GEN IV International Forum

Expertise | Collaboration | Excellence

ANNUAL REPORT 2020

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Foreword from the Chair

It is an honour for me to compose this foreword to the *Generation IV International Forum 2020 Annual Report*, which outlines progress in Generation IV reactor systems and collaboration on developments, even during these difficult times caused by the COVID-19 pandemic around the world. Gen-IV reactor systems are the next generation following the current Gen-III and Gen-III+ light water reactors, and they will ensure the sustainable use of nuclear energy in future.

Since 2001, GIF has been promoting international collaboration for the research and development (R&D) of six types of Gen-IV reactor systems using sodium, lead, gas, molten salt, and supercritical water coolants. 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 beginning in 2030. These systems follow the common development goals established by GIF since its very beginning 20 years ago: “safety” and “economics” are two key goals, together with “sustainability” and “proliferation resistance and physical protection”. These goals remain essential if we are to achieve a breakthrough in nuclear energy, which is even more important today to contribute to a carbon-neutral energy sector alongside renewable energy sources. In recent years, small modular reactors (SMRs) are increasingly being advocated as one of the most innovative nuclear technologies. Gen-IV reactor concepts have been adopted in many SMR developments because of their high level of safety and high temperatures for heat applications, as well as of their nuclear fuel cycle capabilities.

“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.” The priorities of GIF are: 1) safety and regulation: continue the development of international safety design criteria to facilitate future licensing activities; 2) market opportunities 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; and 4) youth: continue to attract the younger generation to the nuclear sector. These priorities have been championed by our three outstanding vice-chairs; Ms Alice Caponiti (United States) for safety; Mr Sylvestre Pivet (France) for market opportunities; and Mr Jong Hyuk Baek (Korea) for R&D.

Given the importance of international safety standards for the licensing of Gen-IV reactor designs, GIF has developed safety design criteria and guidelines, initially for the sodium-cooled fast reactors but eventually for other reactor systems, for example very-high-temperature gas-cooled reactors, lead-cooled fast reactors and molten salt reactors. To continue this work, GIF is also engaging with the nuclear safety community at the international level (i.e. with the Organisation for Economic Co-operation and Development/Nuclear Energy Agency [NEA] and the International Atomic Energy Agency [IAEA]). Risk-informed and performance-based approaches, and the reduction of emergency planning zones (EPZ) for SMRs – an important part of discussions at the IAEA – are great concerns for GIF in terms of the early deployment of Gen-IV reactors. The IAEA-GIF interface meeting in 2020 provided a good opportunity to enhance our collaboration in this area and in many others.

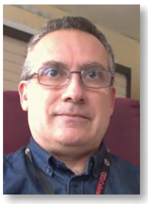
GIF is convinced that stronger cooperation is needed between R&D bodies and the private sector. It will be especially important to integrate future market opportunities and constraints at the design stage. GIF is also working towards expanding cooperation with international organizations, regulatory bodies and the private sector. To this end, GIF recently contributed to a report entitled *Flexible Nuclear Energy for Clean Energy Systems*, released by the Clean Energy Ministerial (CEM) and Nuclear Innovation: Clean Energy Future (NICE Future) initiative.

Through GIF orientations, more than ever we would like to highlight organizational output to policymakers, industry and to people around the world so as to promote progress achieved on the Gen-IV reactor systems during such difficult and complex times in the energy market, with the expanding use of renewable energies, but also with growing concerns for safety after the Fukushima Daiichi Nuclear Power Plant accident, and increasingly violent weather issues resulting from global warming. We have had several occasions to express our opinions and demonstrate our results to the world. For example, GIF had the privilege of talking about fast neutron reactors and their benefits for a clean energy society at the 2020 IAEA Scientific Forum.

This annual report covers our overall activities in 2020 in relation to six reactor systems and the cross-cutting issues overseen by GIF task forces and working groups. I hope that this annual report will be meaningful for all of our readers, encouraging good relations and collaboration.

Hideki KAMIDE
GIF Chair

Foreword from the Technical Director



The year 2020 has been a complicated and unexpected one for all of us, with the global sanitary crisis upsetting our personal lives, our relationships with the world, and our way of working and communicating.

The Generation IV International Forum (GIF) has of course been deeply affected by this situation, especially when an important part of its organization is based on expert exchanges, promoting direct contact and face-to-face meetings.

We have had to invent new methods of working, therefore, promoting digital tools, and in general showing that we can adapt to diverse situations thanks to our collective agility.

In this context, it has not been easy to stay aligned with the GIF perspectives that were presented in the 2019 Programme Plan of the Technical Director. Together, we managed together to avoid any breaks in the GIF workflow among experts, and to pursue our fundamental goal of repositioning GIF in the new context of a deregulated energy market and a decarbonized, future society.

Together, we have experienced some great achievements, despite the COVID-19 situation that has hit all countries hard, and I would like to highlight some of these achievements below:

- We have made a first step towards the nuclear private sector and start-ups by organizing a very successful "workshop on Advanced Manufacturing and Research Infrastructure". The workshop was held in February 2020, and it was close to the last live event for GIF, before the general lockdown in many countries around the world. The workshop was a significant milestone because: 1) it will be the first in a long series of planned events for the coming years (see Chapter 2 of this report); and 2) it was during this workshop that we started the experience of a hybrid conference (i.e. a face-to-face meeting and an online video link for our Chinese colleagues). I am certain that the future transition towards normal, global working and travelling conditions will again allow us to make use of such hybrid events.
- We have pursued our studies on all systems, and in all working groups and task forces. Some of our work has reached a successful conclusion (e.g. the R&D Infrastructure Task Force, sodium-cooled fast reactor safety design criteria), while significant milestones have been achieved in others (e.g. the White Paper on Flexibility, the White Paper on Proliferation Resistance and Physical Protection for some systems, and the synthesis review of fuel qualification for all six systems).
- We have paved the way for the near future action towards creating a GIF brainstorming group on the non-electrical application of nuclear heat, producing recommendations to increase the link between GIF and the SMR sector, and proposing cross-cutting actions between molten salt reactor and safety experts.

In this context, we have even found opportunities to completely refresh GIF branding with a new logo, a completely redesigned GIF website and a new communications approach. We have also confirmed our unique position in education and training, with our webinars seeing a very significant increase in participation levels.

I am thus very grateful for all of the efforts that have been undertaken to synthesize the chapters that follow in this GIF 2020 Annual Report because the context was particularly unfavourable. We have nonetheless found the resources to keep our motivation, willingness and enthusiasm.

Gilles RODRIGUEZ
GIF Technical Director

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1 GIF membership, organization and R&D collaboration

In 2020, the Generation IV International Forum (GIF) structure and organization remained the same: composed of 14 member countries, all of which are signatories of the GIF founding document.

The present chapter focuses only on major changes that occurred in 2020. A detailed explanation of the GIF membership and organization can be found in the corresponding chapter of the *GIF 2019 Annual Report* (see www.gen-4.org/gif/jcms/c_119025/gif-2019-annual-report), or on the GIF website.

GIF global governance remained unchanged in 2020 and is summarized in Figure 1-1 below.

