ANNUAL REPORT 2019

GEN IV International Forum℠

THE GENERATION IV INTERNATIONAL FORUM

Established in 2001, the Generation IV International Forum (GIF) was created as a co-operative international endeavour seeking to develop the research necessary to test the feasibility and performance of fourth generation nuclear systems, and to make them available for industrial deployment by 2030. The GIF brings together 13 countries (Argentina, Australia, Brazil, Canada, China, France, Japan, Korea, Russia, South Africa, Switzerland, the United Kingdom and the United States), as well as Euratom – representing the 28 European Union members – to co-ordinate research and development on these systems. The GIF has selected six reactor technologies for further research and development: the gas-cooled fast reactor (GFR), the lead-cooled fast reactor (LFR), the molten salt reactor (MSR), the sodium-cooled fast reactor (SFR), the supercritical-water-cooled reactor (SCWR) and the very-high-temperature reactor (VHTR).

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1 February 1958. Current NEA membership consists of 33 countries: Argentina, Australia, Austria, Belgium, Canada, the Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Poland, Portugal, Korea, Romania, Russia, the Slovak Republic, Slovenia, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The European Commission and the International Atomic Energy Agency also take part in the work of the Agency.

The mission of the NEA is:

- to assist its member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally sound and economical use of nuclear energy for peaceful purposes;
- to provide authoritative assessments and to forge common understandings on key issues as input to government decisions on nuclear energy policy and to broader OECD analyses in areas such as energy and the sustainable development of low-carbon economies.

Specific areas of competence of the NEA include the safety and regulation of nuclear activities, radioactive waste management and decommissioning, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information. The NEA Data Bank provides nuclear data and computer program services for participating countries.

The Nuclear Energy Agency serves as technical secretariat to GIF.

Foreword from the Chair

It is a great honor for me to pen this foreword to our Annual Report about the progress of Generation IV reactor systems and collaborations on the developments. The Gen-IV reactor systems are the next generation for the sustainable use of nuclear energy from the current light water reactors of Generation III or III+.

Since 2001, the Generation IV International Forum (GIF) promotes international R&D collaboration for the development of six types of Gen-IV reactor systems, using sodium, lead, gas, molten salt, and supercritical water. With the right level of international policy support, and ambitious R&D funding, the objective is to reach commercial deployment of the most advanced systems from the 2030s onwards. These systems follow common development goals: safety and economics are two key goals together with sustainability and proliferation resistance and physical protection, which are principles followed by GIF since the beginning. Today, the relevance of these goals remains essential to achieve breakthrough in nuclear energy.

I am the sixth Chair in the 20 years of GIF history since January 2019, succeeding Mr François Gauché of France. GIF directions were set in 2019 as follows:

"We have roadmaps to develop Gen-IV reactor systems and methodologies to assess their compliance to the GIF goals. We will also need to show how these advanced nuclear technologies can integrate into and support future clean energy systems." Current priorities of GIF are: 1) Safety and regulation: Continue the development of international safety design criteria to facilitate future licensing activities; 2) Market opportunity and challenges: Integration of Gen-IV systems (flexibility, economics) and renewable energy systems in clean energy systems; 3) R&D collaboration: Enhancement of international R&D collaboration; 4) Attracting the young generation.

These priorities are led by our three outstanding vice chairs; Ms Alice Caponiti (United States) for safety and regulation, Mr Sylvestre Pivet (France) for market opportunity and challenges, and Mr Jong Hyuk Baek (Korea) for R&D collaboration.

Given the importance of international safety standards for the licensing of Gen-IV reactor designs, GIF has developed Safety Design Criteria and Guidelines. To take this work further, GIF is also engaging with the nuclear safety community at the international level (with the OECD Nuclear Energy Agency [NEA] and the International Atomic Energy Agency [IAEA]). For example, risk-informed and performance-based approaches, reduction of emergency planning zones (EPZ) for small modular reactors (SMRs) discussed at the IAEA are great concerns to GIF about the early deployment of Gen-IV reactors.

GIF is also convinced that stronger co-operation is needed between R&D bodies and the private sector. This is especially important in order to integrate future market opportunities and constraints at the design stage. On this point, a significant value of the Gen-IV systems is to contribute to the reliable clean energy systems through new sources of flexibility. Higher temperatures of sodium, lead, or gas allow coupling of electricity production with heat storage or hydrogen production as part of a hybrid energy system. Some Gen-IV systems could broaden the flexibility of existing nuclear reactors, for instance load-following capabilities.

Especially from the GIF directions, we will show GIF outputs to the world, policy makers, and industries more than before in order to promote the opinions to make progress with the Gen-IV reactor systems in difficult and complex energy market situations, expanding renewable energies, concerns for safety after the Fukushima Daiichi nuclear power plant accident (1F accident), and the increasing violent weather issues coming caused by global warming. We had several opportunities to express our opinions and results to the world. The CEM10 side event of nice future initiative in Vancouver, Canada was a good occasion for GIF. We also joined the IAEA worldwide meeting of "Climate change and Role of Nuclear Power" in Vienna and gave a keynote presentation.

Dr Hideki Kamide, GIF Chair, representing GIF at the 10th Clean Energy Ministerial (CEM 10) in Vancouver, Canada in May 2019

In parallel to these world conferences, GIF had several workshops with private sectors, the GIF meeting in Canada in May 2019, the GIF meeting in China in October 2019, and also in Paris in February 2020 to identify collaborations on promoting the Gen-IV reactor developments and deployments including SMRs. I believe that these workshops will contribute to the higher opportunity of Gen-IV reactors in the energy market.

This annual report covers our overall activities in 2019 about six reactor systems and also cross-cutting issues of Task Forces and Working Groups. I expect that this annual report is meaningful for all of readers to refer our GIF activities and to find a good relation and collaboration with us.

Hideki Kamide

GIF Chair

Foreword from the Technical Director

It is a pleasure and a big responsibility to serve the Generation IV International Forum (GIF) as Technical Director. In 2019 a big change in all the GIF Organization occurred. I am taking this opportunity to thank and to send my regards to the previous team.

This new Governance Structure is facing new challenges and has to define new horizons. Indeed, Generation IV International Forum will be 20 years old soon (in 2020). For the GIF organization, it is the right time to review its objectives in the new context of a deregulated energy market and a decarbonized future society.

GIF has become an unavoidable organization for people concerned by Gen-IV reactors systems and their related safety, economics and Proliferation Resistance & Physical Protection (PR&PP) items. The key position of systems must be pursued and reassessed, under the prism of the new economy market, the future energy mix, and the transition towards a low-carbon society. The six Gen-IV systems have significant assets to be part of this new energy paradigm. And it is important to highlight the GIF systems coherence in the energy mix through their flexibility and load following; their ability to ensure energy cogeneration (e.g. heat, desalination) and/or a coupling with large-scale energy storage (hydrogen); their deployment flexibility (large – small or micro-scale reactors, small modular reactor type), their siting adaptability.

Thus, it is important to pursue the involvement in these reactor systems, and also to confirm the key role of cross-cutting working groups in main domains: Safety and PR&PP, Economy and Market, Market Opportunities, vision from Industrials, compatibility with renewables, and also future innovative techniques deployable to nuclear systems.

At the same time, GIF has to promote its works on Gen-IV systems, and its views on future energy challenges to the large audience that it can reach due to its reputation. It has started thanks to symposium and webinar initiatives; and we will make efforts to pursue this dynamism and give direction towards a targeted and efficient GIF communication plan.

2019 was a transition year between the great success of the 4th GIF Symposium in October 2018, the large change in GIF Governance, and the new direction where we want to pave the way for the years to come. It was also an interesting year for the various technical progresses that all GIF members achieved. I am convinced that this Annual Report will give you a proper overview of the excellence of these results, and some keys towards our future challenges. The last GIF Annual Report was in 2017. In 2018 – due to the GIF Symposium Event and large involvement of all GIF members – it was decided skip the report. This report will therefore synthetize in some extents the two last years activity report.

Gilles Rodriguez

GIF Technical Director

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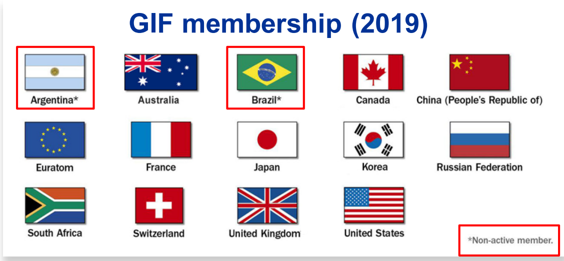
Chapter 1. GIF membership, organization and R&D collaboration

GIF membership



The Generation IV International Forum (GIF) has 14 members, which are signatories of its founding document. Among the signatories to the charter, 12 members (Australia, Canada, France, Japan, China, Korea, Russia, South Africa, Switzerland, the United Kingdom, the United States and Euratom) have signed the Framework Agreement (FA) and its extension. Signatories of the charter are expected to maintain an appropriate level of active participation in GIF collaborative projects. They formally agree to participate in the development of one or more Generation IV systems selected by GIF for further research and development (R&D). The UK was among the first countries to join the GIF (in 2005) but did not ratified the Framework Agreement and UK R&D teams were involved in GIF projects through Euratom. In 2019, UK ratified the Framework Agreement and signed the VHTR and SFR system arrangement. Argentina and Brazil have signed the GIF Charter but have not ratified the FA. They are designated as "non-active members".

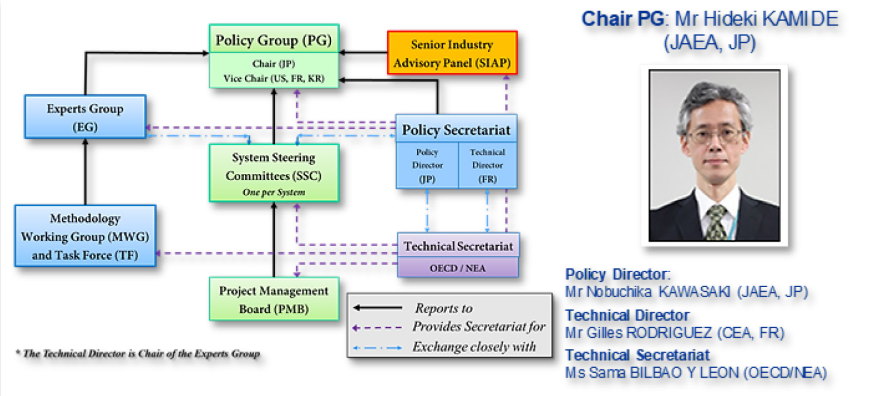
Members interested in implementing co-operative R&D on selected systems have signed corresponding System Arrangements (SA) consistent with the provisions of the FA. This is the case for the Sodium Fast Reactor (SFR), the Very-High-Temperature Reactor (VHTR), the Super Critical Water Reactor (SCWR) and the Gas Fast Reactor (GFR) systems. All four SAs were extended in 2016 for another ten years. Co-operation on the Molten Salt Reactor (MSR) and the Lead Fast Reactor (LFR) systems takes place under Memoranda of Understanding (MoU). The process of SA signature for the MSR system is ongoing.

GIF organization

The GIF Charter provides a general framework for GIF activities and outlines its organizational structure. 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. The PG usually meets twice a year. In 2019, the PG met in **Vancouver (Canada)** in May, and in **Weihai (China)** in October.

The Experts Group (EG), which reports to the Policy Group (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 Systems Arrangements. The EG meets twice a year. These meetings are held back-to-back with the PG meetings in order to facilitate exchanges and synergy between these two groups. Every GIF members nominates two representatives in the PG and two in the EG. **In 2019 the whole GIF Governance was renewed** (see Chapter 2 and Figure 1).

Figure 1. **GIF Governance as of 2019**



Signatories of each SA have formed a System Steering Committee (SSC) to plan 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 addresses the R&D needs of the corresponding system in a broad technical area. The project activities are described in a multiannual Project Plan (PP).

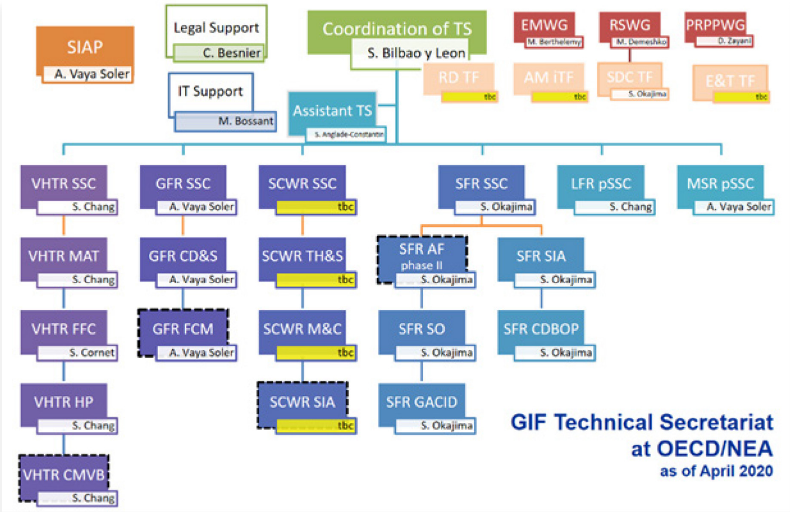
The GFR is composed of two Project Arrangements: Conceptual Design and Safety (CD&S) and Fuel and Core Materials (FCM). The SCWR has three PAs: Materials & Chemistry (M&C), Thermo-hydraulics and Safety (TH&S), System Integration and Assessment (SIA). The SFR contains four PAs: Advanced Fuel (AF), Component Design & Balance-of-Plant (CDBOP), Safety and Operation (SO), System Integration and Assessment (SIA). And the VHTR has four PAs: Hydrogen Production (HP), Fuel and Fuel Cycle (FFC), Materials (MAT) & Computational Methods Validation and Benchmarking (CMVB). R&D carried out under an MoU (the case of LFR and MSR systems) is co-ordinated by a provisional System Steering Committee (pSSC). The GIF Charter allows the participation of organizations from public and private sectors of non-GIF members at the PAs level, with the unanimous approval of the corresponding SSC.

Three Methodology Working Groups (MWGs): the Economic Modelling Working Group (EMWG), the Proliferation Resistance & Physical Protection Working Group (PR&PP WG), 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, PR&PP, and risk and safety. The Methodology Working Groups report to the Expert Group that provides guidance and periodically reviews their work plans and progress. The specific reports of each Working Groups are in Chapter 5.

Figure 2. **GIF Policy Group at the 2018 GIF Symposium in Paris, France**

From left to right: Jong-Hyuk Baek, Henri Paillère, Yun Jung, Steve Napier, John Kelly, Linhao Chen, Thomas Fanghänel , Diane Cameron, Jinlei Jia, Lyndon Edwards, Alice Caponiti, William Magwood, Didier Gavillet, Suibel Schuppner, Kiyoshi Ono, Konstantin Kornienko, Hideki Kamide, Nobuchika Kawasaki, Daniel Brady, Sylvestre Pivet, Gilles Rodriguez, Bessie Mokgopa, and Victor Ignatiev

Figure 3. **Structure of the GIF Technical Secretariat**



In addition, the PG creates dedicated Task Force groups (TFs) to address specific goals or to produce specific deliverables within a given time frame. In 2019, three Tasks Forces arrived to completion: Advanced Manufacturing and Materials Engineering (AMME), R&D Infrastructure (RDTF) and Safety Design Criteria/Safety Design Guidelines (SDC/SDG) Task Forces. Their respective 2019 results are described in Chapter 6. In 2020, the Policy Group will discuss the opportunity to continue to pursue these items into new Task Force groups.

Due to its important and key role in the continuous dissemination of GEN-IVIV scientific knowledge, the Education and Training Task Force (ETTF) was transformed at the end of 2019 into the **Education & Training Working Group** (see the corresponding Report in Section 6.1).

A Senior Industry Advisory Panel (SIAP) comprised of executives from the nuclear industries of GIF members was established in 2003 to advise the Policy Group 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 influencing future potential introduction of commercial Generation IV systems are taken into account (see Chapter 7).

The GIF Secretariat ensures the day-to-day co-ordination of GIF activities and its communication. It includes two groups: the Policy Secretariat and the Technical Secretariat. The Policy Secretariat assists the Policy Group and Experts Group in the fulfilment of their responsibilities. The Policy Director specifically assists the PG on policy matters. The Technical Director serves as Chair of the Expert Group and assists the PG on its technical strategy and vision. The Technical Secretariat team (NEA) assists all the GIF technical boards (the six Systems, Working Groups, Task Force, SIAP), maintains the public and protected GIF websites, and organizes the GIF main initiatives: conferences, symposium, workshops and communication events. The NEA is entirely resourced for this purpose through contributions financial from all GIF members.

Sama Bilbao y León

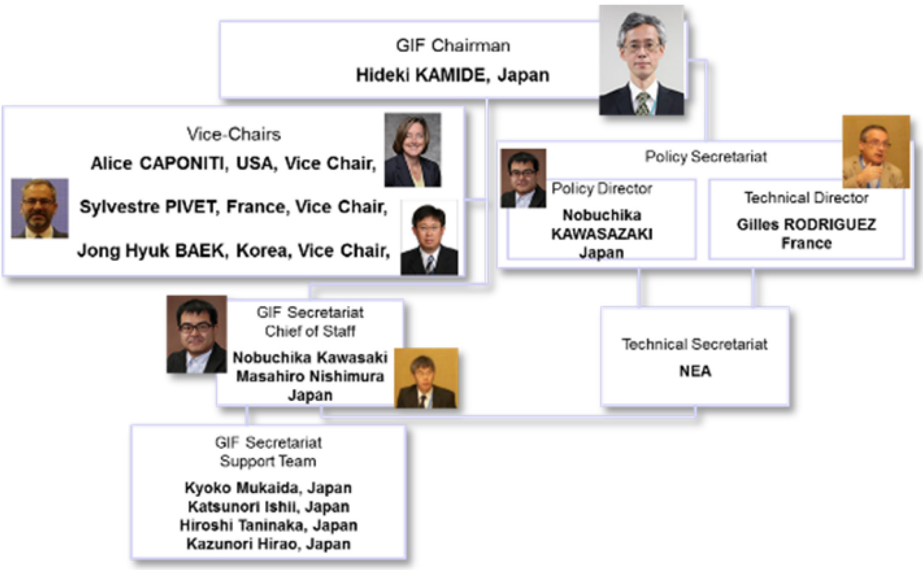
Head of the Generation IV Technical Secretary

Chapter 2. Highlights from the year

General overview

In 2019, Generation IV International Forum has completely renewed its Board with new key members in all the governance position. This **new governance** was completed and accepted in May 2019 during the 47th Policy Group meeting in Vancouver (Canada).

Figure 4. **GIF Leadership (2019)**



For the first time in the GIF Organization, each Vice Chair was assigned a three-year mission to assist the GIF Chairman to better understand the drivers, opportunities, and constrains related to three key cross-cutting topics related to all Gen-IV systems:

- regulatory issues (assigned to Alice Caponiti, United States);
- market opportunities and challenges (assigned to Sylvestre Pivet, France);
- enhancement of R&D collaborations (assigned to Jong Hyuk Baek, Korea).

On the **regulatory issues**, the mission for the Vice Chair is to co-ordinate the GIF's efforts with various regulatory bodies to achieve harmonized regulatory requirements. Routine dialogue between the international research and development community, and regulatory community is mutually beneficial to facilitate a collaborative approach to:

- identify and resolve key regulatory issues;
- identify and address safety research needs;
- further harmonize design, safety and regulatory requirements.

The work will therefore include continuing to promote the external view of Safety Design Criteria and Guidelines (SDC/SDG) reports for sodium fast reactors (and by extension to other Gen-IV systems), and working towards IAEA Safety Standards. The role is also to lead GIF engagement with the OECD/NEA WGSAR (Working Group on the Safety of Advanced Reactors) and with the related IAEA Sections.

On **Market Opportunities and Challenges**, one important topic identified is dealing with decarbonized hybrid energy systems. It is anticipated that in future low-carbon energy systems, variable renewable energy systems will have an increasing share of the overall energy mix, and will need to be complemented by energy storage and dispatchable energy technologies. Gen-IV energy systems in that sense could play an important role both as a low-carbon source of electricity and as a source of low-carbon heat for industrial or other applications in the decarbonized hybrid energy systems. In this context, the proposed Vice Chair mission is to engage with external GIF stakeholders (private sector, policy makers, investors) and various GIF bodies (Systems Steering Committees, Economic Modelling Working Group, Senior Industrial Advisory Panel) on how Gen-IV systems could address future energy market needs and challenges, and support the development of innovative reactor concepts. The role of the Vice Chair is also to co-ordinate GIF-related activities with other multilateral initiatives promoted by international organizations (IAEA/INPRO, NEA/IFNEC, NEA/NI2050). This mission could contribute to further investigate and assess through position papers, the cost and value of Gen-IV nuclear systems, the role of the private sectors for the deployment of Gen-IV systems, and to adjust the policy to develop Gen-IV systems.

The scope of the Vice Chair mission on **Enhancement of R&D Collaboration** is to assist the GIF Chairman with the help of the GIF R&D Infrastructure Task Force and all the GIF system bodies in:

- better understanding the drivers, opportunities and constrains related to the use of large-scale facilities for qualification purposes;
- examining the various means to share such resources and also R&D results in order to optimize joint research and shared results;
- investigating other R&D topics (cross-cutting issues) that are relevant to the research challenges or technical gaps, to further boost the development of Gen-IV systems positioned in the future energy mix.

The Vice Chairs reports the progress of their mission during the Expert Group/Policy Group meetings.

During 2019, GIF participated to The Clean Energy Ministerial (CEM10) and Mission Innovation (MI-4) in Vancouver (Canada) with a Showcase Poster booth in the Clean Energy sector, and the participation to a GIF join Expert Panel on "Breakthroughs – Insights on Nuclear Energy Innovation". The Vancouver meeting was also the opportunity to hold a Round Table discussion between all **GIF Policy Group members and the SMR vendors** attending this event (more than 15 companies gathering all the six Gen-IV concepts).

GIF also presented a position talk in the IAEA International Conference on Climate Change and the Role of Nuclear Power (October 2019, Vienna). It was the opportunity to highlight the growing interest from the private sectors for advanced reactors – essentially on Small or Micro Modular Reactor – which is changing the vision of future Gen-IV deployment. Consequently, the Gen-IV systems have now to reply and adjust their solutions to some increased flexibility requirements. Advanced nuclear energy systems can provide solutions underpinning economic growth, offering additional features in terms of performance and sustainability compared to existing concepts. The GIF calls on policymakers to acknowledge the real contribution that nuclear energy is making today to the mitigation of carbon emissions from the power sector, and to consider supporting the deployment of advanced nuclear reactors and related innovative applications.

At ICAPP 2019 Conference (Juan les Pins, France), GIF signed with other 42 members the **"Declaration from Nuclear Societies"**[1] to reassess, and "*that Nuclear Energy can make its full expected contribution, as part of the clean energy portfolio, towards decarbonization goals*".

1. www.sfen.org/sites/default/files/public/atoms/files/declaration-icapp-2019_002.pdf.

Turkey presented its bid to join the GIF at the 46th PG meeting (October 2018). GIF Policy Group agreed to review this bid through technical discussions (phone conferences) between GIF and Turkish experts on areas of common interest: mainly on MSR chemistry, materials and design, and Innovative Power conversion. A GIF delegation of experts realized Technical visits to Turkish National Scientific and Nuclear Organisms (TUBITAK, TAEK and Istanbul University) in August 2019, and reported to the Policy Group (48th in Weihai, China) their technical audit and conditions to pursue the Turkey bid process. This process will continue in 2020.

Highlights from the Experts Group and the Cross-cutting Working Groups and Task Force

The Experts Group advises the Policy Group on priorities and methodology thanks to the work carried out in specific Task Force and Working Groups. The progress report of all these cross-cutting actions are Chapters 5, 6 and 7. However, it worth mentioning some main achievements carried out in 2018 and 2019 such as:

- the 4th GIF Symposium held in Paris, France on 16-18 October with about 300 participants, plenary sessions and 8 technical sessions. Proceeding will be published in 2020 and accessible on the GIF website;
- the completion of the SFR SDC/SDG Document;
- the pre-production of a Gen-IV Position paper on Flexibility of GEN-IV Systems produced by the EMWG (final diffusion in 2020);
- the preparation of a joint Workshop with Nuclear Industry on Advanced Manufacturing and R&D Infrastructures needs and opportunities, as a completion of the actions carried out respectively in the Advanced Manufacturing and Materials Engineering (AMME) and the R&D Infrastructure (RDTF) Tasks Force (18-20 Feb. 2020 at OECD/NEA);
- the great success of the GIF Webinar series reaching a total of 36 webinars in 2019. The 2020 program is almost fully planned (one per month).

2018 GIF Symposium Program Cover

For 2020, according to the Program Plan proposed by the New Technical Director, GIF will pursue its efforts towards the following items:

- GIF initiative towards private sector.
- Flexibility of Generation IV systems + position paper on application of heat valorization.
- GIF initiatives towards technical innovations and advanced material and manufacturing.
- Position papers on safety standards for the licensing of advanced reactors.
- Promote education & training trying to enlarge this effort towards knowledge capitalization.

In 2020, for its 20th anniversary, GIF intends to be represented at several key events such as the **World Nuclear Exhibition** (Paris, 8-10 December).

Gilles Rodriguez

Technical Director of the Generation IV Forum

Nobuchika Kawasaki

Policy Director of the Generation IV Forum

Chapter 3. **Country reports**

Australia

As the newest member, Australia is an active and enthusiastic member of the Generation IV International Forum (GIF) and continues to increase its engagement with the activities of the Forum. Australia remains committed to undertaking research and development into the next generation of nuclear reactor technologies to advance the peaceful use of the atom.

Within GIF, this includes ongoing contributions to the VHTR System Steering Committee. Australia's contributions to the VHTR Materials Project Arrangement have now been included in the Materials Project Plan, and the signature of the Project Arrangement is in due course.

Similarly, as an active participant in the MSR provisional System Steering Committee, Australia supports the goal of this group to advance to a System Arrangement, and takes the lead on the MSR Materials and Components Project Arrangement.

Turning to other nuclear news, Australia's new nuclear medicine production facility, ANM, is fully operational. On 24 May 2019, ARPANSA, Australia's nuclear regulator, amended its initial conditional hot commissioning licence to allow ANSTO to commence routine production of molybdenum-99 in the ANM facility for both Australian and International markets.

In parallel, construction has started on ANSTO's SyMo nuclear waste plant that will treat the liquid intermediate waste from the ANM facility using ANSTO Synroc technology. The plant will be the first full-scale implementation of Australia's innovative Synroc technology and is expected to be completed by 2021.

The project to select a site for, and establish, the National Radioactive Waste Management Facility (NRWMF) continues, with further detailed site characterization and community consultation. Two sites have been nominated in South Australia, two in the district of Kimba and one in the district of Hawker. Community engagement is ongoing, with consultative committees established and operating in both areas. Unfortunately, community votes in both districts have been delayed by legal action. The facility will be located only where it is broadly supported. The NRWMF will receive Australia's low-level waste for disposal and will temporarily store Australia's intermediate level waste pending the establishment of a separate ILW disposal facility.

Second 2019 semester, there has been increased governmental interest in the potential role of both uranium mining and nuclear energy in Australia. Within the Federal Government, on 2 August 2019, the Hon. Angus Taylor MP, Minister for Energy and Emissions Reduction, established a Parliamentary inquiry into the prerequisites for nuclear energy in Australia. The inquiry is being conducted by the House of Representatives Standing Committee on the Environment and Energy. The Terms of Reference for the inquiry require that members of the Committee specifically investigate, and report on, the circumstances and prerequisites for any future Australian government's consideration of nuclear energy generation, including small modular reactor technologies. The Committee is expected to report by the end of the 2019 year. A separate examination of the nuclear industry in Australia was conducted by the House of Representatives Standing Committee on Industry, Innovation, Science and Resources in Sept. 2019. There has been similar activity within the State Parliaments of New South Wales and Victoria.

On 6 June 2019, the New South Wales Legislative Council's Standing Committee on State Development established an inquiry to consider a bill to repeal the State's ban on uranium mining and the establishment of nuclear facilities. It is anticipated that this inquiry will conclude in 2020.

On 19 August 2019, the Victorian Legislative Council voted to establish an inquiry to examine the merits in lifting that State's ban on nuclear power, with reference to the benefits of nuclear power in mitigating climate change. The 12-month inquiry will investigate whether nuclear power is feasible and suitable for Victoria in the future, and will consider waste management, health and safety issues, and possible industrial and medical applications.

ANSTO plays a vital role in providing expertise and technical advice to government on all matters related to nuclear science and nuclear technology, including nuclear power. In this capacity, ANSTO was asked and has provided, technical advice related to nuclear power and other fuel cycle activities to both the Federal and State-level inquiries.

Australia will host the Policy and Expert Group meetings in Sydney (May 2020). That means Australia will have hosted every GIF committee on which it have representation since joining in late 2017.

Canada

Nuclear Energy in Canada: There are currently 18 operational nuclear power reactors in Canada and one unit undergoing refurbishment. Today, 15% of Canada's electricity comes from nuclear. In the Province of Ontario, home of 18 of Canada's reactors, approximately 60% of the province's electricity comes from nuclear. The Province of New Brunswick, home of the other operational reactor, approximately 35% of its electricity comes from nuclear.

On 21 October 2019, Canada will hold its federal election. In this context, Canada's non-partisan public service is getting ready to brief new ministers.

Refurbishments in Canada: Refurbishments are the number one priority for Canada's nuclear sector. Ontario Power Generation (OPG) – the largest nuclear operator in Canada – is investing CAD 13 billion dollars in the refurbishment of the Darlington nuclear generating station. The project is progressing ahead of time and presently below budget. The first unit was taken offline in October 2016 for refurbishment, and will be reconnected to the grid in 2020.

Canada's subnational government of Ontario has given OPG permission to proceed with refurbishment of the next unit because of the progress being made one the first unit at Darlington.

At the same time, Bruce Power – operator of the largest operational nuclear station in the world – is proceeding with its plans to refurbish their remaining six units at the Bruce Generating Station, with the first unit due to come off line early in 2020.

Small modular reactors (SMRs): Canada's SMR Roadmap was released in November 2018 and can be found https://smrroadmap.ca. In brief, the development of the Roadmap took a national approach over a ten-month period with extensive engagement with industry, initial dialogues with Indigenous peoples, and expert analysis. The process was driven by four provincial and two territorial governments and interested utilities.

There were six workshops held across Canada involving 55 organizations and over 180 participants. This included three Indigenous engagement sessions, with a commitment to continuing the conversation.

Supporting the development of the Roadmap, five expert working groups were established and supported by 18 organizations. These working groups were focused on: 1) Technology assessments; 2) Regulatory readiness; 3) Waste management regime; 4) Economics and finance; 5) Public and indigenous engagement. Each working group developed a report that feed into the Roadmap. These reports are available to the public and can be found online on the above-mentioned SMR site.

Multiple tracks are being pursued to advance small modular reactors (SMRs) development in Canada, notably, 11 SMR vendors have engaged the CNSC under the optional pre-licensing Vendor Design Review Process, and more than 5 SMR vendors are involved in CNL's "Invitation for SMR Demonstration Projects". Canada's nuclear operators (Ontario Power Generation, NB Power, Bruce Power) are also in various stages of engagement with vendors, ranging from technical advisory boards to commercial partners.

In July 2019, the Canadian Nuclear Safety Commission (CNSC) commenced the environmental assessment (EA) of Global First Power's project proposal, in collaboration with Ultra Safe Nuclear Corp. (USNC) and Ontario Power Generation (OPG), for the demonstration of a Micro Modular Reactor at Chalk River, Ontario. The proposed project includes the site preparation, construction, operation, and decommissioning of a single 15 MWth Micro Modular Reactor (MMR) with an expected 20-year core life at Chalk River Laboratories.

On 1 October 2019, Advanced Reactor Concepts (ARC) Nuclear Canada completed the first phase of CNSC's Vendor Design Review (VDR) and will now move to the second phase of the process. The second phase will involve a more detailed review of the reactor concept and will take between 18 to 24 months. To date, three SMR vendors have moved to the second phase of CNSC's VDR process: ARC, Terrestrial Energy and Global First.

Regulatory update: Eleven vendors have engaged with the Canadian Nuclear Safety Commission (CNSC), Canada's nuclear regulator, on their Vendor Design Review process – with the latest application received in March 2019.

In addition, on March 20, Global First Power submitted an application for a license to prepare site for an SMR on Atomic Energy of Canada Limited's land at Chalk River Laboratories. This is the first step in the formal licensing process. The next step would be for the CNSC to issue a notice of commencement, after which the project description would be made available for public comment as part of an environmental assessment process

At the same time, Canada's experienced nuclear operators are working with SMR vendors to vet potential demonstration projects. The Province of New Brunswick, which is open to hosting a demonstration project, has launched a nuclear research cluster with two SMR vendors, ARC Nuclear and Moltex Energy Canada. OPG has recently started a process to extend a site license that it has available now to host new nuclear reactor projects at its Darlington site.

In May 2019, the Province of Saskatchewan announced that it is considering SMRs as a replacement for its coal-fired generation fleet.

Canada's new Impact Assessment Act came into force on 28 August 2019, overhauling the federal environmental assessment system to better protect the environment, respect Indigenous rights and rebuild public trust in how project decisions are made. The new legislation includes a new impact assessment process and a revised list of activities that will trigger an impact assessment. Key features of the new system include:

- Proactive strategic and regional assessments would evaluate big-picture issues (e.g. climate change, biodiversity, species at risk), the cumulative effects of development and provide context for impact assessments;
- An early planning and engagement phase for all projects would build trust, increase efficiency, improve project design, and give companies certainty about the next steps in the review process;
- Indigenous engagement and partnership throughout the process;
- Increased public participation opportunities;
- Legislated timelines to provide clarity and regulatory certainty; and
- Strengthened monitoring, follow-up, and enforcement.

New nuclear reactor projects will be a designated project and trigger an impact assessment if:

- that activity is located within the licensed boundaries of an existing Class IA nuclear facility and the new reactors have a combined thermal capacity of more than 900 MWth; or;
- that activity is not located within the licensed boundaries of an existing Class IA nuclear facility and the new reactors have a combined thermal capacity of more than 200 MWth.

Previously all nuclear reactors would have been designated projects, regardless of size and location. A new project involving a nuclear reactor not designated a project and does not trigger the new impact assessment process is still subject to the existing environmental assessment process.

Canadian Nuclear Laboratories: Five SMR vendors are participating in Canadian Nuclear Laboratories' process to site SMR demonstration projects at CNL sites. In February 2019, CNL announced that two vendors, Starcore Nuclear and Terrestrial Energy, have been invited to advance to the second stage of their four stage process. The second stage has greater focus on due diligence of their technical and economic merits, financial viability, and safety and security requirements.

Meanwhile, Global First Power and its key partners, OPG and Ultra-Safe Nuclear Corporation, have progressed through CNL's second stage and have been invited to participate in preliminary, non-exclusive discussions on siting with CNL.

GIF update: In May, NRCan endorsed Terrestrial Energy's signing of the Memorandum of Understanding for Collaboration on the Molten Salt Reactor System. In addition, NRCan is supporting CNL's participation as an observer in the MSR provisional System Steering Committee as well the Very-High-Temperature Reactor System Steering Committee.

Moltex Energy Canada is seeking observer status for GIF's Molten Salt Reactor system. Canada is supportive of the request. As such, Moltex Energy Canada's participated in the recent Molten Salt Provisional Steering Committee Meeting.

Canada has renewed its participation in the Supercritical Water-Cooled Reactor Thermal-Hydraulics and Safety Project Arrangement, with Atomic Energy of Canada Limited as signatory and Canadian Nuclear Laboratories as the performing organization. In May 2019, Canada hosted the Policy and Expert Group meetings in Vancouver.

China

Nuclear Energy Policy: China's Nuclear Safety Law strengthens China's industry standards. The legislation includes more than 90 items, went into effect in January. It ensures the appropriate treatment of nuclear materials and facilities, and reduces risks and nuclear waste. It is the legal foundation that clarifies protocols, responsibilities and punishments for various government agencies, businesses and civilians when dealing with nuclear-related subjects. The Atomic Energy Law of the People's Republic of China has been included in the national legislative plan, as a basic law in nuclear field.

China issued a white paper to introduce its approach to nuclear safety on 3 September. The white paper titled "Nuclear Safety in China" has been published by the State Council Information Office to elaborate on China's basic principles and policies in the field, share the concepts and practices of regulation, and clarify its determination to promote global nuclear safety governance and the actions it has taken to achieve this. The document says China has always regarded nuclear safety as an "important national responsibility, and integrated it into the entire process of nuclear energy development and utilization". The industry, it says, has "always developed in line with the latest safety standards and maintained a good safety record, pursuing an innovation-driven path of nuclear safety with Chinese characteristics."

The Nuclear Energy Development: By the end of September, there are 47 nuclear power units in operation, with the total installed capacity of 48.73 GW; 11 nuclear power units are under construction, with the total installed capacity of 12.14 GW.

Cold hydrostatic testing has begun on 27 April at unit 5 of the Fuqing nuclear power plant in China's Fujian province, the first of two demonstration HPR1000 under construction at the site. The tests mark the first time the reactor systems are operated together with the auxiliary systems.

In a first for China, China National Nuclear Corporation (CNNC) has uprated its oldest power reactor, Qinshan 1, to 350 MWe (net) from its original 300 MWe in mid of April. The engineering work "*has important reference significance for the power enhancement of subsequent power stations, and plays an exemplary role in the prolongation management of domestic nuclear power plants*".

Long-term irradiation testing of CF3 pressurized water reactor (PWR) fuel has been completed in March. CF3 fuel assemblies are designed for use in the HPR1000.

Members of the World Association of Nuclear Operators (WANO) have voted to establish a new branch office and support centre in Shanghai, China on 21 February during its General Assembly in South Africa. Over the past 30 years, China has become a key player in the commercial nuclear sector. The decision to proceed with establishing a WANO Branch Office and Support Centre in China has the overwhelming support of the worldwide membership.

CNNC announced the launch of a project to construct an ACP100 small multipurpose modular reactor at Changjiang in Hainan Free Trade Pilot Zone on 18 July. Construction of the demonstration unit-also referred to as the Linglong One design – is scheduled to begin by the end of this year.

Having completed a 168-hour test run, Unit 2 of Taishan Nuclear Power Plant became the world's second European pressurized reactor (EPR) qualified for commercial operation on 7 September.

China has started mass production of fuel assemblies for its first self-developed large-scale advanced pressurized water reactor for commercial use. The assemblies can be used for long-cycle refueling and are suitable for the Hualong One reactor and the Yanlong low temperature heating reactor.

Gen-IV nuclear energy systems activities

SFR: CEFR restarted and was operating at low power since February 2019. Pre-conception design of CFR1200 with SCO_2 system as the candidate of power conversion system is in process, and major research work is focused on thermal-hydraulic, SCO_2/Na reaction and code development of SCO_2 system. SFR is planning to conduct irradiation test in CEFR. The design of CN-1515 irradiation rig has finished. Experimental facility for the research on interaction between sodium and supercritical CO_2 has been constructed. Sodium-supercritical carbon dioxide heat exchanger has been designed and is under manufacturing currently.

VHTR: HTR-PM demonstration project will connect to grid in 2020 in accordance with the current plan. The installation is now in the final stage and commissioning test has started already. The process for joining HP-PMB is undergoing. With the contributions from all members, the project plan for CMVB was finished, and was approved by VHTR SSC, the formal signing process can be started. The R&D in FFC and MAT PMB progress as planned.

SCWR: The R&D on SCWR and pre-conceptual design of the China SCWR CSR1000 is ongoing. The small SCWR named CSR-150 is being developed to meet flexible and wide demands. In terms of co-operation in SCWR, a new international benchmark exercise is just set up based on the SCW parallel pipe density wave instability tests from NPIC for assessing the system analysis code. China has taken part in the TH&S PMB and M&C PMB and work is proceeding according to the project plan. The MOST (Minister of Science and Technology) of China is planning to fund the Chinese universities and institutes in the TH&S and M&C PMBs field.

LFR: The Institute of Nuclear Energy Safety Technology, China Academy of Science has been actively participating in GIF LFR activities as an observer and made its contribution to the development of LFR pSSC since 2013. Considering widely involving of Chinese institutions in R&D of LFR and its importance, China was interested in acceding to GIF LFR. The signature was done in October 2019.

Euratom

Research

The Euratom contribution to the Generation IV International Forum relies on three main pillars: Indirect Actions, which are research projects carried out by research institutions of EU Member States, co-funded with EU budget; direct actions, which are research projects carried out directly by the European Commission's Joint Research Centre, and activities carried out by EU Member State Institutions.

Indirect and direct actions are both defined and funded by the multiannual Euratom Research and Training Programme. The 2019-2020 extension of the Euratom Research and Training Programme (2014-2018), complementing the Horizon 2020 European Research Programme was adopted on 15 October 2018.

The ongoing collaborative projects are progressing steadily and cover Molten Salt Reactors (MSR), Lead-cooled Fast Reactors (LFR), Sodium-cooled Fast Reactors SFR, Very-High-Temperature Gas Reactors (VHTR), Gas-cooled Fast Reactors (GFR), as well as cross-cutting fuel and material topics. Additional direct action contributions to GIF: co-ordination, co-operation with Working Groups (Risk and Safety – RSWG, Proliferation Resistance and Physical Protection – PRPPWG), Task Forces (TF) (Advanced Manufacturing Methods – AMME TF and Research and Development Infrastructure – RD TF), and committee work (SSC, PMB, and others).

The last call for proposals of around EUR 140 million was published in December 2018 and proposals will be selected for financing by February 2020. Funded reactor systems will include advanced nuclear systems for increased safety (Gen-IV), SMRs, Partitioning and Transmutation, and support to the Jules Horowitz Research Reactor currently under construction in France. Overall, 62 proposals were submitted, 15 projects are competing for EUR 40 million whereof 6 projects are expected to be funded in relation to advanced nuclear systems. The winning projects will be launched by mid-2020.

The European Commission has presented the most ambitious framework programme for research and innovation ever. The Horizon Europe (2021-2027) proposed budget of EUR 100 billion also includes EUR 2.4 billion for the Euratom Research and Training Programme and EUR 3.5 billion from the InvestEU Fund (gathering several risk sharing financial instruments). The proposed financial envelope for the implementation of the Euratom Research and Training Programme for the period 2021-25 is EUR 1.6 billion, in current prices, with the following indicative distribution: (a) EUR 724 million for fusion research and development; (b) EUR 330 million for nuclear fission, safety and radiation protection; and (c) EUR 619 million for direct actions undertaken by the Joint Research Centre.

The Joint Research Centre has consolidated its activity in the domain of Generation IV systems in three major projects. SEAT-GEN IV (Safety of Advanced Nuclear Systems and Innovative Fuel cycles) project, SAETEC (System Analysis of Emerging Technologies) and WAIF (Waste from Innovative fuel). The topics covered are: Reactor Safety of Generation IV reactor designs, including modular reactors (safety analysis including severe accident modelling); Materials R&D programme with focus on LFR and Supercritical Water Reactor (SCWR); safety of fuel for the SFR, LFR, VHTR, MSR systems, conditioning matrices for waste from innovative fuels, and safeguards. Activities in support to the GIF PRPP WG are carried out in the project MEDAKNOW (Methods, data analysis and knowledge management for Nuclear Non Proliferation, Safeguards & Security).

Euratom continues to rely on its High Flux Reactor (HFR) in Petten, The Netherlands. The governments of the Netherlands and France have agreed to support the Supplementary Research Programme for the period 2020-2024 with about EUR 30 million, which will be implemented by the European Commission's Joint Research Centre. Relevant for GIF are in particular irradiation tests of fuel and materials as well as the ensuing post-irradiation examinations.

The Nuclear Research and consultancy Group (NRG, The Netherlands) has announced the completion of the SALIENT-01 test, the first irradiation experiment in the MSR research programme that started in 2015. Research under this programme is partly funded by the Dutch Ministry of Economic Affairs and is being carried out in collaboration with the Joint Research Centre (JRC) of the European Commission.

Romanian utility Nuclearelectrica has signed a Memorandum of Understanding (MoU) with FALCON, the consortium constituted by Italian and Romanian entities aiming at the construction of Advanced Lead Fast Reactor European Demonstrator (ALFRED). The MoU addresses the pre-project works and the research and development activities which are to be implemented in order to develop the ALFRED project: exchange of information and data regarding the technology; the co-ordination of the research activities; the in-kind contribution of each party; the studies and analyses independently conducted by each party for their organization; and the planning of the necessary framework for preparing the demonstration activities.

The Sustainable Nuclear Energy Technology Platform (SNETP) is currently updating its Strategic Research and Innovation Agenda. Generation IV (within the European Sustainable Nuclear Industrial Initiative) will continue to be an important pillar and new topics will be introduced such as SMRs. SNETP is divided into three main areas: Current Reactor Research (NUGENIA), Research on fast reactors with closed fuel cycle (ESNII), and Nuclear Co-Generation (N2CI), primarily with high-temperature gas-cooled reactors. The priorities for ESNII are in Sodium Fast Reactor and a specific (ADS – Accelerator-Driven System) of Lead/Bismuth Fast Reactor (MYRRHA) to support the technology development. In this frame, the Belgian Federal Government decided in September 2018 to invest EUR 558 million during the 2019-2038 period in phase 1 of MYRRHA which includes the construction of the 100 MeV MINERVA linear accelerator, Proton Target Facility and Fusion Target Station. Other priorities are the Lead Fast Reactor (ALFRED) in the shorter term and Gas Fast Rector (ALLEGRO) in the longer term. Within the third area NC2I, the Very-High-Temperature Reactor has a special role in cogeneration. The Polish Ministry confirmed full interest at the October 2018 HTR conference in Warsaw (Poland) which was confirmed after elections in November 2019. The Polish HTR Committee for Analysis and Preparation of Conditions for Deployment of High-Temperature Nuclear Reactors recommends beginning preparation of HTGR deployment as the Polish HTR strategy is being written into the Polish Energy Policy.

The Euratom FISA 2019 and EURADWASTE '19 conferences took place in Pitesti, Romania, on 4-7 June 2019. The conferences addressed all Euratom Fission safety research and training, innovative projects of reactor systems and radioactive waste management within the frame of Horizon 2020 framework programme. Overall, some 95 projects co-funded by Euratom (with around EUR 350 million out of EUR 500 million) were presented. A significant number of these projects are contributing to GIF. More than 400 scientists from 200 organizations from 40 European countries and worldwide attended. These conferences provided additional opportunities to address and engage with all relevant stakeholders, to further strengthen the EU's innovation potential, careers' attractiveness to the young generation, the research community, policy makers and civil society.

Policy

On 28 November 2018, the European Commission presented its strategic long-term vision for a prosperous, modern, competitive and climate-neutral economy by 2050 – A Clean Planet for All. The strategy shows how Europe can lead the way to climate neutrality by investing into realistic technological solutions, empowering citizens, and aligning action in key areas such as industrial policy, finance, or research – while ensuring social fairness for a just transition. The strategy envisages that, by 2050, more than 80% of electricity will be coming from renewable energy sources, which – together with a nuclear power share of approximately 15% – will be the backbone of a European low-carbon power system.

In May 2019 European Parliament elections took place in all EU Member States after which a new European Commission was set up. The Ursula von der Leyen Commission for the next five years took office on 1 December 2019, and will focus on six main priorities:

- a European Green Deal;
- an economy that works for people;
- a Europe fit for the digital age;
- promoting our European way of life;
- a stronger Europe in the world; and
- a new push for European democracy.

