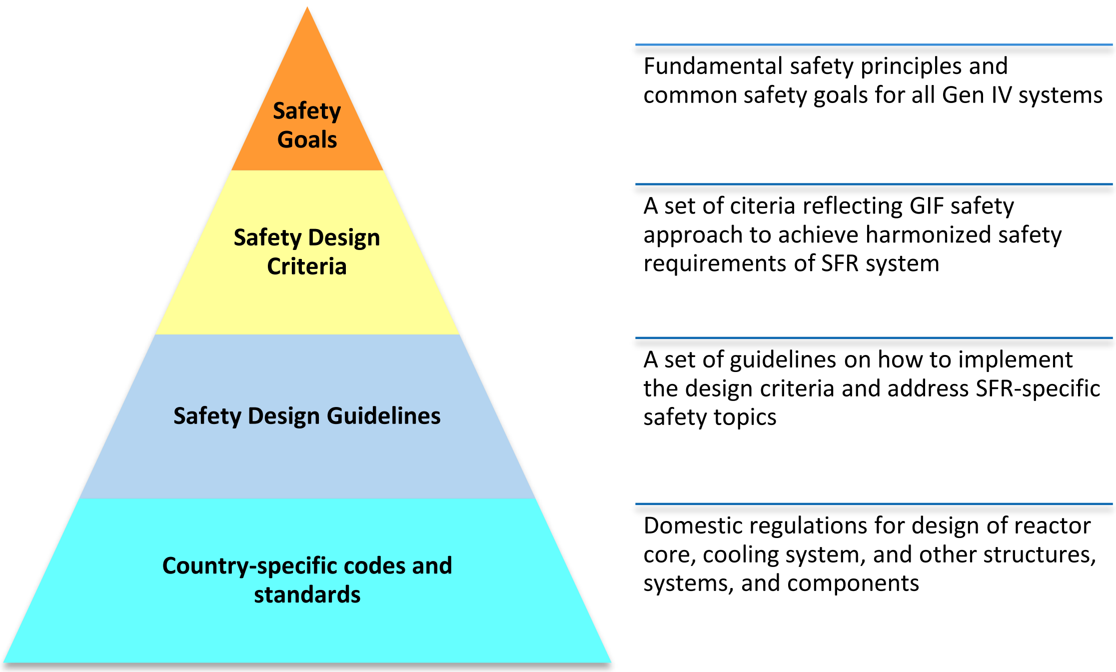
The SFR Safety Design Criteria (SDC) report was completed 2013 and distributed to international organisations, namely IAEA, MDEP, NEA/CNRA, and regulatory bodies of the GIF member states with active SFR development programmes (China, EC, France, Japan, Korea, Russia and the United States). In 2017, the SDC-TF responded to feedback from IAEA, US Nuclear Regulatory Commission (NRC), National Nuclear Safety Administration (NNSA) China and the IRSN France and was finalised. The SDC report was updated based on the comments received ranging from general matters (e.g. safety approach for the Gen IV reactor systems, differences with the Gen III systems, and interface between safety and security) to suggestions for specific criteria (e.g. sodium fires, design-basis accidents and design extension conditions). The SDC-TF updated 22 criteria based on the feedback from NRC, four criteria based on feedback from IAEA, and 15 criteria based on feedback from IRSN. In addition, the SDC-TF adopted most of the technical points from the IAEA SSR 2/1 revision 1, issued in 2016, for the updated SDC, including the new provisions based on lessons learnt from the Fukushima Daiichi nuclear power plant accident, it excluded the new requirements applicable only to light water reactor systems (such as the requirements for containment cooling and alternative power source for emergency coolant injection). The TF's response to comments on the external review and recommendations are summarised in a report. The updated SFR Safety Design Criteria (SDC) report will be published after final approval by the GIF Experts and Policy Groups.

In parallel with the SDC report update, the SDC-TF continued to prepare the SFR safety design guidelines as a set of recommendations on how to implement the design criteria and address SFR-specific safety topics as shown in Figure 5.3. The aim is to facilitate practical application of the SDC to the Gen IV SFR design tracks by clarifying technical issues and providing recommendations with a variety of design options. The first Safety Design Guidelines (SDG) report "Safety Approach SDG" describes prevention and mitigation of severe accidents (i.e. issues related to fast reactor core reactivity) and situations that will practically be eliminated (e.g. issues related to loss of heat removal). It also provides sets of design guidelines for these two issues and provisions with designs options. General design approaches defined in the first SDG report are: (1) the use of multiple redundant engineered safety features (such as independent and diverse scram systems, multiple decay heat removal systems) to cope with design-basis accidents, (2) passive/inherent features for cooling and reactor shutdown for design extension conditions. The Safety Approach SDG was distributed to the NEA GSAR and the IAEA for external review and feedback. The GSAR invited the SDC-TF to attend their meeting in October 2017 and provided the consolidated comments from the GSAR member states. The SDC-TF started the analysis of GSARs extremely important and constructive comments in 2017 and will continue finalising their resolutions along with corresponding revisions to the Safety Approach SDG in 2018.