With the aim of stabilizing and reinforcing the GIF Technical Secretariat, several staff changes occurred in the NEA/GIF Technical Secretariat, and these changes are outlined in the GIF October newsletter (see www.gen-4.org/gif/jcms/c_121997/-5-gif-newsletter-rev2) written by Sama Bilbao y León, who was Head of the Technical Secretariat until mid-September 2020:

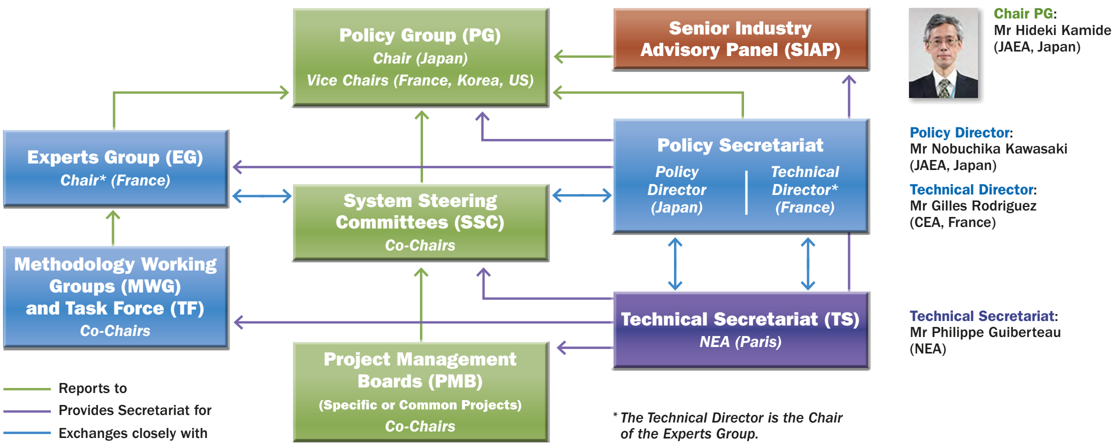
- Mr Philippe Guiberteau replaced Sama Bilbao y León when she left the NEA in September 2020. He also provides secretariat support for the supercritical-water-cooled reactor (SCWR) system and sub groups.
- Ms Kathryn Obisesan also joined the GIF secretariat in mid-September, supporting the Education and Training Working Group (ETWG), the Advanced Manufacturing Material Engineering Task Force (AM TF), and the Research and Development Infrastructure Task Force (RDTF).
- In early November, Dr Masahiro Nishimura took over the secretariat for all of the sodium-cooled fast reactor (SFR) system groups from Mr Satoshi Okajima, who was a key member of the GIF team for the last two and a half years.
- Also in early November, Dr Stephanie Cornet left the NEA and the GIF Technical Secretariat of the Very-High-Temperature Reactor (VHTR) Fuel and Fuel Cycle (FFC) project. She was replaced by Dr Davide Costa, who is also from the NEA Division of Nuclear Science.

The structure of the GIF Technical Structure, as of November 2020, is shown in Figure 1-2.

In terms of the GIF structure, the main changes and objectives are:

- Activities of the SFR Safety Design Criteria Task Force (SDC TF) merged into the Risk and Safety Working Group (RSWG), with the SDC TF disbanded in late 2020.
- Creation of the Molten Salt Reactor System (System Research Plan, System Agreement, Project Plans and Project Arrangements for three new project management boards) from the current, provisional System Steering Committee (pSSC) is a main objective for 2021.
- Reflections on the potential continuation of the RDTF, which completed its work at the end of 2020.
- Because of its important and key role in the continuous dissemination of Gen-IV scientific knowledge, at the end of 2019 the Education and Training Task Force (ETTF) was transformed into the Education and Training Working Group (ETWG), which entered into force in 2020.
- A first meeting was organized at the end of 2020 to explore the possibility of undertaking activities

Figure 1-1: GIF Governance in 2020



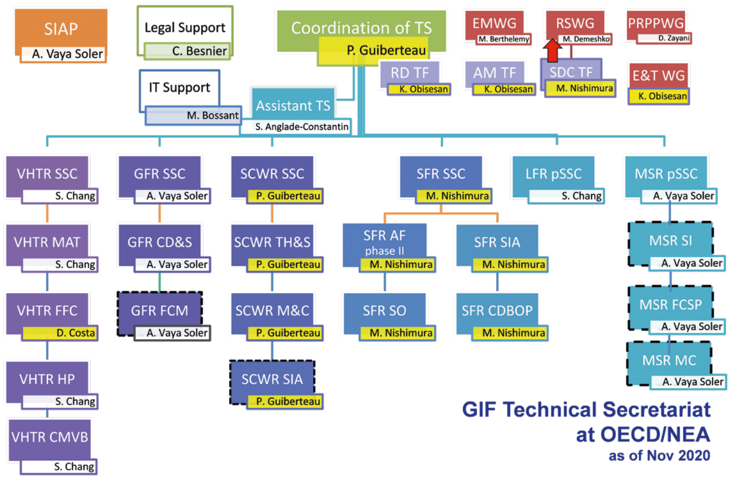


Figure 1-2: Structure of the GIF Technical Secretariat[1]

Figure 1-3: The GIF Policy Group virtual meeting in October 2020

within the GIF framework in relation to non-electrical application of nuclear heat (NEANH) (see the related article in the GIF December newsletter n° 6 at: https://www.gen-4.org/gif/jcms/c_176431/newsletters). A roadmap and a new organizing body will be proposed in 2021.

Despite the unprecedented challenges posed by the COVID-19 pandemic, members of GIF have managed to progress in their work, staying on course in terms of the GIF program of work and defining innovative ways to adjust to this situation. The GIF structure has adapted swiftly to the new situation and has become effective in its use of virtual meetings and other electronic means to advance its work. Semi-annual Policy Group and Experts Group (PG/EG) meetings were held at the end of May and October 2020, both in virtual mode. The 2020 events prepared for GIF 20th anniversary celebrations in Amboise (France) were unfortunately cancelled. More than ever, we are all looking forward to a return to normal, being able to meet face to face and have technical exchanges in person.

To improve GIF communication, GIF leadership has been working with a consultant and the Nuclear Energy Agency (NEA) Information Technology (IT) team to refresh and relaunch the GIF brand, with a new logo (see the cover page), a new website and new collateral materials.

GIF communication objectives are:

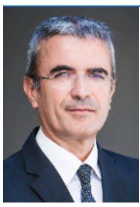
- Refresh the "look and feel" of how GIF is presented to the world so as to reinforce the GIF position as a leading, collaborative organization at the international level, with technical expertise focused entirely on 4th generation nuclear energy systems.
- Combine and encourage engagement in GIF to promote a common understanding of the policy, economic and scientific issues related to 4th generation nuclear energy systems.
- Celebrate GIF's pioneering spirit, growing expertise and excellence in the area of 4th generation nuclear energy systems.
- Remind members and their key stakeholders of the benefits that can be gained from being part of GIF.
- Help raise the profile of GIF in scientific and education circles.
- Capitalize on GIF's long history to further advocate a common understanding of the above issues among partner members.

The first phase of the GIF website enhancement project was implemented in early January 2021, as scheduled. We will continue in 2021 to improve the contents and style of the GIF website.

The objectives were to refresh the existing logo and to reassess our values

Perspectives and objectives for GIF in 2021 are largely focused on continuing to improve the Technical Secretariat functions (see the detailed article in the GIF February 2021 Newsletter at: https://www.gen-4.org/gif/jcms/c_121997/-5-gif-newsletter-rev2) despite the difficulties resulting from the COVID-19 situation, while:

- stabilizing and reinforcing the overall organization of the GIF Technical Secretariat;
- improving links between the GIF Technical Secretariat/Board and the NEA;
- developing cross-cutting projects inside GIF and with the NEA;
- improving GIF links with industry and with non-profit international organizations (e.g. the World Nuclear Association, the International Atomic Energy Agency).
- continuing to develop GIF communication (i.e. technical, external and internal).



Philippe Guiberteau
Head of the GIF Technical Secretary

1. See the list of abbreviations and acronyms in Appendix 2 of the present report.

2 Highlights from the year

The *GIF 2020 Annual Report* is divided along the lines of two main periods:

- First semester: when objectives were completely reshuffled so that the GIF organization could adapt to the global COVID-19 crisis and the consequences on working methods;
- Second semester: when the working process was stabilized, and a great deal of the GIF Board's time was spent on reassessing in depth some of the GIF foundations: our communications approach and 2021 perspectives.

In 2019, the governance of the Generation IV International Forum was completely renewed. In 2020, this new governance entered into force. The vice-chairs began to fulfil their respective missions on regulatory issues (Alice Caponiti, United States), market opportunities and challenges (Sylvestre Pivet, France), and enhancement of R&D collaboration (Jong Hyuk Baek, Korea). They are now strongly connected with the corresponding GIF task forces or working groups, and can follow or influence their actions.

On regulatory issues, the mission of the Vice-Chair is to co-ordinate GIF efforts with various regulatory bodies so as to achieve harmonized regulatory requirements. GIF engagement was confirmed through the Reactor Safety Working Group (RSWG), improving interfaces towards the Nuclear Energy Agency (NEA) Working Group on the Safety of Advanced Reactors (WGSAR) and the International Atomic Energy Agency (IAEA) Safety Sections (see the corresponding chapter for the RSWG).