The European Parliament presented a motion to the COP25: The European Parliament "believes that nuclear energy can play a role in meeting climate objectives because it does not emit greenhouse gases, and can also ensure a significant share of electricity production in Europe; considers nevertheless that, because of the waste it produces, this energy requires a medium and long-term strategy that takes into account technological advances (laser, fusion, etc.) aimed at improving the sustainability of the entire sector". European Parliament resolution of 28 November 2019 on the 2019 UN Climate Change Conference in Madrid, Spain (COP 25) (2019/2712(RSP)).

France

French energy policy: In November 2018, President Macron presented the ten-year Multiannual Energy Plan (PPE), the government released the complete document (January 2019). This steering tool presents the path to be followed in terms of energy policy and ecological transition. The two intricately linked main objectives are to reduce French fossil fuel consumption and to ensure clear, fair and sustainable transition for all. On power generation, the government sticks to the goal of diversifying the energy mix, with the development of renewables. The target of achieving a 50% nuclear power share in the electricity mix is set by 2035 – instead of 2025 as planned in the 2015 Energy transition law. A correlative reactor shutdowns should be progressively decided, subject to conditions related to the electricity market and to the evolution of the electricity system in the neighboring countries. At the same time, the government and the nuclear industry are committed to release a plan by mid-2021 in order to enable a fact-based political decision on coming new reactor construction in France.

The PPE confirms the strategy for nuclear fuel treatment and recycling, at least until the 2040's horizon. To this end, a certain number of 1 300 MW reactors are going to be adapted for the use of MOX fuel, considering the to-be-decided closure of 900 MW reactors, which currently use MOX fuel. Alongside this adaptation of a part of the existing fleet, the French nuclear industry and CEA have launched the feasibility study of nuclear fuel multi-recycling in PWR. This option is considered as a possible intermediate step.

Thus this new law will integrate the revised objective to reduce the share of nuclear power to 50% in 2035, as well as a new objective for France to reach net carbon neutrality by 2050. It also implements new tools for monitoring, governance and evaluation of the climate policy, such as the introduction of a five-year plan setting out the main energy policy objectives.

Regarding radioactive materials and waste management, Emmanuelle Wargon, French Government Secretary to the Minister of Ecological Transition, signed on 4 October the territory project that will support the construction of CIGEO, the geological storage centre for radioactive waste. A wide national debate on the National Plan for the Management of Radioactive Materials and Waste (PNGMDR) took place in 2019, underlining namely the importance of the recycling of nuclear materials.

Status of the French R&D program on closed fuel cycle: The Multiannual Energy Plan confirms the long-term sustainability objective of a fully closed fuel cycle, implying the deployment of fast reactors in the power generation mix. However, the implementation time frame to meet French power generation needs has been reassessed through a review of fast neutron reactors and fuel cycle strategy jointly conducted by CEA and French nuclear industry. The conclusion was jointly reached that the industrial deployment of fast reactors is very likely to be more remote in time and that the option must be kept open. Thus with regard to the updated deployment timeline, the CEA and the nuclear companies have proposed to postpone the ASTRID project. The Government has endorsed this position, as expressed in the Multiannual Energy Plan. But keeping the option open requires maintaining skills, progressing on technological barriers and further developing knowhow. Based on the results and knowledge that stemmed from the ASTRID program, the challenges to be dealt with in order to prepare the future sodium-cooled fast neutron reactor development have been listed by CEA. Consistently with the revised timeline of fast reactor commercial deployment, the R&D roadmap towards a fully closed fuel cycle has been established in continuity with the ASTRID program. It is mainly dedicated to SFR – which remains the most mature technology and the reference option. It involves basic research, modelling, numerical simulation, technological development and experiments, benefiting from useful innovative methodologies (advanced manufacturing, massive data processing, digital design). It also seeks for reactor design innovations. Evaluation of other fast reactor technologies and systems is also a part of the program, with very preliminary design studies. France has now a revised R&D roadmap to implement and support its energy policy for low CO_2 energy system including nuclear energy and renewables.

Other projects: As for the short-term future, considering the need for a decarbonized baseload offer in small and medium power segments internationally, the French nuclear industry leads a light water SMR development project, currently starting the basic design phase. This project (NUWARD) is taking place through a consortium made of CEA, EDF, Technicatome and Naval Group. Consequently, during the IAEA General Conference, the French joint industrial initiative for the development of a small modular reactor (SMR) was unveiled. The NUWARD-TM project is a PWR technology-based solution, designed to meet the growing need of low-carbon electricity market worldwide in the 300-400 MWe power segment. In addition to that, CEA envisions to explore other SMR-concepts both for power generation and non-electrical application purpose, in the framework of an integrated approach of energy systems.

Jules Horowitz Reactor: Significant progress have been made with reactor components manufacturing of the future international user facility JHR. The project has gone through a thorough governmental review process and a set of decisions are going to be implemented to secure the revised overall schedule.

In May, after a high-level review, the French government reinforced the significance of the JHR as a key tool for the nuclear sector, not only for France but also internationally. The reactor has entered a new phase, dedicated to the installation of major components. On-site, the completion of the reactor pool liner was reached in August. This major milestone paved the way to the ongoing reactor pile-block implementation and to the coming installation of other key components such as the heat exchangers of the primary circuit.

EDF nuclear new build activities in France: Progress are being made for the commissioning of Flamanville with the start of hot tests in February 2019. In parallel, the safety authority should deliver its ruling regarding how to proceed further with the deviations detected in welds on the main steam transfer pipes. At the end of June, the ASN announced the discovery of eight non-conforming welds in hard-to-reach areas on the four steam evacuation pipes that need to be taken over. EDF has recently unveiled how it will proceed to correct these welds, which will take a few months. The start-up of the reactor is consequently planned for 2022.

CEA energy integrated approach: The CEA is highly committed on developing an integrated approach of the energy system as a whole, taking into account all the energy needs. This renewed approach considers the increasing penetration of intermittent renewable energies and the emergence of new electrical uses, and the subsequent need for energy storage and conversion systems development. This will lead to a multi-vector and multi-network energy system. It also requires optimization by developing closed material cycles, and a strong digital monitoring (smart grids). The integrated approach of the comprehensive energy system, either at the global or at the local scales, is also taking into account the capacities to meet the future needs and their mutual articulation, in order both to advise the French government and to contribute to the technological developments.

Japan

Current Status of Nuclear Policy of Japan: In December 2016, the Cabinet of Japan approved the policy on the fast reactor development. Based on this, the government built in December 2018 the Strategic Roadmap which determines development activities in the coming decade. This Roadmap highlights the significance of nuclear fuel cycle technologies for Japan, focusing on the efficient use of resources, as well as the minimization of the volume of high-level radioactive waste and its potential toxicity. As part of it, JAEA has been making efforts for the fast reactor development.

Ensuring the highest level of safety is top priority. Japan will also aim to reduce the generation cost of nuclear power. To achieve this, flexible approaches are necessary in deal with uncertainties in the future, including cost competitiveness with other energy sources and the social environment. The Roadmap specifies the roles of driving forces who lead the nuclear development, namely, the government, electric utilities, JAEA, and manufacturers. In addition, it declares that Japan will co-operate with other countries by, in particular, efficiently using the network of the GIF to further advance Japan's technology platform and innovation. The Advisory Committee for Natural Resources and Energy of Ministry of Economy, Trade and Industry of Japan discussed how to further facilitate the nuclear innovation in April 2019. Japan has already started government-funded projects towards it. Thus, movements that encourage nuclear research, development, and demonstration are increasing in private sectors in our country.

Another international co-operation ongoing is a project with Poland. The fifth Strategic Energy Plan approved by the Cabinet in July 2018 states that "*While watching global market trends, Japan, in co-operation with other countries, will further develop technologies that contribute to improving nuclear safety, such as the high-temperature gas-cooled reactor (HTGR) which features its safety*". In January 2019, JAEA jointly held a seminar with the National Centre for Nuclear Research of the Republic of Poland on the technology of the HTGR.

Current Status of Fukushima Daiichi Nuclear Power Station: In Fukushima, the damaged reactors are in cold shutdown status. Based on the Mid-and-Long-Term Roadmap for the plant, Tokyo Electric Power Company (TEPCO) has been inspecting the inside of the cores and developing the procedure towards the fuel debris retrieval scheduled in 2021 and the decommissioning of the plant. TEPCO removed spent fuel from the pool in Unit 4 in 2014, and started removing fuel from Unit 3 in April 2019.

Safety Review of Nuclear Power Stations and Nuclear Fuel Cycle Facilities by Nuclear Regulation Authority (NRA): Among nuclear power plants in Japan, 27 units of 16 plants, in total, applied for the conformity assessment of the Nuclear Regulation Authority (NRA) to restart their operations. As a result, the NRA granted the permission for 15 units of 8 plants to alter their installations. As of today, nine units are in operation.

Current Situation of Facilities of Japan Atomic Energy Agency (JAEA): JAEA is working on the restart of High Temperature Engineering Test Reactor (HTTR), and the experimental fast reactor JOYO. The conformity assessment of the upgrading of HTTR's installations is in the final stage. Regarding JOYO, JAEA submitted the amendment of the facility upgrading to the NRA in October 2018, and is waiting for the result.

Russia

Nuclear Power in Russia: In 2018, Russian NPPs produced 204.3 billion kWh that is 18.7% of the total electricity production. The increase in power production was 0.7% as compared to 2017, with the load factor equal to 80%. Currently, 36 power units with total electrical capacity of all Russian nuclear power plants is 28.9 GW. At the same time, the share of nuclear generation in the country's total energy balance is 18.6% of the total electricity production in Russia. In 2018, 2 power units were put into commercial operation: power unit 4 of the Rostov NPP with the WWER-1000 reactor and power unit 1 of the Leningrad NPP-2 with the WWER-1200 reactor of GEN 3+, which safety is justified taking into account the lessons of the accident at the Fukushima-1 NPP. Power unit 2 of the Novovoronezh NPP-2 also with the WWER-1200 reactor was put into operation on 1 May 2019. Two power units were decommissioned: power unit 1 of the Bilibino NPP with the EGP-6 reactor and power unit 1 of the Leningrad NPP with the RBMK-1000 reactor.

In November 2018, the first criticality was reached for both reactors of the floating nuclear power plant "Academician Lomonosov". In August-September 2019, it is planned to transport the floating NPP to Pevek in Chukotka Region and to provide its putting into operation in December 2019. The deployment of the FNPP in Pevek will create conditions for accelerated social and economic development of neighboring regions and Chukotka as a whole. In addition, it will become one of the key elements of the infrastructure of Northern Sea Route. An extensive program for the construction of an icebreaker fleet, including nuclear power plants, has been adopted, which will ensure reliable year-round operation of the Northern sea route. Delivery of the nuclear icebreaker "Arctic", which is the head in the new series of nuclear icebreakers of the project "Leader", planned to May 2020 after additional tests of the steam power plant, respectively, other representatives of the project – icebreakers "Siberia" and "Ural", are planned to pass in 2021 and 2022.

Today, Russia is a leader in new nuclear construction abroad. Rosatom ranks first in the number of simultaneously implemented projects for the construction of nuclear reactors (6 in Russia and 36 abroad). The following projects are mentioned here, they moved to the practical implementation stage:

- 4-unit Akkuyu NPP in Turkey with WWER-1200 reactors;
- power units 3 and 4 of the second phase of the Kudankulam NPP and power units 5 and 6 of the third phase of the Kudankulam NPP in India with WWER-1000 reactors;
- 4-unit El-Dabaa NPP in Egypt with WWER-1200 reactors;
- 2-unit Ruppur NPP in Bangladesh with WWER-1200 reactors;
- power units 7 and 8 of the Tianwan NPP in China with WWER-1200 reactors;
- 2-unit Belarusian NPP of the "NPP-2006" type.

The competitiveness of Russian proposals can be explained by advanced and the latest technologies developed by Russian scientists and designers. The projects proposed for construction are based on modern reactor facilities of a modernized design of VVER (Russian light water power reactor under pressure), which have long-term good performance indicators. The construction projects of the Russian nuclear power plant are Generation III+ reactors equipped with active and passive safety systems. All design projects comply with current international requirements and IAEA recommendations.

Improvement of VVER technology is necessary for the transition from an open to a closed fuel cycle. The innovative development of VVER projects includes: reduction of capital and operating costs, taking into account the experience gained in construction and licensing; ensuring competitiveness in the domestic and foreign markets; compliance with the achieved level of security; providing the ability to work in the conditions of the short and medium-term nuclear strategy (combination of an open and closed nuclear fuel cycle); fuel tolerance development program. VVER-1200 is the flagship nuclear reactor and the main product of the integrated solution of Rosatom State Corporation. As a development of the VVER-1000 reactors, which were recently built in Iran, India and China, the new design has improved characteristics in all respects.

Perspective nuclear technologies: Russia is a recognized leader in the field of fast sodium reactors (FSR). At present, two power units of the Beloyarsk NPP with BN-600 and BN-800 reactors, as well as the BOR-60 research reactor in NIIAR, Dimitrovgrad, are in operation. The total operational experience of the FSR, accumulated in Russia, exceeds 158 reactor years as of

September 2019. Next year, the life of a power unit with a BN-600 reactor will reach 40 years. The task of BN-800 is to demonstrate the possibility of a closed fuel cycle, improve the technology of fast neutron reactors, and also test new design solutions for machines and reactors designed to increase its economic efficiency, reliability and safety. BN-800 can operate on uranium or mixed uranium-plutonium fuel. The use of MOX fuel helps to dispose of weapons-grade plutonium and burn long-lived radioactive isotopes (actinides) from irradiated fuel from thermal reactors. The initial fuel load of the BN-800 reactor was formed mainly from traditional uranium oxide fuel. At the same time, part of the fuel assemblies contains MOX fuel manufactured at the pilot plants of other Rosatom enterprises – NIIAR (Dimitrovgrad, Ulyanovsk region) and Mayak Production Association (Ozersk, Chelyabinsk region). Currently, the second batch of industrial fuel assemblies based on MOX fuel produced by the mining and chemical plant (Zheleznogorsk, Krasnoyarsk territory) has passed acceptance tests.

In the framework of the project “Proriv” on the BN-600 reactor tests are conducted, and subsequent studies of the mixed nitride uranium-plutonium fuel (MNUP-fuel) production to the Siberian chemical combine, which plan to use in the project BREST-OD-300 and BN-1200. As of now, 18 experimental fuel subassemblies with more than 1 000 fuel elements of various types are under irradiation. For 11 experimental fuel subassemblies, irradiation studies already completed, the maximum burn-up level has reached 7.5% of heavy atoms.

The development of two projects which meets the requirements for the 4th generation reactor systems is continuing: the BREST-OD-300 Lead Fast Reactor – the construction is planned to start on 2020 at the site of the Siberian chemical plant in Tomsk –, and the commercial power unit BN-1200 fast sodium reactor.

At the NIIAR site in Dimitrovgrad, the MBIR research fast reactor with sodium coolant is being constructed, which is intended to replace the BOR-60 reactor, which has been operating for almost 50 years. Russia organized an international research center on the basis of the MBIR reactor, and now the process of legalizing the partnership relationship is ongoing. Key areas of research for the project IRC MBIR are: materials (new fuels, structural materials and coolants, verification data), safety (the rationale for the new security system, studies under transient and abnormal conditions), physical examination (study on closed nuclear fuel cycle, reprocessing of minor actinides and other long-lived radionuclides, verification codes), and endurance tests (fuel, elements of CPS and the active zone, system monitoring and cooling circuits diagnostics).

The transition to the closed nuclear fuel cycle in the transition period allows to stop the rate of accumulation of spent nuclear fuel (SNF) of thermal reactors and its increasing cost of handling. Replacing one thermal reactor with a Fast Reactor prevents the formation of about 1 000 tons of spent fuel during VVER operation for its design lifetime period of 60 years and the cost of its storage until reprocessing; increases by ~15 times the yield of a commercial product – Pu during processing (15% Pu in SNF FR). The use of reprocessing products is an effective way to solve the problem of already accumulated VVER SNF. One new FR can utilize all SNF during the life of one VVER; replacing 10 GW of thermal reactors with fast ones almost completely solves the problem of accumulated Russian VVER SNF (~10 thousand tons), and also ensures the economic result of its reprocessing.

Generation IV update: In 2018, Rosatom has signed the GIF Project Arrangement on SFR Advanced Fuel, and early in this year it agreed extension of the GIF Project Arrangement on SFR Safety and Operation for the next ten years. As part of revision of the System Research Plan on Sodium Fast Reactor, it was added with the BN-1200 concept as a design track meeting the Gen-IV requirements. In addition, Rosatom is actively participating in preparation to signing the GIF System Arrangement on Molten Salt Reactor.

In 2018 – early in 2019, representatives of Rosatom have delivered lecturers at GIF webinars on topics:

- Molten Salt Actinide Recycler & Transforming System (MOSART) with and without TH-U support;
- BN-600 and BN-800 Operating Experience;
- Scientific and Technical Problems of Closed Nuclear Fuel Cycle in Two-Component Nuclear Energetics.

In 2019, Rosatom agreed to extend for the next ten years the validity of the GIF Project Agreement on Safety and Operation of FNR.

South Africa

Energy planning and the nuclear program: After substantive and extensive stakeholder consultation and engagement processes, South Africa's electricity generation master plan, the Integrated Resource Plan was tabled before and approved by Cabinet in October 2019. The approved 2019 Integrated Resource Plan (IRP) calls out for preparations to commence on the 2 500 MW nuclear programme, in particular "*Decision 8: Commence preparations for a nuclear build programme to the extent of 2500 MW at a pace and scale that the country can afford because it is a no-regret option in the long term*". The IRP proposes that the nuclear power programme must be implemented at an affordable pace and modular scale (as opposed to a fleet approach) and taking into account technological developments in the nuclear space. The IRP further advocates for energy system requirements with incremental capacity addition (modular) and flexible technology, to complement the existing installed inflexible capacity. In addition, lessons learnt from the procurement under the Independent Power Producer (IPP) programme has shown that there is a business case for modular and smaller power plants (300 MW and 600 MW) hence spelling clearly South Africa's stance to deploy small modular reactors.

Legislative and policy developments

a. New legislation and policies: The draft position paper on the Decommissioning Policy was developed and currently subjected to stakeholder consultation. This policy is necessary to guide the decommissioning of aging infrastructure including major nuclear installations such as the Koeberg Nuclear Power Station and the SAFARI-1 research reactor as well as future nuclear installations.

The development of a national policy and strategy on nuclear research, development and innovation is work in progress. The R&D policy and strategy is envisaged to span areas of power and non-power applications of nuclear energy.

The safe and long-term management of radioactive waste and spent fuel is pivotal and as such, South Africa is developing a Fund Bill for radioactive waste and spent fuel management. The Draft Bill is subjected to consultation with key stakeholders and in parallel, a socio-economic impact assessment of the Bill is undertaken. Radioactive Waste Management Fund Bill is informed by the polluter pays principle where levies and taxes will be collected from the operators of nuclear installations and facilities to fund management of radioactive waste.

b. Legislative amendments: South Africa also continues with the review and amendments to the National Nuclear Regulator Act to among others strengthen nuclear security, enhance on regulation of radioactive sources and ensure effective independence of the nuclear safety regulator in light of lessons learnt from the Fukushima Daichii accident. The Draft Amendment Bill is subjected to stakeholder consultation and in parallel a socio-economic impact assessment undertaken.

Aging management and plant life extension: The South African Nuclear Energy Corporation (Necsa) continues to implement an aging management programme for the SAFARI-1 research reactor in line with the IAEA SSG-10 Safety Guide on Aging Management for Research Reactors. SAFARI-1 research reactor has an excellent record of accomplishment of operational safety and ranks among the world's highly utilized and available research reactors. Aging Management for SAFARI-1 reactor also continues and in parallel, the Minister of Mineral Resources and Energy in April 2019 commissioned a Task Team to oversee the delivery of the Multi-Purpose Reactor Project, a replacement for the SAFARI-1 Research Reactor. The Task Team is expected to complete the Project Initiation Report for Cabinet consideration by April 2020.

The twin-unit Koeberg Nuclear Power Station continue to implement plant life extension programme. The plan is to extend the life of the plants from an original design lifetime of 40 years to 60 years. The Koeberg Plant Life Extension projects includes Steam Generator

Replacement, Thermal Power Upgrade, Reactor Pressure Vessel Head Replacement, and Refueling Water Tank Replacement. As per regulatory requirements, Koeberg plans to submit a Safety Case for long-term operation to the Nuclear Safety Operator in 2022.

Nuclear safety and operation: NTP Radioisotopes, a subsidiary of the South African Nuclear Energy Corporation has been operating intermittently since the issuance of directives by the National Nuclear Regulator ceasing production operations at the subsidiary following repeated deviations from safety protocols. The cessation of operations at NTP Radioisotopes is taken in serious light due to the adverse impact on the medical fraternity relying heavily on nuclear medicine for cancer diagnosis and therapy; however, safety remains an overriding factor for the nuclear industry. NTP Radioisotopes continues to work closely with the National Nuclear Regulator to address safety concerns, comply with regulatory requirements and ensure uninterrupted supply of molybdenum-99.

Spent fuel management: Work continues on the establishment of the Centralised Interim Storage Facility for Spent Nuclear Fuel to mainly address storage capacity challenges faced by Koeberg and also in line with international best practice to have away from reactor storage of spent nuclear fuel. The Ministerial Task Team overseeing this project is led by the Department of Mineral Resources and Energy for implementation by the National Radioactive Waste Disposal Institute upon Cabinet approval.

Research & development: Eskom continues with research and development for the Advanced High Temperature Reactor towards a “Proof of Concept” machine to demonstrate a set of technical aspects prior to commercialization.

During early 2019, the Department of Mineral Resources and Energy established an inter-departmental task team to oversee and co-ordinate the delivery of the Multi-Purpose Reactor project aimed as a replacement reactor for SAFARI-1. This is mainly to continue nurture nuclear research, development and innovation as well as sustain radioisotope production.

In addition, South Africa undertakes research and development in non-power applications of nuclear under the IAEA Technical Cooperation Project and the African Regional Cooperation Agreement for Research, Development and Training related to nuclear science and technology.

Korea

Nuclear power in Korea: 25 nuclear power plants are being operated in Korea by July 2019. The nuclear power plants generated 11 678 GWh of electricity, which is responsible for 23.5% of the total electricity production in Korea. The generating capacity of the 25 plants accounts for 18.04% (21 850 MWe) of the total generating capacity. Four nuclear power plants, Shin-Hanul units 1 & 2 and Shin-Kori units 5 & 6, are under construction and will be completed by 2020 and 2024 each of two.

Shin-Kori unit 4 (APR1400) obtained an operation permit on 1 February 2019 and started commercial operation in August. APR1400 is designed by Korean state-owned companies Korea Electric Power Corp. (KEPCO) and Korea Hydro and Nuclear Power Co. (KHNP). The US Nuclear Regulatory Commission (NRC) has issued key safety and design approvals for the APR1400. APR1400 recently received DC (Design Certificate) from the US NRC in September 2019 and registered on 10 CFR Part 52 subpart B. Meanwhile, the first of four APR1400 reactors at the Barakah site in the United Arab Emirates (UAE) was completed in March 2018 and unit 1 is preparing for a fuel loading into the initial core in February 2020.

Nuclear Energy Policy in Korea: An energy transition policy was announced in October 2017 that implies lowering the share of coal and nuclear energy in Korea. The new policy includes shut down of aged coal power plants over 30 years and the expansion of the share of renewable energy to 20% in total electricity generation by 2030. At the end of 2018, a strategy to enhance future safety technologies was newly established. This strategy focuses on expanding investment and developing new technology to ensure the safety of nuclear facilities and spent fuel management. This strategy means that Korea still maintains activities in promoting international collaboration for peaceful and safe uses of nuclear science and technology. The

Korean government actively supports the transfer of domestic nuclear technology to other countries in accordance with the global non-proliferation framework. The exporting nuclear technology includes advanced power reactors, small modular reactors (SMRs), and other diverse applications.

The construction of a new research reactor, Gijang Research Reactor, was approved by the Nuclear Safety and Security Commission (NSSC) on 10 May 2019. This research reactor will responsible for producing radioactive isotopes for medical and industrial purposes, and providing for R&D platform. The construction of the research reactor unit in Gijang, about 450 kilometers southeast to Seoul, will be completed by 2024, with further cost assessment in accordance with the decision of the Nuclear Safety and Security Commission. The reactor will be the first of its kind in having a fission molybdenum (Mo-99) production facility.

In May 2019, the spent nuclear fuel (SNF) management policy re-examination commission has been launched to review the previous national policy (submitted in 2016). The commission will submit policy recommendations to the government on the management of SNF including the construction of intermediate storage and final disposal.

R&D on nuclear energy system in Korea: In 1997, the Korean government established the Comprehensive Nuclear Energy Promotion Plan (CNEPP), which includes the national policy on nuclear energy utilization and promotion and its sectoral tasks. As a part of the plan, a national nuclear R&D plan has been formulated every five years since 1997. The national nuclear R&D plan from 2017 to 2021 was set up with the vision of the advancement on nuclear technology development for reassuring people and the goal of nuclear safety enhancement and core technology completion. It focuses on five research fields: 1) nuclear safety; 2) radioactive waste management; 3) advanced reactors and fuel; 4) application of radiation and radioisotopes; and 5) fundamental technologies. A technology innovation project for operating nuclear power plants has also been developed for the nuclear industry.

Future nuclear technology development strategy was established to support for the R&D part of the Energy Transition Policy and expand the socio-economic application of nuclear technology. Five specific R&D strategies were suggested for successful achievement: 1) Secure plant safety and decommissioning technology; 2) Expand use of nuclear and radiation technology; 3) Overseas export promotion; 4) Secure new future engine sources such as fusion energy; 5) Commercialization of nuclear technology. In line with this future nuclear technology strategy, the Ministry of Science and ICT established a Strategy for the Strengthening Future Nuclear Safety Capabilities at the end of 2018.

Under the energy transition policy (lowering the share of coal and nuclear energy gradually, and expand the use of renewable energy), it is the most important to secure the safety of operating nuclear power plants that will be run at least for the next 60 years. The strategy also presents the direction in which the accumulated nuclear capabilities in the power sector can be expanded to securing nuclear safety and technology innovation. Based on this change of direction, the strategy for the strengthening future nuclear safety capabilities promotes three development strategies: 1) Support for the safe operation of domestic NPPs for the next 60 years; 2) Expanded utilization of safety based technology capability; 3) Securing and spreading innovative capability of future nuclear safety technology and establishment of foundation on sustainable safety innovation.

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

Sodium-cooled Fast Reactor (SFR): The long-term development plan for the future nuclear energy systems was authorized by the Korean Atomic Energy Commission in 2008 and updated by Korea Atomic Energy Promotion Council in 2011; it includes a construction of a prototype SFR by 2028 for demonstration of TRU transmutation technologies. The national project to develop the Prototype Gen-IV sodium-cooled fast reactor (PGSFR) was initiated to achieve the national mission stated above in 2012. For this, the SFR Development Agency dedicated to the PGSFR development was established in the middle of 2012. KAERI is in charge of the design and the validation of the nuclear steam supply system (NSSS) and fuel development, and domestic participants were responsible for balance of the plant system design. Argonne National Laboratory (ANL) supported KAERI with their experiences in SFR development through international co-operation programs.

The electric power of PGSFR was determined to be 150 MWe suitable for technology demonstration and can be classified as a small modular reactor (SMR), and can be developed as a new non-light water reactor SMR in the near future. The first design phase of the PGSFR was done at the end of 2015 by issuing a preliminary safety information document (PSID). The second phase of the development was done at the end of 2017 by issuing the specific design safety analysis report (SDSAR) with design documents and safety analyses results sufficient for assessing safety of PGSFR. Ten Topical Reports for keys technical issue such as major design codes and methodologies were also published at the end of 2017 and submitted to regulatory body in 2018. All of the basic design concepts of structures, systems and components were determined and incorporated into the specific design safety analysis report (basic design requirements, system descriptions, results of safety analysis for postulated accident scenarios).

To support and demonstrate the safety performance of the PGSFR, verification and validation activities are being performed in parallel with the design progress. A large-scale sodium thermal-hydraulic test program called STELLA is being progressed in 2016. First the sodium component tests of the PDHRS (STELLA-1) has been completed, the data obtained from which are to be used for validating computer codes for thermal sizing, and system transient analysis. As the second step, an integral effect test loop (STELLA-2) has been started to demonstrate the plant safety and to support the PGSFR design certification. The construction of STELLA-2 facility is scheduled by the end of 2019 and the demonstration of the integral effect test will be completed in the middle of 2020.

Various R&D activities are being performed, including verification and validation of computational codes and development of the metal fuel fabrication technology. The reactor mock-up physics experiment in the BFS facility was completed in 2015 in collaboration with Institute of Physics and Power Engineering (IPPE) in Russia. The irradiation test of advanced cladding material (FC92) and test fuel was started at the BOR-60 experimental fast reactor.

In 2017, it was decided to suspend the design intensification of PGSFR in consideration of the national energy environment and select a new policy direction after 2020. The new SFR development program will be decided by reassessing future schedules and discussing rational directions based on the research outcomes so far obtained. Accordingly, Korean SFR developments focus on further improvements of strategic key technologies, the construction and validation of the STELLA-2 facility, and the development of the licensing environment through the review of topical reports.

Very-High-Temperature Gas-cooled Reactor (VHTR): In preparation for the advent of the hydrogen society, research on the nuclear hydrogen key technologies using VHTR has been developing with the government support. Key technology developments for VHTR performance improvement have been performing since 2017. The purpose is to improve the level of key technologies to support high temperature nuclear cogeneration system. The key technologies are the design analysis codes, thermo-fluid experiments, TRISO fuel (tri-structural isotropic), high-temperature materials database, and high temperature heat applications. These technologies are related to GIF VHTR projects such as Fuel and Fuel Cycle (FFC), Hydrogen Production (HP), Materials (MAT), and CMVB (Computational Methods Validation and Benchmark). KAERI signed the extension of FFC and HP Project Arrangements. KAERI will also participate in the CMVB project.

In the fuel research, ZrC/SiC coating technology is under development in order to improve TRISO fuel performance. Inner ZrC layer will have an effect on protecting SiC layer from Palladium attacks in high temperature. As GIF collaboration, a round robin leach-burn-leach test to validate the detection technology of defected TRISO fuel particles is almost finished and the resulting data from KAERI was delivered to Idaho National Laboratory (INL).

Research on high-temperature heat utilization is performed. It is focused on a cogeneration technology that VHTR system is coupled to both hydrogen production system and electricity generation system to maximize heat utilization. Hydrogen and electricity production costs and economics are evaluated for each combination of reactor outlet temperatures and three different hydrogen production methods: S-I thermo-chemical process, high-temperature steam electrolysis, and steam-methane reforming process.

In high temperature materials, the research is focused on securing data on nuclear-grade graphite, high temperature metal materials and high-temperature composite materials. In 2019, tests for compression strength of nuclear graphite is conducting in the high temperature up to 1 400°C. For the high temperature metal, mechanical and creep properties of thermally aged high nickel alloy (Alloy 617) and creep properties of the weld metals (Alloy 800H) has been investigated. Most of these data will be contributed to the development of GIF VHTR materials database.

KAERI has performed the development of VHTR design analysis codes and its validation and improvement. A hybrid RCCS (Reactor Cavity Cooling System) test facility has been built to simulate the safety of a hybrid RCCS concept developed by KAERI. Several tests have been carried out to verify this concept. It will contribute to the CMVB project for thermo-fluid system code validation.

The Korean government announced its plan for hydrogen economy which focus on two axes of hydrogen powered vehicles and hydrogen fuel cell in early 2019. The plan increases the supply of hydrogen vehicles by 6.2 million units in 2040 and the number of charging station to 1 200. The plan also boosts the supply of fuel cells and the capacity of fuel cell batteries will be 17.1 GW in 2040. The required hydrogen in 2040 is expected to reach 5.26 million tons in a year. In order to support and realize the hydrogen economy plan, the government has launched a joint private-public committee to draw a roadmap for hydrogen technology development. Nuclear hydrogen production using VHTR is reviewed as one of green hydrogen production technologies but decision has not been made yet. Regardless of the roadmap to hydrogen technology development, VHTR R&D will continue to focus on technologies needed to realize the core outlet temperature of 950°C for economical hydrogen production.

Switzerland

GIF activities: Activities for GIF are ongoing as planned. Switzerland organized the 26th GIF VHTR System Steering Committee (April 2019). The main contribution of Switzerland to the VHTR system is on material side. The materials of interest are metals and ceramics. A new study was started recently regarding the additive manufacturing of ODS. The microstructural investigations of the so produced samples are ongoing and will be followed by micromechanical testing.

M. Pouchon presented new results on “Oxide dispersion strengthened steels via additive manufacturing” at the 27th GIF VHTR System Steering Committee meeting beginning of October with a detailed characterization of the strengthening particles.

Politics and regulation: The discussion about the implementation of the energy strategy plan 2050, incorporating the phase out of the running reactors, are still ongoing. The strategic plan for energy research (2021-2024) is under discussion at the government level. The conservation of nuclear competence should be included as a priority.

The draft version of the Swiss strategic plan for energy research (2021-2024) has been published. The relevance of the nuclear plants for helping a smooth transition to a zero emission energy production is stated in the paper. The need to conserve nuclear competences in Switzerland is also clearly recognized.

Operation of the Swiss nuclear power plants and waste management: All units are in operation with KKL (BWR) still running with limited Power (about 92%) due to unexpected CRUD formation (CRUD for corrosion and wear products (rust particles, etc.) that become radioactive (i.e. activated) when exposed to radiation) on some fuel elements. Post-Irradiation Examinations (PIE) and theoretical analyses are still ongoing in order to better understand the root cause of this very local CRUD formation.

The preparation for the definitive shut down of the Mühleberg reactor (BWR) end of 2019 are ongoing according to plan. The regulator has approved the shutdown and decommissioning plan. Its implementation should start soon after the definite shut down of reactor operation.

The process to find the best site for a deep geological waste repository is ongoing according to plan. Nagra, the company in charge of realizing the final repository for nuclear waste in Switzerland has started deep drillings to acquire detailed information on the geology of the three possible locations for a geological waste repository. These extensive studies will allow the final choice for the location of the site and support the safety analysis.

Nuclear power related research in Switzerland: The focus of the NES division is to deliver a strong contribution to the education of the next generation of nuclear experts, the scientific support for the safe operation of light water reactors (LWRs), the delivery of the scientific basis for the assessment of the deep geological repositories safety and the technology monitoring including research work on Gen-IV concepts.

The financing of a Professorship on Nuclear Engineering at the Polytechnic School of Zürich has been finalized. This insures the further teaching of nuclear engineering at ETHZ after the retirement of Professor M. Prasser. The search for candidates is ongoing. Two professor positions and laboratory heads in the division are also open (Laboratory for simulation and modelling/Laboratory for system analysis). Interview of candidates for the three open Professor positions in the Nuclear Energy and Safety division at PSI are ongoing. The final selection and nomination is expected for the end of the year or beginning 2020.

On June 2019, the EPFL Laboratory of Reactor Physics and Systems Behavior was officially designated as a Collaborating Centre of the International Atomic Energy Agency (IAEA) in the fields of open-source data and code development for nuclear applications. This is the second Swiss collaborating center after the Spiez Laboratory.

United Kingdom

Nuclear energy: Nuclear energy continues to be one of UK's largest low-carbon energy sources, producing around 10% of primary energy and around one fifth of the UK's electricity. The amount of nuclear generation capacity is expected to decrease in the 2020s, as the majority of existing nuclear power stations reach the end of their operational lives. One PWR power station (Sizewell B) has a projected end of life beyond the 2020s and one new plant is currently under construction developed (Hinkley Point C). A rapid recent rise in other solar photovoltaics (PV) and wind power maintains, along with nuclear energy, a significant amount of low-carbon electricity generation on the UK's grid.

The UK has set into law a move to zero net emissions by 2050, this government legislative commitment to zero carbon is the priority policy driver and along with the recent rise in UK solar photovoltaics (PV) and wind power, the UK is planning for a future of significant amount of low-carbon energy. Part of this future energy mix requires replacement of existing nuclear power plant with other Generation III systems and as part of this, the UK has launched a consultation on the use of a new financial model, the Regulated Asset Base, as a way of financing new power plants. This approach has already been highly successful in other large infrastructure projects and is now being investigated for use in nuclear construction projects.

GIF Framework Agreement: In October 2018, the UK submitted its instrument of ratification for the Generation IV International Forum (GIF) Framework Agreement for International Collaboration on Research and Development of Generation IV Nuclear Energy Systems. UK participation in GIF R&D activities has started in 2019.

Following the accession in 2019 of the UK to the Generation IV International Forum (GIF) Framework Agreement, the UK has initially engaged in a minimum of two systems, the SFR and HTGR systems, UK appointments to these systems and programme arrangements have been made, formal agreement from the groups are in progress. The UK has also nominated experts to the SIAP, and various Working groups and Task Forces, these nominees are now participating in these meetings. We believe these appointments are bringing significant nuclear industry experience and expertise to these groups.

Nuclear R&D: The UK perceives nuclear energy as a contributor to secure, low-carbon energy supply in the future and recognizes the importance of investing in innovation to support this. The current UK nuclear innovation programme runs from 2016 to 2021, the UK is investing ~GBP 180 million in nuclear innovation over this period and covers a number of areas. Of particular note are recent programmes under delivery.

Advanced nuclear fuels and fuel cycles: A further programme of Advanced Fuel has recently been confirmed by the Department of Business Energy and Industrial Strategy. This Fuel development work extends beyond LWR fuels to cover research into improved manufacturing processes for coated-particle fuels, exploration of a range of coatings and deposition and fabrication techniques for the fuel kernels. The fuels programme includes improved fast reactor fuels, including plutonium containing fuels. 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 as part of their validation prior to reactor testing. Research into fuel recycling processes is also being undertaken to reduce future environmental and financial burdens. The research aims to demonstrate radical improvements in economics, proliferation resistance, waste generation and the environmental impact of nuclear fuel recycle technologies.

Developing materials, advanced manufacturing and modular build for future reactors: An integrated programme of R&D on advanced materials and manufacturing is underway. This programme encompasses 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 for use in the development of Gen-IV reactors. It also includes the modularisation and more effective manufacture of reactors in general.

Research to underpin the development, safety and efficiency of the next generation of nuclear reactor designs: This research and innovation 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.

Advanced nuclear technologies: The Department for Business, Energy and Industrial Strategy (BEIS) has established an Advanced Modular Reactor (AMR) feasibility and development programme. For this competition, AMRs are defined as a broad group of non-LWR advanced nuclear reactors. The aim is to target improvement on current technology through:

- Generating low cost electricity.
- Increasing flexibility in delivering electricity to the grid.
- Increasing functionality, such as the provision of heat output for domestic or industrial purposes or facilitating the production of hydrogen.
- Alternative applications that may generate additional revenue or economic growth.

Nuclear Innovation and Research Advisory Board (NIRAB): NIRAB was reconvened in 2018 to provide independent expert advice to Government. It published key messages to the UK Government:

- A broad role for nuclear that extends beyond baseload, brings a flexible supply, heat and hydrogen.
- Urgent action is needed to accelerate the development and demonstration of technologies that can service new applications and markets.
- Government support is already having an impact through the Nuclear Innovation Program (NIP). NIRAB recommend over the next spending review (2021-2026) that government consider investing up to GBP 1 billion to accelerate and enable the private sector to commercialize new products.
- Effective delivery of the NIP should occur through a delivery body with responsibility for the strategic direction, delivery and integration of the NIP creating maximum value for money.

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. The Department of Energy (DOE) is aggressively working to revive, revitalize, and expand nuclear energy capacity. One of DOE's top priorities is to enable the deployment of advanced nuclear energy systems, including advanced light water and non-water-cooled reactor concepts being pursued by US nuclear developers. The development of improved advanced nuclear reactor designs and technologies, as well as application of advanced reactor technologies to improve the operation of the existing domestic fleet of nuclear power plants is critical to ensuring that nuclear power will be a viable option for the United States (US) energy requirements for generations to come. By focusing on the development of innovative advanced reactors – such as small modular reactors – and investing in the existing fleet, we can ensure a clean, reliable, and secure power source for our nation.

Congressional support for nuclear energy is apparent by the many acts going through the House and the Senate. In addition to the Nuclear Energy Innovation Capabilities Act (NEICA), which was signed by the President on 28 September 2018, and the Nuclear Energy Innovation and Modernization Act (NEIMA), signed by the President on 14 January 2019, the following Acts are currently being considered by the House or the Senate.

The Advanced Nuclear Fuel Availability Act (H.R.1760) was passed by the House on 9 September 2019 and directs the Office of Nuclear Energy in the Department of Energy to develop and deploy high-assay low-enriched uranium for domestic commercial use and to develop a schedule for recovering costs associated with such development. If this Act becomes law, it will pave the way for many advanced reactor fuel types which require fuel with enrichment greater than five per cent.

The Nuclear Energy Leadership Act (S.903, H.R.3306), was introduced on 6 September 2018, and reintroduced in the Senate on 27 March 2019. This bill extends the allowable period of federal power purchase agreements from 10 to 40 years and requires the Secretary to enter an agreement to purchase commercial nuclear power by December 2023 with priority placed on new nuclear technologies. This bill would also direct the Secretary to carry out at least two advanced nuclear reactor design demonstration projects by the end of 2025, and two to five more by the close of 2035. Additionally, the bill states "Not later than 1 year after the date of enactment of this section, the Secretary shall establish a program to make available high-assay, low-enriched uranium, through contracts for sale, resale, transfer, or lease, for use in commercial or non-commercial advanced nuclear reactors".

Additionally, the Advanced Nuclear Energy Technologies Act (H.R.3358), introduced in the House on 19 June 2019, moves to amend the Energy Policy Act of 2005 to direct the Secretary of Energy to carry out demonstration projects relating to advanced nuclear reactor technologies to support domestic energy needs, and for other purposes.

The Nuclear Energy Renewal Act of 2019 (S.2368) was introduced in the Senate on 31 July, 2019 and moves to amend the Atomic Energy Act of 1954 and the Energy Policy Act of 2005 to support licensing and relicensing of certain nuclear facilities, nuclear energy research, demonstration and development.

Finally, the Nuclear Powers America Act of 2019 (S.1134, H.R. 2314) allows a tax credit for investments in qualified nuclear energy property placed in service before 1 January 2026. The credit applies to any amounts paid or incurred for refueling or other specified expenditures for a nuclear power plant for which an application for license renewal was or will be submitted to the Nuclear Regulatory Commission before 1 January 2026.

The FY20 Presidential Budget Request asked for USD 75M specifically for advanced reactor technologies. The House Committee on Appropriations congressionally directed projects with USD 105M for advanced reactor technologies. The Senate Committee on Appropriations did not congressionally direct projects for advanced reactor technologies specifically but did direct USD 10M for a MW-scale reactor, USD 40M for the versatile test reactor, and USD 22M for continuation of two performance-based advanced reactor concepts which refer to Southern Company's project to develop a molten chloride fast reactor and X-energy LLC's high temperature gas reactor. Separately, the Senate directed funds for proposals from industry to build two demonstration advanced reactors. The Committee recommended USD 200 000 000 for the first year of the two demonstrations.

DOE continued an industry-focused, comprehensive, multi-year funding opportunity announcement (FOA) to support innovative, domestic nuclear reactor designs and technologies that have high potential to improve the overall economic outlook for nuclear power. These projects address first-of-a-kind nuclear demonstration readiness, advanced reactor development, and regulatory assistance. In the sixth round, this FOA awarded three projects in three states for a total of approximately USD 15 million in funding including an award to FirstEnergy Solutions Corporation which will develop a light water reactor integrated energy system. The proposed project installs an electrolysis (LTE) unit at the Davis-Besse Nuclear Power Station. The total provided to date for all six rounds of awards is approximately USD 195 million. Subsequent quarterly application review and selection processes will be conducted over the next three years. Additionally, in continuation of the Gateway for Accelerated Innovation in Nuclear (GAIN) effort to move innovative nuclear energy technologies towards commercialization, two companies, Analysis and Measurement Services Corporation, and HolosGen, LLC, were awarded funding during the fourth quarter of 2019.

In the area of light water reactors (LWRs), construction of two Westinghouse AP1000 pressurized water reactors at the Alvin W. Vogtle Electric Generating Plant in Georgia continues, with completion of construction expected by 2021 and 2022 for Units 3 and 4, respectively. On 22 March 2019, Secretary Perry visited the Vogtle plants and announced that DOE reached financial close for USD 3.7 billion in additional guarantees of loans. The Department will now guarantee a total of up to USD 12 billion in loans for the project, including existing guarantees of up to USD 8.3 billion in loans to Georgia Power Company, Oglethorpe Power Corporation, and the Municipal Electric Authority of Georgia Power subsidiaries provided in 2014 and 2015.

The DOE LWR Sustainability (LWRS) program conducts research and development to enhance the safe, efficient, and economical performance of our nation's nuclear fleet and extend the operating lifetime of this reliable source of electricity. The program is currently focused on plant modernization, flexible plant operation and generation, physical security, risk-informed systems analysis, and materials research. With respect to extending operating lifetimes, Florida Power & Light became the first utility to submit a subsequent license renewal for their Turkey Point plant in January 2018. Approval of this license renewal would allow these units to operate until 2052 and 2053. The Nuclear Regulatory Commission has set an 18-month review period for the Turkey Point application with a final decision likely in 2020. Exelon and Dominion also submitted subsequent license renewal applications for the Peach Bottom plant in Pennsylvania and the Surry plant in Virginia, respectively, which would mean a total of up to 80 years of operation for these reactors. Dominion also expects to submit a subsequent license renewal application for the North Anna reactors at the end of 2020. Additionally, Duke Energy announced in September 2019 that it intends to renew the operating licenses of 11 reactors for an additional 20 years. Duke Energy plans to submit the license renewal application for Oconee Nuclear Station in 2021, followed by its other nuclear stations. Oconee is the company's largest nuclear station, with three generating units that produce more than 2 500 megawatts (MW).

A number of plants are under economic pressure to close. Eight units have shut down since 2013, leaving 96 operating commercial nuclear reactors in the United States. In response to the economic pressure, state and local governments and regional electricity markets are considering changes to properly value nuclear power's contributions to clean energy production and grid stability. Following successful actions by New York, Illinois, Connecticut, and New Jersey, a draft law updating the Pennsylvania Alternative Energy Portfolio Standards (AEPS) Act to include nuclear energy was introduced to the state's legislature in March 2019. However, the efforts in Pennsylvania were not successful and this led to the shutdown of Three Mile Island Unit 1 in September 2019. Separately, in July 2019, Ohio passed a bill that would charge new fees to consumers statewide to create a fund to help keep FirstEnergy Solutions' two nuclear power plants open. However, there are petitions to put this bill up for a public referendum vote in November 2020 so it is uncertain whether the bill will remain law in the next year.

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 aging fossil plants, or for deployment in remote locations where electricity demand is lower. From 2012 to 2017, the DOE SMR Licensing Technical Support (LTS) program provided cost-shared financial support to accelerate the design, certification, and licensing of innovative SMR technologies that have the potential to improve SMR safety, operations, and economics of these designs. Among SMR LTS program 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, and the submittal of its design certification application (DCA) to the NRC on 12 January 2017. A significant outcome of this review involved NRC acceptance of the NuScale position regarding eliminating the need for Class 1E power for its SMR design. This is the first time that a reactor designer, large or small, has established a basis for safe nuclear reactor operations without reliance on, or requirement for, any safety-related electrical power. The NRC completed the 3rd phase of the DCA review in July 2019, completing an Advisory Committee on Reactor Safeguards (ACRS) review on all chapters of the Safety Evaluation Report (SER) with some open items remaining, and is on track to complete the review on or before the January 2021 schedule.