The TF is currently developing the second and final safety design guidelines report, SSC SDG, which provides recommendations to consider in design of structures, systems, and components (SSC) important to safety and support practical application of the SDC and Safety Approach SDG to the safety-related SSC designs. The SSC SDG specifies 14 focal points related to three fundamental safety functions: (1) Reactor core system: fuel element, reactor core, passive shutdown or inherent feedback, active reactor shutdown, and prevention of significant mechanical energy; (2) Coolant system:

components of reactor coolant system, reactor cover gas boundary, ensuring reactor coolant level, measures against sodium leak and combustion, measures against sodium water reaction, application of natural circulation, reliability for reactor coolant system; and (3) Containment system: containment boundary and load factor, and containment function of intermediate coolant system. The TF will summarise the SSC SDG report and issue it after the EG and PG approve in the coming year.

Figure 5.3: Hierarchy of GIF safety standards, including safety design criteria and safety design guidelines



CHAPTER 6

Market and industry perspectives on Gen IV systems

6.1 Market issues

Background

Market issues for future deployment of Gen IV reactors are a common concern between developers and users. The Senior Industry Advisory Panel (SIAP) put forward two important recommendations at the PG meeting held in October 2015; 1) Identify the attributes of Gen IV systems that are the most attractive for industry (vendor/utility), 2) Investigate market conditions and timelines for commercialisation of Gen IV reactors. Scope of the Market Issue is as follows:

- Better understanding of the drivers, opportunities and constraints related to the market environment for appropriate ways in carrying out GIF activities.
- Close work with the SIAP, SSC chairs, and related TFs in carrying out their 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.

Work plan

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

- **Phase 1**: Survey of key points on market issues.
- **Phase 2**: Build figures of merit to explain how attractive Gen IV systems are in terms of the market drivers.
- **Phase 3**: Understand and value the Attributes of Gen IV systems.

Status of activities

The following issues were identified as the key points on the market issues through the discussions with the SIAP and EMWG; 1) National and International Market Drivers, 2) Market-related opportunities (e.g. SMRs, integration of renewables, non-electric applications to replace fossil fuel based heat production), 3) Market-related constrains, 4) Analysis of the key issues related to political decision making with regard to the energy mix and the role of advanced reactors in each country (e.g. international agreement on the 2-degree C scenario in terms of global warming, and energy security). Based on these points, a questionnaire was sent to PG members to investigate key issues for political decisions of energy mix and role of advanced reactors as a market driver. The feedback from PG members highlighted two key issues for deployment of Gen IV reactors: safety and economic competitiveness with LWRs, and two for political assistance: energy security and economy of energy source. In addition to the questionnaire, SIAP drafted the position paper to the market issues. In this paper, it was pointed out that the safety is most important issue for the public acceptance, and economic competitiveness with LWR is desirable attribute for Gen IV reactors. These views were consistent to those of PG members, but SIAP also suggested another view that reduction of amount/lifetime of high-level radioactive waste and flexibility or applicability for other systems are also important issues for public acceptance and desirable attribute respectively. Also, they mentioned the importance of the cost comparison with renewable and other low-carbon energy sources. These aspects gave new view points for GIF activity, which will be discussed in the 2nd phase of this activity.

6.2 Senior Industry Advisory Panel report

Throughout the year, SIAP pursued its work along the three tracks which had been defined and agreed by the Policy Group in 2016:

- questionnaire for Design Review of mature systems by SIAP;
- support to the Vice-Chair for market issues;
- yearly charge.