On market opportunities and challenges, the Vice-Chair overseeing the Economic Modelling Working Group (EMWG) chose two priorities at the 42nd meeting, with new actions planned in this regard: advanced nuclear technology cost reduction and advanced nuclear technology private financing. Cross-cutting activities are also planned, in particular on the design-to-cost process, with a presentation of case studies, such as the French ASTRID project (see the corresponding chapter for the EMWG).

The mission focus of the Vice-Chair for the enhancement of R&D collaboration is to strengthen such collaboration within GIF, sharing innovative technology ideas and advancements among GIF members, and promoting collaboration with outside organizations. The Vice-Chair proposed the drafting of a white paper that would outline the activities of R&D collaboration in GIF and action items to enhance this collaboration. The objectives of this white paper would be to: 1) better understand R&D collaboration activities among GIF members; 2) find practical ways of co-operating within GIF members and other organizations; and 3) conduct action items that can promote R&D collaboration in GIF. This white paper initiative was launched in the summer of 2020 through a collective drafting effort. The final draft of the white paper will be produced in 2021.

On 18-20 February 2020, GIF organized a three-day workshop at the NEA entitled "GIF International Workshop with Nuclear Industry, including SMR vendors and supply chain SMEs". The first half of the workshop was dedicated to advanced manufacturing, and was organized by the Advanced Manufacturing and Materials Engineering (AMME) Task Force leader, Lyndon

Figure 2-1: Co-working sessions on advanced manufacturing

Figure 2-2: Panel discussion on R&D infrastructure needs with the private sector

Figure 2-3: First page of the Newsletter No. 6

Edwards. The second half was dedicated to R&D infrastructure needs and opportunities, and was organized by the R&D Infrastructure Task Force (RDTF) leader, Roger Garbil. This successful event gathered over 120 specialists from around the world, who exchanged on recent progress, potential solutions and practical ways of increasing interactions, particularly with the private sector (i.e. SMR vendors).

The conclusions of this workshop are presented in the corresponding AMME TF and RDTF chapters. It should be underlined that the workshop was video recorded and seen by our Chinese colleagues via streaming. This initiative allows the seminar to be seen by a larger audience, even after the event has already taken place.

Highlights from the Experts Group and the cross-cutting working groups and task forces

The Experts Group advises the Policy Group on priorities and methodology thanks to the work being carried out by specific task forces and working groups. Some of the achievements of 2020 include:

- The GIF 2018 Symposium Proceedings were published on the GIF website.
- A chapter on GIF flexibility was produced for the Nuclear Innovation: Clean Energy Future (NICE Future) initiative. This chapter provides the GIF position on flexibility, for which interest is steadily increasing.
- The RDTF final report was produced internally. An open version of this report will be delivered in early 2021.
- The process of writing a white paper on proliferation resistance and physical protection (PRPP) for all of the six systems.
- The Education and Training Working Group (ETWG) webinar series was regularly scheduled and produced on a monthly basis. Topics for these webinars are fully booked until April 2021. All are now available on the YouTube platform.

The second part of the year 2020 was dedicated to enhancing GIF communications with a new logo that expresses GIF values – expertise, collaboration and excellence – new, visual branding and a completely refreshed GIF website. The regular production of the GIF Newsletter appears to be an efficient tool for maintaining a connection between GIF members and other international organizations (i.e. the NEA, IAEA).

During the year 2021, and in accordance with the Programme Plan proposed by the Technical Director and discussed in the Experts Group and Policy Group, GIF will pursue its efforts in the following areas:

- The GIF initiative towards the private sector will go beyond the February 2020 workshop, with a dedicated, international workshop under preparation for 2022, again partnering with the private sector.
- The non-electrical application of nuclear heat (NEANH) will become an important cross-cutting subject that will be further investigated in 2021.
- The promotion of education and training towards the younger generation of scientists will continue with a "Pitch Your Generation IV Research Competition" being prepared for February 2021.

Finally, in 2020 GIF had been preparing for a significant event to celebrate its 20th anniversary. The event was to take place at a highly symbolic place, the Clos Lucé Castle, or the home of Leonardo Da Vinci in Amboise, France. Unfortunately, the COVID-19 crisis postponed these initiatives until better times.

Nobuchika Kawasaki
GIF Policy Director

Gilles Rodriguez
GIF Technical Director

3 Country reports

Australia

Australia remains an active and enthusiastic member of the Generation IV International Forum (GIF). It continues to increase its engagement in the activities of the forum, via the full signing of the amendment to the project agreement, when Australia's Nuclear Science and Technology Organisation (ANSTO) became a full member of the GIF Very-High Temperature Reactor (VHTR) Materials Project. In a similar vein, Australia continues to support the move of the molten salt reactor (MSR) provisional System Steering Committee (pSSC) to a system arrangement through the development of its Materials and Components Project and the Advanced Manufacturing and Materials Engineering Task Force (AMME TF). Following the successful February workshop at the Nuclear Energy Agency (NEA), the Policy Group supports changes to the time frame and scope of this task force.

There has been significant progress in the establishment of Australia's National Radioactive Waste Management Facility. In September, a Federal Parliamentary Committee concluded its inquiry into proposed legislative amendments, which are an essential precursor to the establishment of such a facility, and recommended that the amended legislation pass without changes. The Federal Government Parliamentary Inquiry had three main recommendations:

- that the Australian government consider the prospect of nuclear energy technology as part of its future energy mix;
- that the Australian government undertake a body of work to progress the understanding of nuclear energy technology;
- that the Australian government allow partial and conditional consideration of nuclear energy technology by maintaining its moratorium on nuclear energy in relation to Generation I, Generation II and Generation III nuclear technology; but by lifting its moratorium on nuclear energy in relation to Generation III+ and Generation IV nuclear technology, subject to the results of a technology assessment and a commitment to community consent as a condition of approval.

An official government response to the recommendations of this parliamentary inquiry is expected to be delivered in the near future.

The New South Wales government's inquiry into the repeal of a state-wide ban on uranium mining and nuclear power was released in September. That government response supported the repeal of the uranium mining prohibition but stopped short of endorsing a move towards nuclear power – although it did also express a continuing interest in overseas developments in nuclear technologies, such as those advanced through this forum.

The Victorian government's Legislative Council's inquiry to examine the merits in lifting the state's ban on nuclear power is currently ongoing.

In 2020, the Australian government published the "Technology Investment Roadmap Discussion Paper", which states that new nuclear technologies have potential but require R&D and identified deployment pathways. Its more recently published "First Low Emissions Technology Statement 2020" has identified small modular reactors as a prospective low-emission technology that could play an important role over the long term.

Canada

COVID-19 pandemic response: Canada's nuclear energy sector continues to play an important role in responding to the COVID-19 pandemic. A combination of national and sub-national government measures and private sector action has ensured that Canada's nuclear power fleet has continued to operate at full capacity despite the pandemic. It is important to note that 40% of medical devices around the world are sterilized with Cobalt-60, over 50% of which is supplied by Canada. Measures have been taken to ensure strong and resilient nuclear supply chains in Canada, which include hundreds of small- and medium-sized enterprises. Moreover, Canada's nuclear sector has stepped forward to help in the pandemic response, for example by re-tooling manufacturing and making donations of critical personal protective equipment, masks and face shields. Canadian Nuclear Laboratories (CNL) is contributing its world-class facilities and expertise to Canada's response, including in an effort to develop an easy-to-produce ventilator using easily accessible parts. The CNL is also producing face shields using 3D printing technology.

Nuclear Power in Canada: A total of 19 CANDU nuclear power reactors currently operate in Canada, generating 13.7 gigawatts electric (GWe) of power in two provincial jurisdictions. Of these 19 reactors, 18 supply 60% of the Province of Ontario's electricity, and one reactor supplies 36% of the Province of New Brunswick's electricity demand. Nuclear energy displaces over 50 million metric tonnes of carbon emissions a year across

Canada, and Canadian exports of uranium displace global emissions by up to 551 million tonnes of carbon dioxide per year.

Nuclear energy remains an important contributor to Canada's electricity mix. While the government of Canada has important responsibilities with respect to nuclear energy, investment decisions on the energy supply mix and generation capacity, including the construction of new nuclear power reactors and the refurbishment of existing reactors, fall under provincial jurisdiction. In a speech given by the Honourable Seamus O'Regan, Canada's Minister of Natural Resources,[1] he delivered a clear message that nuclear power is a priority for Canada and a necessary source of clean energy to meet Canada's climate change objectives.