In FY 2018 and FY 2019, NuScale received two separate awards (Phases 1 and 2) from the Department through the industry-focused FOA (mentioned above) to continue the licensing work, finalize the design, and develop the supply chain required for commercialization. Phase 1 of this effort was completed in March 2019, and Phase 2 will be completed in September 2019. Additional Phases of work are expected to be proposed to have the plant fully commercialized and available for deployment in FY 2026.

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 a business case to inform its decision on whether to proceed in the development of a combined license application (COLA) for the proposed site. If favorable, a COLA will be developed and submitted to the NRC sometime in the 2023-2024 time frame with commercial operation projected for the mid-to-late 2020s. On 25 September 2018, NuScale Power announced that they selected Virginia-based BWX Technologies, Inc. (BWXT) to start the engineering work to manufacture NuScale's small modular reactor (SMR). BWXT immediately started work on the first manufacturing phase of NuScale's SMR, which is expected to continue through June 2020.

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 currently under development, was docketed by the NRC on 30 December 2016. On 3 April 2019 the ESP review was completed and the final Environmental Impact Statement was issued by the NRC, the final Safety Evaluation Report was issued on 14 June 2019, and the NRC Mandatory Hearing occurred on 14 August 2019. Pending approval from the commission, it may be possible for the ESP to be issued in the 1st or 2nd quarter of fiscal year 2020.

Another important initiative within DOE involves the development of accident-tolerant fuels, a new fuel for the current generation of light water reactors with higher performance and greater tolerance for severe, beyond-design-basis accidents. In addition to enhanced performance, these fuels would give operators additional time to respond to conditions such as those experienced at Fukushima Daiichi. The congressionally directed program 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 Framatome, Westinghouse, and General Electric, are conducting irradiations of their proposed fuels at the INL Advanced Test Reactor and other facilities in the United States. Several US nuclear utilities are interested in accelerating the development and use of accident-tolerant fuel concepts and in arrangement with the industrial research teams have initiated installation of lead test rods in commercial reactors in 2018 and commercial lead test assemblies continue in 2019. Commercial batch loads may start as early as 2023.

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 March 2019, the Nuclear Energy University Program (NEUP) program announced awards of more than USD 5 million for 45 undergraduate scholarships and 33 graduate fellowships to students pursuing nuclear energy-related disciplines. Through this program, undergraduates receive a USD 7 500 scholarship, while fellowship winners receive up to USD 50 000 annually over the next three years. The graduate fellowships also include USD 5 000 towards a summer internship at a US national laboratory. For FY19, DOE also awarded more than USD 28.5 million through NEUP to support 40 university-led nuclear energy research and development projects in 23 states. NEUP seeks to maintain US leadership in nuclear research across the country by providing top science and engineering faculty and their students with opportunities to develop innovative technologies for civil nuclear capabilities. Additionally, DOE continues to run the Millennials for Nuclear Caucus, a nuclear energy outreach and communications group to further engage the next generation of nuclear engineers.

As 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 modernizing the US energy portfolio. The Department recognizes the need to reinvigorate and revitalize the US nuclear industry to ensure that nuclear power can remain a part of the domestic energy mix for decades to come.

Chapter 4. **System reports**

Gas-cooled Fast Reactor (GFR)

The Gas-cooled Fast Reactor (GFR) system features a high temperature helium-cooled fast spectrum reactor that can be part of a closed fuel cycle. The GFR cooled by helium is proposed as a longer-term alternative to liquid metal-cooled fast reactors. This type of innovative nuclear system has several attractive features: the Helium is a single-phase, chemically inert, and transparent coolant. The high core outlet temperature above 750°C (typically 800-850°C) is an added value of GFR technology.

The reference concept for GFR is a 2 400 MWth plant operating with a core outlet temperature of 850°C enabling an indirect combined gas-steam cycle to be driven via three intermediate heat exchangers. The high core outlet temperature places onerous demands on the capability of the fuel to operate continuously with the high power density necessary for good neutron economy in a fast reactor core. This means the biggest challenge in the development of GFR system. Less significant challenge for GFR is to ensure the decay heat removal in all anticipated operational and fault conditions. Therefore, in the development of commercial GFR it is necessary to establish a type of experimental demonstration reactor for qualification of the refractory fuel elements and for full-scale demonstration of the GFR-specific safety systems. Actually, the ALLEGRO project reactor is to be the proposed demonstration reactor for the reference GIF GFR concept.

The ALLEGRO Gas-cooled Fast Reactor Demonstrator project

The objectives of ALLEGRO project are to demonstrate the viability and to qualify specific GFR technologies such as fuel, the fuel elements, helium-related technologies and specific safety systems (in particular, the decay heat removal function), together with demonstration that these features can be integrated successfully into a representative system. The demonstration of the GFR technology assumes that the basic features of the GFR commercial reactor can be tested in the 75 MWth ALLEGRO project.

The original design of the ALLEGRO consists of two He primary circuits, three decay heat removal (DHR) loops integrated in a pressurized cylindrical guard vessel (see Figure GFR.1). The two secondary gas circuits are connected to gas-air heat exchangers. The ALLEGRO reactor would function not only as a demonstration reactor hosting GFR technological experiments, but also as a test pad for using the high temperature coolant of the reactor in a heat exchanger for generating process heat for industrial applications and a research facility which, thanks to the fast neutron spectrum, makes it attractive for fuel and material development and testing of some special devices or other research works.

The 75 MWth reactor shall be operated with two different cores (see Figure GFR.2). The starting core with UOX or MOX fuel in stainless steel claddings will serve as a driving core for six experimental fuel assemblies containing the advanced carbide (ceramic) fuel. The second core will consist solely of the ceramic fuel and will enable to operate ALLEGRO at its high target temperature.

Central European members of the European Union, the Czech Republic, Hungary and the Slovak Republic are traditionally prominent users of nuclear energy. They intend to use nuclear energy on the long run and besides the lifetime extension of their nuclear units, each country decided to build new units in the coming years. Therefore, four nuclear research institutes and companies of the Visegrad-4 region (ÚJV Řež, a.s. – Czech Republic, MTA EK – Hungary, NCBJ – Poland, VUJE, a.s. – Slovak Republic) decided to start joint preparations aiming at the construction and operation of the ALLEGRO demonstrator of the concept of Generation IV Gas-cooled Fast Reactor (GFR) based on a Memorandum of Understanding signed in 2010. CEA, France, as promoter of the GFR concept since 2000, supports the joint preparations, bringing its knowledge and its experience in building and operating experimental reactors, in particular fast reactors. In order to study safety and design issues and also the medium and long-term governance and financial issues, the four aforementioned organizations created in July 2013 a legal entity, the V4G4 Centre of Excellence, which performed the preparatory works needed to launch the ALLEGRO Project. V4G4 Centre of Excellence is also in charge of the international representation of the project. As a result of the preparatory works it turned out that during the earlier works certain safety and design issues remain unsolved and in several aspects a new ALLEGRO design has to be elaborated. Therefore in 2015, when the ALLEGRO Project was launched, a detailed technical program was established with a new time schedule.

Figure GFR1. The GFR reactor system

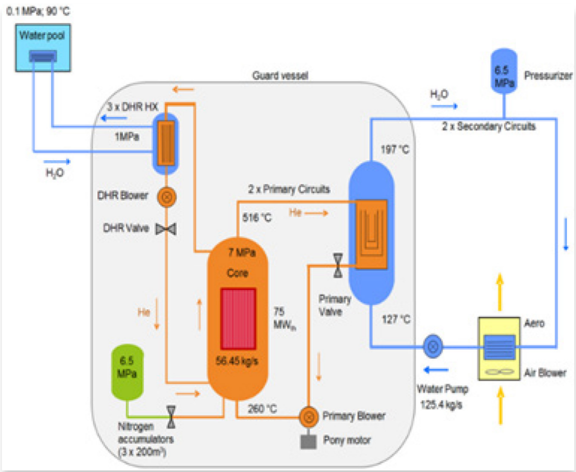
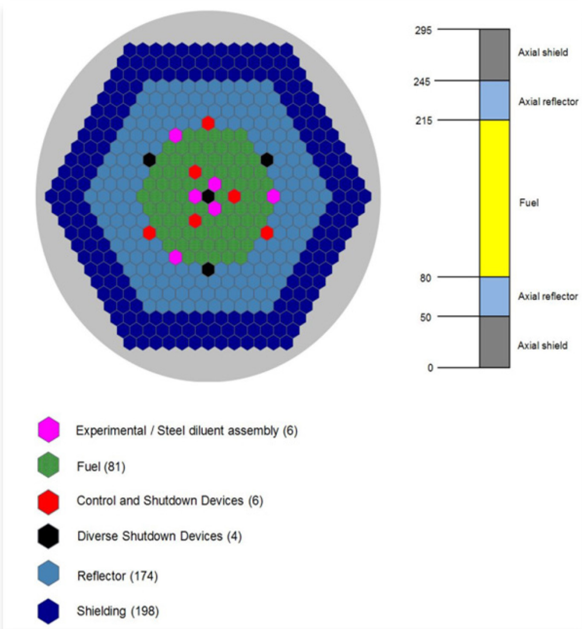


Figure GFR 2. The GFR core concept



Fuel cycle and fuel

Fuel development efforts must be conducted in close relation with reactor design efforts so that both the fuel meets core design requirements and that the core operates within fuel limits. Technology breakthroughs are needed to develop innovative fuel forms, which

- preserve the most desirable properties of thermal gas-cooled reactors, particularly to withstand temperatures in accidental situations (for the HTR up to 1 600°C, but to be confirmed by design and safety studies for the GFR);
- resist fast neutron induced damage, to provide excellent confinement of the fission products;
- accommodate an increased heavy metal content.

Alternative geometries of the fuel and innovative claddings should be investigated. The path to the GFR fuel development is intricately bound with ALLEGRO, and an iterative approach will be necessary. ALLEGRO start-up core will consider MOX or UOX fuel pellets deployed in conventional steel cladding tubing, necessitating its own design and licensing programme. An iterative step to a full ceramic demonstration core in ALLEGRO is an essential part of the R&D required for the GFR.

Candidate fuel types already identified are:

- UOX and MOX pellets in 15-15 Ti tubular steel cladding for the ALLEGRO start-up core;
- pin/pellet type fuels characterized with solid solution fuel pellets in a ceramic cladding material, whereby such pins and eventually assemblies would be introduced into the ALLEGRO starting core and eventually as a demonstration.

A significant knowledge is available on MOX fuel but needs to be available to establish the ALLEGRO start-up core.

Data on potential ceramic (particularly SiCf/SiC) and refractory alloys for cladding materials are patchy. These materials need to be adapted in order to cope with the different loads (thermal gradients, interaction fuel-barrier, dynamic loads, etc.), which means that their composition and microstructure need specific dedicated developments.

The main goal of the high temperature experiments is the investigation of the behavior of 15-15 Ti alloy in high temperature helium. Beyond the testing of small tube samples ballooning and burst experiments will be performed at high temperature. Mechanical testing will be carried out to investigate the change of load bearing capability of cladding after high temperature treatments. The cladding microstructure will be examined by SEM and metallography.

The development of qualification procedure for start-up fuel will include the specification of the steps for MOX/UOX fuel with 15-15 Ti cladding including irradiation in reactors with fast spectrum and post-irradiation examination of irradiated fuel samples.

Numerical model development for the start-up core will focus on the extension of the "FUROM" code with fast reactor fuel properties and models in order to simulate fuel behavior for the ALLEGRO start-up core. The validation of the code should be based on sodium-cooled fast reactor fuel past histories.

Testing of SiC claddings in high temperature helium will be carried out to track the potential changes. Mechanical testing and the examination of microstructure with SEM and metallography is planned with the samples after high temperature treatment. In particular, Ion irradiation effect on SiC composites will be investigated in order to evaluate the importance of the significant volume change observed for Hi-Nicalon type-S fiber and C fiber coating. High dose ion irradiation will be carried out with various temperature range including GFR operating temperature for SiC composites. High dose irradiation effect on SiC composites will be examined.

The investigation of high temperature oxidation behavior of SiC composites is important in order to model severe accident studies with air inlet. Various kinds of silicon carbide composites and monolithic SiC ceramics will be oxidized up to 1 500°C. Surface modification of SiC will be carried out based on understanding of oxidation behavior.

The following topics will be analyzed in short term:

Design of the ALLEGRO reactor core:

- UOX core feasibility study using ERANOS, MCNP, SERPENT validated Codes.
- Determination of total reactor power and power density to satisfy both safety limits and irradiation capabilities.
- Formulation of selection criteria to choose an optimal core.

Development of fuel behavior codes for ALLEGRO fuel:

- Collection of material data for fast reactor materials.
- Derivation of reactor physical parameters needed for the FUROM code.
- Implementation of fast reactor material data in the FUROM code.

Tasks related to ALLEGRO fuel qualification and specification:

- ALLEGRO fuel related acceptance criteria.
- Review of fuel candidates for the first core of ALLEGRO.

- Selection of the components of optimal ceramic fuel for ALLEGRO.
- Development of ceramic fuel qualification procedure.

Tasks related to research on fuel materials:

- Review of SiCf/SiC cladding materials.
- Testing UOX/MOX fuel cladding in high temperature He.
- Mechanical testing of UOX/MOX fuel cladding.

Advanced components and materials

Concerning in-core structural materials for the GFR (cladding, reflector, control rod guides, etc.), the main challenge is to develop materials able to withstand fast neutron induced damage together with high temperature ranges. Thus, ceramic materials (monolithic, composite) are the reference option and as a back-up selected composite cermet structures, refractory alloys, and inter-metallic compounds. In addition, the reflector should have specific neutronic properties to reduce neutron leakage efficiently and to protect the surrounding vessels; an inter-metallic compound of Zr and Si is the favored at this stage for this component.

Special issues and technologies

The improvement of decay heat removal capabilities aims at defining and optimizing a simple and robust combination and sequencing of complementary fail-safe solutions during a fast or slow depressurization transient. The main topics to be addressed are:

- Increase core thermal inertia.
- Optimization of key design parameters to enhance natural convection. Determination of required back-up pressure.
- Optimization of DHR Systems:
  - Coupled primary-secondary turbomachines;
  - Injection systems, Heavy gas accumulators.
- Guard vessel and system containment.

Branislav Hatala

Chair of the GFR SSC and all Contributors

Lead-cooled fast reactor (LFR)

Preamble: For a sake of homogeneity among all the system reports within this Annual Report, this chapter has been intentionally synthesized in a reduced number of pages. The full extended version of the 2019 LFR system report with the complete list of publications can be uploaded on the GIF website.

Main characteristics of the system

The LFR features a fast neutron spectrum and a closed fuel cycle for efficient conversion of fertile uranium. It can also be used as a burner of minor actinides, both self-generated and from reprocessing of spent fuel from light water reactors (LWR), and as a burner/breeder with thorium matrices. An important feature of LFR is the enhanced safety that results from the choice of a relatively inert coolant.

The system identified by GIF includes three reference concepts. The options considered are a large system rated at 600 MWe (ELFR EU) intended for central station power generation, a system of intermediate size (BREST 300 Russia), and a small transportable system of 10-100 MWe size (SSTAR US) that features a very long core life. The expected secondary cycle efficiency of each of the LFR reference systems is at or above 42%. These three GIF LFR reference concepts cover the full range of powers. It has therefore the potential to provide wide electricity needs: from remote or isolated sites or to serve as large inter-connected power stations. Important synergies exist among the different LFR systems, so that a co-ordination of the efforts carried out by participating countries is a the key point of LFR development. The typical design parameters of the GIF LFR systems are summarized in **Table LFR.1**.

Table LFR.1 **Key design parameters of GIF LFR concepts**

Parameters	ELFR	BREST	SSTAR
Core power (MWt)	1 500	700	45
Electrical power (MWe)	600	300	20
Primary system type	Pool	Pool	Pool
Core inlet T (°C)	400	420	420
Core outlet T (°C)	480	540	564
Secondary cycle	Superheated steam	Superheated steam	Supercritical CO_2
Net efficiency (%)	42	42	44
Turbine inlet pressure (bar)	180	180	200
Feed temperature (°C)	335	340	402
Turbine inlet T (°C)	450	505	550

R&D objectives

The System Research Plan (SRP) for the LFR is based on the use of liquid lead as the reference coolant and lead-bismuth as the back-up option. Given the R&D needs for fuels, materials, and corrosion control, the LFR system is expected to require a two-step industrial deployment: demonstration reactors operating at relatively low primary coolant temperature and low power density by 2025; then high-performance reactors by 2040. Following the reformulation of GIF LFR PSSC in 2012 the SRP was completely revised, and an updated final draft is in preparation. The approach taken in the SRP is to consider the research priorities of each member entity, and to propose a co-ordinated research programme to achieve the objectives of each member while avoiding unnecessary duplication of effort.

The committee notes that there are significant potential commonalities in research and design among these three system thrusts. The plan proposes co-ordinated research along parallel paths leading to one (or more) pilot facilities that can serve the R&D needs of the reference concepts. The needed research activities are identified and described in the SRP. Co-ordinated efforts can be organized in four major areas and formalised as projects: System Integration and Assessment (SIA); lead technology and materials; system and component design and fuel development.

System integration and assessment (SIA) project: The ultimate goal of the SIA project is to ensure the feasibility of the LFR system to meet GIF objectives, taking into account schedule and cost. The LFR SIA activities are to be carried through an iterative process aimed at ensuring that the R&D projects, either individually or jointly, satisfactorily address the GIF criteria of safety, economy, sustainability, proliferation resistance and physical protection.

System and component design project: System design activities are envisioned in the preliminary design of central station and small-scale plants, design of prototypes and demonstration plants, and co-ordination of cross-cutting activities including safety approach, component development, balance-of-plant, etc.

Fuel development project: The LFR fuel development project is a continuing long-term process consisting of tasks designed to meet progressively more ambitious requirements. It includes efforts in the areas of core materials development, fuel fabrication, fuel irradiation and tests aimed at fuel qualification. Strong synergies are existing with the parallel SFR fuel development.

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

In the mid-term, it is necessary to confirm the possibility of using advanced minor actinide-bearing fuel at levels representative of the specified equilibrium fuel cycle in order to assure minimization of long-lived nuclear waste and fuel cycle closure. This second goal is therefore to confirm the possibility of achieving higher fuel burn-up compared with that reached in current liquid metal cooled reactors.

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

Lead technology and materials project: In the near term it is necessary to maximize the use of available materials thereby limiting material qualification activities only in their new environment. To establish reactor feasibility, it is necessary to provide a technologically viable structural material capable of withstanding the rather corrosive/erosive operating conditions of an LFR. In the mid- and long term, the high boiling point of lead is advantageous for high temperature operations of the reactor, extending the LFR mission towards higher efficiency cycle and hydrogen production. Those missions require the development of new materials both for mechanical components and fuel cladding, or industrial processes to protect existing materials (coatings). These material developments will be time consuming and will be carried out with a flexible schedule depending on investments and technological achievements. Peculiar is the development of a fuel cladding resistant to high neutron doses (increased fuel burn-up) and at high temperature (increased coolant temperature and power density).

Main activities and outcomes

During 2019, the LFR pSSC has been strongly involved in the drafting or revising of several GIF reports that are expected to be issued in 2020:

- *LFR – System Safety Assessment (SSA)*. In collaboration with RSWG, the first SSA draft report was finalized in December 2018 and sent to GIF experts in 2019. The final agreed version of the report will be issued early in 2020.
- *LFR – Safety Design Criteria (SDC)*. Throughout 2019, the LFR pSSC has worked on a revision of the LFR – SDC draft report based on comments received from RSWG members. The report has been updated and finalized. It will be transmitted back to RSWG in early 2020.
- *LFR – PRPP white paper*. A first draft of the PRPP paper has been developed in strong collaboration with PRPPWG. Following a dedicated meeting in December 2019, the document is now under finalization by the LFR pSSC. It is expected to issue it in 2020.

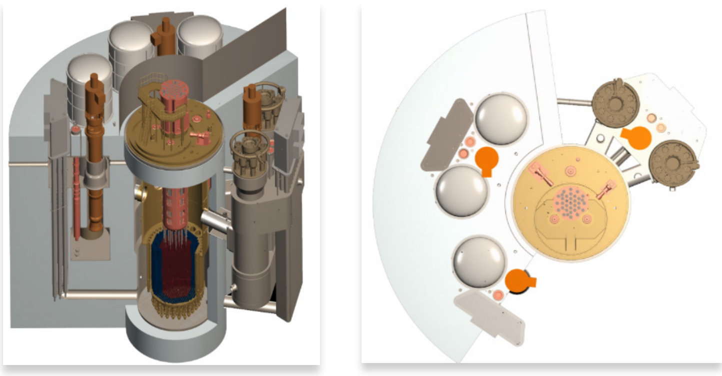
The LFR – pSSC has been also working actively with the GIF Task Force on Research Infrastructures and provided input to the Advanced Manufacturing Task Force (AMME). Finally, the LFR – pSSC was enlarged first in February 2018 by the LFR-MoU signature of USDOE, and then in Oct. 2019 by the signature of INEST (on behalf of the Chinese Academy of Sciences).

Main activities in Russia

The BREST-OD-300 fast neutron lead-cooled reactor (see **Figure LFR 2**) has been developed as the pilot and demonstration prototype of a baseline commercial reactor facility for future nuclear power. The BREST-OD-300 unit is intended for:

- practical confirmation of the key design approaches used in lead-cooled reactor facilities operating as part of a closed nuclear fuel cycle, and the fundamental principles of the inherent safety concept;
- phased justification of reactor component endurance for future commercial lead-cooled reactors;
- electricity generation.

Figure LFR 2. **BREST-OD-300 reactor**



The baseline principle behind the inherent safety of LFR consists of the preferential use of the favorable inherent neutronic and physicochemical properties of the incorporated fuel, coolant and structural materials, as well as design solutions that allow full realization of these properties to exclude entire classes of severe accidents (uncontrolled power excursions and loss of heat removal). The BREST-OD-300 reactor power level has been selected with regard for the feasibility to use the associated design concepts as a reference for future larger output reactor facilities.

The reactor core design uses mixed uranium-plutonium nitride fuel; low-swelling ferritic-martensitic steel as fuel cladding; and fuel elements contained in shroud-less fuel assemblies. The selected dense nitride fuel, in combination with the lead coolant, makes possible to have complete breeding of fissile material in the reactor core with a constant low reactivity margin, thus preventing any rapid or large neutron-power excursion during reactor operation.

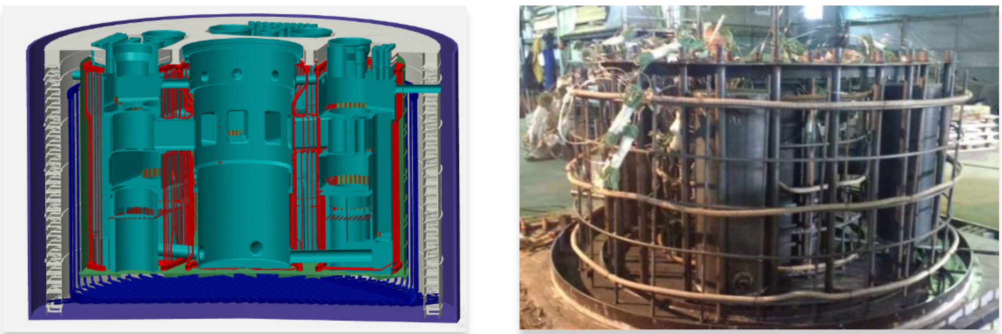
Until now, the dense nitride fuel technology has been implemented in pilot process lines. Technological processes are being improved and industrial production is being created for BREST-OD-300 (i.e. the fuel fabrication and re-fabrication module). To confirm serviceability of the fuel and the structural materials, fuel elements are being tested in BN-600 power reactor and in BOR-60 research reactor. Some of the fuel elements irradiated in BN-600 and BOR-60 have been completed and Post-Irradiation Examination (PIE) has been undertaken. They confirm in principle the fuel serviceability. The maximum burn-up achieved to date is -7.4% heavy atom (h.a.). Results required for fuel code verification have been obtained in these PIEs. The behavior of the fuel elements under irradiation meets, in principle, the pre-test analytical prediction. The obtained data demonstrates the feasibility of safe operation of the BREST fuel elements up to the parameters of the initial stage of operations (fuel burn-up of 6% h.a.).

Full-scale mock-ups have been manufactured for all types of fuel assemblies, reflector and shielding blocks. Hydraulic and vibration tests in water and liquid lead were performed. Data has been obtained which is required for updating reactor core calculations.

Neutronic calculations performed using a certified code have shown that the reactivity margin of the BREST-OD-300 reactor core life is in a range of 0.45-0.68 βeff. This reactivity margin can be ensured with regard for existing experience in fuel fabrication (fabrication accuracy is 1.2% δK/K), and the neutronic characteristic studies have been conducted, including with nitride fuel, at the BFS bench at IPPE (estimated error is 0.7% δK/K).

The specific design concepts used in the BREST-OD-300 reactor include an integral layout, absence of shutoff valves in the primary circuit, and use of passive and active-passive safety-related devices and systems. The integral layout, in a combination with the multilayer metal-concrete vessel, excludes accidents from loss of lead coolant. The justification of the metal-concrete vessel strength and serviceability (**Figure LFR 3**) is being performed based on data obtained by testing medium-sized metal-concrete structures (with typical dimensions of up to 7 m). Tests have been conducted to determine the properties of HT concrete grades under working temperatures and irradiation; the chemical inertness of the coolant with respect to concrete has been shown, and calculation procedures have been verified.

Figure LFR3. **Reactor vessel computational model and mock-up**



The BREST-OD-300 has a submerged-type once-through steam generator with a coiled heat exchanger. Silicon-containing austenite steel is used as the material of the heat-exchange tubes. Experiments were conducted to study the dependent failure of tubes caused by the rupture of one tube. The results of the experiments in lead coolant conditions simulating full-scale ones (temperature, pressure) have shown no dependent failure. Serviceability of the assembly for the heat-exchange tube embedment into the tube sheet (see **Figure LFR 4**) was confirmed based on a thousand cycles of thermally loading a model (heat-up to 540°C, cooling down to 220°C). Studies of the weld and tube metal have not revealed intolerable defects.

Figure LFR 4. **Examination of the assembly for the heat-exchange tube embedment into the tube sheet**

The Main Circulation Pump (MCP) is vertical with an electric drive, axial type. The flow path has been optimized at water and lead test benches. The required head – flow rate characteristic has been obtained to ensure the pump operation in a range from 30 to 100 %. The full-scale MCP lower bearing has been designed and tested for endurance in liquid lead. No damage has been detected based on four intermediate withdrawals of the lower radial bearing's stator and rotor (30% of the design life achieved). A positive serviceability prediction has been formulated.

Radiation safety for the reactor facility conditions is based on data obtained as the result of out-of-pile and in-pile experiments using lead. Experiments have been completed and dependences have been determined to justify the release and transport of activation and fission products from the coolant at different temperatures (up to 680°C). The results of the radiation safety analysis have confirmed the implementation of target indicators, including no need for evacuation and resettlement of the public outside the site during anticipated operational occurrences with multiple failures (e.g. loss of power supply with scram failure, full reactivity insertion). The calculation results show that the FP released from the reactor for the first day is not more than $4.3{\cdot}10^{8}$ Bq (i.e. does not exceed the reference level for the allowable daily release during normal operation) in anticipated operational occurrences accompanied by multiple failures for a scenario with full reactivity insertion. The probability of core damage at the NPP with the BREST-OD-300 does not exceed $8.6{\cdot}10^{-9}$ 1/year, which makes it possible to ensure the acceptable level of safety for such type of nuclear power based reactor.

The reactor facility detailed design was developed subject to the fundamental requirements set forth in Russia's nuclear regulatory documents. The entire set of standards and regulatory documents, which take into account the peculiarities of lead-cooled reactors, are being developed in parallel with the detailed design and R&D performance. At the present time, the federal standards and rules have been updated based on the comments received and have been sent to Rostekhnadzor. Studies show that the BREST-OD-300 concepts can be used in large commercial reactor facilities while ensuring their competitiveness. The BREST-OD-300 unit design received a positive conclusion of the Glavgosexpertiza and currently is in the process of licensing with Rostekhnadzor.

Main activities in Japan

Fundamental experimental and theoretical studies for the LFR have been carried out by the Tokyo Institute of Technology. Experimental studies on chemical control and material compatibility of heavy liquid metal coolants (HLCMs) have been performed. Chemical compatibility of structural concrete materials with the HLMCs is important topic for the development of LFRs, especially in the case of a coolant leakage accident. The chemical compatibility of various cement materials with liquid Pb and Pb-Bi was investigated by means of corrosion tests at 773 K. The coupon specimens made of Portland cement having different water/cement ratio were prepared and immersed into Pb and Pb-Bi at a static condition for 250 hours. After the tests, the chemical interaction between the cement specimens and the liquid metals was analysed. The results indicated that the chemical interaction between the HLMCs and the cement was limited (only small chemical interaction and mass transfer). These chemical behaviors were reasonable (cement materials are thermodynamically stable in the HLMCs at this temperature) and these results indicated the potential of the structural concrete as a coolant boundary.

In a theoretical study, innovative LFR concepts have been studied. The use of lead-alloy can provide for good neutron economy in fast reactors. The study on a new concept of a breed-and-burn reactor has been started utilizing the attractive features of lead-alloy. The new concept of this reactor is be based on a conventional reactor design. The reactor needs only natural uranium or depleted uranium for fuel once they come into an equilibrium condition. It is possible to achieve high burn-up of fuel without the movement of the burn-up region in the core in the equilibrium condition.

Main activities in Euratom

In June 2019, the European Commission (EC) co-organized the FISA 2019 and EURADWASTE ’19 conferences in Pitesti (Romania) with the Ministry of Research and Innovation of Romania and the Institute for Nuclear Research (RATEN-ICN). The conference gathered some 500 stakeholders, presenting progress and key achievements of around 90 projects which are or have been carried out as part of the 7th and Horizon 2020 Euratom Research and Training Framework Programmes (FP). In that frame, a side workshop organized by the FALCON consortium on the ALFRED infrastructure attracted a significant number of participants stimulating the discussion of the state of R&D of heavy liquid metal technology and a road map for the LFR demonstrator in Europe.

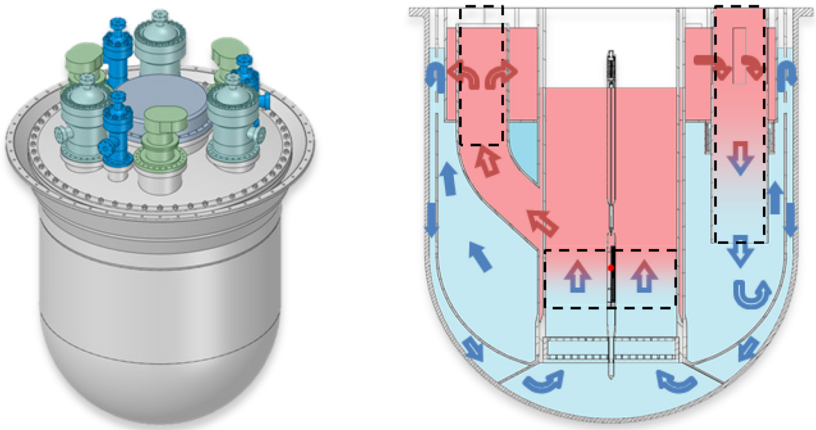
With regard to Euratom R&D projects, the already-running main projects related to LFR technology and Gen-IV fuels are GEMMA, M4F, INSPYRE and the LFR SMR INERI project (involving JRC and US DOE). A new project named PIACE has started related to the passive safety freezing prevention in LFRs. The project has had its kickoff meeting at the ENEA research lab in Brasimone and is presently under execution, expecting some experimental results to be delivered within 2020.

At the end of 2018 MYRRHA defined its road map for implementation of LBE technology for an ADS system. Belgium allocated EUR 558 M for the period 2019-2038 as follows:

- EUR 287 million for phase 1: building MINERVA (linear accelerator up to 100 MeV, 4 mA + proton target facility /PTF/) in the period 2019-2026;
- EUR 115 million for phase 2 and 3: phase 2 being the design and R&D of the second section of accelerator up to 600 MeV and phase 3 for further design and licensing activities related to LBE-cooled subcritical reactor, both to be carried out in the period 2019-2026;
- EUR 156 million for operation and experiments of the MINERVA for the period 2027-2038.

For the ALFRED project (LFR European demonstrator), the FALCON consortium made important steps during the period 2018-2019. First, a main step of the design review was completed, and a new system configuration was defined, consisting of three steam generators (SG) using single wall bayonet tubes, three dedicated dip coolers for the second decay heat removal (DHR) system, and three primary pumps (PP). Additional changes have been made in the primary system configuration by the definition of a hot and cold pool and a special arrangement of the primary flow path to completely eliminate the thermal stratification on the vessel (for forced and natural circulation conditions). The new configuration and its main characteristics are presented in the following **Figure LFR.5**. The DHR-1 is constituted by Isolation Condensers connected to steam generators (three units) and equipped with the anti-freezing system which is investigated in the PIACE project. A similar system is used for the DHR-2 system connected to a dip cooler using double-wall bayonet tubes.

Figure LFR 5. **ALFRED primary system flow-path configuration and externa view**



The FALCON consortium enlarged the community and extended the ALFRED project with the signature of several Memorandum of Agreements with partners willing to support in-kind the technical activities related to ALFRED development. The FALCON consortium also reached an important decision with regard to ALFRED operation and licensing: it was decided to approach both the operation and licensing using a step-wise approach to better face the known limits concerning materials in a representative environment. The idea is to follow a staged approach characterized by a constant primary mass flow and increasing power levels resulting in an increase of the maximum lead temperature:

- 1st stage (low temperature): Use of proven technology, proven materials, oxygen control, low T°, and Hot Fuel Assembly (FA) for in-core qualification of dedicated coatings for cladding;
- 2nd stage (medium temperature): Requires FA replacement, but uses the same SGs and PPs, and Hot FA for in-core qualification at higher temperature;
- 3rd stage (high temperature): Replacement of the main components for improved performance, representative of First-Of-A-Kind (FOAK) conditions for LFR deployment.

Consequently each stage is used to qualify the operation that will be carried out in the following stage. Each stage of the operation will need to be separately licensed but, using the confidence gained in the previous stage, the licensing process is expected to be a continuous process. The following table provides the main parameters of the envisaged staged approach:

Table LFR 2. **ALFRED staged approach main parameters**

Normal operation – full power	Units	Stage 1	Stage 2	Stage 3
Thermal power	*(MW)*	*100*	*200*	*300*
Core inlet temperature	*(°C)*	*390*	*400*	*400*
Core outlet temperature	*(°C)*	*430*	*480*	*520*
Pump head	*(MPa)*	*0.15*	*0.15*	*0.15*

During 2019, the Romanian government awarded RATEN-ICN (the Romanian research lab) funding of EUR 2.5 million in the frame of a project dedicated to "Preparatory activities for ALFRED infrastructure development in Romania". The project will last 15 months from September 2019 to November 2020. RATEN-ICN also responded to a call for proposals from the Romanian government with a project "ALFRED – step 1, experimental research support infrastructure: ATHENA (Lead pool-type facility) and ChemLab (Lead chemistry laboratory)".

A budget of EUR 133.9 million has been allocated in the 2019-2020 Euratom Fission Call for H2020 project proposals. Several project proposals were submitted in domains related to LFRs. The selected projects are expected to start by mid-2020. Finally, the SESAME Euratom collaborative project was finalized in 2019 with a workshop and the issue of a book dedicated to thermal-hydraulics aspects of liquid metals.

Main activities in Korea

In Korea, the LFR R&D has been redirected towards marine propulsion and space power development, by taking advantage of the excellent safety, very long refueling intervals and economic potential of LFR. Since the first Korean study begun in 1996 at Seoul National University (SNU), a new university consortium named Micro Nuclear Energy Research and Verification Arena (MINERVA) was formed to carried out a four-year conceptual design development of a non-refueling marine propulsion reactor cooled by LBE, in support of the Ministry of Science, Information and Technology (MSIT). The Ulsan National Institute of Science and Technology (UNIST) leads the MINERVA consortium with the participation of SNU, the Korea Advanced Institute of Science and Technology (KAIST), Kyunghee University, Ulsan University, KEPCO International Nuclear Graduate School (KINGS) and Moojin-Keeyeon Company. The Korean LFR Program has presently two main objectives:

- micro-modular reactors for marine propulsion, including ice breakers for opening the Northern Sea Route (NSR) that will cut CO_2 emission up to 40% for civilian vessels between Europe and Northeast Asian countries (including ROK). It is envisaged to expand it to container ships and bulk carriers, in support of the International Maritime Organization (IMO) resolution to ameliorate climate change.
- a technology development requirement for sustainable power generation using energy produced during nuclear waste transmutation has been reformulated towards increased safety.

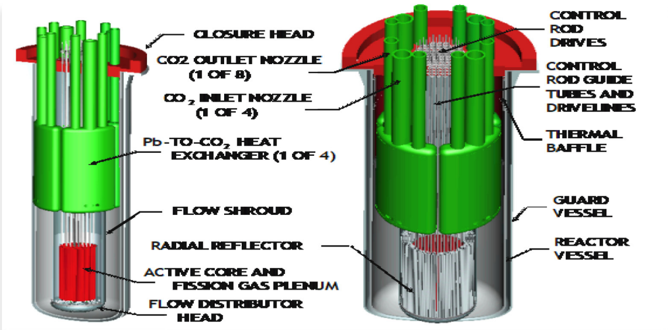
To meet the first goal, a non-refueling micro-modular reactor called MicroUranus has been designed by MINERVA consortium based on URANUS as the reference. MicroUranus has innovative features including a compact core with the help of pony pumps and inherent natural circulation while keeping the reactor core life up to 40 years covering the entire life cycle of icebreakers and container ships without refueling. The power rating of MicroUranus is being optimized in the range between 15 MWe and 30 MWe. In order to assure the reliability of reactor systems overcoming aging phenomena including corrosion, Functionally Graded Composite (FGC) materials are envisioned to be used. As part of this material development, a group of researchers designed a FGC tube pilgering process using three-dimensional finite element analysis (FEA).

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 into short-lived low-intermediate level wastes. In 2008, the Korean Ministry of Science and Technology (MOST) selected the SFR as the technology for long-lived waste transmutation. Recently, 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) by a consortium led by SungKyunKwan University with the participation of Seoul National University and UNIST. For both objectives, large-scale test loops are employed for materials and thermo-hydraulic testing and model benchmarking. Korea’s first large-scale LFR test facility, HELIOS, has been moved from SNU to UNIST where MINERVA is led. At SNU, a URANUS 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 2018.

Main activities in United States

Work on LFR concepts and technology in the United States has been carried out since 1997. In addition to reactor conceptual design, past activities included work on lead corrosion and thermal-hydraulic testing, 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 limited, past and ongoing efforts at national laboratories, universities and the industrial sector demonstrate continued interest in LFR technology. With regard to reactor design concepts, of particular relevance is the past development of the Small, Secure Transportable Autonomous Reactor (SSTAR) shown in **Figure LFR 6**, carried out by Argonne National Laboratory (ANL), Lawrence Livermore National Laboratory (LLNL) and other organizations 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 and sealed. Some notable features include reliance on natural circulation for both operational and shutdown heat removal; a very long core life (15-30 years) with whole reactor or cassette refueling; and an innovative supercritical CO_2 Brayton cycle power conversion system. This concept represents one of the three reference designs of the GIF LFR pSSC. Even if this concept is no longer under development, it is still retained as a reference system by the pSSC to represent the small/very small size category for LFRs.

Figure LFR 6. **ALFRED Small, Secure Transportable Autonomous Reactor (SSTAR) view**



Past national laboratory efforts related to the LFR, in addition to the SSTAR reactor design efforts, include lead and lead-alloy performance and material compatibility studies activities at Los Alamos National Laboratory (LANL) with the Delta Loop. This facility has since been discontinued.

Current national laboratory activities include conceptual design, advanced material development and performance research, and instrumentation for monitoring steam generator status, principally conducted as industry-government partnerships under the USDOE GAIN (Gateway for Accelerated Innovation in Nuclear) program, at Oak Ridge National Laboratory (ORNL) and the Pacific Northwest national Laboratory (PNNL) in association with industry participants Westinghouse, Hydromine and Columbia Basin Consulting Group (CBCG). In the US industrial sector, current LFR reactor initiatives include the three companies mentioned above. Westinghouse Corporation maintains an ongoing initiative to design and commercialize a new advanced LFR system. Hydromine, Inc. is developing a new LFR reactor concept identified as LFR-AS-200 (Amphora-Shaped) in the 200 MWe size range as well as a family of smaller (microreactor) systems, and CBCG is developing a new conceptual design for a LBE reactor concept.

The Westingouse LFR aims at economic competitiveness, even in the most challenging global markets, through a simple and robust design, passive safety and life cycle requirements embedded in the design from the early design phase. It is a 950 MWt (~450 MWe) reactor, being developed starting with a lower-power prototype unit for technology demonstration. It utilizes a hybrid, micro-channel type heat exchangers to reduce vessel size/weight, and a thermal energy storage system to provide load following with minimum variations in-core thermal power. Additionally, it features a supercritical CO_2 power conversion system with air as the ultimate heat sink. The prototype unit will use oxide fuel and a pure lead coolant maintained at temperatures below 550°C. Advanced fuels and higher temperatures will be implemented after the prototype demonstration phase.

Hydromine's LFR-AS-200 concept is a compact 200 MWe LFR in which a high degree of compactness (<1 m^3 primary system volume/MWe output) is achieved by elimination of components and other design optimizations utilizing the favorable characteristics of pure lead as a coolant. This compactness metric is estimated to be from 2 to 5 times lower than other metal-cooled fast reactors previously designed or in current design stages. In addition to this 200 MWe concept, Hydromine also envisions a family of very small (micro) reactors (5-20 MWe) known as the LFR-TR-X family with similar compactness and simplification of design. In these designs, control and shutdown rods are located outside the core, and the reactors are able to operate continuously for 15 years without refueling. The LFR-5 could be deployed in the near term, owing to its lower operating temperatures and use of qualified materials.

CBCG is taking an integrated approach to clean energy production by developing a nuclear plant design with load-following capabilities as an integrated grid-scale battery concept. Both the nuclear plant and the grid-battery are new designs by CBCG – when paired as an integrated facility, demand load-fluctuations are accommodated by the battery, while the nuclear plant remains at baseload operations. The nuclear plant uses LBE coolant with operation in a fast reactor spectrum. Initial efforts are focusing on licensing and regulatory requirements. As part of its ongoing research, CBCG is developing a Polonium mitigation system to reduce containment building requirements and off-site release potentials by eliminating the principal radiological release hazard associated with this technology.

Main activities in China

In February, 2019, INEST, the Chinese Academy of Sciences (CAS) was appointed as the Chinese representative for the Lead-cooled Fast Reactor (LFR) program in the Generation IV International Forum by the Ministry of Science and Technology of China. In this role, INEST co-ordinates the GIF LFR activity framework of domestic organizations inside China. On 18 October 2019, INEST signed the LFR memorandum of understanding (MoU) on behalf of China during.

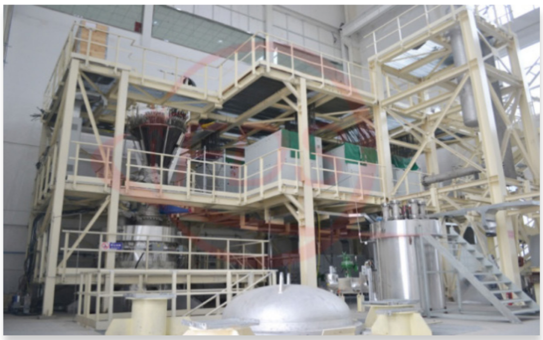
The Chinese government has provided continuous national support to develop lead-based reactor technology since 1986, by the CAS, the Minister of Science and Technology, the NSF. Following the last 30 years of research on lead-based reactors, the China LEAd-based Reactor (CLEAR) was selected as the reference reactor for both ADS and fast reactor systems, and the program is being carried out by the INEST/FDS Team, CAS. The activities on CLEAR are 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 the government related to lead-based reactor have been published. The CLEAR-M project aiming to construct a small modular energy supply system has been launched. The engineering design for the first prototype mini-reactor CLEAR-M10a with power of 10 MWth was carried out. To promote the engineering and commercial application of CLEAR-M, the China Industry Innovation Alliance of Lead-based Reactor (CIIALER) and the International Co-operative Alliance for Small LEad-based Fast Reactors (CASLER), both led by INEST, were established and supported by over 100 companies, and a related industrial park began to be built.

For an ADS system, several concepts and related technologies are under assessment. For example, the detailed conceptual design of CLEAR-I with the final goal of MA transmutation having an operational capability of subcritical and critical dual-mode operation has been finished. An innovative ADS concept system as an advanced external neutron source driven traveling-wave reactor for energy production, CLEAR-A, was proposed. The CiADS project aiming at building a 10 MWth subcritical experimental LBE-cooled reactor coupled with accelerator was approved, and preliminary engineering design is underway. The project was conducted by the collaboration of CAS and other industrial organizations.

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 reactors, three integrated test facilities have been built and commissioned since 2017, including the lead-based engineering validation reactor CLEAR-S (see **Figure 12 LFR.7**), the lead-based zero power critical/subcritical reactor CLEAR-0 coupled with HINEG neutron generator for reactor nuclear design validation, and the lead-based virtual reactor CLEAR-V. In 2019, a loss-of-flow benchmarking test based on the pool-type CLEAR-S facility was prepared, and is planned to be conducted in 2020.

Figure LFR 7. **Lead-based Engineering Validation Reactor CLEAR-S**



In recent years, several other organizations started paying greater attention to LFR development. China General Nuclear Power Group (CGN) is carrying out CLFR reactor conceptual design and related research. China National Nuclear Corporation (CNNC) is developing LFR technologies such as core neutronics characteristics testing. The State Power Investment Corporation (SPIC) is focusing on the 100 MWe BLESS reactor conceptual design. Several universities, such as Xi'an Jiaotong University (XJUT), the University of Sciences and Technology of China (USTC), are carrying out fundamental LFR technologies R&D, including materials testing, thermal-hydraulic analysis, safety analysis, etc., to support LFR development in China.

In December 2019, the domestic co-ordination meeting of GIF LFR was held in INEST. Representatives from more than ten Chinese organizations who were involved in LFR R&D attended this meeting. The domestic LFR joint working group was proposed and INEST was suggested as the lead of the working group to co-ordinate the participation and co-operation of related organizations and activities in China.