In 2016, SIAP had drafted an open-ended questionnaire to be sent to the SSCs, giving them an opportunity to provide information on mature systems designs, for review by SIAP to issue recommendations for further needed R&D activities, taking account of industry perspectives and expectations. The notion of maturity had been clarified as linked with a time frame of 2030/2035 for the pre-FOAK (pre-First-Of-A-Kind), 2037/2040 for the FOAK and 2045 for commercial deployment readiness. The pre-FOAK stage is corresponding to the demonstration phase of the GIF 2014 roadmap, aiming at demonstrating the technological, industrial and licensing feasibility of the proposed design, and providing elements for the economic evaluation to be further confirmed by the FOAK stage.

The draft questionnaire was presented first time to the Policy Group in October 2016 and it was decided to proceed to a pilot test of the questionnaire for some volunteering systems, and to use the outcome to validate and improve the questionnaire before it to be offered at large for use by all systems.

Two designs, both coming from the VHTR system, were proposed and filled questionnaires sent back to SIAP in September 2017: the Chinese HTR-PM and the Areva GT-MHR.

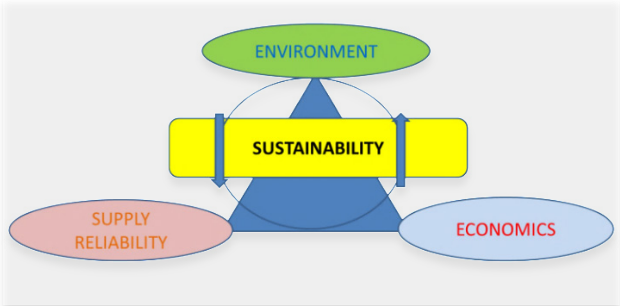
The replies were analysed in depth by SIAP and extensively discussed at the 2017 meeting in Cape Town. The main outcome, beyond design specific recommendations, was to confirm that, apart from minor changes and the need to add a question on PRPP issues (resulting from interaction with the PRPPWG [Proliferation Resistance and Physical Protection Working Group]), the questionnaire was appropriate. Indeed the proposed questions allowed the users to provide enough information, at the right level of quality, for analysis by SIAP to provide recommendations.

Following the meeting in Cape Town, the questionnaire was finalised by SIAP and sent to the Technical Director for further distribution to all SSCs for their use. SIAP is now waiting for systems to provide filled questionnaire for its review. Their analysis will constitute one main activity of SIAP in the coming years.

To support the work of the Vice-Chair in charge of Market Issues, SIAP drafted a paper on Gen IV and Market Issues. Using a broad reading of the notion of attractiveness for the “market”, the paper starts from the concept of sustainability in its wider understanding: environment, economics, reliability of energy supply, and from there proposes more precise areas for targeted R&D. The SIAP vision of sustainability, which is further explained in the paper, can be illustrated by the picture below and may support a broader discussion within the GIF on this concept of sustainability.

Beyond the production of this paper, SIAP has in the course of 2017 strengthened its links with the EMWG (Economic Modelling Working Group), also much connected with the responsibilities of the Vice-Chair for Market Issues. A number of SIAP members attended and contributed to EMWG meetings. This will continue in the future and in particular SIAP has indicated its willingness to review EMWG papers on opportunities and challenges for Gen IV reactors integration into systems with increasing shares of variable renewable sources.

Figure 6.1: Energy sustainability triangle



In addition, SIAP is ready to support the mission of the Vice-Chair for Regulatory Issues. Indeed SIAP considers it appropriate to have all three stakeholder communities (research, regulators and industry) involved when discussing long-term R&D perspectives. Early and continuous engagement of all stakeholders is expected to foster and accelerate the innovation process, as it has been developed by the NEA in its NI2050 (Nuclear Innovation 2050) concept. Indeed ensuring a better alignment of the Technology Readiness Levels with the Licensing Readiness Levels of innovative technologies should help gain time. These concepts are illustrated in the graphs below. Within NEA they will be used to foster a cross-committee dialogue on nuclear innovation, involving in particular the Nuclear Science Committee (NSC), on one hand, and the Committees for the Safety of Nuclear Installations (CSNI) and on Nuclear Regulatory Activities (CNRA), both being the overarching bodies of the GSAR (Joint CSNI/CNRA ad hoc Group on the Safety of Advanced Reactors). GSAR is the main framework serving today as a dialogue platform between GIF and the regulators community.