In October 2020, Canada's Federal Minister of Innovation, Science and Industry announced that the government of Canada would invest CAD 20 million in Canada-based company, Terrestrial Energy, to accelerate development of the company's integral MSR, representing the first such investment support for an SMR in Canada. Also in October 2020, Ontario Power Generation (OPG) announced that it is advancing engineering and design work with the SMR developers, GE-Hitachi, Terrestrial Energy and X-energy. X-energy is developing the Xe-100 SMR (80 megawatt-electric [MWe] high-temperature reactor), and has initiated a vendor design review (VDR) for the reactor with the Canadian Nuclear Safety Commission (CNSC). Terrestrial Energy's 192 MWe integral MSR has completed the first phase of the CNSC's review process. GE-Hitachi also entered the CNSC's VDR for its BWRX-300 water-cooled SMR. GE-Hitachi announced that it had signed memoranda of understanding with the Canadian companies, Aecon Nuclear, BWXT Canada, Hatch, Black & Veatch and Overland Contracting to establish a Canadian supply chain covering cooperation in construction, engineering, modularization and manufacturing of safety-related components, and to support potential BWRX-300 construction, as well as provide future services and components.

Ontario Power utility, Bruce Power, and Westinghouse Electric Company announced an agreement to pursue applications of Westinghouse's eVinci micro-reactor program within Canada. Work between Bruce and Westinghouse will focus on furthering the public policy and regulatory framework; assessing the economic, social and environmental contribution of the eVinci technology compared to alternatives such as diesel or other fossil fuels; identifying potential industrial applications; and accelerating the roadmap for Canada to host a globally recognized demonstration.

Global First Power, Ultra Safe Nuclear Corporation (USNC) and OPG announced the formation of a joint venture, the Global First Power Limited Partnership, which will build, own and operate the proposed micro modular reactor project at the Chalk River Laboratories site. The joint venture is owned equally by OPG and USNC-Power, the Canadian subsidiary of USNC.

Small modular reactor activities: Since the release of *A Call to Action: A Canadian Roadmap for Small Modular Reactors*, the federal government and other essential, enabling partners have moved forward both individually and collaboratively on priority areas, such as advancing SMR R&D and exploring business partnerships for potential deployment in the late 2020s. In early 2020, the Minister of Natural Resources, Seamus O'Regan, announced that, with partners from across the country, Canada would launch its SMR Action Plan to position Canada as a world leader in an emerging global SMR market that is expected to exceed CAD 150-300 billion per year by 2040. The SMR Action Plan will report on actions taken by governments and partners, and chart a path forward for the next wave of nuclear innovation in Canada. It brings together more than 100 organizations to outline the progress and ongoing efforts across Canada to be the world leader on this emerging clean energy technology. It will be launched by Minister O'Regan at Canada's 2nd International SMR Conference on 18 November 2020.

The CNSC continues to work towards ensuring its readiness for SMRs in Canada. As noted above, the CNSC undertakes an optional preliminary step before the licensing process called a vendor design review (VDR). The VDR is completed at a vendor's request and expense to assess their understanding of Canada's regulatory requirements and the acceptability of a proposed design. There are currently 12 SMR proposals in the VDR process – ten VDR service agreements in force between vendors and the CNSC and two under development. Vendors include ARC Nuclear, GE-Hitachi Nuclear Energy, LeadCold Nuclear, Moltex Energy, NuScale, SMR (Holtec), StarCore Nuclear, Terrestrial Energy, U-Battery Canada, Ultra Safe, Westinghouse and X-Energy.

Global First Power is seeking CNSC approval for a license to prepare a site for a micro modular reactor at Chalk River Laboratories in Renfrew County, Ontario. In June 2020, the CNSC held its first hearing on this application, which focused on the scope of the environmental assessment. After taking into consideration public comments, the CNSC has outlined the factors to be considered in the environmental assessment project.

Over the past six months, the CNL, Canada's premier nuclear science and technology organization, entered into several collaboration agreements with SMR developers under CNL's Canadian Nuclear Research Initiative. Earlier in 2020, CNL and Moltex Energy partnered on SMR fuel research to explore innovative fuel processing and development. Then,

1. For a copy of the speech, see www.nrcan.gc.ca/keynote-address-minister-seamus-oregan/22882.

in September 2020, the CNL announced that it would work with Kairos Power and Terrestrial Energy on two distinct SMR research projects to enhance tritium safety, storage and management, and to study molten salt fuel behavior, respectively.

Internationally, in March 2020, Natural Resources Canada (NRCan) led a Canadian delegation to London, United Kingdom for a week-long Canada-UK Nuclear Energy Dialogue. The program was anchored around the Canada-UK Nuclear Energy Summit and the UK Department of International Trade's Civil Nuclear Showcase. The Canada-UK Nuclear Co-operation Action Plan, launched in November 2019, was signed in 2020. It focuses on enhancing collaboration on SMRs, with identified key areas of partnership including waste minimization, SMR fuel supply chain, regulatory collaboration, advanced manufacturing and financing. In addition, a memorandum of understanding between the Canadian Nuclear Association and the UK Nuclear Industry Association was signed, focusing on nuclear collaboration.

Other nuclear advancements: In May 2020, Alternative Radioisotope Technologies for Medical Science (ARTMS) Inc., a leader in isotope production technology, announced USD 19 million financing from a consortium led by Deerfield Management Company and Quark Ventures. This funding builds on a USD 3 million seed investment made in 2017 by Quark Ventures and GF Securities. ARTMS is a spin-off of TRIUMF, which is a Canadian particle accelerator centre that received government funding to develop cyclotron methods so as to produce molybdenum-99.

With the current interest in SMRs in Canada, NRCan has conveyed its interest in signing the MSR System Arrangement and is considering re-joining the VHTR system. For better alignment of project responsibilities, NRCan reassigned the supercritical-water-cooled reactor (SCWR) project arrangements to performing organizations, namely Atomic Energy of Canada and Canadian Nuclear Laboratories.

People's Republic of China

Nuclear energy policy: On 10 April 2020, the "Energy Law of the People's Republic of China (Draft)" was announced by the National Energy Administration for public comment. In terms of nuclear power, it is clearly stated in the Energy Law Draft that the country adheres to the following principles: safe and efficient development of nuclear power with safety as the priority, reservation of potential nuclear power sites for future new build, promoting R&D and the innovation of advanced nuclear power technology and equipment, as well as proven nuclear power technology to achieve safer and more economic performance, advancing nuclear power technology and industrial development, and accelerating the training of nuclear power professionals.

In order to standardize the disclosure of nuclear safety information and protect the rights of citizens, legal persons and other organizations to know participate express and supervise, in accordance with the relevant provisions stipulated by the Nuclear Safety Law of China the Ministry of Ecology and Environment has formulated "The Measures for Disclosure of Nuclear Safety Information", which is effective from 1 October 2020.

Nuclear energy development: As of the end of September 2020, there were 48 nuclear power units in operation with a total installed capacity of 49.87 GW, 13 nuclear power units under construction with a total installed capacity of 14.93 GW. In September 2020, Hainan Changjiang phase II and Zhejiang San'ao phase I were approved by the State Council.

Fuqing unit 5, the first reactor of the HPR1000 technology in the world, started to load fuel on 4 September 2020 and reached criticality on 21 October.

China's first commercial nuclear heating project entered into service at the Haiyang nuclear power plant in Shandong province on 15 November 2019. The system will initially heat 700 000 square metres of housing during the winter, including the dormitory for the staff of the nuclear power plant and some residents in Haiyang city. The Haiyang Nuclear Energy Heating Project is expected to eventually provide heating to the entire city of Haiyang by 2021. In addition, the Haiyang nuclear power plant had reached a strategic cooperation framework agreement with the Yantai city government regarding desalination and clean energy use of nuclear power.

Radiation technology has played an important role in the sterilization of personal protective equipment, which has been in high demand during the COVID-19 pandemic. The irradiation sterilization technology was used to replace the traditional ethylene oxide sterilization method, and efficiently shortened the sterilization time of individual medical protective clothing from seven to ten days to about one day. In addition, nuclear technology will continue to play a greater role in the treatment of medical waste in the later stages, making its own contribution to pandemic prevention and control.