Alessandro Alemberti

Chair of the LFR SSC
and all Contributors

Molten Salt Reactor (MSR)

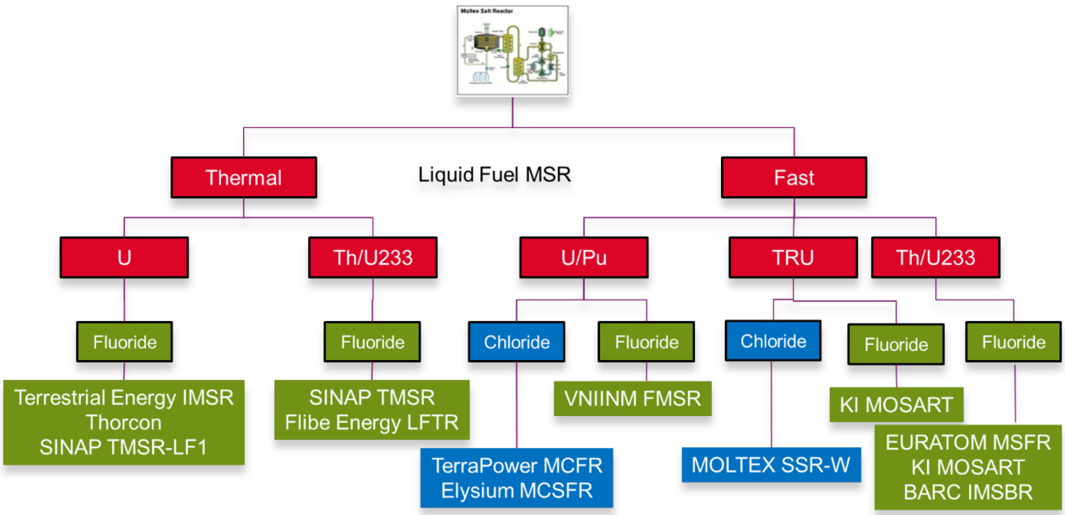
Introduction

Molten Salt Reactor (MSR) concepts have been studied since the early 1950s, but with only one test reactor operated at ORNL in the 1960s. For about 15 years, there is now a renewal in the interest of such a reactor technology, in particular for its acknowledged inherent reactor safety and its flexibility.

MSR uses molten salts as fuel and/or coolant. When a fluoride salt is the coolant only, such concepts are named under FHR (Fluoride salt-cooled high-temperature reactor). Today, in the GIF pSSC MSR, most, not to say all, the studied concepts are actual MSR with a liquid fuel.

MSR is a concept and not a technology. Indeed, the MSR generic name covers thermal and fast reactors, operated with a U/Pu or a Th/233U fuel cycle, or as TRU burners, with a fluoride or a fluoride carrier salt. An illustration of the most studied concept is given in **Figure MSR 1**.

Figure MSR 1. **The most studied MSR concepts, with the key players (RTOs or Vendors)**



Depending on the fuel cycle, MSR can reuse fissile and fertile materials from LWR or they can burn high enriched uranium, plutonium or minor actinides. They have an increased power conversion efficiency (the fission directly occurs in the carrier salt, which transfers its heat to the coolant salt in the heat exchangers). MSR are operated under low pressure, slightly above the atmospheric pressure. They can be deployed as large power reactors or as small modular reactors (SMR). Their deployment is today limited by technological challenges such as high temperature, structure materials, corrosion, etc.

The MSR pSSC includes today seven full members (Australia, Canada, Euratom, France, Russia, Switzerland and United States) and Observers (China, Japan, Korea) and is moving towards a System Arrangement. Mission of 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.

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 MSR technically challenging technology will require concerted, long-term international R&D efforts, namely:

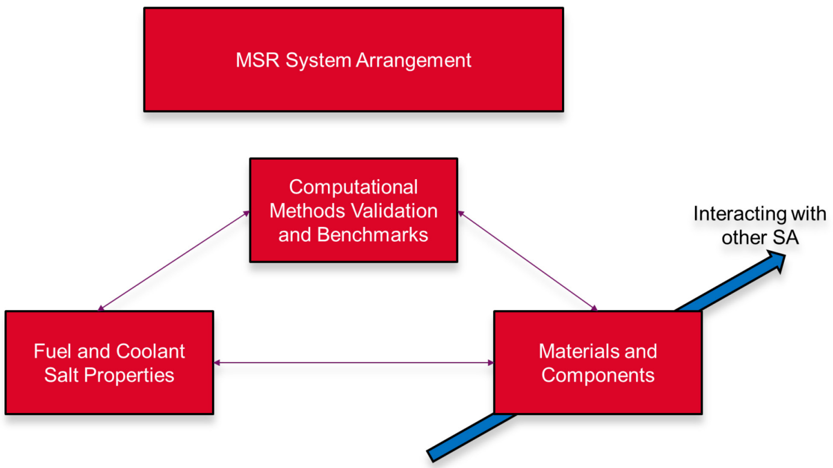
- studying the salt chemical and thermodynamic properties;
- system design: Development of advanced neutronic and thermal-hydraulic coupling models;
- studying materials compatibility with molten salt;
- Salt Redox control technologies to master corosion of reactor vessel;
- development of efficient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

Main activities and outcomes

MSR pSSC activity

In 2019, the key activity was the preparation of the System Arrangements with the definition of three potential Projects Arrangements, which would allow the community to contribute widely. Therefore, these PAs are quite transversal and not concept dependent but can support the development of any concept (see **Figure MSR 2**). They address the salt behavior, the materials properties and the system integration. The SA should enter into force in 2021.

Figure MSR 2. **Foreseen structure of the MSR SA including three PAs**

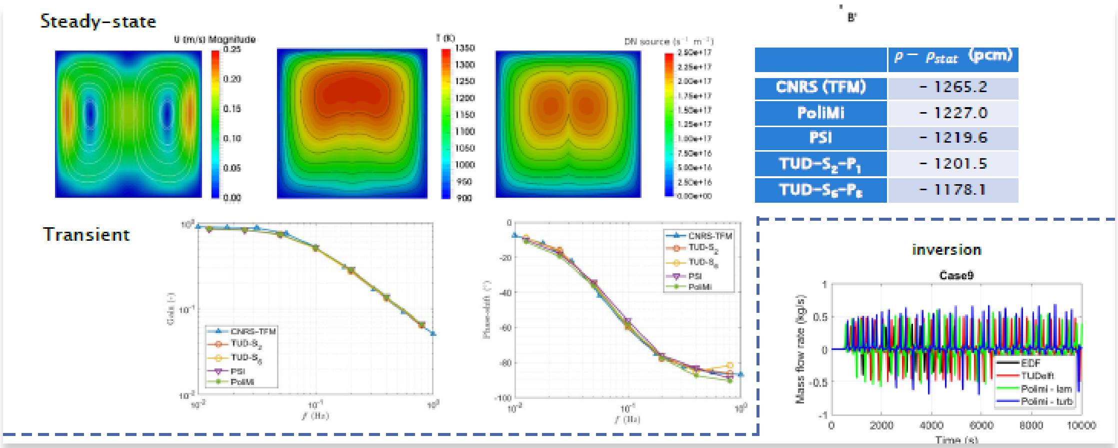


Euratom

European Project SAMOFAR/SAMOSAFER: In the European Union the SAMOFAR project has run successfully under co-ordination of TU Delft for a period of four years from 2015 to 2019 and has been closed by a festive meeting in the theatre of Delft, The Netherlands. This meeting consisted of a students' boot camp at TU Delft organized by the US project NuStem and the EU project SAMOFAR, followed by a two-day meeting presenting the final results of SAMOFAR and with presentations of representatives from Canada, China, EU, Russia and the United States on the progress made in their respective national research programmes. The boot camp was attended by about 40 students worldwide, while the final meeting was attended by about 100 participants.

At 1 October 2019, the new SAMOSAFER project started aiming at new simulation models and tools, and on new safety barriers for the Molten Salt Reactor. The goal of this new project is to develop and demonstrate new safety barriers for more controlled behavior of Molten Salt Reactors in severe accidents, based on new simulation models and tools validated with experiments. The grand objective is to ensure that the MSR can comply with all expected regulations in 30 years' time. After successful completion of this project, the simulation models and tools can be used by the nuclear industry, and the innovative safety barriers can be implemented in new MSR designs. This will lead to increased safety margins in future Gen-IV Molten Salt Reactors to ensure they will comply with the latest and future safety standards. SAMOSAFER is co-ordinated by TU Delft and will run until 2023.

Figure MSR 3. **Multi-physics code for V&V. Benchmark of PoliMi and PSI codes**

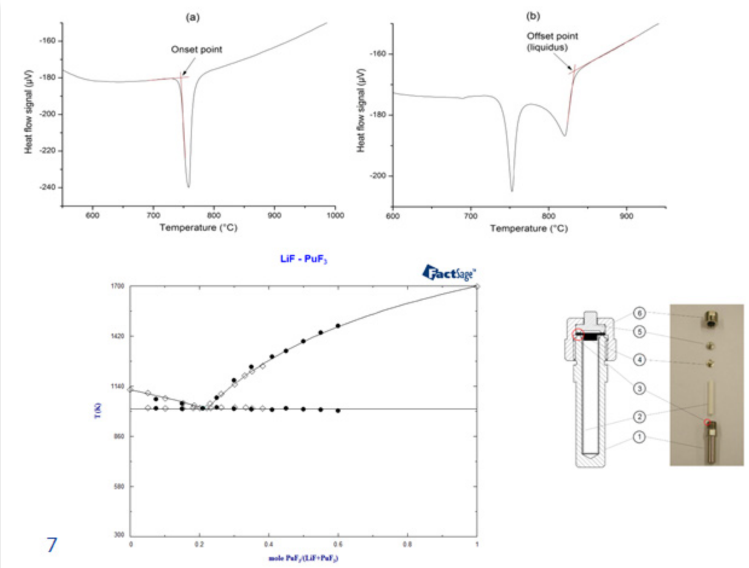


A special session on the progress in the Science & Technology of Nuclear Reactors using Molten Salts was organized in collaboration with the European Physical Journal, Nuclear Sciences & Technologies. Guest editors were Jan Leen Kloosterman (TU Delft), Elsa Merle (CNRS) and Jean Ragusa (Texas A&M).

In the Netherlands, the Salient-01 irradiations in the Petten High Flux Reactor were finalized. The samples are currently being investigated in the framework of SAMOSAFER in laboratories at NRG and JRC Karlsruhe. Follow-up irradiations are underway.

JRC Karlsruhe: After successfully establishing the method to synthesise highly pure actinide fluoride salts using HF fluorination line at JRC Karlsruhe, focus was put on development of a chlorination technique to synthesise actinide chloride salts with same high purity. The first tests were done on innovative synthesis of uranium chloride salts from uranium oxide, using a mixture of Cl_2 and CCl_4 gases (carbo-chlorination to convert UO_2 to UCl_4) with successive reduction by H_2 (to convert UCl_4 to UCl_3). By the end of 2019, the first step of the conversion was successful and small quantities of highly pure UCl_4 were obtained.

Figure MSR 4. **Phase diagram study pf the LiF-PuF3 system**



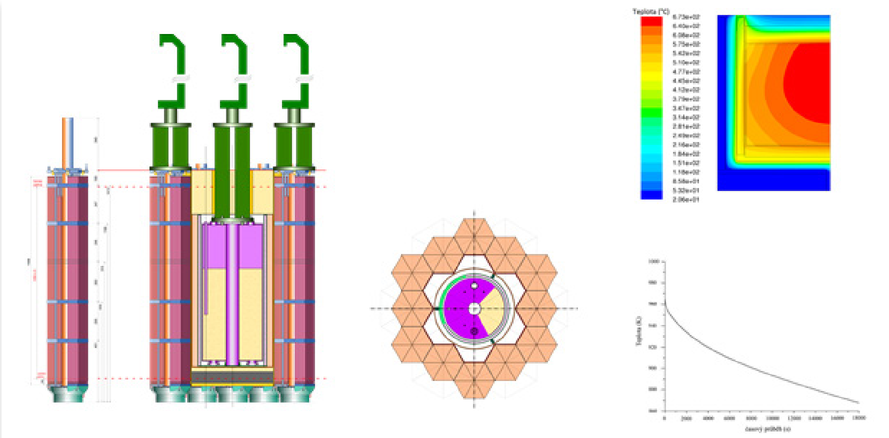
Work on high temperature properties investigation of fluoride salt systems has continued at JRC Karlsruhe with the following highlights:

- First measurements of the melting point of pure PuF_3 have been done. This study still needs optimization, but the experimental set up to determine melting points of actinide fluorides that melt above 1 250°C, which was a limit using standard encapsulation technique, has been established.
- A method to measure thermal conductivity of solid salts, simulating precipitates on the reactor vessel surface, has been improved, achieving successful and reliable measurements using laser flash technique. Thermal conductivity of the series of alkali fluorides has been measured, as well as solidified complex fuel mixtures containing UF_4 and ThF_4 components.
- First trial to test the method to determine solubility of gases in molten salts have been made utilizing high temperature infusion facility with successive quantitative determination of the dissolved gases using effusion cell coupled with quantitative mass spectrometry.

Among experimental studies on basic thermo-chemical properties, JRC is significantly involved in synthesis and fabrication of the fluoride fuel salt for the planned irradiation experiment SALIENT-03 in the HFR (High Flux Reactor) in Petten in a major collaboration with NRG.

Centre Research ŘEŽ: In 2019, research and development of MSR technology continued in the Czech Republic also as a part of a national MSR project supported by the Ministry of Industry and Trade and by the Technology Agency of the Czech Republic. The main effort was devoted to the preparation of neutronic measurements with the so-called "Hot Inserted FLIBE Zone" in the experimental reactor LR-0 of the Research Centre Řež. This is a new demanding experimental program whose aim is to determine the reactivity temperature feedback coefficients of the FLIBE melt in the working temperature range of MSR system (500-750°C). By the end of 2019, all the main components of the Hot Inserted FLIBE Zone had been produced. The active experimental program will start in 2020.

Figure MSR 5. **Design of the heated zone for FLIBE salt**



In addition, the development and testing of impeller pumps for fluoride melt media, in the program of MONICR alloy welding technology optimization and in the long-term MONICR alloy corrosion test program. Research work also continued in the field of the development of electrochemical separation methods from fluoride melt media and a new activity focused on experimental studies and verification of MSR volatile fuel components extraction by Fused Salt Volatilization method was launched in second half of 2019.

France

In addition to the work carried out in SAMOFAR and SAMOSAFER, French partners (CNRS, CEA, FRAMATOME…) worked at building a MSR community around a common roadmap including neutronic and its modelling, fuel salt selection, materials and corrosion to develop a concept of fast reactor using a U/Pu fuel cycle in molten fluoride. In particular, a new project was created at CEA to co-ordinate their activities in the field.

Australia

The widespread deployment of molten-salt-based energy systems, including Molten Salt Reactors requires the development and qualification of materials and components capable of withstanding their challenging operation conditions. Hence, with a view to shortening the time to deployment of MSR low-emission energy generation systems, Australia, is working on the development, manufacturing and testing of suitable structural materials and coatings. In particular, ANSTO continues to collaborate with GIF partners to study and understand the corrosion in FLiNaK of candidate Stainless Steels and Nickel-based alloys, in particular, using ANSTO's large-scale infrastructure, (the OPAL reactor, the Australian Synchrotron, and the Centre for Accelerator Science).

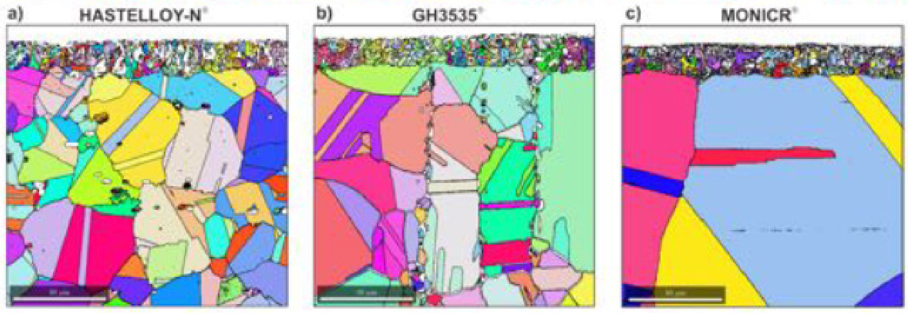
During 2019 Australia continued its initiatives to increase cross-cutting materials related research within GIF. In addition to chairing the GIF Advanced Manufacturing and Materials Engineering Task Force (AMMETF) a joint VHTR/MSR Materials and Components R&D Workshop was held between the MSR pSSC and the VHTR PMB meetings in Avignon in September 2019 designed to identify areas of common research interest.

The workshop was very successful and identified the following potential areas of collaborative R&D:

- advanced manufacturing;
- comparison of ion/neutron radiation damage design methodologies;
- development of creep, and creep-fatigue models;

- high-temperature performance, and radiation damage of graphite, C/C, SiC/SiC;
- small sample testing, and advanced surveillance methodologies;
- welding and joining.

Figure MSR 6. **FLINAK molten salt corrosion for GH3535, Hastelloy N and MoNiCr alloys**



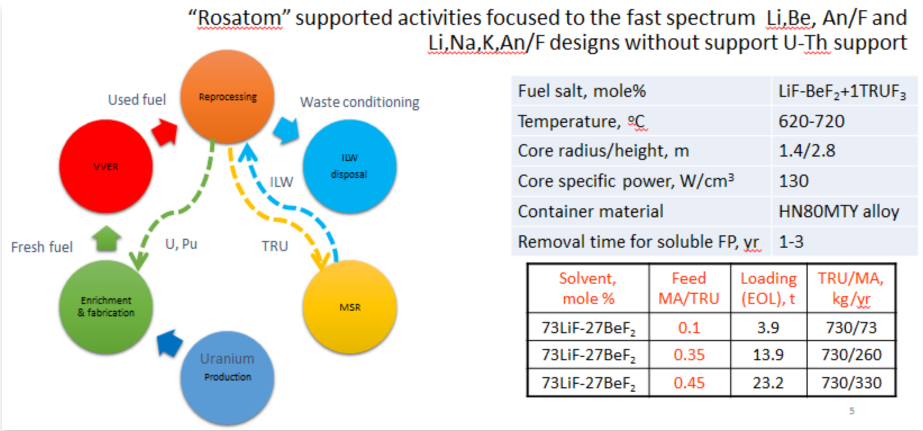
- EBSD orientation maps of HASTELLOY-N®, GH3535® and MONICR® alloys after the exposure to the FLiNaK molten salt at 650°C for 500h. The top edge represents the surface exposed to the molten salt.

Russia

A significant effort on the molten salt reactor development was performed in the RF in 2019. The State Corporation "Rosatom" has continued to provide support through a number of R&D programs to the single fluid MOlten Salt Actinide Recycler & Transmuter (MOSART) system, where minor actinides and plutonium from spent VVER-1000/1200 fuel are dissolved in the liquid fluoride salt. Last developments concerning MOSART design addressed advanced large power Li,Be,An/F and Li,Na,K,An/F units with the main objective to close the nuclear fuel cycle for all actinides, including Np, Pu, Am and Cm. The work was also continued on development of the test 10 MWt Li,Be,An/F MOSART design coupled with fuel salt clean-up unit at the site of the Mining and Chemical Combine (Zheleznogorsk) in order to demonstrate the control of the reactor and fuel salt management with different TRU loadings for start-up, transition to equilibrium, drain-out, shut down, etc. The range of MOSART technology developments currently underway includes:

- high neutron fluence and salt tolerant alloy design property evaluation for fuel Li,Be,An/F and Li,Na,K,An/F salt mixtures;
- both high and low fidelity computational modelling and tool development;
- physical and chemical property measurement for fuel Li,Be,An/F and Li,Na,K,An/F salts;
- molten salt pump & heat exchanger designs and its demonstration;
- instrumentation development;
- highly automated remote operations and maintenance technology development and demonstration;
- fuel salt clean-up demonstration and both solid and gaseous waste stream assessment.

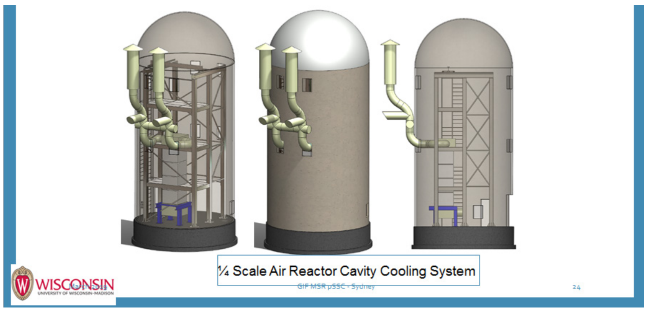
Figure MSR 7. **The MOSART Concept as an integral element helping to solve the problem of long-lived actinides**



United States

A broad range of both molten salt fueled and cooled reactor activities were performed in the United States in 2019. Notably, Kairos Power Inc. continued to file topical reports with the US Nuclear Regulatory Commission (NRC) in preparation for a license application. Also, the Canadian Nuclear Safety Commission and the NRC announced that they have selected Terrestrial Energy's Integral Molten Salt Reactor for the first joint technical review of an advanced, non-light water reactor under a memorandum of co-operation on advanced reactor and SMR reviews. The US government continues to provide support to the emerging US MSR industry through a number of cost-shared R&D programs. The NRC continues the process of modernizing its licensing requirements to better reflect the safety characteristics of advanced reactors. The US Department of Energy (DOE) continues to support both university and national laboratory activities at limited scale to overcome the remaining technical hurdles to MSR deployment. The US government has also continued work to develop MSR models and associated tools for safeguards analysis. The range of MSR technology developments currently underway includes high neutron fluence and salt tolerant alloy design and property evaluation, both high and low fidelity computational modelling and tool development, graphite testing, fuel salt thermo-physical and thermo-chemical property measurement, fuel salt thermodynamic database development, molten salt pump design and demonstration, instrumentation development, highly automated remote operations and maintenance technology development and demonstration, and both solid and gaseous waste stream assessment. Oak Ridge National Laboratory hosted the annual DOE-Gateway for Innovation in Nuclear supported MSR workshop which featured ~250 participants from industry, the national laboratories, government agencies, international organizations, and academia.

Figure MSR 8. **Air cooled RCCS designed and constructed at University of Wisconsin**



Canada

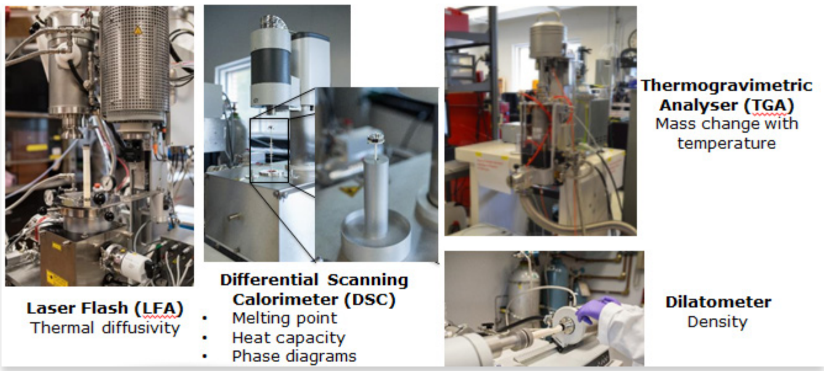
In 2019, Canadian Nuclear Laboratories (CNL) continued to develop expertise and capabilities in support of SMRs and launched a new cost-shared R&D program called the Canadian Nuclear Research Initiative (CNRI). The CNRI program was established by CNL to accelerate the deployment of SMRs in Canada by enabling research and development, and connecting the SMR industry with the facilities and expertise within Canada's national nuclear laboratories. Among the many benefits of the program, participants are able to optimize resources, share technical knowledge, and gain access to CNL's expertise to help advance the commercialization of SMR technologies. In the first intake to this new program three molten salt reactor vendors have submitted proposals with a varied program of work including electrochemical separation methods, tritium management, reactor physics, thermal-hydraulics and safeguards studies.

Under the auspices of the Canadian Federal Nuclear Science and Technology Program, CNL continued to develop molten salt capabilities across a wide range of areas including:

- development of actinide molten salt fuel synthesis;
- fission product retention in molten salt experiments; evaluation of passive cooling during a station blackout with experiments on coupled natural circulation heat transfer between water and molten salt loops and evaluation of molten salt plug melting in accident conditions;
- corrosion loop development for measuring corrosion of structural materials;
- modelling and simulation of Molten Salt Reactor Designs including evaluation of codes for advanced reactor coupled transient simulation toolset against ORNL MSRE: Physics (SERPENT, Rattlesnake); TH (RELAP5-3D, ARIANT); CFD (STAR-CCM+) and atomistic simulations to predict molten salt properties.

Finally significant efforts have continued in further developing nuclear qualified measurement techniques of thermo-physical properties of molten salts.

Figure MSR 9. **Sample encapsulation and measurement technique development at CNL**



Switzerland

The Swiss MSR research is co-ordinated by Paul Scherrer Institute (PSI). In 2019 the PSI continued to develop expertise and capabilities in the three selected areas: fuel cycle, system behavior, and thermo-dynamics of molten salts. The major aim of these simulation activities is the assessment of MSR safety and sustainability. Since PSI is a member of the SAMOFAR and SAMOSAFER projects, part of the PSI activities contribute to the EU progress report.

In the area of fuel cycle assessment, several past studies were published in 2019. The result of 16 different reactors comparison in equilibrium fuel cycle, inclusive 4 MSR types, were published. The performance in both U-Pu and Th-U cycles was compared. The breed-and-burn fuel cycle study from B. Hombourger PhD thesis was published, taking the advantage of special MSR edition of the EPJ Nuclear Sciences & Technologies journal. The breed-and-burn fuel cycle study was further extended by parametric study within MSc thesis and the results will be presented at Physor 2020 conference.

The system behavior study with Open-FOAM based solver continued in 2019 for MSFR core as a H2020 project SAMOFAR contribution. The activities were still ongoing, after the project was finished, and the respective freezing model developed within the project still requires validation. The Open-FOAM based solver was also applied for conceptual designing of own breed-and-burn core thermo-hydraulics layout. The results will be also presented at Physor 2020 conference.

The thermo-dynamics simulation of molten salts was continuing with GEMS TM code, focusing on fluoride-based salts. The liquidus temperature and its evolution during transition from initial to equilibrium fuel cycle was simulated and the results published at ICAPP 2019 Conference. The thermo-dynamics code was also coupled to severe accident code MELCORE to improve the simulation of compounds evaporation from the fuel salt in severe accidents. Towards the end of the year the preparation of the GEMS database for chloride salts started.

Figure MSR 10. **Transition from initial to equilibrium cycle**

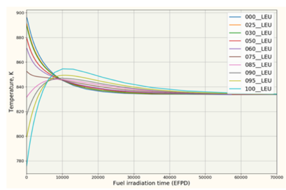
5 major existing fissile materials to start the cycle

Material	RG_Pu	LEU	~~HEU~~	~~U233~~	~~WG_Pu~~
Fissile isotope(s)	^{239}Pu, ^{241}Pu	^{235}U	^{235}U	^{233}U	^{239}Pu
Fissile isotope share	~60%	1-20%	21-95%	60-100%	>93%
"Availability"	medium	high	high	low	medium
Proliferation risk	medium	medium	high	high	high

Material	RG_Pu	LEU	HEU	U233	WP_PU
Single fluid MSBR	×	×	✓*	✓*	✓*
Double fluid MSBR	×	×	✓*	✓*	✓*
MSFR	✓**	✓**	✓*	✓*	✓*

* Unsuitable material from proliferation perspective. ** PuF_3 concentration above the solubility limit.

- The convergence to the closed Th-U fuel cycle takes more than 100 years for LEU more than 70 years for RG_Pu
- The LEU started case has quite low initial temperature (bellow 780 K)
- However, the peak temperature reach 875 K during the transition
- The best performance from the liquidus temperature perspective has the mixture of 75-85% LEU and 25-15% RG_Pu
- In this case the maximal and final liquidus temperatures differ only by 12 K.
- Nonetheless, even the 100 % LEU case may be acceptable.



China

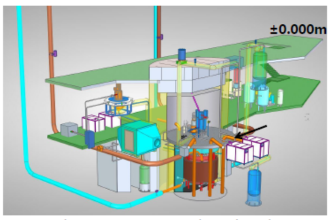
In 2019, Shanghai Institute of Applied Physics, Chinese Academy of Sciences (SINAP-CAS) is steadily promoting the related work of TMSR.

The design of the 2 MWth molten salt test reactor (TMSR-LF1) was completed and the Preliminary Safety Analysis Report (PSAR) submitted by SINAP to the National Nuclear Safety Administration has passed the examination and approval. In addition, the preparation of key components has been basically completed, and the construction of TMSR-LF1 has begun.

Figure MSR 11. **Progress of the TMSF-LF1**

- Complete the preliminary design and pass the expert review organized by the Bureau of Major Tasks in Jun. 2018.
- Start up the processing and manufacturing of key materials and equipment, and determine the manufacturer.
- Design of equipment construction drawings was completed jointly with manufacturers in Feb. 2019.

Expert review meeting



equipment construction drawings

The construction of the scaled experimental device (TMSR-SF0) has been completed, and the key commissioning activities such as primary and secondary circuit molten salt loading and main pump operation have been successfully completed. The experimental program will be launched in 2020.

Conceptual design of the flowsheets for TMSR fuels is ongoing with validation of some key techniques being finished. Fundamental studies on the chemistry of actinides and fission products in molten salt were started. The PIE experiments of several different kinds of nuclear graphite have been completed, and the evaluation method of irradiation life of nuclear graphite by ion beam irradiation was established. Additionally, the alloys used for high-temperature (750-850°C) MSR are being developed.

Stéphane Bourg

Chair of the MSR SSC and all Contributors

Super Critical Water Reactor (SCWR)

Preamble: *For a sake of homogeneity among all the system reports within this Annual Report, this chapter has been intentionally synthesized in a reduced number of pages. The full extended version of the 2019 SCWR system report with the complete list of publications can be uploaded on the GIF website.*

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 the specifics of the core design, these concepts have many similar features (e.g. outlet pressures and temperatures, thermal neutron spectra, steam cycle options, materials, etc.). Therefore, the R&D needs for each reactor type are common; this enables collaborative research to be pursued.

The main advantage of the SCWR is improved economics because of the high thermodynamic efficiency and the potential for plant simplification. Improvements in the areas of safety, sustainability, and PR&PP 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 the Project Arrangements for Thermal-Hydraulics and Safety as well as the Materials and Chemistry in 2017.

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 behavior 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 minimize materials degradation and corrosion product transport and will be based on materials compatibility and an understanding of water radiolysis.

Main activities and outcomes

System integration and assessment

Four SCWR core concepts with thermal spectrum have been proposed. 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. The China Pressure vessel-type SCWR (named CSR1000) has the following characteristics: thermal neutron spectrum, light water as moderator, two flow-pass of coolant in core, direct once-through cycle. The reference CSR1000 has the 9X9 pin by pin fuel assemblies with center 5X5 pin taken by the water moderator box. Recently, the fuel assembly and core structure design are simplified. In the new design, the UO_2 fuel rods are set around the tube to get moderated homogeneously and sufficiently. MOX fuel rods are settled in the outer zone to match spectrum. No Water rod or solid moderator are needed. **Figure SCWR 1** presents the reference and new FA design. Aiming at the reactivity control requirements of the SCWR core and its strong nuclear thermal coupling characteristics, a new type of control rod loading design was invented to overcome the shortcomings of the traditional "checkerboard" control rod loading design method. The new loading method reduces the number of control rod drive mechanism arrangements, reducing the difficulty in designing the SCWR pressure vessel top cover, and simplifying the control rod operation management procedure. Two project proposals have been approved by the China Ministry of Science and Technology in 2019 to promote the China SCWR design. The two projects start from 2020 and end in 2022. The international review of China SCWR design is supposed to be completed during this period.

Figure SCWR 1. **SCWR Thermal Spectrum Core Concepts**

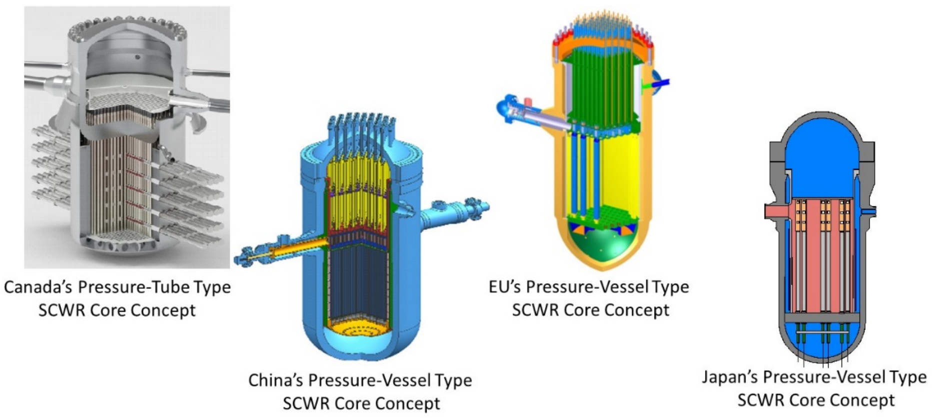
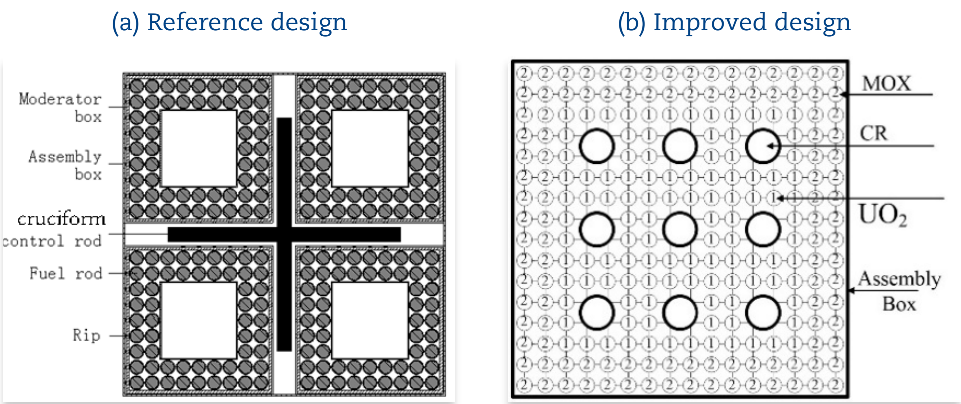


Figure SCWR 2. **China SCWR fuel assembly design improvement**



Canada has developed a preliminary small pressure-tube type SCWR concept with similar core configuration. Currently the 43-Element bundle is the preferred candidate for the fuel bundle housing 170 fuel channels. The operating pressure 25 MPa with outlet temperatures about 450°C. Work on finalizing this concept is ongoing.

One of the main general activities covering almost all fields of the SCWR development was the preparation of the European-Canadian-Chinese Small Modular SC-Water Reactor Technology (ECC-SMART) proposal. The proposal joined the significant institutions working in the field of SCWR development from Europe, China, Canada and Ukraine to create very strong multi-international consortium. The proposal covers the major knowledge gaps in the thermal-hydraulics and safety and materials and chemistry issues for SCWR technology as well as the specific SMR topics related mostly to the scaling of the technology and legislation aspects.

The 9th International Symposium on SCWRs was held in Vancouver, Canada, March, 2019, hosted by Canadian Nuclear Society. The conference was supported by Natural Resources Canada, Atomic Energy of Canada Limited and Canadian Nuclear Laboratories. About 60 participants from Canada, China, EU and Japan attended this conference and 58 presentations covering concept development and technology areas were presented.

Thermal-hydraulics and safety

The TH&S PMB activities include flow and heat transfer experiments and correlations development, critical flow and flow instability investigation, numerical investigation and code development.

A project was established at the Canadian Nuclear Laboratories (CNL) to outline a framework of current prediction methods, the parameters associated with these methods, and the fluids they are applied to. The goal of this project is to complete a summary of literature reviews of heat transfer prediction methods in SuperCritical fluids (SC-fluids). From the literature review, the heat transfer to SC-fluids is dependent on at least six system parameters and four fluid flow parameters. The six system parameters include geometry, flow orientation, fluid flow direction, fluid type, heat flux direction, and power profile. The commonly used geometries and fluid types reflect the class of research and industrial applications where SC-heat transfer occurs. A more in-depth analysis of these parameters reveals that the majority of applications in which SC-heat transfer occurs is restricted to a narrow range of fluid parameters. These four fluid flow parameters are defined as: 1) Fluid pressure; 2) Fluid mass flux; 3) Surface heat flux; and 4) Fluid bulk enthalpy or bulk temperature. Six different SC-heat transfer prediction methods are currently used: 1) Correlations; 2) Semi-empirical models; 3) Look Up Tables; 4) Look Up Lists; 5) Neural Networks; and 6) Numerical/Computational Fluid Dynamics.

CNL conducted a study to investigate the applicability of a break discharge model that was specifically developed for supercritical conditions. To achieve this goal, the model was introduced in the Canadian thermal-hydraulics system code CATHENA. This model is hereafter referred to as the Modified Homogeneous Equilibrium Model (M-HEM). A comparison between the previously used homogeneous Equilibrium Model (HEM) and the Modified Homogeneous Equilibrium Model (M-HEM) model was performed. The assessment of the discharge models was performed by using experimental data in a simple geometry configuration (shows two representative results). The results of the assessment are used as a base to update the LOCA simulations used for the Canadian SCWR conceptualization.

The fuel channels of the Canadian SCWR undergo large density variation along the reactor core as condition of the coolant flow changes across the pseudo-critical point. To verify the stability of the design, CNL created a task aimed to verify, assess and develop a stability map for the Canadian SCWR design. This task was divided in two steps: i) assessment of the tools and ii) development of the Canadian SCWR stability map. Currently, CNL is focusing on pure thermal hydraulics instabilities and assessment of modelling tools. The tool selected was the system code CATHENA. Two datasets were selected to verify the applicability of the code: 1) the two parallel channel instability experiments carried out by NPIC (Nuclear Power Institute of China), and 2) the natural circulation numerical experiments conducted at the University of Manitoba. The simulation results showed that CATHENA is able to predict the flow oscillations, nonetheless the magnitude differs from the experimental data. However, given that the model was simplified significantly, as recommended by the experimentalist, and the flow instability is highly dependent on geometry, this could have an impact on the simulation results. **Figure** SCWR4 shows two representative CATHENA prediction cases.

Figure SCWR 3. **CATHENA Predicted Coolant Mass Fluxes for 1-mm Orifice Diameter (Left) and 1.395-mm Orifice Diameter (Right) Supercritical Discharge Experiment at École Polytechnique de Montréal**

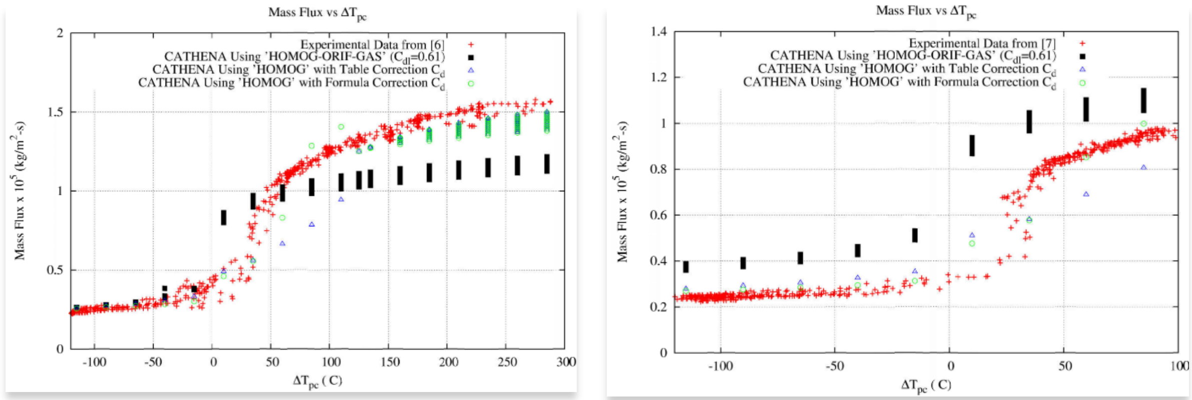
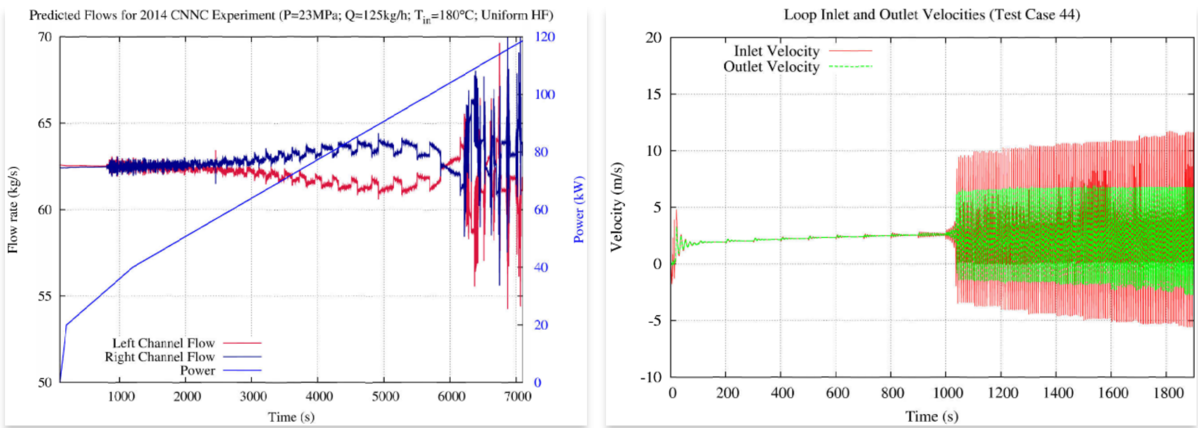


Figure SCWR4. **CATHENA Mass Flow Rate Predictions of Two Parallel Channel Instability (Left). CATHENA Predicted Velocities and Pressure Drops in a Natural Circulation Loop (Right)**



Heat transfer of in-tube SC-fluid cooling accompanying out-tube pool boiling was investigated in Xi'an Jiaotong University (XJTU). A smooth horizontal circular tube with an inside-diameter of 20 mm was submerged in a water pool at atmospheric pressure. Test parameters of in-tube were as follows: Pressure: 23-28 MPa, mass flux: 600-1 000 kg/m^{2}s, fluid temperature: 400-725 K, and the temperature difference between bulk and wall: 300–374 K. A thermal amplification system based on out-tube pool boiling was used to improve the measurement accuracy of local heat duty near pseudo-critical region. According to the experiment, the transition from nucleate boiling to film boiling in the pool occurred near the pseudo-critical fluid region. Sharp variation on thermo-physical properties led to the peak value of heat transfer coefficient in the pseudo-critical region. The pool boiling heat flux increased gradually to 1.19 MW/m^2 near the pseudo-critical point. Based on the experimental data, a modified Gnielinski equation was adopted to predict the heat transfer coefficient of in-tube SC-fluid cooling without-tube pool boiling.

Xi'an Jjiaotong University also performed experiments of heat transfer to supercritical Freon R134a flowing upward and downward in a circular tube with inner diameter of 10 mm with heat fluxes of 20-65 kW/m^2, mass fluxes of 400-1 000 kg/m^2s, bulk fluid temperature of 80-115°C at pressure condition of 4.2 MPa. The influences of heat flux, mass flux, flow direction, buoyancy force and flow acceleration on supercritical R134a heat transfer were discussed respectively. The influence of buoyancy force and flow acceleration on heat transfer were investigated and the non-dimensional parameters were obtained. New heat transfer correlations for upward and downward flow were proposed respectively.

Nuclear Power Institute of China (NPIC) performed Natural Circulation (NC) experiments and numerical analysis with water and carbon dioxide. For the supercritcal water NC instability, the preliminary analysis work has been done with the system analysis code. The code could predict the instability behavior of natural circulation. But some discrepancies exist which need further improvement. For the SC-CO_2 NC instability, based on the theoretical analysis of flow and heat transfer of SC-CO_2, a new explanation of the mechanism of flow oscillation in SC-CO_2 natural circulation has been put forward. The reliability of the new mechanism has been verified by experimental results.

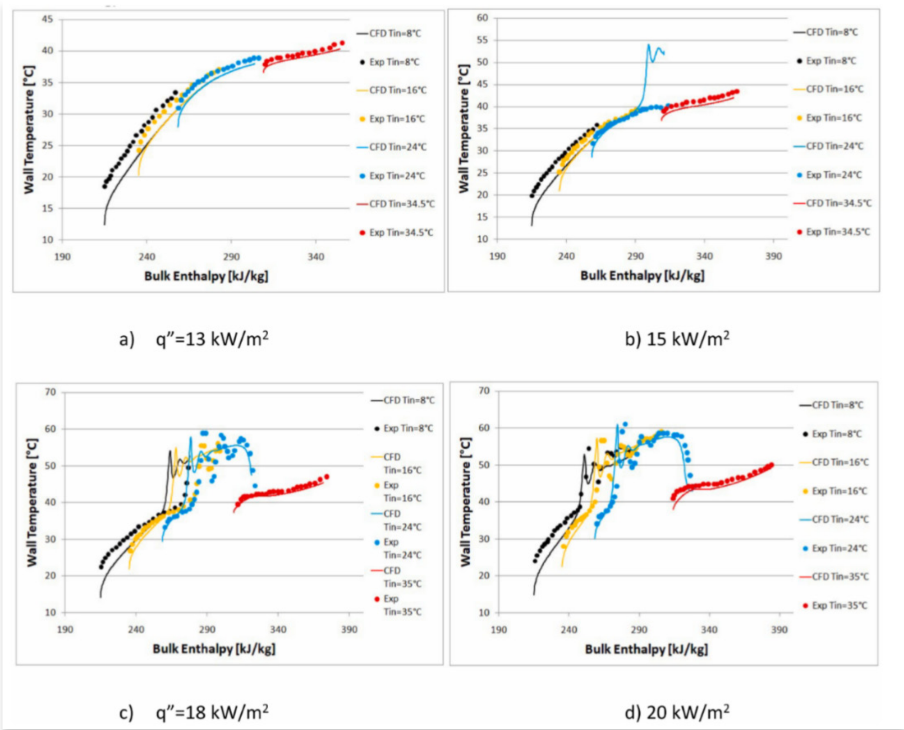
China Institute of Atomic Energy (CIAE) performed the investigation of critical flow model for supercritical pressure condition. The model is derived to calculate discharge flow rate and critical pressure based on isentropic flow and thermal equilibrium assumptions. A correction coefficient of the influence of friction and local resistance is added. The model avoids the calculation of quality and is applicable to wide range which covering the subcooled water, two-phase mixture, steam critical flow under subcritical pressure and SC-pressure. The model calculated results agree well with the experimental critical flow data under SC-pressure.

Numerical investigations have been performed by the researcher of BME NTI in 2019 in the field of SCWR TH&S. The thorium fueled SCWR concept was in the focus through a weakly coupled CFD – neutronics code-system. The thermal-hydraulics of the fuel assembly design of Th-SCWR has been simulated by the ANSYS CFX CFD code by a detailed 3D numerical model with and without wrapped wire spacer concept. The density field of SC-water (provided by the CFD results) was handed over to the MCNP Monte Carlo transport code as a boundary condition in each iteration step. The MCNP code calculated the field of heat source and this field was provided in return to the CFD code as a boundary condition. These calculations have proved that the wrapped wire spacer improves the heat transfer in most of the sub-channels within the fuel assembly and an axial and radial fuel enrichment distribution is essential for a viable fuel assembly design from the TH&S point of view. The linear heat source distribution has been optimized in the axial direction, but the maximum wall temperatures still seems to be higher than the melting temperature of currently available cladding materials. So, further optimization of the fuel enrichment is foreseen in the continuation of this research direction.