Figure 6.2: NEA's Nuclear Innovation 2050 concept

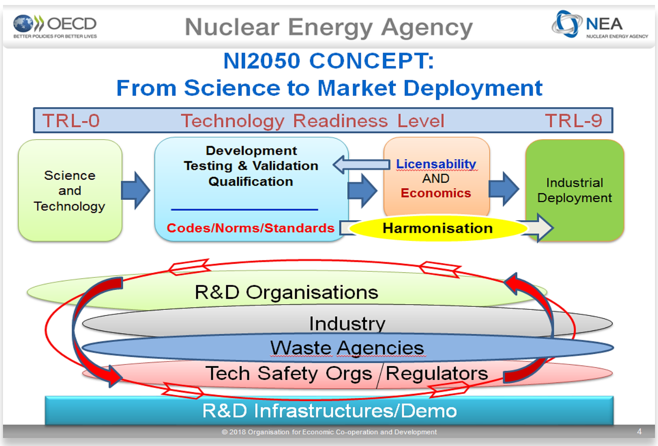
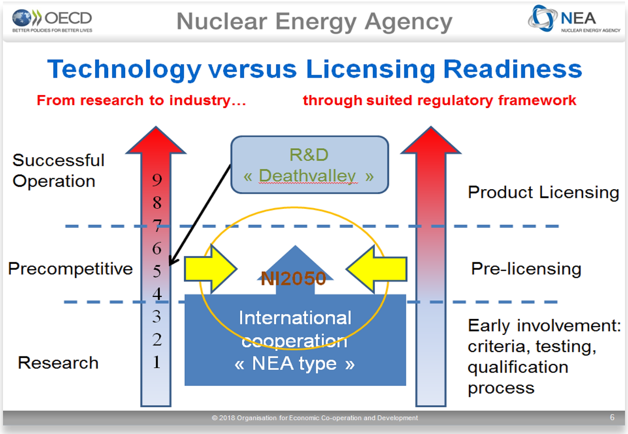


Figure 6.3: NEA's NI2050 Concept: Technology and Licensing Readiness scales



The Charge 2017 focused on the prospects from recent Gen III innovations for Gen IV systems development, covering both technological and organisational innovations of Gen III reactors designs and deployment. The aim was to extract lessons learnt which Gen IV may benefit from at time of moving towards demonstration phases. A non-exhaustive set of issues were proposed for the reflection of SIAP.

Main SIAP general recommendations were presented to the Policy Group along the following lines:

- Engage regulators early, integrate early the safety requirements in the design (i.e. safety classification).
- Engage MDEP like and CORDEL: harmonisation of licensing and design rules.
- Finish research by end of conceptual design and finish detailed design of nuclear island before start procurement and construction.
- Modularisation is to be pursued but there are pro and cons.
- Consider (and optimise) plant long-term maintainability (accessibility, inspectability, online monitoring, predictive maintenance, spare parts and obsolescence, etc.), waste management and decommissioning aspects during the design phase.
- Project management (for large industrial project) have to be improved, learning lessons of past experiences in nuclear and non-nuclear areas.
- Use PLM (product/project life cycle management) throughout project from design to construction, operation and decommissioning.
- Follow closely the market demands and evolution, including in terms of U supply.

Time constraints during the meeting in Cape Town did not allow to go further in extracting R&D recommendations from the above list of general technological and organisational recommendations. This might have to be further pursued in the course of 2018.

This leads to an organisational issue which might deserve further attention: the best use of the limited time during the bi-annual GIF meeting weeks (spring and autumn), where all groups are coming together. Indeed, from the SIAP perspective, time should be used to the maximum extend for "internal SIAP work". But at the same time SIAP is expected to report to the EG and to the PG – requiring to produce nearly in parallel the substance documents and the presentation files, which reduces the time for in depth dialogue between the members, without mentioning the need for interactions with other groups, i.e. the SSCs (on Questionnaire for Design Review) and the EMWG (on Market Issues).

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 subassembly
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 MWth 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 eleventh edition of the Generation IV International Forum (GIF) Annual Report highlights the main achievements of the Forum in 2017. During the year, several of the GIF Project Arrangements were extended for another ten years, new projects were prepared and others terminated, thereby setting the scene for long-term co-operation among GIF members. Australia, which joined the GIF in 2016, formally acceded to the Framework Agreement in 2017, and subsequently signed the Systems Arrangements for very high temperature reactors and the molten salt reactors. The safety design criteria and guidelines first developed for sodium fast reactors were extended to other systems, and the Education and Training Task Force successfully organised twelve webinars. In the context of rapidly evolving energy markets and efforts to reduce global greenhouse gas emissions, the GIF continued to work on assessing and highlighting the benefits of deploying Generation IV systems with the support of the Economic Modelling Working Group and the Senior Industry Advisory Panel.

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