Gen-IV nuclear energy system activities: sodium-cooled fast reactor (SFR) – the China experimental fast reactor (CEFR) restarted and reached full power operation after finishing the related operational commissioning tests; the construction and installation of the CFR-600 are proceeding as planned.

VHTR – the full commissioning of the High-temperature gas-cooled reactor power module (HTR-PM) demonstration project started on 25 July 2020, with the cold functional test starting on 6 October and successfully ending on 19 October. It will be connected to the grid in 2021 and reach full power operation in 2022. The R&D in the VHTR fuel and fuel cycle (FFC) and MATerial Project Management Board (MAT PMB) is going as planned, joining the Hydrogen Production-Project Management Board (HP-PMB) is still in progress.

SCWR – the pre-conceptual design of the CSR1000 and the small SCWR, called the CSR-150, is ongoing. In terms of cooperation on the SCWR, some new international benchmark exercises have been discussed among the members on thermal-hydraulic behavior in complex structures. Two research projects funded by the Ministry of Science and Technology (MOST) have been started by Chinese universities and institutes the 2020 year. China joined the Joint European Canadian Chinese Development of Small Modular Reactor Technology (ECC-SMART) project led by Europe, and attended the kick-off teleconference in September 2020.

Lead-cooled fast reactor (LFR) – on 6 October 2020, The Institute of Nuclear Energy Safety Technology (INEST) attended the 27th LFR pSSC teleconference to discuss and promote the finalization of the LFR proliferation resistance and physical protection (PR&PP) system safety assessment (SSA) and other technical documents. Two small LFR projects have been funded by MOST, which aim to explore innovative, small LFR concepts to provide a flexible energy supply.

Euratom

Policy: On 1 December 2019, Ms Ursula von der Leyenn, the new President of the European Commission (EC), took office with a new program focused on six main priorities: 1) a European Green Deal; 2) an economy that works for people; 3) a Europe fit for the digital age; 4) a protection of the European way of life; 5) a stronger Europe in the world; and 6) a new push for European democracy.

On 11 of December 2019, The European Commission issued a communication that sets out the European Green Deal for the European Union (EU) and its citizens, towards a European climate-neutral economy by 2050, aimed at mobilizing at least EUR 1 trillion of public/private investment over the course of ten years to achieve net zero greenhouse gas emissions for EU countries as a whole. Several initiatives have been launched by the European Commission in the frame of the implementation of the EU Green Deal towards a European climate-neutral economy by 2050. The most important initiative is the European Commission's proposal to cut greenhouse gas emissions by at least 55% below 1990 levels by 2030. This is a substantial increase compared to the existing target, upwards from the previous target by at least 40%. It is in line with the Paris Agreement objective to keep the global temperature increase to well below 2°C, and pursue efforts to keep it to 1.5°C.

In the context of implementing the green deal, the EU Council and Parliament also adopted a regulation (EU-2020/852) in June that establishes the general framework for determining whether an economic activity qualifies as environmentally sustainable for the purposes of establishing the degree to which an investment may be environmentally sustainable (the so-called "Taxonomy"). The regulation empowers the commission to establish, for each of the environmental objectives laid down in that regulation, the technical screening criteria for determining the conditions under which specific economic activities qualify as contributing substantially to that objective and ensuring that those economic activities cause no significant harm to any of the other environmental objectives.

The group that supports the EC in the technical screening for economic activities in relation to their environmental sustainability, called the Technical Expert Group on Sustainable Finance (TEG), has recommended in its report of 10 March 2020 that in-depth technical work be undertaken on nuclear life-cycle technologies and on existing and potential environmental impacts (the so-called "Do No Significantly Harm" [DNSH] criteria). The TEG has recommended that the EC attribute the technical work to an independent group with real technical expertise in the field. The work has thus been attributed to the EC Joint Research Centre (JRC). The JRC will produce a report that will be reviewed by two independent expert groups.

Budget: The EU has been very busy with the adoption of the Multiannual Financial Framework 2021-2027 in the context of COVID-19. EU leaders agreed on an overall budget of EUR 1 824.3 billion, EUR 1 074.3 billion for the multiannual financial framework and an additional EUR 750 billion (known as the Next Generation EU) to help the EU recover from the COVID-19 pandemic.

They agreed on a proposed budget in current prices for "The Horizon Europe" framework program for research and innovation (2021-2027), with a budget of about EUR 84 billion, complemented by EUR 1.98 billion for Euratom research and training and about EUR 5.6 billion for the ITER project. The text of the Euratom research and training program, as well as the ITER text, are being finalized in the EU Council with member states.

The objectives of the Framework program for Euratom research and training remain the same as those for the precedent framework program: to improve and support nuclear safety, security, safeguards, radiological protection, safe spent fuel and radioactive waste management and decommissioning, including the safe and secure use of nuclear power; maintain and further develop expertise and competence in the nuclear field, develop fusion energy, and support the policy of the EU and its member states to continuously improve nuclear safety, safeguards and security. The EU had added a new objective on the safe and secure use of non-power applications of ionizing radiation. The program will also strengthen activities on training, education and open access to nuclear facilities, including the JRC nuclear facilities.

Research activities: Under the current Euratom Research and Training Programme, the selected projects for 2019-2020 amounted to a budget of EUR 140 million. All winning projects have started.

Five projects on advanced system proposals will be co-funded on topics such as: fuel cycle Pu management, safety of Gas Fast Reactors, partitioning and transmutation, safety of SCWR SMR, and the high-performance computing safety evaluation of SMRs. Research Infrastructures Material Testing Reactors include two actions on the Jules Horowitz Reactor (JHR) that will allow for innovative fuel and material testing:

- access rights for Euratom researchers (EUR 6 million, leading today to about EUR 40 million from Euratom in total, about 6% JHR irradiation time);
- the Jules Horowitz operation plan 2040, and optimized use of research reactors (EUR 2.2 million, and EUR 2.6 million in total) to plan Euratom specific irradiations.

The Supplementary Programme for the high-flux reactor (HFR), supported by about EUR 30 million from the governments of the Netherlands and France, has been adopted. The Sustainable Nuclear Energy Technology Platform (SNETP), composed of the Nuclear Generation II & III Alliance (NUGENIA) pillars, the European Sustainable Nuclear Industrial Initiative (ESNII) and the Nuclear Cogeneration Industrial Initiative (NC2I), have evolved to form an International non-profit legal association last October 2019. It is now in the process of updating its Strategic Research and Innovation Agenda.

The JRC has consolidated its activity in the domain of Gen-IV through three major projects. The Safety of Advanced Nuclear Systems and Innovative Fuel cycles (SEAT-GEN-IV), System Analysis of Emerging Technologies (SAITEC) and Waste from Innovative fuel (WAIF). The topics covered are: reactor safety of Gen-IV reactor designs, including modular reactors (severe accident modelling), materials R&D program, safety of fuel: SFR, LFR, VHTR, MSR systems, conditioning matrices for waste from innovative fuels, and safeguards. Activities in support of the GIF-Proliferation Resistance and Physical Protection Working Group (PRPPWG) are carried out in the Methods, data analysis and knowledge management for Nuclear Non Proliferation, Safeguards & Security (MEDAKNOW) project. The JRC has signed the GIF Project Arrangement on Fuel and Core Materials for the International Research and Development of the Gas-Cooled Fast Reactor Nuclear Energy System and is now finalizing the process for the new MSR system arrangement.

France

COVID-19 crisis and recovery plan: French nuclear energy generation has proven resilient through the COVID-19 crisis. Projected production was reduced by nearly 20% in the April-June period (from 375 to 390 TWh, down to 300 TWh), due to revised maintenance-shutdown planning. However, EDF has raised its forecast upwards for 2020 to 325-335 TWh.

In September, the French government presented a recovery plan called "France relance", intended to support the national economy, which has been deeply impacted by the health crisis. This plan identifies three major domains (i.e. ecology, competitiveness and cohesion). Alongside a major program on hydrogen, it includes a significant state budget dedicated to the nuclear industry, to maintain essential nuclear skills and to enhance industrial modernization. More precisely, this budget addresses three challenges: 1) training and skills development; 2) supply chain robustness; and 3) R&D and SMRs (conceptual design of the NUWARD™ product).