The CVR main activity on the thermo-hydraulic for Super Critical water Coolant (SCC) is based on the development and license activity that rotate in the insertion of the Super Critical Water Loop (SCWL) in the LVR-15 Reactor. For this reason, a consistent input deck of the facility was developed in ATHLET3.1A Patch1 code. From the several correlations adopted in ATHLET to simulate the SCC media, three were selected and qualified in Czech Republic: Watts-Chou, Mokry and Gupta. However, ATHLET3.1A Patch1 assessment was submitted to the thermo-hydraulic commission managed by the Regulatory State Office for Nuclear Safety (SONS) (Code and User Qualification) and it was qualified in March 2017. After the first revision of the flow regimes, all scenarios are reconsidered due to the lowering the operational pressure of the SCWL from 25 MPa to 24 MPa. The actual activity will focus in completing the flow regime scenarios according to the new specifications. Those selected scenarios analyses are used to verify the system performance in accordance with the safety criteria. A particular attention was given by providing operating regimes data in these conditions for structural analyses.

The University of Pisa developed RANS analyses of CO_2 data in 2017, making use of an Algebraic Heat Flux Model (AHFM) developed in the STAR-CCM+ code, on the basis of the Lien et al. model available in it. The RANS model is being assessed and improved on a variety of experimental data and the very systematic data by Kline offered the opportunity to understand capabilities and limitations of the improved AHFM as developed in this frame. The results, published in different steps showed a remarkable capability of the model to correctly simulate heat transfer phenomena at relatively low flow rates. In particular, the phenomenon of deteriorated heat transfer termination at the transition to gas-like fluid was observed with reasonable accuracy, as shown in **Figure SCWR 5.**

Figure SCWR 5. **Results obtained for the cases with p=8,35 MPa, ID=4,6 mm and G=300 Kg/m2s**



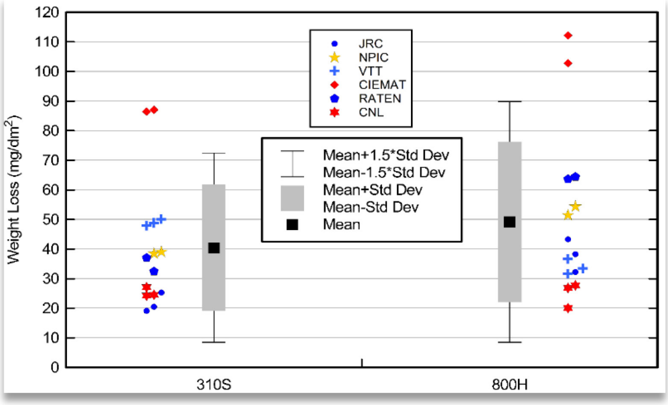
Materials and chemistry

The M&C PMB has been focusing on selection and qualification of candidate alloys for all key components in the SCWR. This includes general corrosion and stress corrosion cracking tests in autoclaves and loops as well as development work on test facilities, ionic irradiation tests of fuel cladding candidate alloy and development of novel alloys for fuel cladding. In addition, modelling of oxide film effects on fuel cladding heat transfer has been performed to better understand the interplay between general corrosion and thermal-hydraulics.

A major activity of the M&C has been the organization of a 2nd Round Robin corrosion test exercise between partners (Canada, China and Euratom) to compare the results of corrosion tests in different test facilities. Each laboratory used a standard test protocol with the coupon materials originating from the same batch and prepared by JRC-IEC. The tests were completed in 2017, and the coupons were sent to CNL for descaling. The results were reported in 2019. After 1 000 h exposure to 550°C supercritical water, coupons of Alloy 800H and Type 310S stainless steel were observed to have (descaled) weight losses of 54±26 mg/dm^2 and 41±22 mg/dm^2, respectively. Interestingly, the data were clustered by participant, with tight agreement between coupons of the same material exposed in the same facility at the same time, shown in Figure SCWR 6. It is not clear if the disagreement among participants is due to differences in flow velocity in the autoclave. It was proposed as a possible explanation.

In 2019, the Canadian materials and chemistry programme focused on expanding the set of high temperature general corrosion data, evaluating the effect of coatings on the corrosion of zirconium and titanium alloys in SC-water (500°C), and developing SC-water test facilities.

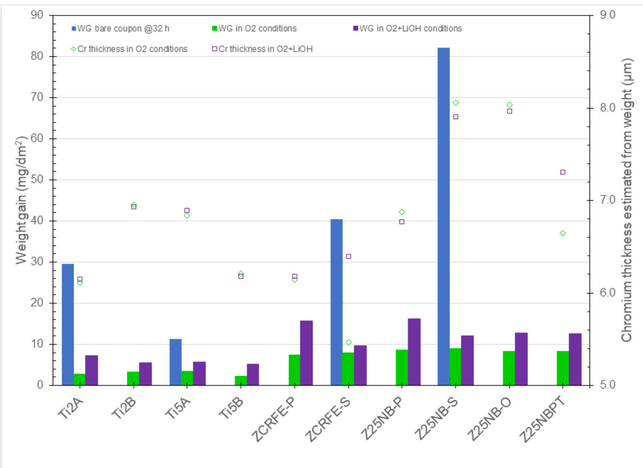
Figure SCWR 6. **Box plot of descaled weight loss data after 1 000 h exposure to deaerated water at 550°C and 25 MPa.**



Alloy 625 is a candidate fuel cladding for the benchmark Canadian SCWR concept with superior corrosion resistance. Experiments were conducted in 2019 that aimed to examine the effects of added hydrogen peroxide on the corrosion of Alloy 625 at temperatures between 650 and 700°C. These experiments also collected thermocouple data from a heated section of the loop that may be helpful in determining the effects of oxide growth on heat transfer. The collected data will be analyzed in 2020.

Corrosion testing of chromium-coated zirconium and titanium alloys was conducted in 2019 in support of the Canadian Small SCWR, a 300 MWe small modular concept. In an effort to improve neutron economy to lengthen the fuel cycle, coated zirconium alloys are being re-evaluated along with titanium alloys; the latter is an unlikely candidate given its poor neutron economy, but a Ti-50 enriched alloy would have good neutron economy and superior corrosion resistance. Coupons of Zr-1.2Cr-0.1Fe (R60804), Zr-2.5Nb (R60901 and R60904), pure titanium (R50400) and Ti-6Al-4V (R56400) were coated with a uniform layer of 10 µm chromium and exposed to 500°C oxygenated SC-water for 150 h, with and without LiOH as a pH-control agent. Weight gain measurements indicate an eight-fold improvement in corrosion resistance of coated coupons compared to the as-received alloys, shown in Figure SCWR 7. Alkaline treatment resulted in much higher weight gains than was found in pure oxygenated SC-water. Microscopic analysis of the coupons, as well as hydrogen uptake measurements, will be performed in 2020.

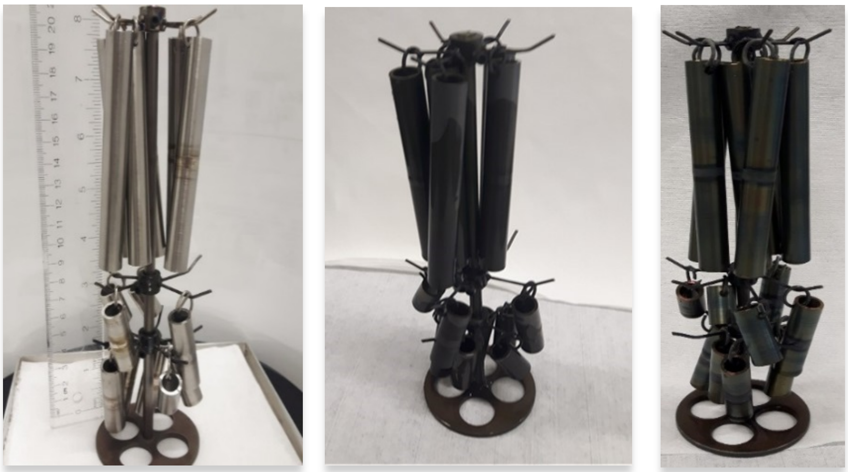
Figure SCWR 7. **Weight gain of chromium-coated zirconium and titanium alloys after exposure to 500°C SC-water for 150 h**



In addition, SC-water test facilities continued to be developed at CNL in 2019. A refreshed SC-water target system for a 2.5 MeV Van de Graaff electron accelerator was commissioned, and continues to be modified and improved. A SC-water hydrocyclone was designed with a MAWP of 29.25 MPa at 649°C (1 200°F) for future studies on activity transport and high temperature purification.

The corrosion and tensile behavior of both Alloy 625 and Alloy 800H welded specimens was examined by exposure to 575°C SC-water for 500 h at CNL. Tube specimens were butt welded autogenously by gas tungsten arc welding using a Swagelok® M200 orbital welding system. For Alloy 800H, post-exposure analysis indicated a 20% increase in the weight gain of welded specimens compared to unwelded specimens. Corrosion and aging of Alloy 800H reduced the ductility of both welded and unwelded specimens by 25% and increased the yield strength by 30%. For Alloy 625, which corrodes very little, welded specimens gained 60% more weight compared to unwelded specimens. Corrosion and aging of Alloy 625 reduced the ductility of both welded and unwelded specimens by 40% and increased the yield strength by 25%. Welded specimens that had been exposed to SC-water were observed to have 10% higher UTS and up to 15% lower ductility when compared unwelded specimens.

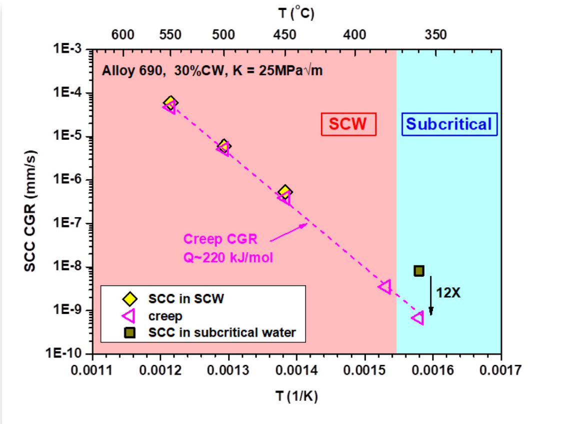
Figure SCWR 8. **Before Corrosion Test (left), 800H After 250 hours of Exposure (center) and Alloy 625 After 500 hours of exposure (Right)**



The SCC tests were conducted on 30% cold worked Alloy 690 in high-temperature pressurized water at temperatures between 360°C and 550°C in Shanghai Jiaotong University (SJTU). Creep induced cracking was measured at each testing temperature in inert argon environment to study the effect of creep on the overall crack growth behavior. The crack growth rates at each condition are summarized in **Figure SCWR 9.** Experimental results showed that creep contributed to more than 80% of the overall crack growth rate at temperatures above 450°C in SC-water, while only 8% at 360°C in subcritical water. It was clearly proved that the dominant mode of cracking from subcritical to SC-temperatures are different for cold worked Alloy 690. Corrosion induced cracking controls the crack growth rate in subcritical water environment, while creep is the major factor that dominates the cracking in SC-water.

The effect of InterGranular (IG) carbides on the cracking behavior of cold worked Alloy 690 was also studied in both subcritical and SC-water environments. The crack growth rate was lower when IG carbides were removed by prior solution annealed (SA) treatment, indicating a detrimental effect of IG carbides. The presence of IG carbides enhances the local strain accumulation at the grain boundary due to the lattice mismatch, thus promotes the crack tip strain rate and increases the crack growth rate. The SCC CGRs of Alloy 690 were compared with those of 310S SS in SC-water, and results show that the SCC CGR of the 310S SS specimen in 550°C SCW was 1.4×10^{-7} mm/s, ~17 times higher than the Alloy 690 specimen (8.3×10^{-9} mm/s) at the same testing condition. The degree of sensitization prior to and after the test was confirmed by both double loop electrochemical potentio-kinetic reactivation (DL-EPR) method and TEM analysis at the grain boundaries. It was found that the degree of sensitization increased dramatically for 310S than Alloy 690 after the SCC test, indicating severe in situ sensitization occurred in 310S during the SCW exposure.

Figure SCWR 9. **Comparison of the SCC and creep CGRs of Alloy 690 in subcritical and supercritical water at the temperatures ranging from 360°C to 550°C**



Two candidate alloys modified from 310S austenitic stainless steels were subjected to ionic irradiation to see their radiation damage effects in Nuclear Power Institute of China (NPIC). The major difference in these two alloys lies in their optimized minor alloying elements, addition of Mo, Nb, W and Ta in alloy SC1, and in SC2, Mo and Zr were added. Proton radiation tests were performed on an ion accelerator at Wuhan University, with implanting energy of 50 keV, and temperature at 290°C up to doses of 0.1 and 0.3 dpa by proton, and at 550°C up to 5, 15 and 30 dpa by Ar ion. Figure SCWR 10 shows the TEM micrographs of specimens irradiated at 550°C and Figure SCWR 11 shows the defect caused by irradiation. The irradiation tests showed that minor alloy elements added to the alloys played different roles after irradiation. At 290°C, Zr modified alloy SC2 showed lower density of void and dislocation loop defects than SC1, which contains Nb, W and Ta. However, at 550°C Zr caused void swelling in SC2 while Nb and Ta in SC1 reduced the density of voids.

Figure SCWR 10. **The TEM micrographs of ally SC1 and SC2 irradiated at 550°C**

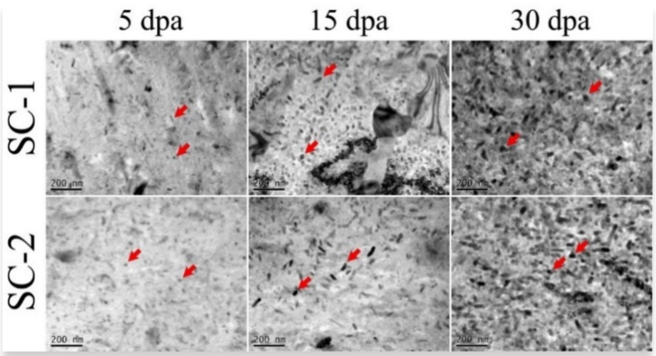
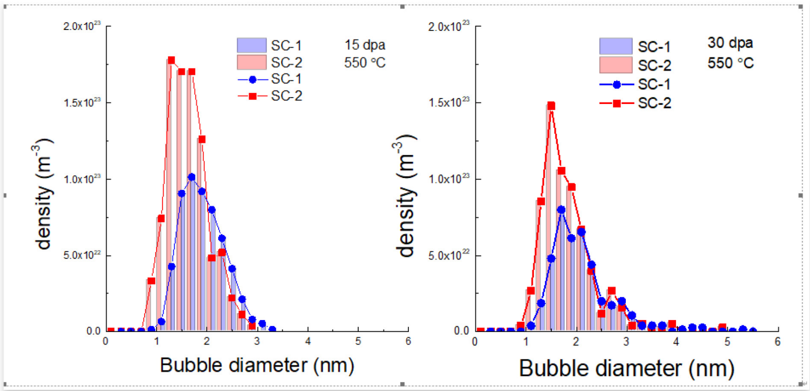
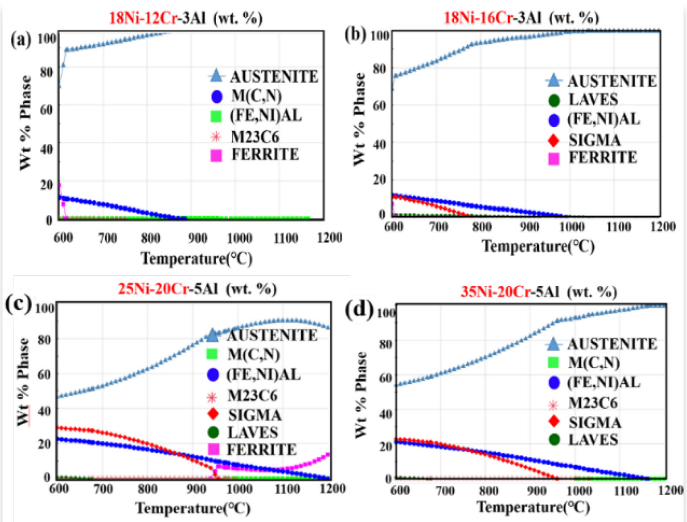


Figure SCWR 11. **The density of defects obtained by TEM of ally SC1 and SC2 irradiated at 550°C**



The major activity at the University of Science and Technology Beijing (USTB) has been focusing on developing of novel candidate alloys for cladding tube application. Alumina Forming Austenitic (AFA) alloy is proposed as a new grade SCWR candidate alloy in view of the existing reported research works, as well as results from the Round Robin corrosion test exercise between PMB partners (Canada, China and Euratom). One of the key challenges for the composition design of AFA alloys is to balance the corrosion resistance and maintain a single austenitic matrix phase for good creep strength. Figure SCWR 12 shows the matrix phases and fractional volume of precipitates in the materials with different Al% at temperatures between 600 and 1 200°C, which was calculated by using the computational thermodynamic calculation program, JMatPro. It is obvious that high Al, Cr content, but insufficient Ni content (25Ni20Cr5Al) will result in the formation of duplex $\gamma+\alpha$ matrix, as Al is a strong ferrite stabilization element. Therefore, the content of Ni should be carefully designed based on the content of Al, Cr and other minor ferritic forming elements to obtain a necessary single γ phase structure. ODS austenitic alloy is the other promising new grade material for the in-core structure application in SCWR. The microstructure stability of a 310 type ODS austenitic alloy after aging at 500°C for different length of time is investigated.

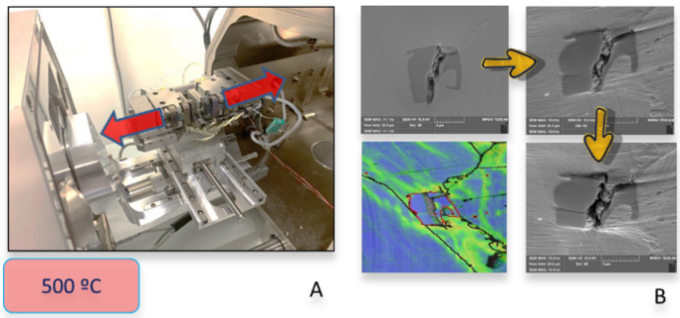
Figure SCWR 12. **Phases in AFA alloys with different Ni, Cr and Al contents based on Jmat- Pro calculations**



Along 2019 Ciemat has performed several in situ tensile tests with a nickel base alloy 690 TT previously tested in deaerated Supercritical Water (SCW) at 500°C. These tests were carried out in collaboration with CVR laboratories. The aim of this work was double, on the one hand, considering this was the first time the staff of CVR in Pilsen work with this material and this specimens geometry, the team tried to write a procedure to study the Alloy 690 TT by means in tensile in situ tests within a Scanning Electron Microscope (SEM) at high temperature (see **Figure SCWR 13**). On the other hand, the second objective was to follow the evolution of precipitates, defects, etc., in the microstructure by means the in situ test. As a result of this first attempt, both laboratories were able to follow the evolution of C,N(Ti) with temperature and strain. This test showed how the plastic deformation gathered around the carbides during the tensile test. This behavior, that has been reported by other groups, could have some meaning in the cracking processes of this material especially if the CNTi are near the surface in contact with SCW at high temperature. After these tests new tensile specimens were designed in order to avoid the problems detected during the first tests. These new specimens will be used to continue this work and to gain more in-depth knowledge into the mechanical behavior of some defects present in this alloy like the vacancies along the grain boundaries that appear as a result of cold work and high temperature. The main results from this work were presented in the EUROCORR congress that was held in Seville (Spain) and in the EPRI meeting EPRI Alloy 690/52/152 Primary Water Stress Corrosion Cracking Research Collaboration Meeting that was held in Tampa (United States).

In addition to this, Ciemat co-ordinates the European funded project MEACTOS in which there is a small task related to SCW. In this task the SCW will be used as an accelerating environment to produce stress corrosion cracks in an austenitic stainless steel 316 L type L.

Figure SCWR 13. **A) Photograph of the device used to perform the tensile in situ tests inside the chamber of a SEM: B) Evolution with temperature and stress of a C,N(Ti) found in a A690 TT specimen previously tested in SCW**



The activities at VTT in 2019 have mainly come through its participation in the EU MEACTOS project where SCW is used as accelerating method of the stress corrosion cracking processes in an austenitic stainless steel. Moreover, VTT participated in the writing of the ECC-SMART proposal for the Horizon 2020 call. In addition to this, VTT has prepared together with Aalto University a project proposal called TAMAT (Towards Advanced Materials for Energy Technologies: Multimetallic Layered Composites and Innovative Cladding Solutions for Nuclear and Beyond) to the Academy of Finland where one work package is dealing with experimental testing and oxide film modelling in supercritical water conditions. The Academy of Finland decision is expected by June 2020.

The M&C in CVR focused on microstructural evaluation of candidate materials for SCWR internals and fuel claddings. First three materials – 800H, T505 (uquivalent of T91) and 08Cr18Ni10Ti (equivalent of AISI 321) were exposed in the supercritical water loop(SCWL) in 2018. In the end of the 2019, next exposure in SCWL started with another three candidate materials (Nimonic 901, Nitronic 60 and In 718). First corrosion exposure up to 550 h and second exposure up to 1 000 h duration were carried out at 400°C/ 25 MPa with deoxygenated water, pH 6, conductivity under 2 μS/cm, Fe < 100 μg/l.

Materials from first exposure were analyzed by SEM technique in combination with EDX for chemical composition and EBSD for crystallography. Final Raman and XRD analysis confirmed compounds of magnetite oxides (Fe_3O_4) on all surfaces. No significant oxide layer occurred on 800H and 08Cr18Ni10Ti, only random not compact oxide particles. Double spinel (3-7 µm) layer occurred on T505: inner passivation layer of chromite $FeCr_2O_4$/trevorit $NiFe_2O_4$ and outer layer of magnetite Fe_3O_4. Other investigations in SCW were development works on two autoclaves with parameters: volume 137 ml, 600°C/25 MPa and volume 850 ml, 700°C/30 MPa. These autoclaves are supposed to work from 2020. One more autoclave is supposed to be developed in hot cells next year, to expose irradiated materials to SC-Water.

Yanping Huang

Chair of the SCWR SSC and all Contributors

Sodium-cooled fast reactor (SFR)

Main characteristics of the system

The primary mission for the SFR is the effective management of high-level wastes and uranium resources. If innovations to reduce capital cost and improve efficiency can be realized, the Generation IV SFR is an attractive option for electricity production. The Generation IV Technology Roadmap ranked the SFR highly for advances it offers towards sustainability goals. The fast reactor closed fuel cycle significantly improves the utilization of natural uranium, as compared to ~1% energy recovery in the current once-through fuel cycle. By recycling the plutonium and minor actinide spent fuel components, decay heat and radiotoxicity of the waste are minimized. The SFR is also highly rated for safety performance.

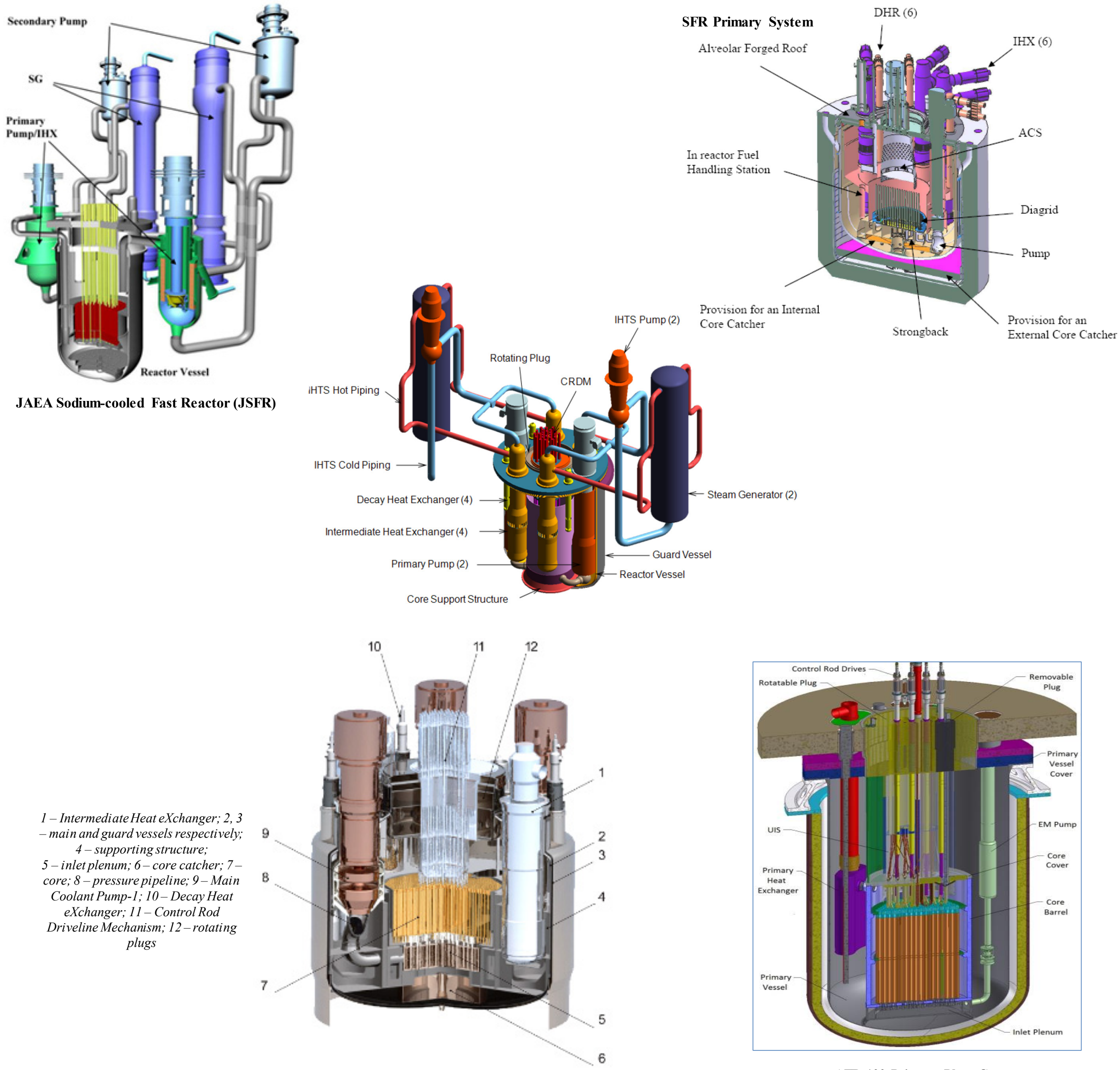
The SFR system uses liquid sodium as the reactor coolant, allowing high power density with low coolant volume fraction. Because of advantageous thermo-physical properties of sodium (high boiling point, heat of vaporization, heat capacity, and thermal conductivity) there is a significant thermal inertia in the primary coolant. While the oxygen-free environment prevents corrosion, sodium reacts chemically with air and water and requires a sealed coolant system. The primary system operates at near-atmospheric pressure with typical outlet temperatures of 500-550°C; at these conditions, austenitic and ferritic steel structural materials can be utilized, and a large margin to coolant boiling at low pressure is maintained. The reactor unit can be arranged in a pool layout or a compact loop layout. Typical design parameters of the SFR concept being developed in the framework of the Generation IV System Arrangement are summarized in **Table SFR 1**. Plant sizes ranging from small modular systems to large monolithic reactors are considered.

Table SFR 1. **Typical Design Parameters for the Generation IV SFR**

Reactor parameters	**Reference value**
Outlet Temperature	500-550°C
Pressure	~1 Atmosphere
Power Rating	50-2000 MWe
Fuel	Oxide, metal alloy, others
Cladding	Ferritic-Martensitic, ODS, others
Average Burn-up	150 GWD/MTHM
Breeding Ratio	0.5 -1.30

There are many sodium-cooled fast reactor conceptual designs that have been developed worldwide in advanced reactor development programs. In particular, the BN-800 Reactor in Russia, the European Fast Reactor in the EU, the Advanced Liquid Metal Reactor (PRISM) and Integral Fast Reactor Programs in the United States, and the Demonstration Fast Breeder Reactor in Japan, have been the basis for many SFR design studies. For Gen-IV SFR research collaboration, several *system options* that define general classes of SFR design concepts have been identified: loop configuration, pool configuration, and small modular reactors. Furthermore, within this structure several *design tracks* that vary in size, key features (e.g. fuel type) and safety approach have been identified with pre-conceptual design contributions by Gen-IV SFR Members: JSFR (Japan), KALIMER (Korea), ESFR (Euratom), BN-1200 (Russia), and AFR-100 (United States) (see **Figure SFR.1**). The Gen-IV SFR design tracks incorporate significant technology innovations to reduce SFR capital costs by a combination of configuration simplicity, advanced fuels and materials, and refined safety systems; thus, they are utilized to guide and assess the Gen-IV SFR R&D collaborations.

Figure SFR 1. **The five Gen-IV SFR design tracks**



JAEA Sodium-cooled Fast Reactor (JSFR)

3D Layout of the BN-1200 Primary System

AFR-100 Primary Plant Concept

Status of co-operation

The system arrangement for the Gen-IV international R&D collaboration for the SFR nuclear energy system became effective in 2006 and was extended for a period of ten years in 2016. Several new Members were added to the original agreement and United Kingdom was welcomed to the system arrangement in 2019. The present signatories are: Commissariat à l'énergie atomique et aux énergies alternatives, France; US Department of Energy; Joint Research Centre, Euratom; Japan Atomic Energy Agency; Ministry of Science and ICT, Korea; China National Nuclear Corporation; Rosatom, Russia; UK Department of Business, Energy and Industrial Strategy.

Based on international R&D plans, the Gen-IV SFR research activities are arranged by the SFR Signatories into four technical Projects: System Integration and Assessment (SIA), Safety and Operations (SO), Advanced Fuels (AF) and Component Design and Balance-of-Plant (CDBOP).

Three Project Arrangements (PAs) were signed in 2007: Advanced Fuel (AF), Component Design and Balance-of-Plant (CDBOP), and Global Actinide Cycle International Demonstration (GACID). The PA for Safety and Operation (SO) was signed in 2009, and the PA for System Integration and Arrangement (SIA) was signed in 2014. The Project Arrangements were agreed for a ten-year term with annual updates of the Member contributions. The PA for AF and the PA for GACID expired in 2017. A new PA (Phase II) for AF for next ten years was entered into force in 2018. The PA for CDBOP and SO extended for another ten years in 2017 and 2019 respectively.

R&D objectives

SFR designs rely heavily on technologies already developed and demonstrated for sodium-cooled reactors and associated fuel cycle facilities that have successfully been built and operated in several countries. Overall, approximately 400 reactor years of operating experience have been logged on SFRs, including 300 years on smaller test reactors and 100 years on larger demonstration or prototype reactors. Significant SFR research and development programs have been conducted in the United States, Russia, Japan, France, India[1] and the United Kingdom. The only SFR power reactors in operation are the BN-600 (Russia) which has reliably operated since 1980 with a 75% capacity factor, and the BN-800 which started commercial operations in 2016. Currently operating test reactors include BOR-60 (Russia), and CEFR (China). The JOYO (Japan) test reactor is in licensing process for restart. New SFR test reactors MBIR (Russia) and VTR (United States) are expected in the next decade. In addition, SFR technology development programs are being pursued by all members of the GIF SFR System Arrangement.

A major benefit of previous investments in SFR technology is that the majority of the R&D needs that remain for the SFR reactor technology are related to performance rather than viability of the system. Accordingly, the Generation IV collaborative R&D focuses on a variety of design innovations for actinide management, improved SFR economics, development of recycle fuels, in-service inspection and repair, and verification of favorable safety performance.

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. The Generation IV SFR system options and design tracks are identified and assessed with respect to Generation IV goals and objectives. Results from the technical R&D projects will be evaluated and integrated to assure consistency.

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 and demonstrating minor actinide-bearing (MA-bearing) high burn-up fuel for SFRs. 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 ODS steels.

1. India is not belonging to GIF.

Component design and balance-of-plant project (CD&BOP): The project includes the development of advanced Energy Conversion Systems (ECS) to improve thermal efficiency and reduce secondary system capital costs. The project also include R&D on advanced in-service inspection and repair (in sodium) technologies, small sodium leak consequences, and new sodium testing capabilities. 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 ECS based on a Brayton cycle with supercritical carbon dioxide or nitrogen as the working fluid. In addition, the significance of the experience that has been gained from SFR operation and upgrading is shared.

Main activities and outcomes

In this Section, recent Member contributions to the Gen-IV SFR collaboration are highlighted.

System integration and assessment (SIA) Project: In 2019, five trade and assessment studies were contributed. CIAE contributed a study that evaluates the CFR1200 design main heat transfer parameters. Key factors that significantly influence the thermal performance were identified (e.g. primary/secondary circuit temperatures). They performed sensitivity analyses for these main factors and quantified the impacts on system efficiency and component design.

During the design phases of the ASTRID demonstrator (2010-2019), CEA continuously assessed and improved the design of ASTRID to enhance its safety. It is a good example of how SFR safety can be improved by design with a core showing favorable natural behavior under multi-failure accident conditions, and with added devoted complementary safety devices to prevent or mitigate severe accidents. In 2019, based on the ASTRID design evolutions, CEA is providing its feedback experience on the SFR safety enhancement by design.

Within the ESFR-SMART project, various safety improvements for ESFR have been proposed, taking into account the safety objectives envisaged for Gen-IV reactors and the recommendations following the Fukushima Daiichi accident. The Euratom contributions provide overviews of the improved ESFR safety approach including the safety requirements for the evaluation of the innovative design options, the assessments of the proposed system safety measures and recommendations for further developments. Safety approaches assessments were performed using the GIF RSWG ISAM methodology relevant tools: Qualitative Safety features Review (QSR) and Objective Provision Tree (OPT). The focus of the contribution for this year is the use of ISAM QSR including a short description of the QSR approach for SFR and a checklist of recommendations developed for a generic SFR concept. The contribution discusses the assessment of the checklist for the ESFR-SMART considering compliance with defence-in-depth, safety objectives, ALARA principle and need for harmonization of safety and security architecture. Included are recommendations and conclusions on the QSR application.

JAEA contributed a study on countermeasures against sodium-water reactions. A single tube helical coil steam generator was evaluated as a design alternative to the JSFR double-wall type. Failure propagation and leak detection behavior was compared for the two concepts. Future work will include a detailed evaluation of sodium-water pressure and system impacts.

KAERI performed a deployment scenario study of large size TRU burners to estimate spent fuel accumulation from PWR operation and to evaluate radiotoxicity reduction of spent fuels by introducing TRU burners. The spent fuel accumulation from PWR operation was estimated based on domestic plans for long-term electricity demand and supply. The spent fuel accumulation of TRU recycle was compared to that of direct disposal, and the radiotoxicity of finally disposed high-level wastes of TRU recycle reaches natural uranium level after about 5 000 years.

Safety and Operations Project: As the topic of the SO project, the common project that consists of two benchmark analyses (EBR-II test and PHÉNIX Dissymmetric tests) have been started in the SO project last quarter of 2019. The first phase of the benchmark analysis (“blind phase”) will take two years.

The SO project is structured in three work packages (WPs): 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”.

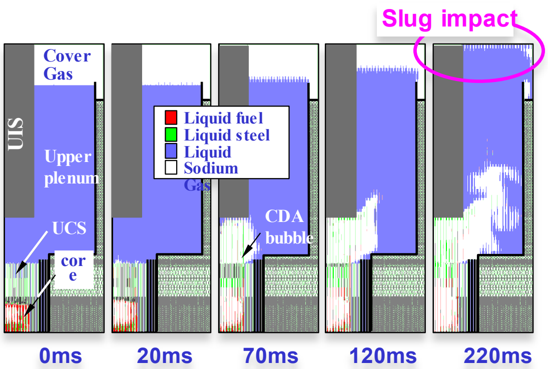
WP SO 1: Methods, models and codes

CIAE conducted benchmark analysis for EBR-II Shutdown heat removal tests SHRT-17 and SHRT-45R, as a part of joint work with ANL from 2017 to 2019.

In order to improve the simulation of an Unprotected Loss of Flow transient, CEA studied the way to calculate the reactivity coefficients to be used in the system code point kinetics module. MACARENa and APOLLO3 Codes were used for this study. As a result, it was confirmed that the point kinetics parameters (in particular the sodium-void reactivity worth) were strongly affected by 3D angular effects. Meanwhile, the transient simulation results were not very different. Hence, the most significant progress is probably to be found in improving the neutronic/thermo-hydraulic coupling. In order to demonstrate that a severe accident is sufficiently unlikely, CEA provided the deliverable which presents the safety demonstration methodology, measurement systems, and reactor protection sub-systems corresponding with each core meltdown initiating events.

JAEA developed the evaluation method of the consequence of energetics in Post-Disassembly Expansion (PDE) phase during an unprotected loss-of-flow accident. They performed the preliminary evaluation of mechanical energy and reactor vessel response using the developed method, see **Figure SFR 2**. As a result, no Slug Impact nor residual strain of the reactor vessel was predicted in the case of realistic temperature condition. Therefore, they obtained the perspective of the robustness of prototype SFR against the energetics in severe accident conditions.

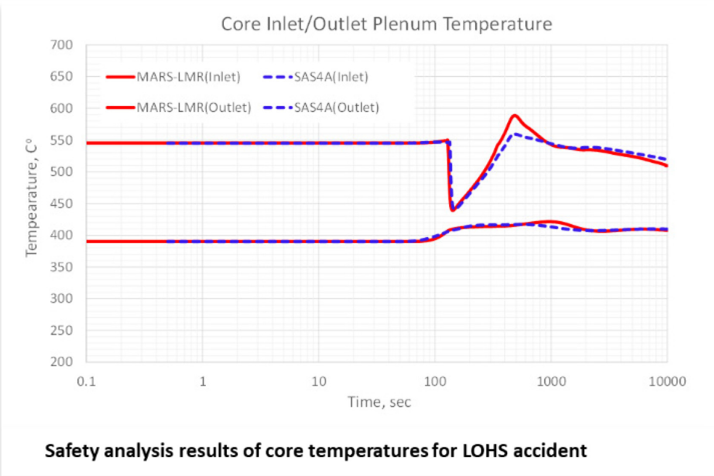
Figure SFR 2. **Material distribution calculated by JAEA (pessimistic temperature condition)**



In order to obtain licensing approval for the developed code (MARS-LMR), KAERI carried out the comparative safety analysis with SAS4A/SASSYS-1 code. In **Figure SFR 3**, sample results from the comparison of the Transient Over Power, Loss Of Flow, and Loss Of Heat Sink cases are shown. The safety analysis results from each code were found to be in good agreement.

IPPE (Rosatom) continued developing 3D severe accident analysis code COREMELT3D. The 3D model of the reactor gas system (from the gas volume under the sodium level in the reactor through the expansion tank up to the ventilation system) has been developed and implemented into the code. This model has been integrated with the primary circuit 3D thermo-hydraulic model, it is necessary for simulating transport of gaseous fission products from disintegrated fuel pins up to ventilation system, and consequently into the environment. IPPE has performed integral analysis of the consequences of the severe accidents in BN-1200. There have been used the following codes: COREMELT3D (core, primary and intermediate circulation loops, emergency system of heat removal, reactor gas system), KUPOL-BR (ventilation system), VYBROS-BN (transport of radioactive products in the environment under different meteorological conditions, doses). IPPE has performed preliminary experiments with thermite compositions to obtain melt of stainless steel with high temperature. This technique will be used in a facility (which is being designed now) to simulate transport of melt in the SFR conditions.

Figure SFR 3. **Comparison between MARS-LMR and SAS4A/SASSYS-1 codes on Loss Of Flow case**



WP SO 2: Experimental programmes and operational experiences

CIAE conducted the experimental research and the code development for Heat Transfer Analysis of CEFR Damaged Spent Fuel Assemblies in Closed Space. The experiment simulated the spent fuel assemblies during transportation and the heat transfer characteristics were investigated.

The Project ESFR-SMART aims to evaluate the safety of a low-void Sodium Fast Reactor (SFR) core design, in particular the analysis of an unprotected loss-of-flow (ULOF) accident. Recent studies on the low-void SFR core show the occurrence of a stabilized chugging sodium boiling regime that can be classified as a new safety measure acting as a level of defence preventing severe accidents. In order to better understand and simulate the chugging boiling regime condition and to gather new experimental data, the ESFR-SMART project envisaged the construction of a new simple facility named CHUG (see **Figure SFR 4)**, designed using water as simulant. The Euratom contribution describes the pre-test calculation results, as well as the facility layout for the first phase of the test, including the main parts and the instrumentation. Preliminary results and main outcomes of first phase of experiments are summarized. Results of analytical simulations of the experiment conducted using the thermal hydraulics code TRACE to assess the validity of the code for the simulation of chugging boiling are shared.

Figure SFR 4. **Layout of the CHUG facility**

Euratom discusses design guidelines for sodium loops. Using liquid sodium at high temperatures in test facilities requires defining rules specific to this technology to ensure that operations are safe and reliable. The purpose of this contribution is to explain the safety rules to be incorporated by the designer during the definition of a project to build a facility implementing liquid sodium. The recommendations take into account European feedback on safety issues related to the design of sodium facilities. However, they do not under any circumstances replace the regulations in force applicable to each subject discussed.

WP SO 3: Studies of innovative design and safety systems

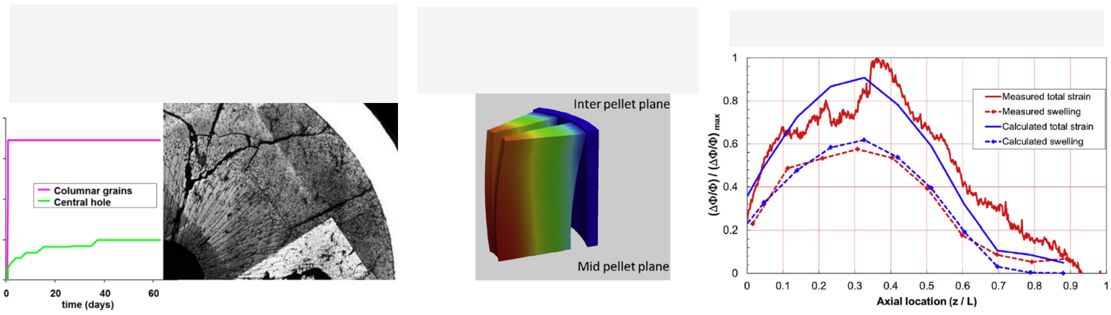
In order to confirm the applicability of Self-Actuated Shutdown System (SASS) in conditions of low power operation, JAEA carried out design modification and 3D thermal-hydraulic analysis and calculated the response time of SASS. By introducing the improvement of design and the required temperature difference, core damage was prevented by the SASS in the case of LOF type ATWS event from low power operation.

Advanced Fuels Project: The AF project consists of three work packages: WP 2.1 "SFR Non-MA-bearing Driver Fuel Evaluation, Opt. & Demo.", WP 2.2 "MA-bearing Transmutation Fuel Evaluation, Opt. & Demo." and WP 2.3 "High-burn-up Fuel Evaluation, Opt. & Demo."

WP 2.1: SFR Non-MA-bearing Driver Fuel Evaluation, Optimization & Demonstration

CEA presented the current capabilities of the GERMINAL fuel performance code, part of the PLEIADES simulation platform, used for (U, Pu) mixed oxide fuel pins calculations. The modelling of GERMINAL and its validation by comparison between calculations and measurements have been shared (see **Figure SFR 5**).

Figure SFR 5. **Examples of calculations and validation studies for GERMINAL code**



DOE continued to develop simulation tools for the evaluation of metallic fuel performance. Additional models were added and improved in the BISON fuel performance code to enhance its ability to model both U-Pu-Zr and MOX fuel for sodium fast reactors. Also, DOE successfully fabricated novel geometries of Pu bearing metal fuel to support accelerated testing. This includes both small diameter samples needed for high fission rate testing as well as more complex fuel geometries that can explore alternative methods to accommodate swelling that eliminate the need for sodium bonding.

JAEA measured the physical properties of non-stoichiometric (U, Pu)O_2 as function of Pu content, Am content, O/M ratio and temperature.

WP 2.2: MA-bearing Transmutation Fuel Evaluation, Optimization & Demonstration

CEA has performed a preliminary thermo-mechanical design of a MA-bearing oxide pin loaded with 10% of americium in UO_2 matrix. The behavior of the pin has been calculated with GERMINAL fuel performance code with specific developments for MA-bearing fuels.

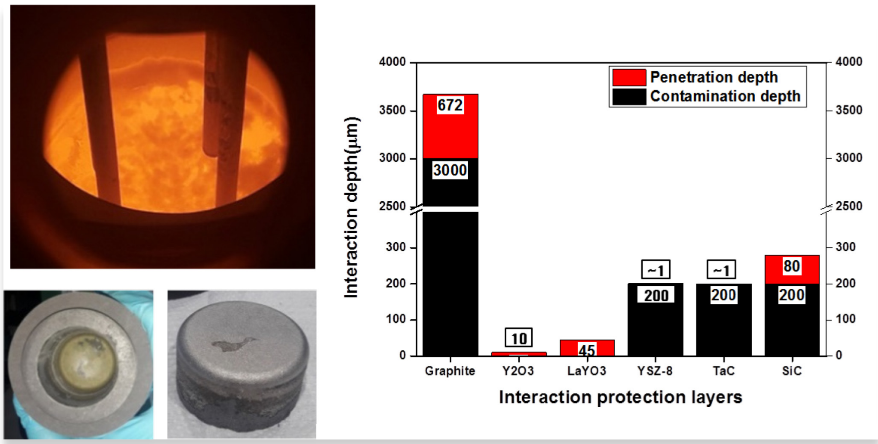
DOE investigated the effect of minor actinide additions to metallic fuel through post-irradiation examination (PIE) and micro-chemical analysis of minor actinide bearing transmutation fuel. Minor actinide bearing fuel irradiated in several different reactors including true fast spectrum reactors (EBR-II, Phenix) and pseudo-fast spectrum tests in the Idaho National Laboratory Advanced Test Reactor were all compared. The fuel performance across these different conditions was fundamentally the same, and the addition of minor actinides did not significantly change the performance of the fuel. The "Metallic Fuel Handbook" which documents the fundamental thermo-physical properties of metallic fuel alloys and their constituents was updated with a significant revision to the U-Zr and U-Pu-Zr system information.

Euratom JRC installed the Cold Finger Apparatus (KüFA) installed into the JRC hot cell facilities for out-of-pile safety transient testing. A temperature transient up to 1 800°C will be applied and the release of gaseous and solid fission products will be quantitatively determined as a function of time. Additionally, JRC studied on Synthesis of Am-bearing MOX fuel for the homogeneous recycling concept ($(U,Pu,Am)O_2$), containing circa 5% americium, 20% plutonium and 75% uranium. The proposed synthesis method synthesize $(U,Th)O_2$ nanopowder (particle size is about 5 nm).

JAEA evaluated effects of Am on MOX fuel temperature using an irradiation behavior analysis code, it was suggested Am-MOX fuel could be irradiated with the same conditions as conventional MOX fuel.

KAERI completed fuel rod fabrication for the 2nd Fuel irradiation test in HANARO (SMIRP-2 test), which will be started from 2020. They also conducted the development of reusable crucible and mold for metal fuel fabrication, see Figure 3.10. In order to reuse casting parts, various new materials were tested for the coating material, which confirmed the effectiveness of the Y_2O_3 coating.

Figure SFR 6. **Development of reusable crucible and mold**



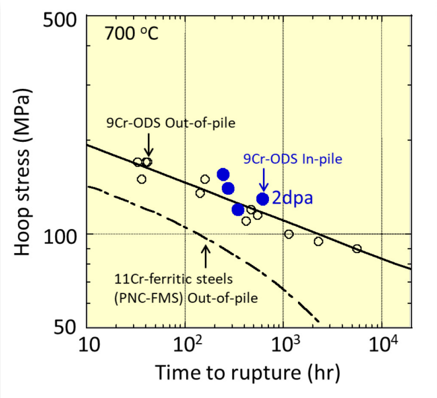
Rosatom created the experimental installation of low temperature sintering of mixed nitride uranium and plutonium fuel by the HVPC-process. Through the test using titanium nitride, they confirmed good repeatability of the HVPC-process. And they started the testing of the installation via uranium nitride.

WP 2.3: High-burn-up Fuel Evaluation, Optimization & Demonstration

CIAE updated the oxide fuel performance code, FIBER, to analyze up to 10at% burn-up fuel. They conducted the verification of the FIBER code and the benchmark analysis with the past CEFR calculation.

JAEA has developed 9Cr-ODS tempered martensitic steel (TMS) as prospective material for high burn-up fuel cladding tube. JAEA confirmed that there was no remarkable degradation in the in-pile creep mechanical strength of 9Cr-ODS TMS cladding tubes irradiated in JOYO (see **Figure SFR 7**). Furthermore, on the basis of knowledge on 9Cr-ODS TMS having prominent mechanical strength and irradiation resistance, JAEA started developing a new type of high Cr-ODS TMS: the 11Cr-ODS TMS for improving corrosion resistance.

Figure SFR 7. **Comparison of In-pile and Out-of-pile creep rupture strength of 9Cr-ODS TMS**



KAERI conducted the parametric study and sample manufacturing for Cr electroplating for nuclear cladding applications. And they also conducted several performance tests (Out-of pile diffusion couple test, mechanical test).

Rosatom have planned the Post-Irradiation Examination (PIE) for the specimens made of EP823 ODS steel in order to increase the burnout level of the fuel. Rosatom fabricated the samples for this PIE and conducted the pre-reactor tests. Additionally, the first stage of irradiation was completed and they are conducting the PIE.

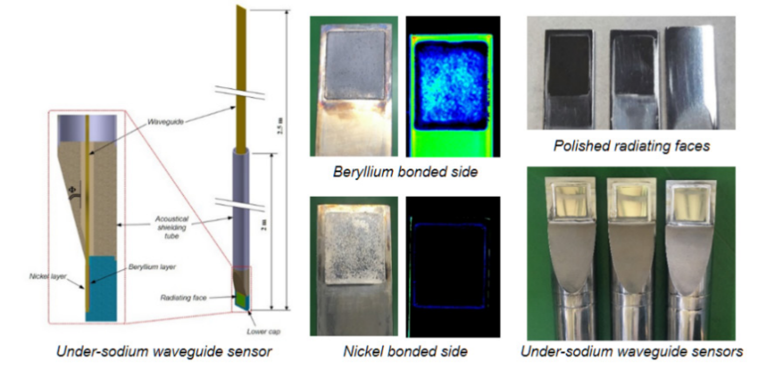
Component Design and Balance-Of-Plant Project: Activities within the CD&BOP project include experimental and analytical evaluation of advanced In-Service Inspection, Instrumentation & Repair technologies (ISI&R), development of Advanced Energy Conversion Systems (AECS), study of sodium Leakages and Consequences (SL), advanced Steam Generator technologies (SG), and study of sodium Operation technology and new sodium Testing Facilities (O&TF).

ISI&R technologies

CEA have studied the capability of the Leaky Lamb Waves on the non-destructive testing from outside of the main vessel. They developed the model to represent the behavior of Leaky Lamb Waves in plates, and validated it by comparing with the literature results. Additionally, for the further validation, they prepared experimental devices consisting of immersed plates, emitter and receiver.

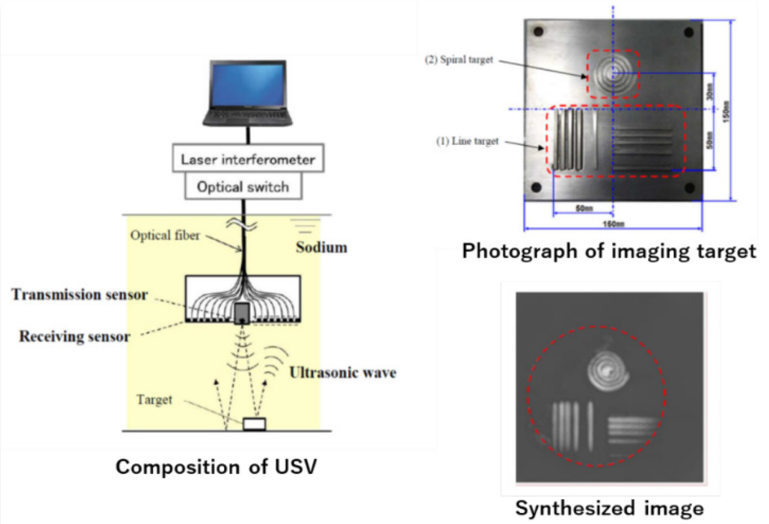
KAERI investigated the sodium-wetting property of the plate-type ultrasonic waveguide sensor under various conditions to improve the sensor performance. KAERI fabricated waveguide sensors with different surface roughness of radiating faces, and prepared a sodium-wetting test facility. They also constructed a new sodium test facility for the verification of under sodium viewing and ranging capabilities of the plate-type ultrasonic waveguide sensor as shown in Figure SFR.8.

Figure SFR 8. **Waveguide sensors fabricated for sodium-wetting tests**



JAEA developed an improved imaging under sodium viewer for a middle distance (see Figure SFR.9). The transmission sensor provided better profile of the wave, and the receiving sensor successfully reduced the noise of the wave profile. The imaging experiment in water showed that higher resolution can be obtained through the improvement of the imaging under sodium viewer.

Figure SFR 9. **Imaging experiment in water for the improved imaging under sodium viewer**

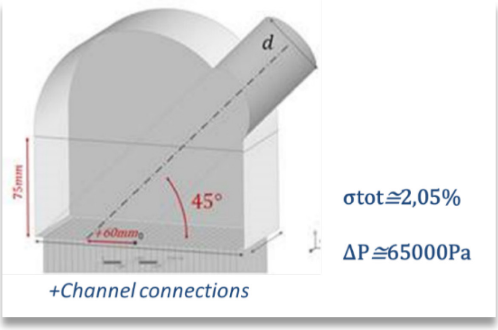


Supercritical CO_2 Brayton cycle

CEA is developing the method to detect the bubbles in the sodium flow by using eddy current flowmeter (ECFM). In 2019, CEA proposed a numerical representation of an ECFM and a bubble. Through the comparison study with experimental result, it was confirmed that this method can detect the bubble effectively.

CEA also conducted the parametric study and the optimization of the design of the heat exchanger's header. Their CFD models validated by the experimental result was used for this optimization. As the result of the optimization of the header and the channels bundle (see **Figure SFR 10**), the maldistribution level was reduced from 25% to about 2%, as compared to the design objective of 5%.

Figure SFR 10. **Design of heat exchanger's optimized header**



Sodium leakages and consequences

In 2019, no specific activity was conducted in this work package.

Steam generators

JAEA studied on the heat transfer coefficient inside tube in case of sodium-water reaction. The overheating tube rupture is one of the considerable failure mode derived from the sodium-water interaction. For the evaluation of the possibility of this mode, heat transfer coefficient of water side is important factor. JAEA conducted the rapid heating experiments for the tube containing water flow and estimated the heat transfer characteristics of inner surface. Based on this experimental results, correlations for RELAP5 code was modified and their conservativeness were confirmed.

KAERI has upgraded the signals analysis software as well as the combined SG tube inspection sensor and signal acquisition device. The upgraded software newly employs several signal transformation functions for MFL image processing, and an automatic defect detection algorithm. They conducted performance tests of the upgraded prototype combined SG tube inspection sensor system, and confirmed its defect detection performance.

Sodium operation technology and new sodium testing facilities

KAERI has completed the installation of the STELLA-2 test section in 2019, and remaining works for a cold test will be finished in early 2020. The first test data for sodium integral effect test using STELLA-2 will be obtained no later than the end of 2020 as well. Besides the sodium thermal-hydraulic test program, KAERI is constructing new test facilities for sodium leak detection and simulation, which are called WALSUM (Water-mock-up test for Advanced Leak Simulation and Upgraded Monitoring system) and SELAAD (Sodium Experimental Loop for Advanced Aerosol Detection). The objectives of the new facilities are to develop highly reliable sodium leak detection and monitoring system as well as performance evaluation of advanced sodium leak detectors.

Bob Hill

Chair of the SFR SSC and all Contributors

Very-high-temperature reactor (VHTR)

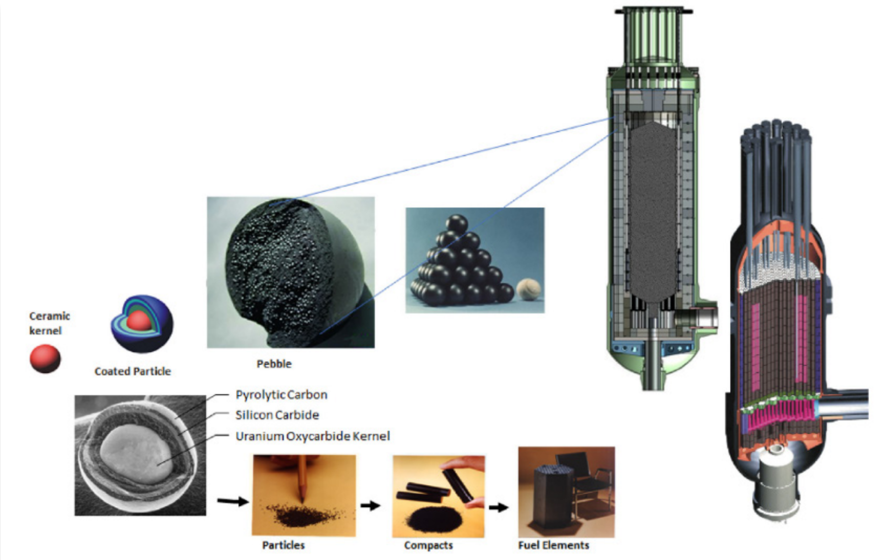
Main characteristics of the system

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

Use of helium as coolant and ceramics as core structure material allows operation temperature at core outlet of 850°C or above allowing for hydrogen production using processes with no greenhouse gas emission, such as thermo-chemical cycles (Sulphur-Iodine process) or High-Temperature Steam Electrolysis (HTSE). Beyond electricity generation and hydrogen production, high-temperature reactors can provide process heat for use in other industries, substituting fossil fuel applications

As previously noted, the basic technology for the VHTR has been established in former high-temperature gas-cooled reactors such as the US Peach Bottom and Fort Saint-Vrain plants as well as the German AVR and THTR prototypes, also the test reactors HTTR in Japan and HTR-10 in China. These reactors represent the two baseline concepts for the VHTR core: the prismatic block-type and the pebble bed-type (see **Figure VHTR 1**).

Figure VHTR 1. **TRISO coated-particle fuel as the basis for hexagonal block and pebble bed core designs**



The fuel cycle will initially be once-through with low-enriched uranium fuel and very-high-fuel burn-up, with plutonium or thorium-based fuels as alternatives. Solutions need to be developed to adequately manage the back-end of the fuel cycle. The potential for a closed fuel cycle needs to be fully established. Although various fuel designs are considered within the VHTR systems, all concepts exhibit extensive similarities allowing for a coherent R&D approach, as the TRISO coated-particle fuel form is the common denominator for all. This fuel consists of small particles of nuclear material, surrounded by a porous carbon buffer, coated with three layers: pyro-carbon/silicon carbide/pyro-carbon. These coatings represent the first barrier against fission product release under normal operation and accident conditions.

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

In the current projects of VHTR, the electric power conversion unit is an indirect Rankine cycle applying the latest technology of conventional power plants, as this technology is available. Direct helium gas turbine or indirect (gas mixture turbine) Brayton-type cycles can be envisaged in the longer term.

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

The technology is being advanced through near and medium-term projects, such as HTR-PM, NGNP, GT-MHR, NHDD, and GTHTR300C, led by several plant vendors and national laboratories respectively in China, the United States, Korea and Japan. The construction of the HTR-PM demonstration plant (two pebble bed reactor modules with one super heated steam turbine generating 210 MWe) is currently being finalized. Each reactor module has a thermal power of 250 MWth. The coolant gas temperature will be 750°C, which represents the current state of the art for materials and the requirement of high-temperature steam generation. High quality steam of 566°C from either reactor will be fed into a common steam header and turbo generator set. The HTR-PM demonstration plant will be connected to the grid in 2020, representing a major step towards a Generation IV demonstration plant.

Status of co-operation

The VHTR system arrangement was signed in November 2006 by Canada, Euratom, France, Japan, Korea, Switzerland and the United States. In October 2008, China formally signed the VHTR SA during the Policy Group meeting held in Beijing. South Africa, formally acceded to the GIF Framework Agreement in 2008, but announced in December 2011 that it no longer intends to accede to the VHTR SA. Canada withdrew from the SA at the end of 2012 but is again an observer and remained active in the Hydrogen Production Project. The new members of the system arrangement was subsequently signed by Australia (December 2017) and the United Kingdom (January 2019).

The fuel and fuel cycle project arrangement became effective on 30 January 2008, with implementing agents from Euratom, France, Japan, Korea and the United States. The project arrangement has been extended to include input from China and was amended in 2013. The project was extended in 2018 for a period of ten years.

Although the term of the original Materials Project Plan (PP) was completed in 2012, the Materials Project Arrangement (PA) continued through 2019 under its 1st amendment, which added China as a Signatory, while simultaneously pursuing a 2nd amendment that would incorporate a new PP for activities from 2018-2022 and add Australia as another Signatory. Contributions to the new PP for 2018-2022 were developed by the current seven Signatories (China, European Union, France, Japan, Korea, Switzerland, and United States), as well as Australia, which will be joining the PA. This 2nd amendment of the PA (incorporating the new PP and Australia) was approved by the SSC in April 2019 and was distributed by NEA for signature on 20 November 2019.

The hydrogen production PA became effective on March 2008 with implementing agents from Canada, France, Japan, Korea, the United States and Euratom. In 2019, the forthcoming five-year Project Plan was prepared to incorporate Chinese contributions and other countries' updated contributions. The finalized Project Plan is expected in early 2020.

The computational methods validation and benchmarks (CMVB) PA remained provisional. In 2019, detailed discussions on finalizing a multi-year work plan were performed by the participants. The PA is now ready for final approval by the VHTR SSC.

R&D objectives

Even if the VHTR development is mainly driven by the achievement of very-high-temperatures providing higher thermal efficiency for new applications, other important topics are driving the current R&D: demonstration of inherent safety features and high fuel performance (temperature, burn-up), coupling with process heat applications, cogeneration of heat and power, and the resolution of potential conflicts between those challenging R&D goals.

The VHTR system research plan describes the R&D programme to establish the basic technology of the VHTR system. As such, it is intended to cover the needs of the viability and performance phases of the development plan described in the Generation IV Technology Roadmap and in the GIF R&D Outlook (2018 Update). From the six projects outlined in the VHTR SRP, three are effective, and one is provisional, as discussed below:

- Fuel and fuel cycle (FFC) investigations are focusing on the performance of the TRISO coated particles, which are the basic fuel concept for the VHTR. R&D aims to increase the understanding of standard design (UO_2 kernels with SiC/PyC coating) and examine the use of uranium-oxicarbide UCO kernels and ZrC coatings for enhanced burn-up capability, best fission product confinement and increased resistance to core heat-up accidents (above 1 600°C). This work involves fuel characterization, post-irradiation examination, safety testing, fission product release evaluation, as well as assessment of chemical and thermo-mechanical materials properties in representative service and accident conditions. The R&D also addresses spent fuel treatment and disposal, including used-graphite management, as well as the deep burn of plutonium and minor actinides (MA) in support of a closed cycle.
- Materials (MAT) development and qualification, design codes and standards, as well as manufacturing methodologies, are essential for the VHTR system development. Primary challenges for VHTR structural materials are irradiation-induced and/or time-dependent failure and microstructural instability in the operating environments. For core coolant outlet temperatures up to around 950°C, it is envisioned to use existing materials; however, the stretch goal of 1 000°C, including safe operation under off-normal conditions and involving corrosive process fluids, requires the development and qualification of new materials. Improved multi-scale modelling is needed to support inelastic finite element design analyses. In addition to other high-temperature heat exchangers, additional attention is being paid to the metal performance in steam generators, which reflects the current interest in steam-based process applications at somewhat lower core outlet temperature of 750 to 850°C. Structural materials are considered in three categories: graphite (for core structures, fuel matrix, etc.), very/medium-high-temperature metals, and ceramics & composites. A materials handbook has been developed and is being used to efficiently store and manage VHTR data, facilitate international R&D co-ordination, and support modelling to predict damage and lifetime assessment.
- For hydrogen production (HP), two main processes for splitting water were originally considered: the sulphur/iodine thermo-chemical cycle and the high-temperature steam electrolysis process. Evaluation of additional cycles has resulted in focused interest on two additional cycles with lower temperature requirements: the hybrid copper-chlorine thermo-chemical cycle and the hybrid sulphur cycle. R&D efforts in this PMB address feasibility, optimization, efficiency and economics evaluation for small and large-scale hydrogen production. Performance and optimization of the processes will be assessed through integrated test loops, from laboratory scale through pilot and demonstration scale, and include component development such as advanced process heat exchangers.

Hydrogen process coupling technology with the nuclear reactor will also be investigated and design-associated risk analysis will be performed covering potential interactions between nuclear and non-nuclear systems. Thermo-chemical or hybrid cycles are examined in terms of technical and economic feasibility in dedicated or cogeneration hydrogen production modes, aiming to lower operating temperature requirements in order to make them compatible also with other Generation IV nuclear reactor systems dealing with a lower temperature range.

- Computational Methods Validation and Benchmarks (CMVB) in the areas of thermal-hydraulics, thermal-mechanics, core physics, and chemical transport are major activities needed for the assessment of the reactor performance in normal, upset and accident conditions and for licensing. Codes validation needs to be carried out through benchmark tests and code-to-code comparison, from basic phenomena to integrated experiments, supported by HTTR and HTR-10 tests or by past high-temperature reactor data (e.g. AVR, THTR and Fort Saint-Vrain). Improved computational methods will also facilitate the elimination of unnecessary design conservatisms and improve construction cost estimates.
- Even though it is not currently implemented, the development of components needs to be addressed for the key reactor systems (core structures, absorber rods, core barrel, pressure vessel, etc.) and for the energy conversion or coupling processes (such as steam generators, heat exchangers, hot ducts, valves, instrumentation and turbo machinery). Some components will require advances in manufacturing and on-site construction techniques, including new welding and post-weld heat treatment techniques. Such components will also need to be tested in dedicated large-scale helium test loops, capable of simulating normal and off-normal events. The project on components should address development needs that are in part common to those of the GFR, so that common R&D could be envisioned for specific requirements, when identified.

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

Milestones

In the near term, lower-temperature demonstration projects (from 700°C to 950°C) are being pursued to meet the needs of current industries interested in early applications. Future operation at higher temperatures (1 000°C and above) requires development of high-temperature alloys, qualification of new graphite types and development of composite ceramic materials. Lower temperature version of VHTR (from 700°C to 950°C) will enter the demonstration phase around 2020, based on HTR-PM experience in China which is scheduled to operate in 2020. A future higher temperature version (1 000°C and above) will require more research.

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 pyro-carbon (PyC), silicon carbide (SiC), then PyC – could evolve along with the improvement of its performance through the use of a uranium oxycarbide (UCO) kernel or a zirconium carbide (ZrC) coating for enhanced burn-up capability, minimized fission product release, and increased resistance to core heat-up accidents (above 1 600°C). Fuel characterization work, post-irradiation examinations (PIE), safety testing, fission product release evaluation, as well as the measurement of chemical and thermo-mechanical material properties in representative conditions will feed a fuel material data base. Further development of physical models enables assessment of in-pile fuel behavior under normal and off normal conditions.

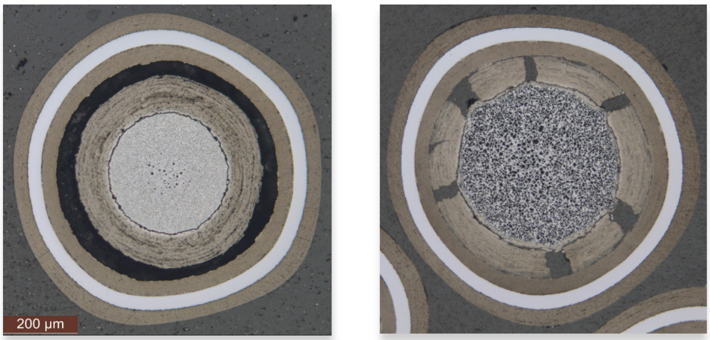
Fuel cycle back-end encompasses spent fuel treatment and disposal, as well as used graphite management. An optimized approach for dealing with the graphite needs to be defined. Although a once-through cycle is envisioned initially, the potential for deep burn of plutonium and minor actinides in a VHTR, as well as the use of thorium-based fuels, will be accounted for as an evolution towards a closed cycle. Recent activity in the various work packages is summarized below.

Irradiation and PIE

This work package includes the activities of fuel irradiation testing, PIE facility and equipment development, and post-irradiation examination of fuel specimens. Work in China has continued to develop domestic fuel post-irradiation examination capabilities. This includes hot cells and equipment for fuel heating tests.

Post-irradiation examination on the AGR-2 fuel (including both UCO and UO_2 TRISO particles) has continued in the United States. This includes destructive examination of fuel compacts and particles. Up to this time, 11 UCO and 2 UO_2 compacts have been examined, providing information on fission product retention in the particles and compacts during irradiation and detailed microstructural information on the condition of the coating layers and the migration of fission products in the layer (see **Figure VHTR 2**).

Figure VHTR 2. **Micrographs of UCO TRISO particles from an AGR-2 compact irradiated to an average burn-up of 12.0% FIMA**



The US AGR-5/6/7 irradiation of UCO TRISO fuel continues in the Advanced Test Reactor. This experiment is both the final fuel qualification irradiation and a separate high-temperature fuel performance margin test (peak temperatures of ~1 500°C) and contains approximately 570 000 fuel particles in 194 fuel compacts. The irradiation is roughly half complete.

The Unites States has also recently developed – and is currently using – the capability to re-irradiate fuel specimens prior to performing heating tests. This capability is essential for measuring the release of short-lived fission products (including 131I) that can be significant contributors to off-site dose during reactor accidents. The fuel specimens (previously irradiated in the Advanced Test Reactor), are re-irradiated in the Neutron Radiograph (NRAD) reactor located at the Hot Fuel Examination Facility at INL, where they can quickly be removed from the reactor and transported to the hot cell for heating tests.

Fuel attributes and material properties

The FFC PMB organized the 5th Workshop on High-Temperature Gas-Cooled Reactor SiC Material Properties in conjunction with the 15th official meeting of the PMB at ORNL in May 2019 (34 people participated from five different countries). The participants ranged from members of academia, industry, national laboratories, and intergovernmental agencies. The meeting was divided into technical sessions including 16 technical presentations along with significant discussion focusing on scientific challenges facing tri-structural-isotopic (TRISO) fuel for HTGR applications. Technical topics broadly covered two different areas: issues surround SiC coating layers in TRISO fuel, and oxidation of TRISO fuel materials. The meeting also included a series of tours focusing on ORNL's past and present nuclear research and development capabilities.

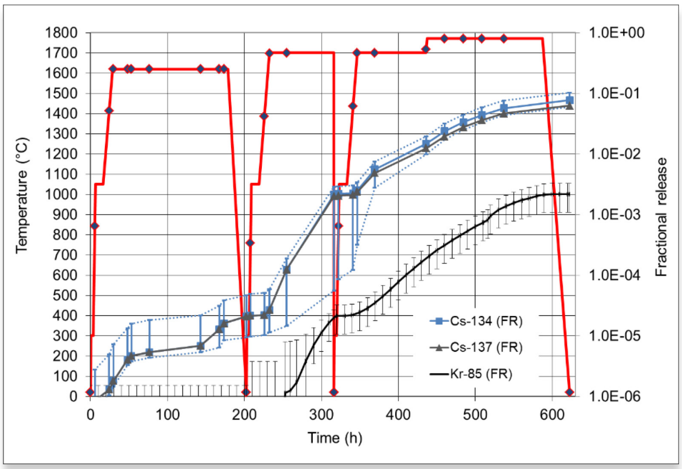
The United States, China, and Korea are completing the final stages of an as-fabricated fuel characterization "round robin" experiment. The work involves performing leach-burn-leach analysis on well-characterized particle specimens to detect defective SiC coatings and complete through-layer coating defects. The specimens were provided by the United States and China. As of the end of 2019, all of the experimental work has been completed. The United States is preparing a final report that summarizes these results.

The United States, Japan, and Korea have also completed a TRISO fuel accident test computational benchmark task. All three participants used fuel performance models to predict fission product release from TRISO fuel during heating tests in pure helium. The results of the predictions were compared with the experimental results from the safety tests performed on fuel in the United States and the EU. A draft report has been prepared and will be issued in 2020.

Safety

High-temperature safety tests are being performed at JRC Karlsruhe on HTR-PM spheres that were previously irradiated in HFR. The tests have been performed at temperatures ranging from 1 620 to 1 770°C for a total of 450 hours for each sphere. A total of four tests have been completed. Krypton release during the tests remained below the level of a single particle, indicating no particle with complete TRISO failure. Cesium releases were below ~2×10^{-5} for 150 h at 1 620°C, but increased at longer exposures and higher temperatures, indicating gradual degradation of the SiC layers. And example of the cesium and krypton release data is shown in **Figure VHTR 3**. In addition to the HTR-PM sphere tests at Karlsruhe, China is also deploying a KÜFA heating test capability at the hot cells at INET. The system has been installed in the hot cells and is undergoing testing.

Figure VHTR 3. **Fractional release of fission products Cs-134, Cs-137, and Kr-85 from an HTR-PM fuel specimen irradiated in HFR Petten and heated in the KÜFA facility. The heating program is shown in red**



In the United States, high-temperature safety tests of AGR-2 UCO and UO_2 fuel compacts in pure helium have continued. A total of 16 safety tests have been performed at temperatures ranging from 1 500°C to 1 800°C. One of these tests was performed with a test temperature that varied over time in a manner similar to the predicted peak fuel temperature in a modular HTGR during a depressurized loss of forced cooling. The results have indicated no TRISO failure during testing of UCO temperatures of 1 800°C for 300 h and testing of UO_2 at 1 700°C for 300 h. Cesium release from UCO fuel remains low during the tests (highest releases are ~3×10^{-4} after 300 h at 1 800°C), but somewhat higher for UO_2 fuel (release as high as 9×10^{-2} observed after 300 h at 1 700°C).

The United States is also performing PIE on the AGR-3/4 irradiation experiment components and heating tests on AGR-3/4 TRISO fuel compacts. These compacts contain about 1 900 TRISO fuel particles, and 20 "designed-to-fail" particles that experience coating failure during the irradiation. Some of these compacts have been re-irradiated in the NRAD reactor to generate short-lived ^{131}I prior to the heating tests. These tests are therefore being used to assess fission product release from exposed kernels.

A dedicated furnace designed to heat irradiated TRISO fuel specimen as high as 1 600°C in oxidizing atmospheres is currently being developed at INL in the United States. The system will be used to test oxidation behavior of fuel and fuel materials in air/He and moisture/He gas mixtures, while monitoring the release of fission products and reaction products in real time. The system is expected to be deployed in 2021.

The United States has also prepared a topical report on UCO TRISO fuel performance in co-operation with the Electric Power Research Institute (EPRI) that describes the results of the AGR-1 and AGR-2 irradiation experiments and subsequent PIE. The report was submitted to the Nuclear Regulatory Commission (NRC) for review. The objective of this report and NRC review is to obtain agreement from the US regulatory authority that the fuel performance data from these experiments can be used by future reactor designers in their licensing submissions.

In Japan, researchers are studying the oxidation of the TRISO SiC layer at the SiC-OPyC boundary. This includes computational modelling of the mechanism of oxidation and the influence of such parameters as temperature, O_2(g) concentration, and transport to the SiC layer through the OPyC. A series of experiments is proposed using TRISO particles with surrogate kernels at temperatures up to 1 600°C and O_2 concentrations of 20 ppm to 20%. JAEA has proposed a new computational benchmarking activity that will focus on the release behavior of short-lived fission gases.

Enhanced and advanced fuel fabrication

Development of fabrication of larger UO_2 kernel sizes that typically used in TRISO fuel is being pursued in Korea. Researchers are targeting sintered kernel sizes of 800 μm, for potential application in accident-tolerant fuels. Experiments have been successful in producing kernels in excess of 800 μm diameter, and work continues to refine the process to improve kernel properties. In conjunction with this effort, coating process for the larger kernels are also being developed. To date this has included computational modelling of the fluidized particle bed, and experiments are planned in the future. Finally, development of double-layer ZrC/SiC TRISO coatings with improved properties continues to be studied in Korea.

China is studying equipment and processes for fabricating ZrC coatings as a potential replacement for SiC in TRISO fuel. Fabrication of UCO kernels is also being pursued.

Significant recent work has been performed on PIE and safety testing of TRISO fuel and new PIE and safety testing capabilities are being developed by several members. The Project Management Board has produced results on two collaborative projects: an LBL round robin experiment and an accident testing computational benchmark. This has led to the creation of a third five-year plan.

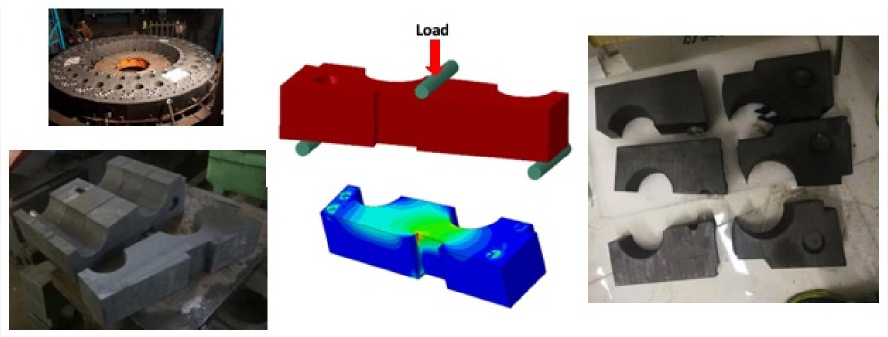
Materials project

As part of the development of the new Program Plan, a thorough review was made of all the High-Level Deliverables (HLDs). All HLDs scheduled for completion prior to the end of 2015 were adjusted for completion during the term of the new PP. Additionally, by the end of 2019, over 420 technical reports and over 10 000 materials test records describing contributions from all signatories had been uploaded into the Gen-IV Materials Handbook, the database used to share materials information within this PMB. This reflects the outstanding technical output of the membership that has now been shared to support system design and codes & standards development.

In 2019, research activities continued focused on near- and medium-term projects needs (i.e. graphite and high-temperature metallic alloys) with limited activities on longer-term activities related to ceramics and composites.

Additional characterization and analysis of selected baseline data and its inherent scatter of candidate grades of graphite was performed by multiple members. Mechanical, physical, and fracture properties behavior were examined for numerous grades. Graphite irradiations and post-irradiation examinations & analysis continued to provide critical data on property changes, 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 interest among signatories is the validation of the anticipated multi-axial loading response of graphite from dimensional changes and seismic events. A figure illustrating large-scale experiments on graphite blocks to validate design models is shown in **Figure VHTR 4.**

Figure VHTR 4. **Fracture testing of large graphite blocks with complex geometry to verify failure probability calculations for HTR-PM construction**

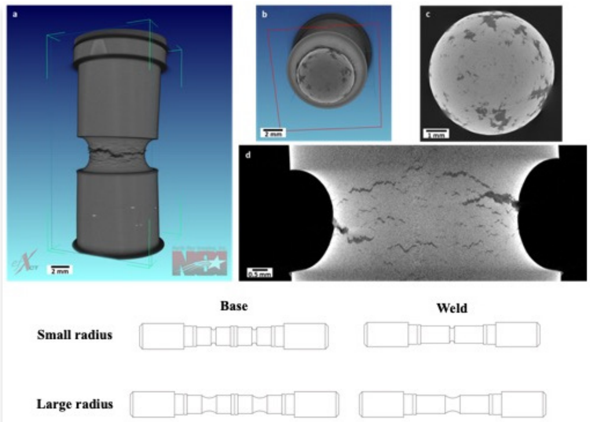


Courtesy of Institute of Nuclear and New Energy Technology.

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. Examination of high-temperature alloys (especially weld behavior in 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 aging, creep, creep-fatigue and creep crack growth rate testing to 950°C. Examination of enhanced diffusion bonding techniques for construction of compact heat exchangers is showing very promising results. The most significant outcome of high-temperature alloy work was approval of the ASME Code Case for the use of Alloy 617 as a new construction material for high-temperature nuclear components at temperatures to 950°C for 100 000 hrs. Data for the Code Case was contributed from multiple Signatories (DOE, KAERI & CEA). Other metallic materials were also examined. Irradiation and irradiation creep was studied on 9Cr-1Mo ferritic-martensitic steels and oxide-dispersion-strengthened steels, plus creep behavior was examined in 2.25Cr-1Mo steel for steam generator applications.

Input for improvements in High Temperature Design Methodology (HTDM) were also contributed by participants. Removing unnecessary conservatism through improvement in analysis procedures and extending the applicability of the rules to longer life times or to a wider selection of materials could provide greater design flexibility and potential innovative designs to enhance safety or reduce construction costs. During 2019, constitutive models and inelastic analysis methods were developed to better define and extend the applicability of simplified design methods to maximum upper temperature limits. An example of experimental support needed for the HTDM improvements concerning multi-axial loading is illustrated in **Figure VHTR 5**. Creep testing and failure process assessment of different V- and U-notched specimens were performed to determine if a crossover from notch-strengthing to notch-weaking occurs in Alloy 617 base and weld metal at times up to 100 000 hrs.

Figure VHTR 5. **Specimens and example examination for creep testing of Alloy 617 at 800°C to assess effects of notch strengthening versus notch weakening**



Courtesy of Idaho National Laboratory.

In the near/medium term, metallic alloys are considered as the main option for control rods and internals in VHTR projects, which target temperatures below about 850°C. However, future projects are considering the use of ceramics and ceramic composites where radiation doses, environmental challenges, or temperatures (up to or beyond 1 000°C) will exceed capabilities of metallic materials. This is especially true for control rods, reactor internals, thermal insulation materials, and fuel cladding. Limited work continues to examine the thermo-mechanical 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. A significant milestone in this area occurred in 2019 with drafts on all articles related to General Requirements and Design Rules for Ceramic Components of ASME Code Section III Division 5 (Rules for Construction of High Temperature Reactor Components) having been completed and submitted for ballot.

Hydrogen production project

In 2019, the international hydrogen community saw a tremendous interest towards bringing hydrogen economy to reality through a range of applications led by the transportation sector. This enthusiasm was also apparent in the progress reported by the signatories at 19th and 20th Official Hydrogen PMB meetings held in Grenoble and Shanghai, respectively, during the year.

Canadian efforts on hydrogen production focused mainly towards the demonstration of an integrated Copper-Chlorine Cycle (hybrid thermo-chemical process) at a laboratory scale (50 L/h H_2 production) by 2021 March. Experimental development of equipment required to carry out each of the four steps of the process led to the following advances during the year: an electrolyser (electrolysis of CuCl/HCl producing H_2) design capable of producing up to 100 L/h H_2 was operated over several weeks with consistent performance; separation of CuCl/$CuCl_2$ was simplified; and an innovative method for the decomposition of Cu_2OCl_2 (an intermediate product) to produce O_2 was successfully demonstrated. Development of an efficient method for the hydrolysis step (reaction between $CuCl_2$ and steam) is being investigated to complete the integration of all the steps involved. In parallel, modelling of the process using Aspen Plus V9 is being carried using an updated database for physical properties of the various species involved in the process.

In China, the HTGR reactor development continued, and when completed is expected to provide the power and heat required for the hydrogen production processes being developed, namely Sulphur-Iodine (S-I), High Temperature Steam Electrolysis (HTSE) and Hybrid Sulphur (HyS) Processes. In the development of the S-I process, focus has been on the construction and simulation of the sulphuric acid Bayonet-type decomposer, the construction of the hydrogen iodide decomposer for hydrogen production at 1 Nm^3/h and the intermediate He loop heat exchanger (100 kW) to satisfy the heat requirement of the S-I process. In the development of HyS process, efforts have gone into simulation of the process, fundamental studies, including simulation of the SO_2 depolarized electrolyser (SDE). A facility for testing a stack of six units (each 200 cm^2) of the SDE has been designed and built. In the meantime, an agreement has been signed by Tsinghua University, China National Nuclear Corporation and China Baowu Steel Corporation to jointly advance nuclear hydrogen technologies for application in steelmaking – an exciting development.

CEA in France is taking an integrated R&D approach for nuclear and renewable energy integration in establishing their overall energy system. The main emphasis of their Low Carbon Energies Division on hydrogen production is in the development of HTSE. The generic development of the cells and stacks have included optimization of the solid oxide cells through thicker oxygen electrodes and thinner barrier layers for performance enhancement and minimization of degradation of cell components for long-term durability. They have also adapted the original thick-cell stack designs to thinner cells and Solid Oxide Fuel Cell (SOFC) operation. During the year they focused on the development of reversible systems for nuclear coupling to allow switching of the electrolyser operation to low-power fuel cell mode when the nuclear reactor is not producing power. Their first reversible industrial system (supplied in 2018) with one stack producing 1 Nm^3/h of hydrogen and 1 kWe in fuel cell mode has continued to operate with electrical efficiency at 84% in electrolyser mode and 55% in fuel cell mode.

Hydrogen production technologies related developments from EU have focused on HTSE and HyS processes. Although the reported work focused on coupling of these processes to solar power production, the actual technical aspects of these hydrogen production processes apply equally to nuclear systems. A steam electrolyser system producing 6.7 NL/min of hydrogen has been built and operated at ~750°C at DLR (Deutsches Zentrum für Luft- und Raumfahrt). Developments on the hybrid sulphur process has progressed under the European research project SOL2HY2. In the first of the two main steps, sulphuric acid is decomposed at high temperatures forming oxygen as a product and SO_2 for the subsequent electrolysis step. SO_2 is then electrolysed at about 80°C with water to produce hydrogen as the main product. Because of the low voltage requirement for this electrolysis step, the power consumption is significantly lower compared to conventional water electrolysis, leading to a significant efficiency gain.

JAEA has been developing various corrosion resistant components for the S-I process, and have incorporated them for the latest 150 h test (completed in January 2019) of the integrated system for hydrogen production at 30 L/h. Following the test, they have been carrying out inspection of materials of components to investigate any corrosion that may have taken place during the test and its impacts. Initial observations have revealed that the improvements made on the glass-lined sheath in HI sections functioned well.

During the year, two roadmaps were released by the Korean government: 1) "Hydrogen Economy Roadmap" in January 2019 to drive a new growth engine and turn Korea into a society fueled by eco-friendly energy, and 2) "Hydrogen Technology Development Roadmap" in October 2019 for technology development across ministries to support the implementation of the hydrogen economy by enhancing domestic technological competitiveness in the hydrogen energy sector. This establishment of the roadmaps on hydrogen economy provided impetus to activities on hydrogen production reported at the Hydrogen PMB meetings during the year. Simulations have been carried out on coupling various hydrogen production processes to a 350 MWth HTGR. The hydrogen production processes included Steam Methane Reforming, HTSE and S-I Process. Component test facilities with nitrogen (**Figure VHTR 6**) and helium loops operating at 60 kWe and 600 kWe respectively, and 950°C have been used to derive databases for Code Verification and Validation. Studies included on sulphuric acid decomposer, corrosion resistance and SiC coating of fluid channels. Emphasis has also been placed on manufacturing of components and transfer of technologies to private industry.

Figure VHTR 6. **Small-scale nitrogen gas loop used for studies involving sulphuric acid decomposition and SiC coated process heat exchangers**

During the year, the American activities under DOE-NE Nuclear-Renewable Integrated Energy Systems (IES) have focused on modelling and simulation, demonstration/experimental systems and stakeholder engagement. INL has established a Dynamic Energy Transport and Integration Laboratory (DETAIL) that will consist of multiple heat and electricity producers, thermal and electrical storage, and multiple heat and electricity customers coupled via a thermal and electrical network (**Figure VHTR 7**). The combined system will provide a demonstration of real-time integration with electrical grid, renewable energy inputs, energy storage and energy users. The entire energy network can be simulated to understand how to optimize energy flows while maintaining stability and efficient operation of all assets in the system. Related to advanced hydrogen production research, a 25 kW high-temperature electrolysis research and demonstration facility (**Figure VHTR 8**) has been designed, installed and commissioned with initial testing at 5 kW scale. Focus has been applied to the actual electrolyser stack components production (interconnects, electrolyte and cells), stack assembly and testing with cycling and long-term operations. The plan is to couple a NuScale SMR Module to DETAIL for R&D activities.

Figure VHTR 7. Systems Integration Laboratory at INL

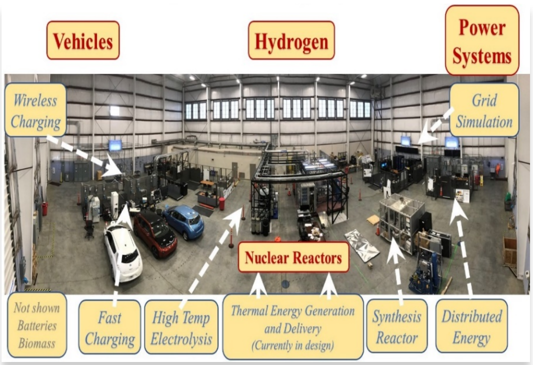


Figure VHTR 8. 25 kW HTSE Test Facility in DETAIL within the INL Energy Systems Laboratory

Computational methods validation and benchmarks Project

The Computational Methods Validation and Benchmarks (CMVB) project was restarted in 2014. From 2015 to 2018, eight meetings organized by the CMVB Provisional Project Management Board (pPMB) were held in turn in different participating countries. The main activities resulting from these meetings include discussion and confirmation of the research tasks in each work package (WP), review and approval of the draft project plan (PP) of which the final version is the indispensable annex of the project arrangement (PA), the discussion of some common topics and potential test facilities which will be the fundamental ressources of this project.

Table VHTR 1. **Work Package organization of the CMVB**

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)

All the above-mentioned efforts were made to launch the signing process of the PA in 2019 focused on the review of the PA and discussion of how to carry out the PP. Through the pPMB meetings, the past, current, and new test facilities and projects have been identified, proposed and confirmed as fundamental resources to perform the development and assessment of codes and models covering HTR physics, thermal-hydraulics, CFD, fission products transport, etc.

In China, the demonstration project HTR-PM is under construction and commissioning. The installation of pressure vessels, steam generators, reactor internals and other important components have been finished. The standard design of the HTR-PM600, a commercial plant with an electricity power 600 MW, has been performed and reviewed by an independent nuclear engineering company. The engineering verification tests have been completed to support the HTR-PM project and such tests involve the main components of the HTR-PM, such as the helium circulator, the fuel handling system, the control rod driving system. Some benchmarking cases were defined and expected in the CMVB PP based on the HTR-PM future first criticality and low power physics tests. The HTR-10 was restarted and a temperature measurement experiment has been completed, whose purpose is to detect the temperatures inside the fuel elements. In addition, key operation parameters were monitored and one instance was the RCCS experimental data which will be used in the CMVB project to evaluate the capabilities of the system analysis tools to calculate the water-cooled RCCS behavior.

Figure VHTR 9. HTR-10 reactor coupled with RCCS

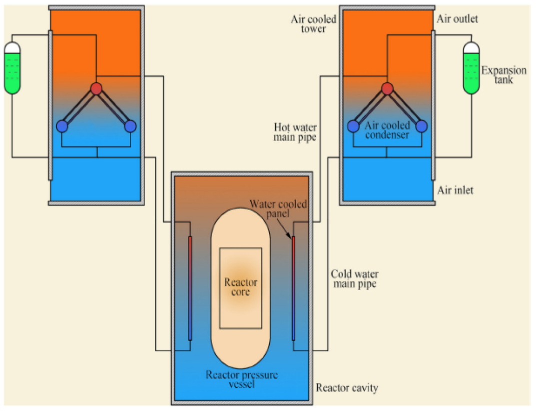
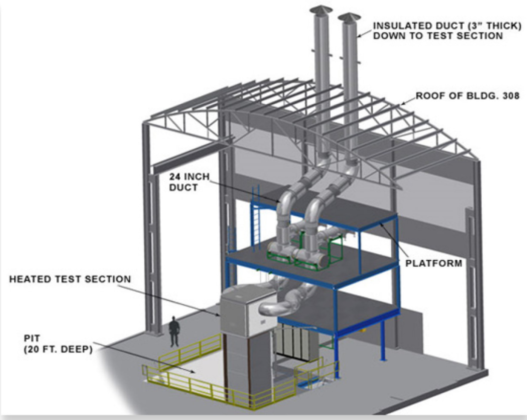


Figure VHTR 10. Natural Circulation Shutdown Heat Removal System at Argonne National Laboratory



In the United States, the advanced reactor technologies (ART) program is a strong support to CMVB project, since this program could provide data from the irradiation tests of fuel, graphite and also alloys. Regarding the design methods and validation, many concerned tests and benchmarks have been done through different facilities, e.g., the ANL Natural Circulation Shutdown Heat Removal Facility (NSTF, **Figure VHTR 10**), the High Temperature Test Facility (HTTF), Matched Index of Refraction Facility (MIR). Data from NSTF experiments is available for validation of air- and water-cooled RCCS models.

EU activities related to HTGR and CMVB include the GEMINI+ project which now is performing design iterations with thermal-hydraulics, neutronics and balance-of-plant calculations, previous Euratom Framework Program projects such as ARCHER, RAPHAEL, PUMA and NC2I-R, and also some past experiment projects such as NACOK, HELOKA, EVO, and HeFUS3. A new proposal for the Horizon 2020 Framework Program has been submitted: HYDRO-GeN-IV. If awarded funding Spring 2020, it will enable to continue and expand the work initiated in GEMINI+ after August 2020.

The VHTR R&D program in Korea aims at improving the VHTR key technologies in terms of the design codes development and assessment, and also high-temperature materials and component technologies. Some code validation work falling in the scope of the CMVB WPs has been completed, including scale-down standard fuel block tests, code-to-code comparisons for key design parameters.

JAEA is making a strong effort to restart the HTTR as early as possible. Based on design, construction and previous as well as future operation database of the HTTR, JAEA is developing and benchmarking various models and analysis methodologies and codes for reactor physics, thermal fluids, etc. The JAEA R&D in these areas is expected to support planning the CMVB co-operative activities such as benchmark activity using ATR irradiation data.

Michaël Fuetterer

Chair of the VHTR SSC and all Contributors

Chapter 5. Methodology Working Groups

EMWG: Economic Modelling Working Group

The Economic Modelling Working Group (EMWG) was established in 2003 to provide a methodology for the assessment of the Generation IV systems against the two economic-related goals as follows:

- To have life cycle cost advantage over other energy sources (i.e. to have a lower levelized unit cost of energy); and
- 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)

In 2007, the EMWG published the Cost Estimating Guidelines and an Excel-based software package G4ECONS v2.0 for calculating two figures of merit; the levelized cost energy and the total investment cost, to assess the Generation IV systems against GIF economic goals. These EMWG tools were made available to the public trough GIF Technical Secretariat which resulted in several publications demonstrating the EMWG methodology for economic assessments of Generation III and Generation IV systems, as well for the cogeneration applications such as hydrogen production.