Nuclear fleet: Fessenheim power plant unit 1 was definitively shut down in February, which should be followed by unit 2 in June, resulting of the industry's strategic contract, signed by the government and the nuclear industry in January 2019.

CEA Energy Division: A reorganization took place at the French Alternative Energies and Atomic Energy Commission (CEA) in February, that saw in particular the evolution of the Nuclear Energy Division into a new Energy Division, with the ambition of better addressing the comprehensive energy system.

Status of the French Fast Reactor program: The CEA is implementing its new fast neutron reactor (FNR)-related activities. It is focused on a strong R&D program dedicated to further progress on fast-reactor technology and the associated fuel cycle. The priority is still being given to the sodium-cooled fast-reactor technology, which is considered the most mature. The program also includes other FNR-concept assessments.

The overall objective is to increase the maturity of the SFR technology. The five-year work plan focuses on high-stake topics: 1) basic physics, modelling and simulation, especially the physics of severe accidents, sodium chemistry and sodium risks assessments; 2) increase of fuel performance, structural materials in service behavior and codification, qualification and the 60-year lifetime justification; and 3) technological developments of some components, especially monitoring and inspection techniques.

Another pillar is sketch studies, focusing on power threshold effects from a safety and economics point of view (assessment of SFR breakthrough designs with intrinsic safety). Other concepts of FRs (e.g. fast spectrum molten salt reactors) will be studied in order to identify key feasibility issues, as well as their specific features and potential performance.

In this newly oriented endeavour, partnerships and collaboration remain essential for further skill and capability development. The CEA will play a role in the GIF and push bilateral collaboration among leaders of the SFR development. The program benefits from the large feedback of the Advanced Sodium Technological Reactor for Industrial Demonstration (ASTRID) program, which is significant from different perspectives. New design solutions for the prevention and mitigation of severe accidents have been proposed. Innovations have been introduced in the design, for example concerning the core, reactor components, power conversion system, fabrication processes and in many other areas. Digital modelling has been improved in several fields, based on the phenomenological approach. Methodologies for verification, validation and uncertainty quantification (VVUQ) and safety demonstration have been formalized. New experimental platforms have been put into operation to run model validation and "proof of concept" tests. A large collaboration with industrial partners has been implemented, as well as methodologies for simultaneous engineering, interface data consistency and product life-cycle management.

SMRs: The French NUWARDTM initiative is based on the pressurized water reactor (PWR) technology. It is being designed to meet the growing needs of a low-carbon, safe and competitive electricity market worldwide, in the 300-400 MWe power range. NUWARDTM is a concept that combines proven solutions, innovative licensing and manufacturing, innovative solutions to gain competitiveness: simplicity and compactness of an integrated design, flexibility in the construction and operation phase, an innovative approach to compliant safety to the best world standards. The ambition is to offer the world market a competitive product by 2030. Being open to international cooperation in the field of SMRs can thus be a key milestone on the path towards regulatory harmonization, and the standardization and optimization of design.

Japan

Current energy and nuclear policy: Japan has started reviewing its Basic Energy Plan. On 13 October, an advisory committee for natural resources and energy to the Ministry of Economy, Trade and Industry (METI) kicked off discussions on how Japan can reach this goal. As a result, first, they will identify issues that need to be solved in order to achieve the concept of the Basic Energy Plan. It is being called 3E+S, which represents energy security, economic efficiency, the environment and safety. Second, they will identify issues that need to be solved to achieve a zero-carbon society at the earliest possible time in the latter half of this century. Finally, they will verify the progress of the energy mix targeted for the year 2030 and how efforts towards each energy source have progressed. Then, further efforts or approaches will be discussed.

In the meantime, Japan has continued to reform its electricity market: fossil fuel and non-fossil fuel electricity were treated uniformly in the past, but Japan opened a dedicated non-fossil fuel electricity market in May 2018 — the values of non-fossil fuel electricity have thus emerged. This suggests a growing role for nuclear energy in energy conversion and decarbonization.

Development of advanced reactors: Regarding R&D of advanced reactors, a governmental initiative called Nuclear Energy Innovation Promotion is being carried out to promote nuclear technology innovation in the private sector. A new initiative entitled Nuclear Energy x Innovation Promotion (NEXIP) aims to boost this development. It adopted 28 inventive proposals from the private sector in September 2019. For advanced non-light water reactors, the government is looking at proposals for high-temperature gas-cooled reactors (HTGRs), SFRs, and MSRs.

The Energy Plan has demonstrated that Japan is further advancing the technologies of HTGRs and fast reactors in cooperation with international partners. To achieve this, the Japan Atomic Energy Agency (JAEA) plays an active role in developing safer fast reactors and HTGRs, and in establishing global standards in the safety and the design of high-temperature structures.

Current status of Fukushima Daiichi Nuclear Power Station: All of the reactors at the power station have been in a cold shutdown state since the accident in 2011. Based on the "Mid-and-Long-Term Roadmap" for the station, which was updated in December 2019, Tokyo Electric Power Company Holdings (TEPCO) is working towards decommissioning. TEPCO is removing fuel pins from the spent-fuel pool in unit 3, and expects to complete this work in March 2021. TEPCO will provide examinations in relation to fuel debris removal from unit 2, which would start in 2021, and the handling of water from the Multi-Nuclide Removal Facility (ALPS), produced through purifying contaminated water.

Safety review of nuclear power stations and nuclear fuel cycle facilities by the Nuclear Regulation Authority (NRA), and regulatory inspections under the state of emergency over the COVID-19 pandemic: To date, the NRA has given its green light to 16 units at 9 sites, among 27 units at 16 sites, that have applied for the new conformity assessment. As of October 2020, three units are in operation. On 13 May 2020, the NRA approved the review report that certifies the compatibility of the Rokkasho reprocessing plant of Japan Nuclear Fuel Limited (JNFL) with the conformity assessment. The JNFL will address public comments, reinforce the construction, and explain the plant to the community before they begin operation.

Current situation of JAEA facilities: the JAEA is preparing to restart the high-temperature engineering test reactor (HTTR) and the experimental fast reactor, Joyo. The NRA completed the assessment of HTTR conformance to Japan's new regulatory requirements, and issued an amendment to the reactor installation permit in June. The JAEA will reinforce the HTTR against internal and external events, aiming at restarting it as early as possible. Once restarted, the JAEA will carry out a loss of forced coolant (LOFC) experiment in international cooperation under the framework of the NEA.

The JAEA submitted the amendment to the reactor installation permit for Joyo to the NRA in October 2018. The NRA is currently reviewing the amendment in terms of beyond-design-basis analyses (BDBA) and internal fires, as well as other factors.

Regarding Monju, which will be undergoing decommissioning, a new experimental research reactor with an output capacity of about 10 000 kilowatts (kW) will be constructed at the Monju site. Designing of the new reactor will start in detail during Japanese fiscal year 2022.

The JAEA will restart the Japan research reactor number 3 (JRR-3) in 2021. The NRA issued an amendment to the reactor installation permit in 2018. Since then, the JAEA has been implementing work for seismic resistance and reinforcement of the reactor. In August 2020, the JAEA announced that the operation schedule for three years from the restart. The JAEA is striving to ensure the restart.

The JAEA is also working on the development of an evaluation platform called the Advanced Reactor Knowledge- and AI-based Design Integration Approach (ARKADIA), which will cover the entire plant lifecycle, and on innovative technology to further enhance safety and economic efficiency. In addition, it is supporting the private sector to develop innovative technologies. Using ARKADIA platform, in March 2020, the JAEA coupled the analysis codes of the core, heat, and (deformed) structure of an SFR and successfully developed an analysis method for transient characteristics in an SFR core.

Korea

Nuclear power in Korea: Twenty-four nuclear power plants (20 PWRs and 4 CANDU reactors) are in operation as of July 2020, providing 13 721 GWh of electricity, which corresponds to 29.3% of total electricity production in Korea, a 2% increase from last February. The installed nuclear capacity of 24 NPPs accounts for 18.2% (23 250 MWe) of total capacity. Four PWRs (i.e. Shin-Kori units 5 and 6; and Shin-Hanul units 1 and 2) are under construction. Construction is expected to be completed for Shin-Hanul units 1 and 2 by August 2021.