G4ECONS v2.0 was also benchmarked against the economic tools of the International Atomic Energy Agency (IAEA); namely the Nuclear Economics Support Tool (NEST) and the Hydrogen Economic Evaluation Program (HEEP) and the results have been published in peer-reviewed publications [2, 3]. The lessons learnt from the benchmarking exercise and the feedback from the users informed the refinement of the G4ECONS tool. The EMWG released the new version, G4ECONS v3.0, with improved user interface, in October 2018.

In 2016, the EMWG started to investigate the challenges and opportunities for deployment of the Generation IV systems in the emerging energy markets with increasing share of renewable energy resources. The terms of reference for the EMWG were amended in 2018 to incorporate the expanded mandate to inform the GIF Policy Group and the Experts Group on the policies and the research and development needs for the future deployment of Generation IV systems.

Since October 2016, the EMWG worked collaboratively with the Senior Industry Advisory Panel (SIAP) to investigate challenges and opportunities for deployment of Generation IV systems in the electricity markets with significant penetration of renewable energy resources, and produced a position paper for the Policy Group. An abridged version of the EMWG position paper on the impact of increasing share of renewables on the deployment prospects of Generation IV systems was presented at the 4th GIF Symposium and an executive summary has been posted on the GIF external website. The study found that the Generation IV systems will have to be more flexible compared to the current reactors for deployment in the low-carbon energy systems and such requirements are already being proposed by the utilities. Large-scale energy storage and cogeneration applications would allow flexible dispatch, while ensuring high capacity utilization. Nuclear-renewable hybrid energy systems with adequate energy storage and cogeneration applications could, thus, meet the flexible demand from the grid while operating the power generators to full capacity to ensure overall economically viable operation. However, such flexibility considerations impose additional requirements on the research and development of Generation IV systems.

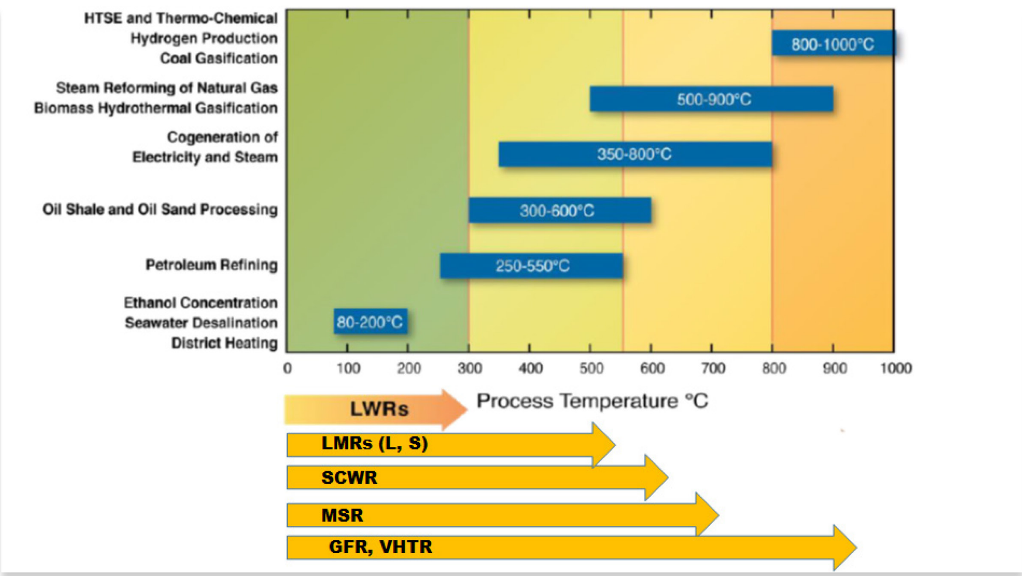
Activities in 2019

Main focus of the EMWG activities in 2019 was on flexibility considerations for the Generation IV systems. The advanced Generation IV reactors are significantly different compared to Gen III reactors. The Generation IV reactors use different fuels, different coolants and operate at higher temperatures, making the reactors suitable for applications beyond electricity production. Therefore, to evaluate the flexibility of Generation IV type reactors, Electrical Power Research Institute (EPRI) developed expanded flexibility criteria and proposed Technology Readiness Scales for Advanced Reactors, such as Generation IV systems. EPRI's expanded flexibility criteria consists of a set of three sub-criteria or attributes, as follows:

- operational flexibility;
- deployment flexibility;
- product flexibility.

Using these as a basis, EMWG developed a questionnaire to gather information on the extent to which the flexibility aspects are being addressed in the research and development of the six Generation IV systems. Subsequently, a workshop was held in May 2019 with the joint participation of the representatives of the six System Steering Committees, the SIAP, and the EMWG to discuss the R&D needs for flexibility and to identify opportunities for cross-cutting R&D. All Generation IV systems are being developed to be more flexible compared to the Generation III systems in terms of deployment flexibility (scalability and constructability) and the product flexibility (cogeneration applications). All Generation IV systems have higher outlet temperatures and thus are amenable to provide thermal energy for multiple industrial applications as shown in **Figure EMWG 1**.

Figure EMWG 1. **Product Flexibility of Generation IV Systems**



The evaluation of the operational flexibility requires validation through multi-dimensional physics calculations and can be performed after the systems are developed to sufficiently high technological readiness level. The EMWG produced a position paper based on the outcome of the questionnaire survey and the joint workshop and made recommendations to the Experts Group to provide guidance to the system developers to include flexibility requirements as part of the R&D, and to identify opportunities for cross-cutting R&D among the six Gen-IV systems. The EMWG also documented the capabilities of various economic models available for optimization of nuclear-renewable integrated systems.

To accompany the latest version G4ECONS v3.0 released in late 2018, training slides were prepared and are available for use by the GIF community.

Finally, the EMWG developed a set of Frequently Asked Questions and Answers for the GIF external website encompassing a wide range of related topics, including the use of the EMWG tools, benchmarking, figures of merit for economic assessment and external factors affecting the economic viability of nuclear, such as, integration with renewables, flexibility requirements and the system costs.

Ramesh Sadhankar

Co-Chair of the EMWG and all Contributors

PRPPWG: Proliferation Resistance and Physical Protection assessment methodology Working Group

The Generation IV Roadmap defined the following Proliferation Resistance and Physical Protection (PR&PP) goal for future nuclear energy systems:

Generation IV nuclear energy systems will increase the assurance that they are a very unattractive and the least desirable route for diversion or theft of weapons-usable materials, and provide increased physical protection against acts of terrorism.

The Proliferation Resistance and Physical Protection Working Group (PRPP WG) was created to develop, implement and foster use of an evaluation methodology to assess Generation IV nuclear energy systems with respect to the GIF PR&PP goals. The current version of the methodology is presented in a document entitled Evaluation Methodology for Proliferation Resistance and Physical Protection of Generation IV Nuclear Energy Systems, Rev. 6, which was released for general distribution in 2011.

The methodology provides designers and policy makers a generic and formal comprehensive approach to evaluate, through measures and metrics, the Proliferation Resistance (PR) and Physical Protection (PP) characteristics of advanced nuclear systems. As such, the application of the evaluation methodology offers opportunities to improve the PR and PP robustness of system concepts throughout their development cycle. 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. The compendium report with white papers on the PR&PP characteristics of each of the six GIF nuclear energy systems prepared with the SSCs, and a set of Frequently Asked Questions and materials from workshops. In 2016 PRPPWG launched a questionnaire for the SSCs to assess the need to update the white papers. A joint SSCs-PRPPWG workshop was then held at the OECD NEA in Paris, April 2017. The PR&PP WG and the six SSCs/pSSCs are currently in the process of updating the six white papers to reflect changing and updated designs and new work on several of these systems.

As a first task the template for the white papers was updated. The SSCs updated first the description of the systems, considering both changes occurred in designs and new designs not considered in the 2011 white papers. After having updated the systems description, the PRPPWG, in collaboration with the SSCs, started updating the parts related to the PR and PP features of the considered designs. For each design option, the PRPP evaluation begins by identifying the relevant system elements with respect to potential adversary targets and applicable safeguards and physical protection approaches. The evaluation then proceeds to assess the design against potential threats using the technical design information to gauge the response of the system. A first draft, providing an overview of technology characteristics and status of design development for each of the six GIF systems was completed by the SSCs/pSSCs in the fall of 2018. A special session with the SSCs/pSSCs was held during the 29th meeting of the PRPPWG, in Oc. 2018. Members of the collaborative team provided status updates on the PR&PP white papers. A roundtable discussion identified information gaps in the white papers and the team developed a work plan to address all parts of the white papers in 2019. In 2019, the PRPPWG focused its activities on:

- continuing collaborative work with SSCs/pSSCs in the updating the white papers on PR&PP aspects of the six GIF systems;
- publicising the methodology and its applications within and outside GIF; and
- monitoring related activities in the areas of proliferation resistance and physical security for their relevance to the GIF programme.

An updated draft of the white papers was completed in November 2019 and an in-depth review of each white paper was planned for the 30th meeting. **Table PRPP 1** presents the high-level structure of the white papers.

During this meeting an extended working session of one and a half days was dedicated to the revision and discussion of the PR&PP white paper updates. Each PRPPWG Point of Contact for the six GIF systems had to:

- introduce the paper, and the reason for the update with regard to the 2011 version; to point out the main differences with regard to the 2011 version;
- illustrate the paper structure and content; drive the discussion on the paper, in general and section by section; execute a deep dive in the papers and get feedback; illustrate missing parts and propose a way forward and timeline;
- propose topics for cross-cutting considerations and availability to lead their investigation.

An observer from the IAEA and a representative from the RSWG also attended the meeting.

Table PRPP 1. **High-level structure of the updated SSCs/pSSCs PR&PP white papers**

Section	Type of Information
Overview of Technology	*Description of the various design options in terms of their major reactor parameters, such as: core configuration, fuel form and composition, operating scheme and refueling mode, fresh/spent fuel storage and shipment, safety approach and vital equipment, physical layout and segregation of components, etc.*
Overview of Fuel Cycle(s)	*High-level description of the type, or types, of fuel cycles that are unique to this Gen-IV system and its major design options. Information such as recycle approach, recycle technology, recycle efficiency, waste form(s)*
PR&PP Relevant System Elements and Potential Adversary Targets	*For each design option, identification and description of the relevant system elements and their potential adversary targets, safeguards and physical security approaches*
Proliferation Resistance Features	*High-level, qualitative overview developed jointly by the SSC and the PR&PP working group, to identify and discuss the features of the system reference designs that create potential benefits or issues for each of the representative proliferation threats. Ideally the section should highlight the response of the system to a) the concealed diversion or production of material, b) the use of the system in a breakout strategy, and c) the replication of the technology in clandestine facilities*
Physical Protection Features	*High-level, qualitative overview developed jointly by the SSC and the PR&PP working group, to discuss those elements of the system design that create potential benefits or issues for potential subnational threats, with specific discussion on the general categories of PP threats (a) theft of material for nuclear explosives or dispersal device and b) radiological sabotage)*
PR&PP Issues, Concerns and Benefits	*Review of the outstanding issues related to PR&PP for the concepts and their fuel cycles, the areas of known strength in the concept, and plans for integration and assessment of PR&PP for the concept. This section would ideally terminate with a bullet list of identified PR&PP R&D needs for the system concept*

The white paper team is preparing a new draft of the white papers incorporating review comments that arose during the review sessions. The team expects to release a final draft for approval by the SSC/pSSC by spring 2020. In addition to the peculiarities of each of the six GIF reactor technologies, addressed by the corresponding White Papers, there are topics that are common to all the six families. The 2011 white papers identified some of the cross-cutting areas, but others are being identified in the course of the update. Cross-cutting topics will be dealt with in the course of 2020.

In 2019, new members were nominated to the PRPPWG, two representatives from the United Kingdom, one additional representative from Canada and two substitute observers from Korea.

The working group continues to publicise its methodology within and outside the GIF through presentations in national and international fora and publications in scientific journals. The group contributed papers on PR&PP to the 4th GIF Symposium 2018, the IAEA Symposium on International Safeguards 2018, the 41st ESARDA Annual Meeting Symposium on Safeguards and Nuclear Material Management 2019. Presentation of the work of PRPPWG, its methodology and its results at these international fora provided opportunities to discuss with other experts and get feedback on its perceived benefits and drawbacks and potentials for its improvement and collaboration.

In support of knowledge management, 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). The 2019 revision of the bibliography is near completion.

Within GIF, collaboration with the Risk and Safety Working Group (RSWG) was strengthened by personal exchanges at each group's meeting. Topics for further discussion between the two groups were identified including: establishment of an integrated framework encompassing the RSWG and PRPPWG methodologies, and identification of synergies and complementarities in the two approaches and evaluations, such as the interface between safety and security.

In its engagement with the IAEA, the PRPPWG maintains regular exchanges with the IAEA INPRO Project and the agency's Department of Safeguards. An observer in the working group from the IAEA, made several presentations for the special session with the SSC at the 29th PRPPWG Meeting covering safeguards needs for Gen-IV reactors, GIF-IAEA interactions and IAEA INPRO update.

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RSWG: Risk and Safety Working Group

The Risk and Safety Working Group (RSWG) was established in 2005 to provide a harmonized approach and consistent methods for risk and safety assessments of six Gen-IV systems. Since its inception, the RSWG proposed a set of broad safety principles, objectives, and attributes based on GIF safety and reliability goals, as input to R&D plans for specific Gen-IV design tracks (see 2008 report on Basis for Safety Approach):

- developed a technology-neutral Integrated Safety Assessment Methodology (ISAM) to ensure a consistent process to address risk and safety;
- supported the implementation of ISAM for specific Gen-IV design tracks as a toolkit for the entire design cycle from concept development to basic design and licensing;
- established technical interfaces with the International Atomic Energy Agency (IAEA), OECD/NEA's Committee on Nuclear Regulatory Activities (CNRA) Working Group on Safety of Advanced Reactors (WGSAR), and other national regulatory stakeholders and designers.

The RSWG membership currently includes representatives from Canada, China, France, Japan, South Africa, Russia, United Kingdom, and United States as a mixture of designers and regulators forum. The group holds biannual meetings. It proceeded to:

- an update of 2008 version of GIF Basic Safety Approach to reflect the lessons learnt from the Fukushima Daiichi accident;
- interface with GIF PR&PP and ETTF working groups; and
- organize a new joint initiative with WGSAR on development of a technology-inclusive risk-informed approach for selection of licensing basis events and safety classification of systems, structures and components common to Gen-IV systems.

The ongoing RSWG collaborations with the SSCs include:

- development of white papers on pilot application of ISAM to assess its usefulness for self-assessment of select Gen-IV design tracks;
- preparation of system safety assessment reports as summaries of the current state of high-level safety design attributes/challenges and overview of remaining R&D needs after the first decade of system development under GIF; and
- contributions to development of safety design criteria for each system.

By the end of 2019, all but one of the white papers are completed and only MSR white paper pending MSR pSSC revision based on RSWG feedback. The system safety assessment reports for SFR, VHTR, SCWR systems are also completed while the LFR and GFR reports are both pending SSC update based on RSWG feedback. The completed white papers and system safety assessments reports are published and can be accessed through the GIF RSWG public web page. Other than the SFR system (as completed by the SDC-TF), the process for development of safety design criteria is in various stages of preparation for other Gen-IV systems.

The ongoing GIF Basic Safety Approach report update aims to capture the needed revisions more than a decade after its first issuance, mainly focusing on integrating post-Fukushima recommendations and requirements to ensure a level of safety compatible with the expectations of the safety authorities. The update also expands the RSWG efforts to harmonize GIF members' safety approach to:

- converge on a common vision;
- provides common definitions for the plant states considered in a design and their alignment with the levels of defence-in-depth;
- reinforce the independence of prevention/mitigation features in different defence-in-depth levels; and
- clarifies the definition of, and the selection process for, the practically eliminated accidents.

Two separate reports are being prepared:

(1) "Basis for the Safety Approach Update for Design & Assessment of Generation IV Nuclear Systems" as a substantial revision of 2008 version but with a similar outline;

(2) a compendium report on "Impact of Fukushima Accident and Recent Regulations on the Safety Approach for Generation IV Nuclear Systems" as an extension of the focus on post-Fukushima Daiichi recommendations and requirements issued by regulators and international organizations since 2011 to provide insights into their applicability in design and safety assessments of Gen-IV systems.

The GIF-WGSAR joint initiative focuses on development of risk-informed approach for selection of licensing basis events and safety classification of systems, structures and components. This technology-inclusive approach is intended to reinforce common understanding of plant states corresponding to different defence-in-depth levels with emphasis on inherent and passive safety features, and to offer a structured approach for incorporating risk insights in safety assessments and regulatory decisions to supplement deterministic approach for increased confidence and improved safety margins.

As a GIF/CNRA joint initiative, it aims to facilitate a structured dialogue among international designers and regulators. Expected outcome is a report on key considerations for applying the risk-informed approach in a way that:

(a) it is inclusive of all six Gen-IV systems with a flexible implementation recognizing unique and varying sovereign regulatory structures;

(b) it builds on existing GIF safety approaches and methodologies (e.g. Basic Safety Approach and ISAM);

(c) it describes the key constituent parts of the risk-informed approach and provides a process description for its implementation. The two-year project is envisioned for completion of the report with co-ordinated input from GIF System Steering Committees and Safety Design Criteria Task Force before it is presented to WGSAR for their subsequent review and feedback.

The RSWG continues to advice the GIF Policy and Experts Groups on interactions with the nuclear safety regulatory community, international organizations and stakeholders relevant to Gen-IV nuclear systems. In 2019, the RSWG also provided a weeklong ISAM training, sponsored and hosted by China, and presented a Gen-IV risk and safety webinar hosted the GIF Education and Training Task Force.

Tanju Sofu

Chair of the RSWG and all Contributors

Chapter 6. Task Force Reports

ETTF: Education & Training Task Force

Background/term of reference

The GIF Education and Training Task Force (ETTF) was launched in 2015 to serve as a platform to enhance open Education and Training (E&T), as well as communication and networking of people and organizations in support of Gen-IV International Forum. The principal objective of the task force is focused on promoting E&T by:

1) identifying and advertising current training courses;
2) identifying and engaging collaboration with other international E&T organizations;
3) developing webinar series dedicated to Gen-IV systems and related cross-cutting topics and advertising these at the national and international level;
4) creating and maintaining a modern social medium platform (LinkedIn www.linkedin.com/groups/8416234) to exchange information and ideas on GIF Research and Development (R&D) topics, as well as related GIF E&T activities.

Main achievements in 2019

The development of webinars is a one of the main activities of this task force. It is intended to inform and stimulate not only young scientists' interest, but also managers, key decision makers and the general public; about advanced reactors introducing foreseen advantages but also key R&D to be developed, knowledge management and preservation with lessons learnt, current research and existing projects.

The ETTF has established collaborative associations with universities and nuclear organizations (**Table ETTF 1**) actively involved in Gen-IV systems to foster the exchange of scientific and technical information for the development of webinars.

Table ETTF 1. **Organizations involved with the development of GIF webinars**

US Department of Energy – Office of Nuclear Energy, United States	Université de Lille 1, France
Institute of Energy and Environment, Youngsan University, Korea	Paul Scherrer Institute, Switzerland
Commissariat à l'Energie Atomique et aux Energies Alternatives, France	Euratom, EU
Argonne National Laboratory, United States	Institute of Physics and Power Engineering, Russia
Canadian Nuclear Laboratories, Canada	Ansaldo Nucleare, Italy
University of California, Berkeley, United States	Kurchatov Institute, Russia
US Naval Postgraduate School, United States	Brookhaven National Laboratory, United States
Nuclear Energy Agency	SCK.CEN, Belgium
Idaho National Laboratory, United States	Los Alamos National Laboratory, United States
Nuclear National Laboratory, United Kingdom	INET, Tsinghua University, China
Research Center Řež, Czech Republic	Japan Atomic Energy Agency, Japan
National Research Nuclear University "MEPhI", Russia	Idaho State University, United States
Colorado School of Mines, United States	Oak Ridge National Laboratory, United States

Because of its easy access, and free of charge, the ETTF has decided to present webinars and exploits this modern internet technology to reach interest of a broader audience. Therefore, to promote training in Gen-IV systems and to ensure a knowledgeable workforce exists, were created and made them available to the public since 2016 a series of webinars on topics specific to advanced reactor systems and cross-cutting subjects. These webinars are intended to be of interest to those already in the workforce who may need a refresher course or a better understanding of a specific topic, to a more general public. World-class webinars presented by Gen-IV Experts (usually GIF members) will be useful to a wide scope of people (like quality assurance officers, data validators, technicians, managers, regulators, and others who require an enhanced understanding of Gen-IV reactor concepts in their work). Thirty-six webinars have been developed, recorded and archived on the GIF Open Website. It is worth to note that during the GIF Symposium (October 2018 in Paris), an elevator pitch challenge (EPiC) contest was organized and the three best students winning this contest were offered to give a webinar presentation (**Table ETTF 2**). During the American Nuclear Society winter meeting (Washington DC, November 2019), a similar event, called "Pitch your PhD" was organized. The winner Dr Coddy Wiggins will present a webinar titled "*Development of Multiple-Particle Positron Emission Particle Tracking for Flow Measurement*", in Dec. 2020.

Table ETTF 2. **GIF Webinar Series (September 2016 to December 2019)**

INTRODUCTION
Atoms for Peace - John Kelly, USA
Introduction to Nuclear Reactor Design - Claude Reanult, France
European Sodium Fast Reactor, An Introduction - Konstantin Mikityuk, Switzerland
GEN IV SYSTEMS
Sodium cooled Fast Reactor - Bob Hill, USA
Lead Fast Reactor - Craig Smith, USA
Gas cooled Fast Reactor, Alfredo Vassile, France
MYRRHA An Accelerator driven System Based on LFR Technology - Hamid Ait Abderrahim, Belgium
Lead Containing mainly isotope Pb-208: New Reflector for Improving Safety of Fast Neutron Reactors - Evgeny Kulikov, Russia
Very High Temperature Reactors - Carl Sink, USA
Supercritical Water Reactors - Lawrence Leung, Canada
Fluoride cooled-High Temperature Reactors - Per Peterson, USA
Molten Salt Reactors - Elsa Merle, USA
Gen IV Coolants Quality Control - Christian Latge, France
Czech Experimental Program on MSR Technology Development, Jan Uhlir, Czech Republic
OPERATIONAL EXPERIENCE
Desig, Safety Features and Progress of HTR-PM - Yujie Dong, China
Feedback Phenix and Superphenix - Joel Guidez, France
Molten Salt Actinide Recycler & Transforming System with and without Th-U Support: MOSART - Victor Ignatiev, Russia
Astrid Lessons Learned - Gilles Rodriguez, France
Safety Of Gen IV Reactors - Luca Ammirabile, EC
Advanced Lead Fast Reactor European Demonstrator, ALFRED Project - Alessandro Alemberti, EC
Russia BN 600 & BN 800 - Ilya Pakhomov, Russia
The ALLEGRO Experimental Gas Cooled Fast Reactor Project - Ladislav Belovsky, Czech Republic
GEN IV CROSS CUTTING TOPICS
Energy Conversion, Richard Stainsby, United Kingdom
Estimating Costs of Gen IV Systems - Geoffrey Rothwell, NEA/OECD
Materials Challenges for Gen IV Reactors - Stu Maloy, USA
FUEL TYPES
General Consideration on Thorium as a Nuclear Fuel - Franco Michel-Sendis, NEA/OECD
Metallic Fuels for SFRs - Steven Hayes, USA
Advanced Gas Reactor TRISO Particle Fuel - Madeline Feltus, USA
SUSTAINABILITY AND FUEL CYCLE
Closing the Fuel Cycle, Myeung Seung, Republic of Korea
Sustainability, A Relevant Approach for Defining Future Nuclear Fuel Cycles - Christophe Poinssot, France
Scientific and Technical Problems of Closed Nuclear Fuel in Two-Components Nuclear Energetics - Alexander Orlov, Russia
Winners of the Elevator Pitch Challenge (EPiC) contest at the GIF Symposium, Paris, October 2018
Formulation of Alternative Cement Matrix For Solidification/Stabilization of Nuclear Waste - Matthieu de Campos, France
Interactions between Sodium and Fission Products in case of a severe Accident in a Sodium-cooled Fast Reactor - Guilhem Kauric, France
Security Study of Sodium Gas Heat Exchangers in Frame of Sodium-Cooled Fast Reactors - Fang Chen, France

The webinars are already planned from January 2020 to June 2020, as shown in **Table ETTF 3**.

Table ETTF 3. **GIF Webinar Planned until June 2020**

Webinars Planned from January 2020 to December 2020
Thermal-hydraulics in Liquid Metal Fast Reactors, Antoine Gerschenfeld, CEA, France – January 2020
SFR Safety Design Criteria (SDC) and Safety Design Guidelines (SDGs), Shigenobu Kubo, JAEA, Japan, February 2020
MicroReactors: A Technology Option for Accelerated Innovation, Jee Gehin (INL and DV Rao LANL), United States, March 2020
GIF VHTR Hydrogen Production Project Management Board, Sam Suppiah, CNL, Canada, April 2020
Performance Assessments for Fuels and Materials for Advanced Nuclear Reactors, Daniel LaBrier, ISU, United States
Comparison of 16 Reactors Neutronic Performance in Closed Th-U and U-Pu cycles, Jiri Krepel, PSI, Switzerland, June 2020

As of August 2019, attendance during the live webcasts totals 1906 and the number of viewings of recorded webinars in the online archive is 3 332 for a total of webinar viewing of 5 238 in 3 years.

The participants in the GIF webinars include representatives from multiple organizations such as federal agencies, national laboratories, various state agencies, universities, international organizations, contractors, and commercial organizations. **Figure ETTF 1** represents the GIF webinar attendance distribution for 35 webinars presented. It is important to note that 35% of webinar attendees are from international organizations. Representatives from state agencies comprise the largest single organization type.

Figure ETTF 1. **Organization type for 35 webinars as of December 2019**

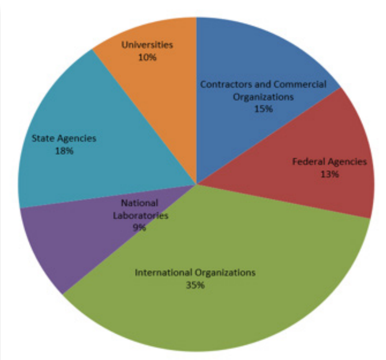


Figure ETTF.2. **Molten Salt Bootcamp, 1-3 July 2019**

The GIF ETTF was represented by Prof. M. Fratoni, from UC Berkeley who participated in the "Molten Salt Summer Boot Camp", 1-3 July 2019, TU Delft, Netherlands.

A paper summarizing the ETTF's activities and titled: "The GIF Webinars, Past, Present and Future" was presented at the international conference Global 2019, September 2019, Seattle. During this conference, Prof. Il Soon Hwang and Dr Patricia Paviet were representing the GIF ETTF at a panel session "Building Next Generation Nuclear: Enabling Succession Planning to Create and Maintain a Well Educated Worksfoce in the Nuclear Energy Sector". This panel addressed some of the key challenges for the nuclear energy sector with respect to maintaining and growing a healthy and diverse talent pipeline of higher-level skills and subject matter experts to drive future thought leadership in fuel cycle.

Due to these excellent results proving the viability and dynamism of this Task Force (see **Figure ETTF 3**), it was decided during the GIF Policy Group meeting in Oct. 2019 to transform this Task Force to a Working Group. The E&T Task Force (ETTF) will therefore move to the E&T Working Group (ETWG). Thanks to this it will start to think deeper – in addition with the webinars series – to some medium-term/long-term actions. Therefore the 1st Face to Face ETWG will occur in 2020 the discuss and propose a common GIF Education & Training vision.

Figure ETTF 3. **The 1st ETTF Selfie event during the Webinar Presentation of Fang Chen who was one of the three "best students winning contest" of the GIF Symposium**

Patricia Paviet

Chair of the ETTF and all Contributors

SDC-TF: Safety Design Criteria Task Force

In 2018 and 2019, the GIF SFR Safety Design Criteria Task Force (SDC-TF) completed the first draft of the SFR Safety Design Guidelines report "Safety Design Guidelines on Structures, Systems and Components for Generation IV Sodium-cooled Fast Reactor Systems" (SSC SDG), which is the second guideline, and revised the "Guidelines on Safety Approach and Design Conditions of Generation IV SFR Systems" (SA SDG), which is the first guideline, by reflecting external feedback from OECD/NEA Working Group on the Safety of Advanced Reactors (WGSAR) and the IAEA.

The SDC-TF completed the SFR Safety Design Criteria (SDC) report in 2013 as the outcome of its phase I activities, distributed it to international organizations (IAEA, MDEP, NEA/CNRA, and regulatory bodies of the GIF member states with active SFR development programmes, namely, China, EC, France, Japan, Korea, Russia and the United States), and revised it on the basis of their comments. To revise it, the SDC-TF adopted many technical descriptions of the IAEA SSR 2/1 revision 1 issued in 2016, including new provisions that reflect lessons learnt from the TEPCO's Fukushima Daiichi nuclear power plants accident. It published the revised SFR SDC report in 2017 after the GIF Experts Group (EG) and Policy Group (PG) have approved it.

The SDC-TF has prepared the SFR safety design guidelines as a set of recommendations on how to meet the SDC and address SFR-specific safety issues. The purpose of the SA SDG is to facilitate the practical application of the SDC to Generation IV SFR design tracks by clarifying technical issues and providing recommendations with a variety of design options. It describes prevention and mitigation of severe accidents, situations that should be practically eliminated (e.g. issues related to the loss of heat removal), and considerations for SFR reactivity characteristics. The SDC-TF distributed the SA SDG to the NEA GSAR (the predecessor of NEA WGSAR) and the IAEA to receive external review. The SDC-TF integrated solutions to IAEA's 23 comments and WGSAR's 128 comments into the revised SA SDG report and sent the revised version to the GIF EG members in 2019 to invite their comment.

The purpose of the SSC SDG is to guide and support SFR designers when practically applying the SDC in design process so that their design can ensure the highest level of safety. The SSC SDG builds bridges between the recommendations of the SA SDG and each SSC design. In addition, the SSC SDG describes recommendations to meet the requirements of the SDC Report which are not covered in the SA SDG. The recommendations in the SSC SDG include measures considering SFR's reactivity characteristics against Anticipated Transient Without Scram (ATWS), and the measures to practically eliminate the core uncovering and the complete loss of decay heat removal function. The recommendations which are not covered in the SA SDG are on fuels and materials under high-temperature, radiation conditions and on measures against various hazards such as sodium fire, sodium-water reaction, and load factors on the containment system, for example **Figure SDCTF 1** shows the consideration process towards the SSC SDG development. The SSC SDG describes the three fundamental safety systems: the core system, the coolant system, and the containment system, which particularly includes selected 14 focal points regarding the SFR-specific safety features as listed in **Table SDCTF 1**. The SDC-TF referred to design features of Generation IV SFR systems, and the descriptions, definitions, and formats of IAEA NS-G series to develop the recommendations. Although the current SSC SDG primarily covers the main components, it will also address other SSCs such as fuel handling and fuel storage systems. The SDC-TF distributed the SSC SDG in 2019 to OECD/NEA WGSAR and IAEA Department of Nuclear Energy to receive external review.

For the next generation advanced LMFRs under development worldwide, GIF and IAEA have a mutual interest in harmonizing safety approaches, safety requirements, the SDC, and the SDGs. This has become a significant topic especially after the TEPCO's accident in 2011, which caused increased attention to nuclear safety and importance of international frameworks for existing reactors currently in operation and for reactors with new designs. In a framework of GIF-IAEA collaboration, there have been eight joint IAEA-GIF technical meetings on SFR safety since 2010. The SSC SDG was introduced in the eighth IAEA-GIF workshop in Vienna in March, 2019.

Figure SDCTF 1. **Consideration process of the SSC SDG**

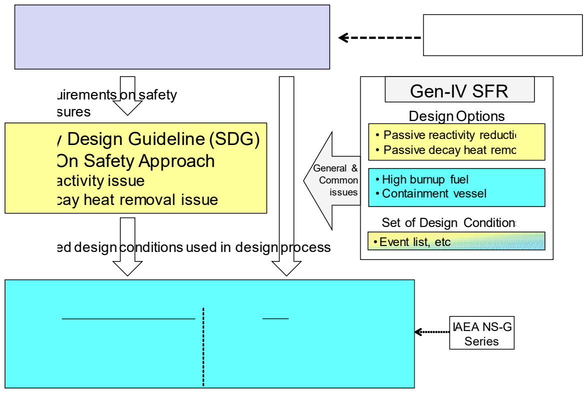


Table SDCTF 1. **SFR-specific safety features**

Systems	Safety features	Focal points	SDC	SDG on Safety Approach
Reactor Core systems	Integrity maintenance of core fuels	1. Fuel design to withstand high temperature, high inner pressure, and high radiation conditions	✓	
		2. Core design to keep the core coolability	✓	✓
	Reactivity control	3. Active reactor shutdown	✓	✓
		4. Reactor shutdown using inherent reactivity feedback and passive reactivity reduction	✓	✓
		5. Prevention of significant energy release during a core damage accident, In-Vessel Retention	✓	✓
Coolant systems	Integrity maintenance of components	6. Component design to withstand high temperature and low pressure conditions	✓	
	Primary coolant system	7. Cover gas and its boundary	✓	
		8. Measures to keep the reactor level	✓	✓
	Measures against chemical reactions of sodium	9. Measures against sodium leakage	✓	
		10. Measures against sodium-water reaction	✓	
	Decay heat removal	11. Application of natural circulation of sodium	✓	✓
		12. Reliability maintenance (diversity and redundancy)	✓	✓
Containment systems	Design concept and load factors	13. Formation of containment boundary and loads on it	✓	
	Containment boundary	14. Containment function of secondary coolant system	✓	

The SDC-TF has produced the SFR safety documents listed below and successfully completed most of its missions: (These are currently being reviewed)

- Safety Design Criteria for Generation IV Sodium-cooled Fast Reactor System.
- Safety Design Guidelines on Safety Approach and Design Conditions for Generation IV Sodium-cooled Fast Reactor Systems.
- Safety Design Guidelines on Structures, Systems and Components for Generation IV Sodium-cooled Fast Reactor Systems.

To discuss remaining topics, SDC-TF members proposed to the GIF PG that the SDC-TF joins the RSWG on the GIF PG meeting (Oct. 2019, Weihai, China), and the PG approved it; the SDC-TF members will join the RSWG as new members from the RSWG meeting in April 2020.

Shigenobu Kubo

Chair of the SDC-TF and all Contributors

AMME TF: Advanced Manufacturing and Material Engineering Task Force

Background

Deployment of future Generation IV reactors will require the successful utilization of both traditional Nuclear Structural Materials and improved material designs and utilize modern advanced manufacturing techniques where they can reduce cost or time. However, most nuclear design codes utilize design by rule philosophies that typically dictate that only qualified materials and processes can be used. Getting new materials or new manufacturing processes qualified can be a long and tortuous process and the long lead times involved produce an effective and consequent barrier to market entry of new or optimized materials and processes at an industrial scale.

Collectively, these issues present a barrier to market entry for Generation IV reactors and the development of materials and manufacturing solutions to benefit the six Generation IV reactor systems. Furthermore, developments in advanced manufacturing are occurring much faster than our ability to introduce new materials and methods into design codes potentially stifling innovation and hampering deployment. The GIF Advanced Manufacturing Materials Engineering Task Force was formed to investigate how collaborative R&D could be used to enable such advances to reduce the time to deployment of Gen-IV and comparable advanced reactors.

The initial primary aims of the Task Force were to undertake a feasibility assessment for a GIF cross-cutting activity in Advanced Manufacturing and Materials Engineering by.

- Assessing the interest of both research institutions and nuclear companies within GIF countries in a cross-cutting activity in GIF supporting Advanced Materials and Manufacturing solutions to a High Technology Readiness Level (TRL).
- Developing and applying a flexible and accessible approach with clearly identified mechanisms for directly involving leading and SME advanced nuclear reactor companies from GIF countries.
- Developing a priority list of R&D areas and initiatives.
- Delivering a white paper discussing the identifying merits and difficulties of such co-operation on this topic and identifying potential ways forward.

Operation

The Task Force consists of members from the GIF countries. Its initial activities focused on identifying mechanisms to reach out to relevant personnel in the nuclear industry. Consequently, the following hypothesis was developed designed to be tested through the use of a questionnaire:

"*That the development of advanced reactor systems to provide clean energy around the world can benefit from international collaborations in the development of advanced manufacturing technologies and techniques*."

A survey was developed using Survey Monkey (www.surveymonkey.com) to obtain data to test the hypothesis. The survey was sent to over 200 relevant nuclear industry contacts, which were identified using input from Task Force, Expert Group and Steering Committee representatives.

The individuals represented the following stakeholder groups:

- Designers and developers of advanced reactor technologies;
- Research institutions and national laboratories;
- University nuclear research departments;
- Safety authorities;
- Manufacturers and suppliers to the nuclear industry;
- Codes and standards organization;
- Nuclear industry policy and trade associations.

There were just under 50 replies. Although it was possible to complete the survey anonymously almost all respondents volunteered contact information to facilitate follow-up. This data showed that the respondent breakdown was:

- 46% research institutions and national laboratories;
- 33% designers and developers of advanced reactor technologies;
- 15% manufacturers and suppliers to the nuclear industry;
- 8% university nuclear research departments;
- 5% nuclear industry policy and trade associations;
- 3% codes and standards organization.

Encouragingly responses came from ten GIF member countries:

Considering the survey as a whole, a number of clear messages emerged. There was strong support for collaborating on establishing codes and achieving regulatory acceptance. 90% of respondents see approval by codes and standards organizations as the largest obstacle to the adoption of Advanced Manufacturing. There was also a clear preference on how to best work collaboratively to address this problem with interest across the board (87%) in participating in workshops and conferences. There was also substantial interest (59%) in pursuing collaborative research and development opportunities but this is balanced by relatively low interest (26%) in investing in advanced manufacturing at this time. In this context, respondents' interests aligned with orientation of their organization, e.g. Universities supported further R&D but did not, in the main want to invest.

Responses to specific questions about areas of interest and priorities provided important insights into the needs and interests of the community. In response to a question asking what type of components are of the greatest interest then fuel cladding, fuel assemblies, reactor internals and heat transfer systems (e.g. IHX, steam generator tubes etc.) were gained equal support with a substantial but lesser interest in reactor pressure vessels. When asked what Advanced Manufacturing techniques hold the greatest potential value, the highest support was for cladding, coating and surface modification techniques with good support for both improvements to welding & joining and metal additive manufacturing and also support for post-manufacturing treatment techniques and new approaches to construction. Indeed, virtually all advanced manufacturing methods were considered opportunities with only 21 out of 143 individual assessments of the techniques listed ascribing them "Low" or "Very Low" value.

As noted earlier, when asked what are the greatest obstacles for the adoption of Advanced Manufacturing approval by code and regulatory bodies was cited by 90% or respondents. Other main concerns centred around uncertainties about the quality and/or maturity of Advanced Manufacturing technologies. Interestingly cost was only seen as a moderate issue indicating the increasing interest in the community in alternatives to the conventional nuclear supply chain.

Clear consensus also emerged when asked what the best pathway for gaining international acceptance for Advanced Manufacturing. Given that the major obstacle was seen as acceptance by regulatory bodies, it is not surprising that by far the greatest support was for collaboration on testing and materials performance combined with demonstrations in real world applications. This was followed by collaboration on codes and standards and Advanced Manufacturing R&D.

As may be expected, collaborations on codes and standards was rated the highest by manufacturers, codes and standards organizations, and industry associations while collaboration on R&D was rated the highest by research entities and national laboratories. Importantly, demonstration in real world applications was supported by all stakeholders; and particularly by codes and standards organizations.

Conclusions (or next steps)

The results from the survey showed that there was a very real interest in both research institutions and nuclear companies within GIF countries in active collaboration supporting Advanced Materials and Manufacturing solutions to a High Technology Readiness Level (TRL).

Consequently, the AMME Task Force prioritized its activities and concentrated on the provision of an international workshop designed to investigate how collaborative R&D in the file of advanced manufacturing can be used to reduce the time to deployment of advanced reactor systems.

Advantage was taken to combine the AMME Task Force Advanced Manufacturing Workshop with another GIF workshop being organized by the GIF RDTF allowing both workshops to be held on 18-20 February 2020 at the NEA in Paris. The structure of the AMME workshop is given below:

Tuesday 18 February	GIF AMME Workshop on Advanced Manufacturing **DAY 1**
0900 - 0910	**Welcome and opening remarks** Sama Bilbao y Leon
Session 1 – Overview of workshop	
09:10 - 09:30	Introduction on the opportunities and challenges of advanced manufacturing, overview of AMME Task Force, purpose of workshop: Why we are here! *Lyndon Edwards, ANSTO, Australia*
09:30 - 10:00	The nuclear supply chain; past, present and future *Andrew Storer, NAMRC, UK* (20mins+10 min discussion)
10:00 – 10.30	Morning Tea
Session 2 – Advanced Manufacturing Technologies	
10:30 - 11:00	Additive manufacturing in the nuclear supply chain *Kurt Terrani, ORNL, United States* (20mins+10 min discussion)
11:00 – 11:30	Innovative fabrication in the nuclear supply chain *Dave Gandy, EPRI, United States* (20mins+10 min discussion)
11:30 - 12:00	Advanced surface coatings in the nuclear supply chain *Alfons Weisenburger KIT, EU* (20mins+10 min discussion)
12:00 – 12:30	Panel Discussion (Presenters) led by Moderator (tbc)
12:30 - 14:00	Lunch
Session 3 – National Advanced Manufacturing Activities	
14:00 – 14:20	Advanced Manufacturing collaboration in the United States *Isabella Van Rooyen/Mark Messner*, DoE, (15mins+5 min discussion)
14:20 – 14:40	Advanced Manufacturing collaboration in the EU *Lorenzo Malerba*, CIEMAT, EU (15mins+5 min discussion)
14:40 - 15:00	Advanced Manufacturing collaboration in France *Eric Abonneau, CEA, France, EU* (15mins+5 min discussion)
Session 4 – Group Activity	
15:00 – 17:00 **includes Afternoon tea**	Split into 3 or 4 groups, which undertake following activities led by Moderator/Rapporteur a. identify potential collaborative AMME activities/projects b. analyze each identified area of collaboration (SWOT analyses?) c. prioritization of identified areas/ideas d. agree communication for Rapporteur to give to meeting (can develop presentation overnight)
17:00	End of Day 1
Wednesday 19 February	GIF AMME Workshop on Advanced Manufacturing **DAY 2**
Session 5 – Group Reporting and Meeting Outcomes	
09:00 - 10:30	Communally undertake following activities: a. Rapporteurs from each group presents group output b. Overall prioritization of potential collaborative AMME activities/projects c. Identification of next steps and way forward
10:30	End of Meeting

Lyndon Edwards

Chair of the AMME TF and all Contributors

RDTF: Research & Development Infrastructure Task Force

R&D infrastructure

Today's research infrastructure needs, from R&D to demonstration and deployment, includes major scientific equipment, scientific collections, structured information and ICT-based infrastructure. They are single-sited or distributed throughout several countries. GIF member countries are faced with a wide spectrum of issues related to infrastructure, many of which are globally unique and regionally distributed. Many stakeholders are involved, from ministries to researchers and industry, with an underlying and growing use of e-infrastructure. They present opportunities for, and yet difficulties in, interactions between basic research and industry. Public and private funding appears always to be lacking, and single countries do not have the critical mass or the dimensions to implement large research infrastructure. There is a real need to co-operate on a broad international level. Substantial research, development and demonstration (RD&D) of systems' conceptual/detailed design and analysis are needed. Refurbishment and/or construction of research infrastructure and facilities are increasingly complex and costly. An opportunity exists, by identifying the latest R&D needs and the mapping of infrastructure, to plan for the shared use of existing facilities and to undertake the development of others. The most important priorities are in the areas of fuel cycle, fuel and material irradiation, reactor safety, dedicated loops, mock-ups and test facilities, advanced simulation and validation tools, transnational access to infrastructures, the E&T and Knowledge Management (KM) of scientists and engineers. GIF members strongly support a co-ordinated revitalization of nuclear RD&D infrastructure worldwide, to a level that would once again quickly move forward a new generation of reactors.

Background/terms of reference

Background: At the 43rd GIF Policy Group (PG) meeting held on 13-14 April 2017 in Paris, France, it was decided to establish a new Task Force (TF) on R&D Infrastructure. The PG tasked the Technical Director (TD) to develop, in collaboration with the PG Vice Chair in charge of external collaboration and with the Technical Secretariat (TS), the Terms of Reference (ToR) for the GIF R&D Infrastructure Task Force (GIF RDTF). This Task Force is expected to accomplish its objectives over a short duration (less than two years) and make maximum benefit of the GIF Symposium held in October 2018.

Objectives: Identify essential R&D experimental facilities needed for development, demonstration and qualification of Gen-IV components and systems, including activities to meet safety and security objectives. To this end, the Task Force should prepare relevant presentations and papers for the October 2018 GIF Symposium.

Promote the utilization of the experimental facilities for collaborative R&D activities among the GIF partners. To this end, identify existing mechanisms and approaches, including organizational points of contact, for obtaining access to relevant R&D facilities in the GIF member countries. This information should be made accessible to GIF participants on the GIF website.

Organization: Each Gen-IV System Steering and provisional System Steering Committee (SSC and pSSC) designated one representative to the GIF RDTF. The task Force reports to the Technical Director (TD), the Expert Group (EG) for review, quality and completeness, and the Policy Group (PG). Members of the GIF RDTF meet as needed, taking advantage of teleconferences and GIF EG/PG venues. Chairpersons and a two-year work plan were agreed since their first meeting at OECD/NEA in Paris in February 2018. It included milestones and deliverables, with a recommendation to take full advantage of any relevant work from IAEA and NEA in the area of infrastructures. First objective was reached on time for presentation at the October 2018 GIF Symposium in Paris. The second objective, originally planned by spring 2019 was delayed, and upon completion of the two objectives of the GIF RDTF, SSCs and pSSCs will be expected to maintain cognizance of infrastructure needs and approaches for their access as work evolves from mid-2020 onwards.

Main achievements in 2019

Identification of existing experimental facilities in response to the aforementioned needs highlighted some gaps. Planned experimental infrastructure constructions, availability of experimental infrastructures outside the GIF countries were discussed.

An opportunity was also taken to propose any update of existing IAEA and NEA databases (including any new infrastructures or facilities launched) with the close support of GIF SSC (or pSSC) and EG groups. The Task Force benefitted from GIF Member State's latest relevant updates and R&D needs outlooks together with: a) IAEA database of Facilities in Support of Liquid Metal-cooled Fast Neutron Systems Facilities and its latest compendium; b) The Advanced Reactor Information System (ARIS); c) The Research Reactor database (RRDB); d) OECD/NEA Research and test facilities database (RTFDB); e) OECD/NEA Task Group on Advanced Experimental Facilities (TAREF) on SFR and GFR but also the Support Facilities for Existing and Advanced Reactors (SFEAR); and f) International Co-operation initiatives and collaborative projects (e.g. IAEA CRPs, ICERR, NEA joint projects, NEST, NI2050, and EU/Euratom projects) for building knowledge and facilities needed for the development of nuclear energy systems e.g. ADRIANA (ADvanced Reactor Initiative And Network Arrangement).

An opportunity to start interacting with NEA Working Group on Safety of Advanced Reactors WGSAR took place as from October 2018, to identify and address safety research needs, and to identify and resolve key regulatory issues.