Nuclear energy policy and R&D in Korea: The goal of the national energy policy in Kora has changed to increasing the portion of power generation from renewable energy sources to 20% by 2030, as well as to gradually reducing the share of nuclear and coal-based electricity production. On the other hand, the government will support both an export promotion of nuclear power plants, and R&D activities relevant to enhancing nuclear safety and nuclear dismantling and disposal technology.

The national policy for spent nuclear fuel (SNF) management in Korea remains undecided. An SNF management policy re-examination commission was launched in May 2019 to review the previous national policy. The commission regularly opens online meetings and has begun to collect public opinions about NPPs. Construction of the MACSTOR SNF dry-storage facility was recently decided in August 2020.

Sodium-cooled fast reactor (SFR): As for SFR development, SFR R&D activities until the end of 2020 focused mainly on obtaining a technical database to support a back-end fuel cycle design option. At present, there is no engineering design development of SFRs. For the purpose of certifying key SFR technologies, ten topical reports (TRs) dealing with key design technologies and safety-related issues have been submitted to the Nuclear Safety and Security Commission (NSSC) and are now under review. For license support and the integral safety validation of the prototype Generation IV sodium-cooled fast reactor (PGSFR), an integral sodium thermal-hydraulic test facility, STELLA-2, has been constructed and the demonstration of the integral effects will be completed. SFR development is focusing on the new nuclear market. One of the main targets is the SMR market. A new SFR project will deal with key technology development on advanced types of SMRs.

Very-high-temperature gas-cooled reactor (VHTR): The Korean government announced its national plan for a hydrogen economy, which centred on the two axes of hydrogen-powered vehicles and hydrogen fuel cells in early 2019. Hydrogen demand in 2040 is expected to reach 5.26 million tons per year. Nuclear hydrogen production using VHTRs was evaluated as one of cleanest hydrogen production technologies.

Key technology developments for VHTR performance improvement were completed in 2019. These developments include design and analysis codes, thermo-fluid experiments, tri-structural isotropic (TRISO) fuel, high-temperature materials database, and high-temperature heat applications. A subsequent project, called “Very-High-Temperature System Key Technology Development”, was launched in April 2020 to develop the performance evaluation technologies of design and analysis codes, the performance verification technologies of high-temperature materials, and the coupled analysis technologies between very-high-temperature systems and a high-temperature steam electrolysis (HTSE) hydrogen production system.

Russian Federation

This year of 2020 is a jubilee year for the Russian nuclear sector. The 20th day of August in 1945 became the starting point of the history of the national nuclear industry, which for more than 75 years has provided global nuclear arms balance, fed cities with energy, and fostered the development of science and technology far beyond traditional “nuclear” objectives.

By 1 September 2020, nuclear electricity generated by Russian nuclear power plants (NPPs) made up 136.5 billion kWh, and aimed to reach 214 billion kWh by the end 2020. This will bring growth up by 2.4% compared to 2019, in spite of the global COVID-19 pandemic. The load factor by the beginning of September was 77.9%, (the 2019 figure was 78.4%).

In total, 38 power units are currently in industrial operation at 11 NPPs in Russia. The total installed capacity of all Russian NPPs is 30.3 GW. Total capacity of nuclear power units in operation is 26.3 GW. The share of nuclear energy in total energy generation in Russia is 19%, with European and north-western parts of Russia having 30% and 37% of nuclear generation, respectively.

In August 2020, a physical start-up of a new nuclear reactor took place at the Leningradskaya NPP-2. Its commissioning is planned for the beginning of 2021. This will be the last nuclear power unit of the first series of VVER-1200 power units of the Russian “AES-2006” design developed in the 21st century. NPPs with VVER-1200 power units are today the main product exported by the State Atomic Energy Corporation, Rosatom.

Test operations of the floating nuclear thermal and electric power plant (FNPP) “Akademik Lomonosov” are underway, and are showing stable and accident-free operation at design parameters. More than 80 million kWh of electricity have been generated by 1 September 2020. The FNPP is recognized as a base element of the Northern Sea route. Rosatom also commenced development of the second generation of FNPP – an optimized floating nuclear power plant (OFNPP), which is planned to be smaller than its predecessor and equipped with two RITM-200M type nuclear reactors of 50 MW each.

Within the Northern Sea route development program, the series-leading ice-breaker, “Arctica”, came out of the shipyard of the Joint Stock company, “Baltiyskiy Zavod” (Saint-Petersburg), in September 2020, and will pass the acceptance “ice-breaking” tests. After passing these tests, Arctica will go to Murmansk in the context of a transfer to Rosatom’s Atomflot company. Two other ships, the “Sibir” and “Ural” icebreakers, are scheduled to be commissioned in 2021 and 2022, ensuring that tank ships are carrying hydrocarbons from Yamal, Gydan peninsulas to Asia-Pacific region markets year-round.

Russia is today a world leader in new nuclear construction abroad. Rosatom also ranks first in the number of simultaneously implemented projects for the construction of nuclear power units (3 in Russia and 36 abroad). The current Rosatom portfolio includes: Hanhikivi-1 (Finland), Akkuyu NPP (Turkey), Kudankulam (India), Paks-2 (Hungary), Ruppur NPP (Bangladesh), Cuidapu (China), Tianwan (China), El Dabaa NPP (Egypt), Belorusskaya NPP (Belorussia) and Busher (Iran). In August 2020, nuclear fuel loading started at the first power unit of Belorusskaya NPP in Ostrovets (Grodno region). Commissioning of the first unit is scheduled for the beginning of 2021, and the second unit in 2022. With both power units in operation, Belorussia will have an additional 2 400 MW of generating capacity.

Russian NPP designs are based on Gen-III+ reactors, equipped with both passive and active safety systems, and in full conformity with the modern international requirements and recommendations of the International Atomic Energy Agency (IAEA). Russia is further improving the VVER technology to enable transition from the open to closed nuclear fuel cycle, and to ensure efficient operation of two-component nuclear power. The VVER-1200 is a flagship nuclear reactor and the main product of Rosatom’s complex offer. Evolving from the VVER-1000 design units that were recently built in China, India, Iran VVER-1200 design units have improved characteristics in all design parameters.

Perspectives of nuclear technologies: Russia is a recognized leader in the field of sodium-cooled fast reactors (BN or SFR). At present, two power units of the Beloyarsk NPP, with the BN-600 and BN-800 reactors, as well as the BOR-60 research reactor in NIIAR, Dimitrovgrad, are in operation. The total BN operation experience accumulated in Russia and the former Union of Soviet Socialist Republics (USSR), as of September 2020, exceeds 160 reactor-years. The lifetime of the power unit with the BN-600 reactor has reached 40 years.

To solve the task of closing the nuclear fuel cycle, along with the transmutation of long-living isotopes, the hybrid core of the BN-800 was designed to have both uranium and uranium-plutonium mixed oxide (MOX) fuel assemblies. The first serial batch of 18 MOX fuel assemblies was loaded into the BN-800 in December 2019, and during 2020 demonstrated an incident-free operation without degradation of economic performance. In July 2020, at the Mining-Chemical Combine (Krasnoyarsk region), a full reload batch of 169 MOX fuel assemblies for the BN-800 was manufactured and tested. Transition of the BN-800 at Beloyarsk NPP to 100% MOX fuel operation is planned for 2022.

In the framework of the PRORYV Project (proryv means breakthrough in Russian), construction of the lead-cooled fast neutron BREST-OD-300 reactor commenced. Irradiation tests continued for the innovative mixed nitride uranium-plutonium (MNIT) fuel; its manufacturing is planned to start in 2022 at the BREST-OD-300 site in Seversk. This type of fuel is aimed to be used for both the BREST-OD-300 and BN-1200 reactors. All of the MNIT fuel elements irradiated in the BN-600, under close to design conditions, have reached 6% burn-up levels, which is a target level for the first loading of the BREST-OD-300. Safe fuel operation up to 8.5% burn-up of heavy atoms has been demonstrated, and testing continued to reach a 9.2% burn-up level. The fuel-element cladding damage dose reached a record value of 110 displacements per atom (dpa).