IAEA Liquid Metal-cooled Fast Neutron Systems (LMFNS) database's update took place throughout the year 2019, by organizing a technical meeting in January, a joint workshop by the end of March, and updates on a case by case basis during the following months. To summarize, LMFNS has been updated as following: a) 43 facilities updated (22 LFR facilities and 21 SFR facilities); b) 34 new facilities added (16 SFR facilities and 18 LFR facilities); and c) now LMFNS Online Catalogue includes 180 facilities (86 SFRs, 80 LFRs, and 14 cross-cutting, for dual applications). IAEA LMFNS Online Catalogue is from now on publicly available at ***https://nucleus.iaea.org/sites/lmfns*** and it is online since August 2019. Any new update is welcome and dealt with on.

Similarly, IAEA Technical Meeting on Knowledge Preservation for Gas Cooled Reactor Technology and Experimental Facilities (GCR and HTR) database was launched in December 2018. IAEA and GIF RTDF members devoted efforts in compiling around 115 facilities identified. A database "GCR and HTR" has been produced in 2019, quality checks are taking place and a database could be available by mid-2020. As such, GIF RDTF participants welcome such existing databases' updates. IAEA should be able to update them on a two years' basis. GIF Policy Group should engage and give their full support.

A dedicated GIF RDTF report was drafted during 2019 and presented at the GIF EG/PG meeting in Weihai (CN). Three major sections still need to be completed namely: a) section IX – cross-cutting R&D infrastructures; b) section X – Mechanisms and approaches for collaborative R&D activities; and c) section XI – key recommendations. The objective will be to have a full draft report available for its review by the EG members, by May 2020, also integrating key recommendations of the following workshops.

GIF International Workshops with Nuclear Industry including SMR vendors and supply chain SMEs were organized successfully, with 60 high-level participants, on 18-20 February 2020, at OECD/NEA, in Boulogne-Billancourt, France. The first and half day, the Workshop was devoted on Advanced Manufacturing (see AMME report). The second half of the Workshop was on R&D Infrastructures needs and opportunities. It included roundtables: Engaging with the private sector, Identification of collaboration opportunities between private and public sectors for Gen-IV systems, a Networking Cocktail gathering both GIF representatives and Industry, Examples of collaboration between governmental organizations and industry, and views from the private sector, an Outlook for SMRs. GIF Policy Group Chair Hideki Kamide concluded the workshop together with representatives of industry, regulators, GIF Member States' and OECD/NEA representatives.

Conclusions (and/or Next steps)

The results show that there is a very real interest in both research institutions and nuclear companies within GIF countries in active collaboration supporting GIF member's organizations at the workshops. The main objective for 2020 is to finalize GIF RDTF report and any related database update. Way forward will also be discussed at the EG/PG meeting in May 2020, in Sydney, Australia.

GEN IV International Forum℠

GIF International Workshops with Nuclear Industry including SMR vendors and supply chain SMEs:

GIF workshop on R&D Infrastructures needs and opportunities

Wednesday 19 February 2020

- **11h00 - 11h15 Welcome**
  - Welcome by *Roger Garbil*, Euratom, DG RTD, Chair of the GIF R&D Task Force and *Sama Bilbao y Leon*, NEA
- **11h15 – 12h30 Engaging with the private sector – Round Table**
  Moderator: Sama Bilbao y Leon
  - R&D challenges for Gen IV systems, *Gilles Rodriguez*, CEA (GIF Technical Director)
  - GIF R&D infrastructures and large scale experimental programmes, *Roger Garbil*
  - GIF Advanced Manufacturing initiative, *Lyndon Edwards*
  - Regulatory challenges to license Gen IV systems, *Raj Iyengar*, Chief of the Component Integrity Branch, NRC's Office of Research
- **14h00 – 16h00 Identification of collaboration opportunities between private and public sectors for Gen IV systems**
  - 14h00 - 14h10 Introduction - *Roger Garbil*, Euratom, DG RTD, Chair of GIF R&D TF
  - 14h10 – 14h30 Example of a LWR-based Advanced Reactor development programme
    Moderator: Sang Ji Kim
    - *Fredrik Vitabäck*, GE-Hitachi, BWRX300
    - *Richard Wain*, UK SMR, Rolls Royce
    - *Jean-Michel Ruggieri*, Program Manager, SMRs, CEA, NUWARD Project
    - *Sang Ji Kim*, SMART Technology Development Division, KAERI, GIF EG member
    - *Marketa Krykova*, Project Manager, CVR, SSC SCWR co-Chair
  - 14h30 – 15h00 Molten Salt Reactors (MSR)
    Moderator: Stéphane Bourg
    - *David Leblanc*, President and CTO, Terrestrial Energy
    - *Stéphane Bourg*, CEA, GIF SSC Chair
    - *Lou Martinez Sancho*, CIO Kairos Power FHR (KP-FHR)
    - *Victor Ignatiev*, IPPE, MOSART project and related infrastructures
    - *Jan Uhlir* Update on Pu/Pr cooperation in CZ
  - 15h15 - 15h45 Liquid Metal Reactors (SFR and LFR)
    Moderator: Alessandro Alemberti
    - *Fausto Franceschini*,Westinghouse LFR,
    - *Alessandro Alemberti*, ANSALDO Nucleare, LFR SSC Chair,
    - *Jean-Marie Hamy*, Framatome, SFR
    - *Ilia Pakhomov*, Head of Laboratory, Russian Federation, Institute for Physics and Power Engineering (IPPE)

GIF International Workshops with Nuclear Industry including SMR vendors and supply chain SMEs:
Workshop on Advanced Manufacturing / Workshop on R&D Infrastructures needs and opportunities
18-19-20 February 2020, OECD/NEA – Boulogne-Billancourt, France

  - 15h45 – 16h15 Gas-cooled High Temperature Reactors (HTR)
    Moderator: Lyndon Edwards
    - *Jean-Marie Hamy*, Framatome, US SC-HTGR program
    - *Dominique Hittner*, USNC
    - *Karl-Fredrik Nilsson*, EU/Euratom JRC, Chair HTR SSC
  - 16h15 – 16h45 Cross-cutting topics, non-electric applications
    Moderator: Taiju SHIBATA
    - *John Jackson*, INL, National Reactor Innovation Center,
    - *François LE Naour*, CEA
    - *Taiju Shibata*, JAEA
    - *Abderrahim Al Mazouzi*, EDF
  - 16h45 – 17h30 Wrap-up and lessons learnt (Tech Director + Moderators)
    - All the moderators – Need to 2 bullet points + ½ page reporting
- **17h30 Networking Cocktail – GIF and Industry**
  - Making connections and fostering exchanges among GEN-IV systems and cross-cutting topics between: Public / Private sectors, R&D Organisms / Industry, R&D platforms.

Thursday 20 February:

- **9h00 – 11h00 Examples of collaboration between Governmental organisations and industry**
  - Welcome by GIF Vice Chair on R&D Collaboration, *Jong-Hyuk Baek*, KAERI
  - Panel discussion - *Moderator: Gilles Rodriguez*
    - *Gilles Rodriguez*, (on behalf of the CEA's Sodium School Director)
    - *John H. Jackson*, Acting Director, Gateway for Accelerated Innovation in Nuclear (GAIN)
    - *Tatiana Ivanova*, FIDES ~~projects~~ to address post-Halden situation / OECD
    - *Stefano Monti*, IAEA
    - *Raj Iyengar*, Chief of the Component Integrity Branch, NRC's Office of Research
    - *Stephen Bushby*, Atomic Energy of Canada Limited
    - *Iuliia Kusina*, Director, Institute for Physics and Power Engineering (IPPE)
- **11h30 – 12h30 Views from the Private Sector, an Outlook for SMRs**
  Moderator: Stefano Monti
    - *Fausto Franceschini*, Westinghouse LFR,
    - *Lou Martinez Sancho*, CIO Kairos Power FHR (KP-FHR),
    - *David Leblanc*, President and CTO, Terrestrial Energy,
    - *Robin Manley*, VP SMR Technology, Ontario Power Generation,
    - *Raj Iyengar*, Chief Component Integrity Branch, NRC's Office of Research,
    - *Dominique Hittner*, USNC,
    - *Richard Wain*, UK SMR, Rolls Royce
    - *Fredrik Vitabäck*, GE-Hitachi, BWRX300,
    - *Arkady Karneev*, Rosatom Western Europe
- **12h30 – 13h00 Workshop conclusions**
  GIF Policy Group Chair *Hideki Kamide*, representative of industry, representative of regulator, representative of OECD/NEA
- **13h00 Closing of the Workshop**

GIF International Workshops with Nuclear Industry including SMR vendors and supply chain SMEs:
Workshop on Advanced Manufacturing / Workshop on R&D Infrastructures needs and opportunities
18-19-20 February 2020, OECD/NEA – Boulogne-Billancourt, France

Roger Garbil

Chair of the RDTF TF and all Contributors

Chapter 7. **Market and industry perspectives/SIAP report**

Market issues

Since the creation of the Generation IV International Forum (GIF) in 2000, market conditions have never ceased to evolve and they are a common concern between users and developers of Gen-IV concepts. In this sense, the Senior Industrial Advisory Panel (SIAP) has actively worked in better understanding the core drivers, opportunities and constraints related to the market environment, with the objective to identify the most appropriate ways to perform GIF activities. It always works in close collaboration with the System Steering Committees (SSCs) chairs and Task Forces (TFs), and with the guidance of the members of GIF Policy Group (PG).

Following the conclusion of the Paris agreement 2015, numerous countries initiated major endeavours to reduce CO_2 emissions related to economic activities. The decarbonization of the electricity sector has concentrated most of the efforts over the last years with the massive capacity additions of variable renewable energy resources (VRE) such as wind and solar power. As recently illustrated by the International Energy Agency (IEA)[1], low-carbon electricity demand is set to increase by 2040 and the mobilisation of all low-carbon technologies will be needed in order to meet the climate engagements. For instance, according to the *Sustainable Development Scenario* set out by the IEA, nuclear capacity should increase of 60% compared to today's levels[2]. Nevertheless, several issues are challenging the economic rationale of nuclear slowing down the development of nuclear power.

The cost of VRE has been steadily decreasing enabling a higher penetration of this type of technologies in the current electricity systems. This trend, combined with cheap and abundant fossil fuels (especially in the United States), is undermining the profitability of nuclear projects in a *Levelized Cost Of Energy* (LCOE) basis. At the same time, and partly due to the long hiatus in nuclear new build since 1990s, recent nuclear project are finding difficulties to being delivered on time and on budget in OECD countries increasing the risks perceived by investors.

On the other hand, it is important to highlight that the irruption of VRE resources is shaping electricity systems and new opportunities are emerging. Dispatchability attributes are becoming more valuable in the light of intermittent electricity generation and the absence of large-scale storage solutions. Distributed generation is also gaining momentum. Furthermore, the decarbonization of the energy systems also involves low-carbon heat generation for domestic and industrial processes or also hydrogen massive production.

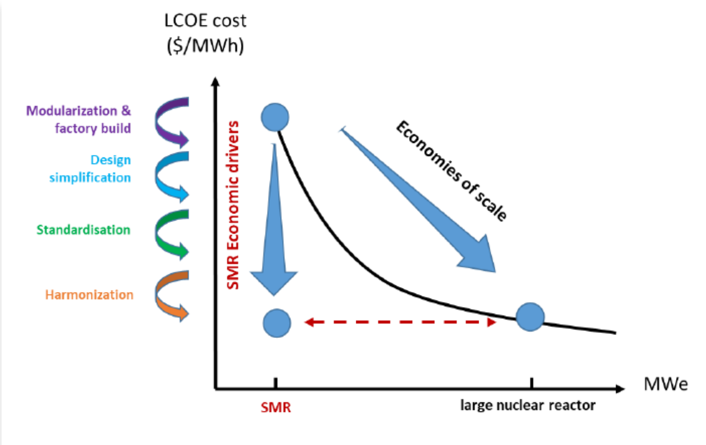
All these aspects were explored during the GIF Workshop on Flexibility held in May 2019 in Vancouver. This event gathered Economic Modelling Working Group (EMWG), SIAP and SSCs members with the objective to assess the flexibility of the different Gen-IV systems. It was a good opportunity for SIAP to share with the GIF community the main findings of the 2018 SIAP charge with a strong focus on the flexibility of Gen-IV systems and the opportunities associated with hybrid systems. The workshop confirmed that all Gen-IV concepts have significant flexibility features to meet emerging energy market needs in terms of load-following, scalability and heat generation and hydrogen production. Those technologies with lower Technology Readiness Levels (TRLs) have the highest potential as they face reduced constrains from a design standpoint. The different flexible options may allow Gen-IV systems to better adjust to more uncertain and turbulent energy markets. Nevertheless, integrating flexibility in Gen-IV designs may come at a cost and should be fairly compensated through adequate market designs.

1. World Energy Outlook 2019, www.iea.org/reports/world-energy-outlook-2019
2. Tracking Clean Energy Progress 2019, www.iea.org/reports/tracking-power-2019/nuclear-power

In this context, small modular reactors (SMRs) are capturing the attention of the nuclear industry as they potentially offer a more attractive business case in current market conditions. SMRs are nuclear reactors with power output ranging between 10 MWe and 300 MWe that integrate by design higher modularisation, standardization and factory-based construction in order to maximize economies of series (or series effect). The different modules can then be transported and assembled on-site, leading to predictability and savings in construction times. More recently, vendors are proposing Micro Modular Reactors (MMRs), with power outputs lower than 10 MWe, capable of semi-autonomous operation and taking advantage of higher levels of transportability compared to larger SMRs.

The series effect, among other conditions, plays a central role in the economic competiveness of SMRs. In fact, the small size of this type of reactors introduces a considerable economic penalty in terms of LCOE (diseconomies of scale). The cumulative effect of modularisation, simplification, standardization and harmonization may drive the series effect, necessary to compensate the scale penalty, and potentially improving the economic performance of SMRs. This effect is illustrated below The potential of the economic drivers of SMRs is supported by experience in other industries (e.g. aviation). Nevertheless, it still needs additional empirical evidence for SMR technology. In this process, the access to a global market allowing the large-scale deployment of SMRs will be essential.

Beyond LCOE issues, the value proposition of SMRs also includes unique features such as access to off-grid/remote areas and non-electric applications. From a financial perspective, SMRs may represent an attractive investment principally due to the lower overall capital outlay compared to large reactors. This implies that private investors will face lower capital at risk, which could make SMRs a more affordable option. In turn, this could attract new sources of financing and lower the cost of capital. The ability to add modules incrementally provides additional financial flexibility, especially under sudden market shifts. Moreover, the shorter construction period may allow for shorter paybacks.



According to the IAEA[3], there are more than 50 SMRs concepts under development with different technology and licensing readiness levels. Among these concepts, around 50% are Gen-IV concepts also called, Advanced Small Modular Reactors (ASMRs). HTR-PM, a 200 MWe model of gas-cooled high-temperature reactor, is under construction in China. Some other are under a licensing process. Even if most of these projects are backed, to some extent, by GIF member states government, the involvement of the private sector has been increasing.

Countries like the United States and Canada have made significant progress in development of policies and licensing frameworks to accelerate the time-to-market of SMRs. Nevertheless, several challenges need to be overcome in order to achieve commercialization. Currently, there is a wide variety of ASMRs and this represents both an opportunity and a challenge. In the near term, the role of the first demonstrators will be crucial, not only to trigger the subsequent investments necessary to build a module factory, but also to downselect the most performant concepts. Additional efforts will be required to revisit the currently licensing frameworks, which rely extensively on the experience developed with Gen III/III+ and Gen II light water reactors. At

3. Advances in Small Modular Reactor Technology Developments 2018 https://aris.iaea.org/Publications/SMR-Book_2018.pdf

the same time, enabling policy frameworks and international collaboration will be continued to be key components for the timely deployment of new reactor concepts.

SIAP 2019 charge and response

Based on the market and industrial conditions described in Section 6.1, particularly those related to the ASMR ecosystem, SIAP was tasked in July 2019 to generate ideas and recommendations on how GIF should envisioned its interaction with private ASMR vendors. The development of the 2019 SIAP charge followed a first contact with private vendors held in May 2019 in Vancouver. Aspects covered in the charge included:

- The evaluation of the mutual benefits of higher involvement of the private sector in GIF activities including the right way for GIF to interact with vendors and addressing the need of reciprocity;
- Identification of R&D areas suitable for mutual co-operation with the objective to accelerate ASMR demonstration phase;
- Initial steps to involve ASMR designers.

One of the first outcomes of the 2019 SIAP charge was the necessity to define a set of criteria to evaluate/classify/downselect ASMR vendors. The selected vendors should be – at least – aligned with the GIF goals[4] and propose a mature design. Using the TRL scale, a level of 4 and 7 was judged as acceptable. SIAP members also highlighted the importance of a commitment from vendors to engage over an extended period of time. A potential way to rapidly engage with vendors may require the preparation of a set of questions using as starting the SIAP questionnaire developed in 2016 complemented, as appropriate by questions coming from SSCs chairs and other GIF TFs.

SIAP concluded that, in order to interact with the ASMRs designers in an effective manner, intellectual property rights (IPR) issues should be handled with care. IPR is a central issue in the setting up and operation of GIF, allowing public funded information to be exchanged among the partners, and this should not be endangered by involvement of the private sector. The same rules should apply to future private partners. Past experience within the GIF, shows that reaching an agreement in IPR aspects may require considerable efforts and this might discourage potential ASMR vendors.

The NEA NI2050 also provides valuable insights on the possible ways to interact with the private sector. Collaboration is easier at low TRLs as few intellectual property (IP) has been generated at this stage. As the concepts move to higher TRLs, IP on technology becomes more relevant hindering international collaboration. According to NI2050 finings, co-operation on the qualification of the technology may be more effective. The term “qualification” covers both industrial (i.e. codes and standards) and regulatory (i.e. licensing) qualification. If indeed countries could work together to reach common approaches (i.e. harmonization) of qualification processes of technologies/designs, this would greatly help reduce the time-to-market and broaden the potential market. In other words, if it would be possible to create a “common qualification pipe” that should attract ASMR vendors. GIF SSCs could also benefit from this *frame* as their activities are essentially technology focused. This topic might constitute the subject for a new cross-cutting task force within GIF. SIAP and NEA could also assist GIF to further develop this ideas and exchange with other organizations cumulating significant experience on the topic such as *Standard Developments Organizations* (SDOs), *Multinational Design Evaluation Programme* (MDEP) and the *Committee for Nuclear Regulatory Activities (CNRA)*.

The main following joint R&D activities may be of interest for the private sector, going beyond the pure technical topics. Potential areas include:

- advanced materials and manufacturing;

4. Sustainability, economics, safety and reliability, and proliferation resistance and physical protection.

- development of a common R&D infrastructure to accelerate qualification and demonstration;
- risk-informed methods and new related requirement to assist the licensing of advance concepts using alternative coolants and fuel arrangements;
- fuels and fuel cycles (both front-end and back-end activities);
- in-service inspection methodologies and their common qualification (i.e. European ENIQ experience).

Additional specific topics could be included after consulting the different SSCs. It is important to note that the aforementioned topics have strong cross-cutting dimensions, in line with the recent trends observed in GIF with the establishment of more horizontal working groups and TFs.

To initiate the interaction with the private sector, SIAP recommended GIF to set up a small group of experts to pre-select a first group of ASMRs vendors while keeping in mind the possibility to broaden later depending on market trends. A questionnaire could be sent then to the selected vendors in order to evaluate their responsiveness and willingness to join GIF communities. Based on their responses, a series of well-designed ad hoc meetings could be held with a first group of ASMRs vendors. Meetings could take place at the SSC levels (vendors could be regrouped by coolant type) or with more cross-cutting working groups or TFs (economics, safety, advanced manufacturing, etc.).

Additionally, and in line with the main R&D areas depicted above, two contiguous workshops will take place in Paris in February 2020: one on advanced manufacturing immediately followed by another on R&D infrastructures needs and opportunities. Using as a base the outcomes of the 2019 SIAP charge detailed in this chapter, these events will provide additional insights to properly assess the potential win-win areas for long-term co-operation between GIF and the private sector.

SIAP intentions for 2020

Since inception, the GIF has keyed on and supported the (necessary) R&D elements to support Gen-IV systems. The commercial SMR thrust has recently awakened more interest in nuclear power. SIAP seeks to advise and support GIF to harness this new momentum.

SIAP stands ready to support GIF looking at streamlining, establish an information campaign, how to convince the power generation community that the ASMR systems are ready to replace fossil fuel plants, and promote make nuclear licensing more international/transportable.

Eric Loewen

Chair of the SIAP and all Contributors

Appendix 1. List of Generation IV specific abbreviations and acronyms

AF	Advanced Fuel
AMME	Advanced Manufacturing and Materials Engineering
ARIS	Advanced Reactor Information System
CD&BOP	Component Design and Balance-of-Plant
CD&S	Conceptual Design and Safety
CMVB	Computational Methods Validation and Benchmarking
EG	Experts Group
EMWG	Economic Modelling Working Group
ETTF	Education and Training Task Force
ETWG	Education and Training Working Group
FA	Framework Agreement
FCM	Fuel and Core Material
FFC	Fuel and Fuel Cycle
GACID	Global Actinide Cycle International Demonstration
GIF	Generation IV International Forum
GFR	Gas-cooled fast reactor
HP	Hydrogen Production
HTR	High-Temperature gas-cooled Reactor
ISAM	Integrated Safety Assessment Methodology
LFR	Lead-cooled Fast Reactor
M&C	Materials and Chemistry
MAT	Materials (VHTR project)
MSR	Molten Salt Reactor
MWG	Methodology Working Group
PA	Project Arrangement
PD	Policy Director
PG	Policy Group
PMB	Project Management Board
PP	Physical Protection or Project Plan
PR	Proliferation resistance
PR&PP	Proliferation Resistance and Physical Protection

PSSC	Provisional System Steering Committee
RDTF	R&D infrastructure Task Force
RSWG	Risk and Safety Working Group
SA	System arrangement
SCWR	Supercritical-Water-cooled Reactor
SDC	Safety Design Criteria
SDG	Safety Design Guidelines
SFR	Sodium-cooled Fast Reactor
SIA	System Integration and Assessment
SIAP	Senior Industry Advisory Panel
SO	Safety and Operation
SRP	System research plan
SSC	System Steering Committee
TD	Technical Director
TF	Task force
TH&S	Thermal-hydraulics and Safety
TS	Technical Secretariat
ToR	Terms of Reference
VHTR	Very-high-temperature reactor
WG	Working group
WGSAR	Working Group on the Safety Advanced Reactors

Technical terms, projects and facility acronyms

ACP	Code for the Chinese SMR (PWR type)
ACRS	Advisory Committee on Reactor Safeguards
ADRIANA	ADvanced Reactor Initiative And Network Arrangement
ADS	Accelerator-driven system
AECS	Advanced Energy Conversion System
AGR	Advanced gas-cooled reactor (United States)
AFA	Alumina Forming Austenitic
AFR	Advanced Fast Reactor
AHFM	Algebraic Heat Flux Model
ALFRED	Advanced lead fast reactor European demonstrator
ALLEGRO	Gas Fast Reactor Project
AMR	Advanced Modular Reactor
ANM	ANSTO Nuclear Medicine
ASTRID	Advanced Sodium Technological Reactor for Industrial Demonstration
ART	Advanced Reactor Technology program (United States)

ASMR	Advanced Small Modular Reactor
ATR	Advanced Test Reactor (at INL)
ATWR	Anticipated Transient Without Scram
AVR	Arbeitsgemeinschaft Versuchsreaktor
BWR	Boiling Water Reactor
CASLER	Co-operative Alliance for Small Lead-based Fast Reactor
CIIALER	Chinese Industry Innovation Alliance of Lead-based Reactor
CEFR	China Experimental Fast Reactor
CFD	Computational Fluid Dynamics
CFR	Chinese Sodium Fast Reactor
CGR	Crack Growth Rate
CLEAR	China Lead-based Reactor
CNEPP	Comprehensive Nuclear Energy Promotion Plan (Korea)
CNRI	Canadian Nuclear Research Initiative
COLA	Combined License Application
CRP	Co-ordinated Research Project
DCA	Design Certification Application
DG	Director-General
DHR	Decay heat removal
EBR	Experimental Breeder Reactor (United States)
ECC-SMART	European-Canadian-Chinese Small Modular SCWR
ECFM	Eddy Current FlowMeter
ECS	Energy Conversion System
ELFR	European Lead Fast Reactor
EPR	European Pressurized Reactor
EPZ	Emergency Planning Zones
ESFR	European Sodium Fast Reactor
ESP	Early Site Permit
E&T	Education & Training
FA	Fuel Assembly
FEA	Finite Element Analysis
FLIBE	mixture of lithium and beryllium fluoride (BeF2)
FLINAK	salt mixture of LiF-NaF-KF
FOA	Funding Opportunity Announcement (United States)
FP	Framework Program
FSA	Fuel SubAssembly
FHR	Fluoride salt-cooled high-temperature reactor
FOAK	First-Of-A-Kind

FSR	Fast Sodium Reactor
FR	Fast Reactor
FY	Financial Year or Fiscal Year
GAIN	Gateway for Accelerated Innovation in Nuclear
GW	GigaWatt
GWD/MTHM	Gigawatt-Days per Metric Tonne of Heavy Metal
HANARO	High-flux advanced neutron application reactor
HEEP	Hydrogen Economic Evaluation Program
HINEG	High Intensity D-T fusion NEutron Generator[2]
HFR	High Flux Reactor
HLD	High-Level Deliverable
HLMC	Heavy Liquid Metal Coolant
HPR	Advanced Pressurized Water Reactor
HTDM	High-Temperature Design Methodology
HTGR	High-Temperature Gas-cooled Reactor
HTR	High-Temperature Reactor
HTR-PM	High-temperature gas-cooled reactor power module
HTSE	High-Temperature Steam Electrolysis
HTTR	High-Temperature Test Reactor
ICT	Information Communication Technology
IG	InterGranular
IHX	Intermediate Heat Exchanger
ILW	Intermediate Level Waste
IPP	Independent Power Producer
IRP	Integrated Resource Plan
IRRS	Integrated Regulatory Review Service
ISI&R	In-Service Inspection and Repair
JHR	Jules Horowitz Reactor
JRC	Joint Research Centre
JSFR	Japanese Sodium-cooled Fast Reactor
KALIMER	Korea Advanced Liquid Metal Reactor
KKL	Kernkraftwerk Leibstadt Reactor (BWR)
KM	Knowledge Management
LBL	Leach-Burn-Leach
LCOE	Levelized Cost Of Energy
LOCA	Loss-Of-Coolant Accident
LWR	Light Water Reactor
LBE	Lead-Bismuth Eutectic

LMFNS	Liquid Metal-cooled Fast Neutron Systems
LTE	Low Temperature Electrolysis
LTS	Licensing Technical Support
MA	Minor Actinides
MAWP	Maximum Allowable Working Pressure
MINERVA	Micro Nuclear Energy Research and Verification Arena
MBIR	Russian multipurpose fast neutron research reactor
M-HEM	Modified Homogeneous Equilibrium Model
MOSART	Molten Salt Actinide Recycler and Transmuter
MoU	Memorandum of Understanding
MOX	Mixed oxide fuel
MMR	Micro Modular Reactor
MSFR	Molten salt fast reactor
MYRRHA	Multi-purpose Hybrid Research Reactor for High-tech Applications
MW	MegaWatt
NC	Natural Circulation or Natural Convection
NEUP	Nuclear Energy University Program
NPP	Nuclear power plant
NSSS	Nuclear Steam Supply System
NSR	Northern Sea Route
NSTF	Natural Convection Shutdown Heat Removal Test Facility
NRAD	Neutron Radiography (NRAD) Reactor
NRWMF	National Radioactive Waste Management Facility
NUWARD	French PWR SMR Project
ODS	Oxide dispersion-strengthened
OPAL	Open Pool Australian Lightwater reactor
OPT	Objective Provision Tree
O&TF	Operation technology and Testing Facilities
PEACER	Prolif.-resistant Environment-friendly Accident-tolerant Continual Energy Economical Reactor
PGSFR	Prototype Generation IV Sodium-Cooled Fast Reactor
PILLAR	Pool-type Integral Leading test facility for lead-alloy cooled SMR
PDE	Post-Disassembly Expansion
PDHRS	Passive Decay Heat Removal System
PIE	Post-Irradiation Examinations
PP	Primary Pump
PPE	Multiannual Energy Plan (France)
PRISM	Power Reactor Innovative Small Module

PSID	Preliminary Safety Information Document
PV	Photovoltaic
PWR	Pressurized Water Reactor
QSR	Qualitative Safety features Review
RANS	Reynolds Analysis Navier-Stokes
RCCS	Reactor Cavity Cooling System
R&D	Research and Development
RD&D	Research Development & Demonstration
RRDB	Research Reactor DataBase
RTFDB	Research and Test Facilities DataBase
SASS	Self Actuated Shutdown System
SCC	Stress corrosion cracking
S-CO_2	Supercritical Carbon Dioxide
SCW	SuperCritical Water
SCWL	SuperCritical Water Loop
SDSAR	Specific Design Safety Analysis Report
SELAAD	Sodium Exp. Loop for Advanced Aerosol Detection
SEM	Scanning Electron Microscopy
SER	Safety Evaluation Report
SFEAR	Support Facilities for Existing and Advanced Reactors
SG	Steam generator
S-I	Sulphur-Iodine process
SMART	System-integrated Modular Advanced Reactor
SME	Small and Medium Enterprise
SMR	Small modular reactor
SNETP	Sustainable Nuclear Energy Technology Platform
SNF	Spent Nuclear Fuel
SSTAR	Small, Sealed, Transportable, Autonomous Reactor
STELLA	Sodium integral effect test loop for safety simulation and assessment
TAMAT	Towards Advanced Material for Energy Technologies
TAREF	Task Group on Advanced Experimental Facilities
TEM	Transmission Electron Microscopy
THTR	Thorium high-Temperature Reactor
TH-U	Thorium-Uranium
TMS	Tempered Martensitic Steel
TMSR	Thorium Molten Salt Reactor (China)
TORIA	Thorium-optimized Radioisotope Incineration Arena
TRISO	Tri-structural isotopic (nuclear fuel)

TRL	Technology Readiness Level
TRU	TransUranic
UCO	Uranium OxyCarbide
ULOF	Unprotected Loss Of Flow
UOX	Uranium Oxyde
VRE	Variable Energy Ressources
VTR	Versatile Testing Reactor (United States)
VVER	Russian light water power pressurized reactor model
WALSUM	Water mock-up test for Advanced Leak Simulation and Upgraded Monitoring system

Organizations, Companies and Agency

ANL	Argonne National Laboratory (United States)
ANS	American Nuclear Society
ANSTO	Australian Nuclear Science and Technology Organisation
ARC	DOE Office of Advanced Reactor Concepts (United States)
ARPANSA	Australian Radiation Protection and Nuclear Safety Agency
ASME	American Society of Mechanical Engineers
ASN	Autorité de Sûreté Nucléaire (France)
BEIS	Business, Energy and Industrial Strategy Dpt (UK)
CAEA	China Atomic Energy Authority (China)
CAS	Chinese Academy of Science
CBCG	Columbia Basin Consulting Group
CEA	Commissariat à l'énergie atomique et aux énergies alternatives (France)
CGN	China General Nuclear Power Group
CIAE	China Institute of Atomic Energy
CIGEO	Centre Industriel de stockage géologique (France)
CNL	Canadian Nuclear Laboratories
CNNC	Chinese National Nuclear Corporation
CNRA	Committee for Nuclear Regulatory Authorities (NEA)
CNRS	Centre national de la recherche scientifique (France)
CNSC	Canadian Nuclear Safety Commission
DOE	Department of Energy (United States)
EC	European Commission
EDF	Electricité de France
ENEA	Italian National Agency for New Technologies, Energy and Sustainable Economic Development
EPFL	École Polytechnique Fédérale de Lausanne

ETHZ	Eidgenössische Technische Hochschule Zürich
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 (NEA)
INET	Institute of Nuclear and New Energy Technology (China)
INEST	Institute of Nuclear Energy Safety Technology (China)
INL	Idaho National Laboratory (United States)
INPRO	International Project on Innovative Nuclear Reactors and Fuel Cycles (IAEA)
IPPE	Institute of Physics and Power Engineering (Russia)
ITU	Institute for Transuranium Elements (Euratom)
JAEA	Japan Atomic Energy Agency
JRC	Joint Research Centre (Euratom)
KAERI	Korea Atomic Energy Research Institute
KAIST	Korea Advanced Institute of Science and Technology
KEPCO	Korea Electric Power Corporation
KINGS	KEPCO International Nuclear Graduate School
KIT	Karlsruhe Institute of Technology (Germany)
LANL	Los Alamos National Laboratory (United States)
LLNL	Lawrence Livermore National Laboratory (United States)
MDEP	Multinational Design Evaluation Program (NEA)
MOST	Ministry of Science and Technology (China)
MSIT	Ministry of Science, Information and Technology
MTA	Hungarian Academy of Sciences Centre for Energy Research
NCBJ	Narodowe Centrum Badan Jadrowych (Poland)
NEA	Nuclear Energy Agency
NEICA	Nuclear Energy Innovation Capabilities Act (United States)
NEIMA	Nuclear Energy Innovation and Modernization Act (United States)
NIRAB	Nuclear Innovation & Research Advisory Board (UK)
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
NSSC	Nuclear Safety and Security Commission (China)
NTPD	Nuclear Power Technology Development Section (IAEA)
OECD	Organisation for Economic Co-operation and Development

OPG	Ontario Power Generation
ORNL	Oak Ridge National Laboratory (United States)
PNNL	Pacific Northwest National Laboratory
PSI	Paul Scherrer Institute (Switzerland)
RATEN	Regia Autonoma Tehnologii Pentru Energia Nucleara (Romania)
RIAR	Research Institute of Atomic Reactors (Russia)
SINAP	Shanghai Institute of Applied Physics
SJTU	Shanghai Jiaotong University
SNU	Seoul National University
SPIC	State Power Investment Corporation (China)
TAEK	Turkish Atomic Energy Authority
TEPCO	Tokyo Electric Power Company
TUBITAK	Scientific and Technological Research Council of Turkey
TVA	Tennessee Valley Authority
UAMPS	Utah Associated Municipal Power Systems
UNIST	Ulsan National Institute of Science and Technology
USTC	University of Sciences and Technology of China
V4G4	Visegrad GEN-4 Centre of Excellence
VTT	Valtion Teknillinen Tutkimuskeskus (Finland)
VUJE	Slovakian engineering company
WANO	World Association of Nuclear Operators
XJUT	Xi'an Jiaotong University (China)

Appendix 2. **Selection of GIF publications (2018-2019)**

This list is not exhaustive because the large amount of scientific papers in relations with GIF systems and cross-cutting actions is far too wide. It must be noted that GIF hold a Symposium in 2018 where all the papers presented are in relation with Generation IV. The proceedings of the 2018 Symposium can be uploaded on the GIF website. This list is therefore highlighting some specific relevant papers from the 4th GIF Symposium and from other publications (scientific journals or papers presented to international conferences).

General Paper

Kamide, H. and S. Pivet (2019), “Development and deployment of advanced nuclear power technologies to increase the use of low-carbon energy”, Intl Conf. on Climate Change and the Role of Nuclear Power, Vienna, Austria, 7-9 Oct. 2019.

Abousahl, S. et al. (2018), “10 years’ overview of a successful contribution of EURATOM to Generation IV International Forum”, *Proc. of the 4th GIF Symposium*, Paris, France, 16-18 Oct.

GFR

Hatala, B. (2018), “Progress in GFR Technology”, *Proc. of the 4th GIF Symposium*, France, Paris, 16-18 Oct.

Hatala, B. et al. (2019), “Progress in GFR Technology”, *Proc. of ICAPP 2019 – International Congress on Advances in Nuclear Power Plants*, Juan-les-pins, France, 12-15 May.

LFR

Frignani, M., A. Alemberti and M. Tarantino (2019), “ALFRED: A Revised concept to improve pool related thermal-hydraulics”, *Nuclear Engineering and Design*, Elsevier, Vol. 355.

Moreau, V. et al. (2019), “Pool CFD modelling: Lessons from the sesame project”, *Nuclear Engineering and Design*, Vol. 355.

Roelofs, F. and SESAME project partners (2019), *Thermal Hydraulics Aspects of Liquid Metal Cooled Nuclear Reactors*, Woodhead Publishing, Elsevier Ltd. ISBN: 978-0-08-101980-1 (print) ISBN: 978-0-08-101981-8 (online).

Kuwagaki, K., J. Nishiyama, and T. Obara (2019), “Concept of breed and burn reactor with spiral fuel shuffling”, *Annals of Nuclear Energy*, Vol. 127, pp. 130-138.

Bak, S-I., S-W. Hong and Y. Kadi (2019), “Design of an accelerator-driven subcritical dual fluid reactor for transmutation of actinides”, *The European Physical Journal Plus*, Vol. 134.

Ferroni, P. et al. (2019), “The Westinghouse Lead Fast Reactor,” International Congress on Advances in Nuclear Power Plants (ICAPP), Juan-les-pins, France, 12-15 May 2019.

Lemekhov, V.V. et al. (2019), “Modeling of the Wearing for Coupling of Tube-Spacer Grid of the Steam Generator of the Lead Coolant Nuclear Reactor”, *Atomic Energy*, Vol. 127, Number 4, pp. 7-11.

Solonin, V.I. et al. (2019), "Metal Liner Reliability Assessment for BREST-OD-300 Reactor Vessel Accounting for Brittle Fracture and Leaks", Herald of the Bauman Moscow State Technical University. Series Mechanical Engineering, Number 5 (128).

MSR

Guidez, J. et al. (2018), "Status of current knowledge and developments in France on Molten Salt Reactor", *Proc. of the 4th GIF Symposium*, Paris, France, 16-18 Oct.

Ignatiev, V. et al. (2018), "Molten-salt reactor as a necessary element of the nuclear fuel cycle closure for all actinides", *Proc. of the 4th GIF Symposium*, Paris, France, 16-18 Oct.

Uhlir, J. et al. (2018), "Current progress in experimental development of MSR and FHR technologies", *Proc. of the 4th GIF Symposium*, Paris, France, 16-18 Oct.

Delpech, S. et al. (2018), "Design and safety studies of the molteb salt fast reactor concept in the frame of the SAMOFAR H2020 project", *Proc. of the 4th GIF Symposium*, Paris, France, 16-18 Oct.

Feng, B. et al. (2019), "Core and Fuel Cycle performance of a Molten Salt Fast Reactor", *Proc. of ICAPP 2019 – International Congress on Advances in Nuclear Power Plants*, Juan-les-pins, France, 12-15 May.

SFR

Ashurko, Y. and P. Fomichenko (2018), "Realization of GEN-IV requirements in the BN-1200 Project", *Proc. of the 4th GIF Symposium*, Paris, France, 16-18 Oct.

Guidez, J. et al. (2018), "New safety measures for European Sodium Fast Reactor in Horizon 2020 ESFR-SMART Project, *Proc. of the 4th GIF Symposium*, Paris, France, 16-18 Oct.

Ohtsuka, S. et al. (2018), "Development of ODS tempered martensitic steel for high burn-up fuel cladding tube of SFR", *Proc. of the 4th GIF Symposium*, Paris, France, 16-18 Oct.

Plancq, D. et al. (2018), "Progress in ASTRID Gas power conversion system development", *Proc. of the 4th GIF Symposium*, Paris, France, 16-18 Oct.

Yamano, H. et al. (2019), "Activities of the GIF Safety and Operation Project of Sodium-Cooled Fast Reactor Systems", *Proc of the 27th ICONE Conf*, 19-24 May 2019, Tsukuba, Ibaraki, Japan.

Baqué, F. et al., (2018), "In service Inspection d Repair development for SFRs", *Proc. of the 4th GIF Symposium*, Paris, France, 16-18 Oct.

SCWR

Conference: The 9th International Symposium on Supercritical-Water-Cooled Reactors (ISSCWR-9), www.cns-snc.ca/events/isscwr9.

Vasić and T.G. Beuthemm (2019), "A Framework of Supercritical Heat Transfer Prediction Method Development", *Proc. of the 9th International Symposium on SCWRs (ISSCWR-9)*, Vancouver, Canada, 10-14 March 2019.

Lv, H., et al. (2019), "Investigation on heat transfer of in-tube supercritical water cooling accompanying out-tube pool boiling", *International Journal of Heat and Mass Transfer*, Vol. 136, pp. 938-949.

Musa, A. et al. (2020), "Licensing activity and code validation for generation IV SCW technology", *Nuclear Engineering and Design*, Vol. 357, 110424, ISSN 0029-5493.

Buzzi, F., A. Pucciarelli and W. Ambrosini (2019), "On the mechanism of final heat transfer restoration at the transition to gas-like fluid at supercritical pressure: A description by CFD analyses", *Nuclear Engineering and Design,* 355, 110345.

Chen, K. et al. (2019), "Characterizing the effects of in-situ sensitization on stress corrosion cracking of austenitic steels in supercritical water", *Scripta Materialia*, Vol. 158, pp. 66-70.

Sun, S. et al. (2019), "Evolution of microstructure and mechanical properties of an oxide dispersion strengthened austenitic steel during aging at 973K", Mater. Res. Express 6 085550.

VHTR

Fuetterer, F. et al. (2018), "Recent advances in the GIF Very High Temperature Reactor System", *Proc. of the 4th GIF Symposium*, Paris, France, 16-18 Oct.

Freism D, et al. (2018), "Burn-up determination and accident testing of HTR-PM fuel elements irradiated in the HFR Petten", *Proceedings of the 9th International Topical Meeting on High Temperature Reactor Technology (HTR-2018)*, 8-10 Oct. 2018, Warsaw, Poland.

Zhao, H. et al. (2019), "A one-dimensional code of the passive residual heat removal system for the modular high temperature gas-cooled reactor", *Progress in Nuclear Energy*, Vol. 110, pp. 374–383.

Zhang, Z. et al. (2019), "HTR-PM: Making dreams come true", *Nuclear Engineering International*, Vol. 64, pp. 16-18.

Gougar, H. et al. (2020), "The US Department of Energy's high temperature reactor research and development program – Progress as of 2019", *Nuclear Engineering and design*, Vol. 358, 110397.

Shin, Y. et al. (2018), "Co-generation of hydrogen and electricity using 350 MWth HTGR-based SMR, HTSE, and SI processes", *The 6th International Conference on Nuclear and Renewable Energy Resources (NURER2018)*, 30 Sep. to 03 Oct. 2018, Jeju, Korea.

EMWG

Moore, M. et al. (2017), "Benchmarking of Nuclear Economics Tools", *Annals of Nuclear Energy*, Vol. 103, pp. 122-129.

Sadhankar, R. et al. (2018), "Benchmarking of Economic Models for Nuclear Hydrogen Production", *Proceedings of the Pacific Basin Nuclear Conference*, San Francisco, United States, 30 September to 5 October 2018.

Mendoza, A., M. Berthelemy and R. Sadhankar (2018), "EMWG Position Paper on the Impact of Increasing Share of Renewables on the Deployment of Generation IV Nuclear Systems", *Proc. of the 4th GIF Symposium*, Paris, France, 16-18 Oct. 2018.

Mukaida, K. et al. (2019), "Levelized cost of electricity evaluation of SFR system considering safety measures", *Proc. of ICAPP 2019 – International Congress on Advances in Nuclear Power Plants*, Juan-les-pins, France, 12-15 May 2019.

PRPP WG

Cojazzi, G. et al. (2018), "The GIF Proliferation Resistance and Physical Protection working group (PRPPWG): achievements and perspectives", *Proc. of the 4th GIF Symposium*, Paris, France, 16-18 Oct. 2018.

Cojazzi, G. et al. (2018), "The GIF Proliferation Resistance and Physical Protection methodology applied to GEN IV system designs: Some reflections", In Building Future Safeguards Capabilities (IAEA-CN-127), 05-08 November 2018, Vienna.

Cheng, L. et al. (2019), "The GIF Proliferation Resistance and Physical Protection Methodology Applied to GEN IV System Designs: An Update". *Proc of 41st ESARDA Annual Meeting Symposium on Safeguards and Nuclear Material Management*, 14-16 May 2019, Stresa, Italy.

Cipiti, B. et al. (2019), "An Update of the GIF Proliferation Resistance and Physical Protection White Papers for the Six Gen IV Systems", JRC117892., Workshop INMM-ESARDA, Tokyo, October 2019.

RSWG

Okano, Y. et al. (2018), "GIF Risk and Safety Working Group: Applications of the ISAM methodology to Gen-IV nuclear systems", *Proc. of the 4th GIF Symposium*, Paris, France, 16-18 Oct. 2018.

Guidez, J. et al. (2018), "Application of the practical elimination approach for GEN IV reactor designs", *Proc. of the 4th GIF Symposium*, Paris, France, 16-18 Oct. 2018.

Cipiti, B. et al. (2018), "Developing a Molten salt Reactor safeguard model", *Proc. of the 4th GIF Symposium*, Paris, France, 16-18 Oct. 2018.

ETTF

Paviet, P. (2019), "The GIF Webinar Initiative: Past, Present and Future", *Proc of Global 2019*, 22-26 Sept. 2019, Seattle Washington, United States.

Mikityuk, K. et al. (2018), "GIF Webinar: An online educational resource", *Proc. of the 4th GIF Symposium*, Paris, France, 16-18 Oct. 2018.

Latgé, C. et al. (2018), "Teaching Sodium Fast Reactors in CEA", *Proc. of the 4th GIF Symposium*, Paris, France, 16-18 Oct. 2018.

SDC-TF

Kubo, S. (2018), "Development of safety design guidelines on structures, systems and components for Generation IV Sodium-cooled Fast Reactor Systems", *Proc. of the 4th GIF Symposium*, Paris, France, 16-18 Oct. 2018.

AMME TF

Muransky, O. (2018), "Development and assessment of materials for the Generation-IV Nuclear reactors", *Proc. of the 4th GIF Symposium*, Paris, France, 16-18 Oct. 2018.

Petesch, C. (2018), "CEN Workshop 64: An innovative way to work on a harmonized set of rules for GEN IV reactors", *Proc. of the 4th GIF Symposium*, Paris, France, 16-18 Oct 2018.

RDTF

Garbil, R. (2018), "Generation –IV systems experimental infrastructure needs", *Proc. of the 4th GIF Symposium*, Paris, France, 16-18 Oct. 2018.

Leung, L. (2018), "R&D experimental capabilities for advancing GIF SCWR system in the next decade", *Proc. of the 4th GIF Symposium*, Paris, France, 16-18 Oct. 2018.

Zheng, Y. (2018), "Introduction on some experimental facilities for VHTR system", *Proc. of the 4th GIF Symposium*, Paris, France, 16-18 Oct. 2018.

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This twelfth edition of the *Generation IV International Forum (GIF) Annual Report* covers actions in 2018 and 2019. In 2018, the *Fourth GIF Symposium Proceedings* was issued in place of the Annual Report.

In 2019 the GIF entirely renewed its Board with new members in all key governance positions. Moreover, for the first time in the history of GIF management, each Vice-chair was granted a three-year mandate, thus assisting the GIF Chairman to better understand the drivers, opportunities, and constraints related to three key cross-cutting topics connected with all GEN IV systems: Regulatory Issues; Market Opportunities and Challenges; and Enhancement of R&D Collaborations. In terms of management, GIF has kept the structure that has proved successful in the past.

This Annual Report also includes a list of selected related scientific publications that show the relevance and the high scientific quality of the research carried out by all GIF members.

For the first time, this Annual Report will only be published in an electronic format, available on the GIF Website.

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