At the NIIAR site in Dimitrovgrad, the MBIR fast research reactor with sodium coolant is being constructed, which is intended to replace the BOR-60 reactor whose operation lifetime has already reached 50 years. Russia established the International Research Center on the basis of the MBIR reactor (IRC MBIR), and has approved the new construction schedule for full development. Construction work of the research reactor thus started at the construction site in accordance with this new schedule. Key research directions enabled by the IRC MBIR are materials science (e.g. new fuel, structural materials, coolants, data verification), safety (e.g. justification of new safety systems, transients and beyond-design conditions research), physics (e.g. closed cycle studies, minor actinides and long-lived fission product treatment, reprocessing, computer code verification), resource tests (e.g. fuel, control system elements, core elements, cooling loops monitoring and diagnosis systems).

An important addition to the IRC MBIR research capacity is the inclusion of the poly-functional radiochemical complex, which will be able to carry out a series of irradiation tests and post-irradiation investigations at the same site.

Under the strategic framework of the future of Russian nuclear scenarios, two-component nuclear power development, with closed fuel cycle based on fast neutron reactors and standard VVERs with thermal neutrons, are scrutinized. Options assuming the use of MSRs are also under consideration.

Transition to the closed fuel cycle and its transient phase (i.e. two-component nuclear power) puts a stop to the accumulation of SNF from thermal reactors and eliminates any increase in the growth of corresponding costs of spent-fuel management. Substituting a thermal reactor with a fast neutron reactor leads to: the elimination of approximately 1 000 tonnes of spent fuel (i.e. for the VVER reactor operating for 60 years) and expenditures for its storage before reprocessing; an increase by 15 times of the plutonium from spent-fuel reprocessing which is used back as a fuel (15% of plutonium in fast reactor spent fuel). Use of nuclear materials from spent-fuel reprocessing for the commissioning of fast reactors, as well as the closed fuel cycle, is an efficient way of coping with the problem of accumulated spent fuel from VVERs: one fast reactor is capable of using a lifetime of spent fuel from one VVER; substitution of 10 GW of thermal reactors by fast reactors almost completely resolves the problem of all accumulated spent fuel from Russian VVERs (~10 thousand tonnes), while at the same time ensuring economical results from fuel reprocessing.

Generation 4: Rosatom is considering the possibility of signing the system agreement on MSRs, as well as taking part in the project agreement for the thermo-hydraulics and safety of supercritical reactor, and the project agreement for fast reactor equipment and conversion module design of SFRs.

In 2019, Rosatom agreed to extend for the next 10 years the validity of the GIF project agreement on safety and operation of SFRs.

South Africa

As part of the implementation of the Integrated Resource Plan 2019-2030, in June 2020 South Africa issued a non-binding request for information (RFI) for 2 500 MW of nuclear energy. The request for information is part of nuclear vendor engagement aimed at obtaining information on the feasibility of the program, for example to assess the financing model, schedule and costs associated with the procurement of conventional NPPs and SMRs that are coming online for the delivery of 2 500 MW capacity of nuclear energy.

South Africa is considering the revival of its High-Temperature Power Reactor Programme based on the pebble bed modular reactor (PBMR) technology.

Eskom's implementation of the Koeberg Long-Term Operation Programme is on track as guided by requirements of the nuclear safety authority that is the National Nuclear Regulator, and supported by the IAEA expert Mission on the Safety Aspects of Long Term Operation (SALTO). In September 2020, the first of six replacement steam generators arrived at Koeberg. These steam generators are designed by the French company, Areva (now Framatome) and manufactured in China under a subcontract with Shanghai Electric Power Equipment Company. In order to enable the long-term operation of the Koeberg NPP, in June 2020 South Africa published regulations on the long-term operation of nuclear installations. These regulations have gone through public consultation and are now in preparation for Gazetting by the Minister of Mineral Resources and Energy.

South Africa is implementing a multi-purpose reactor (MPR) Project aimed at replacing the SAFARI-1 research reactor. The Project Initiation Report for the multi-purpose reactor has been approved by the Necsa Board and endorsed by the Minister of Mineral Resources and Energy for presentation in Cabinet. The Ministerial Task Team is currently working on the pre-feasibility phase of the project, which will later be followed by fully-fledged feasibility, indicating the bankability of the program.

Through the National Radioactive Waste Disposal Institute (NRWDI) and under the oversight of the Ministerial Steering Committee, South Africa is implementing a Centralized Interim Storage Facility project for SNF. The project is currently in the pre-feasibility phase. This project will be linked to Eskom's internal spent-fuel storage facility, the Transient Interim Storage Facility (TISF). In the long run, the idea is to store spent fuel off-site in an above ground storage facility, in line with government policy and decisions aligned with lessons learnt from the Fukushima Daaichi NPP accident.

The Radioactive Waste Management Fund bill is being developed to achieve the polluter-pays principle and to "ensure that the financial burden for management of radioactive waste is borne by the generator of that waste" as per Radioactive Waste Management Policy and Strategy for the Republic of South Africa, 2005. The National Nuclear Regulator Amendment bill is under development to close some gaps in the legislative framework. These bills will strengthen South Africa's nuclear regulatory framework in general.

Switzerland

Operation of the Swiss nuclear power plants and waste management: The Mühleberg boiling water reactor (BWR) was the first Swiss power reactor to be shut down definitively on 20 December 2019. The decommissioning of the plant is ongoing as planned and was approved by the regulator. All other reactors are in operation and running at nominal power. The revision of the reactors were realized on time and according to plan despite the COVID-19 crisis.

Nagra, the company in charge of realizing the final repository for nuclear waste in Switzerland has obtained all the necessary authorizations for deep drillings designed to acquire detailed information on the geology of the three possible locations for a geological waste repository. The results of these studies are meant to determine the detailed geological differences between the possible sites and act as back up of the final choice for the location of the repository.

Nuclear power related research in Switzerland: A new professor and group leader at the Reactor Physics and Thermal-Hydraulics Laboratory of the Nuclear and Safety Division at Paul Scherrer Institute (PSI) has been nominated and will support professor A. Pautz in maintaining and improving the Nuclear Engineering Master's degree in Switzerland. The nomination of two professors and laboratory leaders for the Energy System Analysis and Scientific Computing and Modelling Laboratories has been impacted by the COVID-19 pandemic and is still ongoing.

Some specific capabilities (modelling) of the Nuclear Energy and Safety Division of PSI have been used to realize specific studies on COVID-19.

GIF activities: Operations at PSI, where most GIF research activities are taking place, were maintained at all times in spite of the COVID-19 pandemic. A summary of the main activities during the period on the MSR and SFR research includes:

- The major focus of MSR research was on MSR safety, with no preference for a specific MSR concept. However, the MSFR is considered as a reference system and the molten chloride salt fast reactor (MCFR) as the potentially most promising system for fuel cycle simplicity. Research in 2020 was dedicated to thermo-dynamics modelling of the salt properties and to MCFR fuel cycle and thermal-hydraulic layout.
- SFR research also focused on safety and corresponding investigations of design improvements. PSI is contributing work to two international frameworks: 1) the IAEA Technical Working Group on Fast Reactors (Co-ordinated Research Projects on the United States Fast Flux Test Facility and Chinese Experimental Fast Reactor); and 2) Horizon-2020 European Sodium Fast Reactor Safety Measures Assessment and Research Tools (ESFR-SMART) project co-ordinated by PSI. After three years of participating in the ESFR-SMART project, discussions on a follow-up project have been initiated.

Summary of the main activities during 2020 on the materials side:

- thermal conductivity measurements on silicium carbide (SiC) irradiated materials using laser and infrared based equipment (PhD work) has been completed. The emphasis was on the micro/macro-structure analysis of SiC material, based on X-ray tomography. The existence of very long pores along the fibres has been observed.
- PSI creep data will be included in the materials handbook.
- three virtual VHTR Materials Project Management Board meetings were held virtually, where Mr Pouchon acted as a co-chair and gave an update on the Swiss situation and Gen-IV related materials research. It was decided that the lead of the Ceramics Sub-Working Group would rotate between the main actors, and that Switzerland will start with the next term.

United Kingdom

Nuclear energy: Nuclear energy continued to be one of the United Kingdom's largest low-carbon energy sources, producing around 10% of primary energy and around 40% of the United Kingdom's clean electricity. The United Kingdom 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 United Kingdom is planning a significant amount of low-carbon energy in future.

The Prime Minister recently set out further commitments to ensure that, within the decade, the United Kingdom would be at the forefront of the green industrial revolution, as we accelerate our progress towards net zero emissions by 2050. Confirming offshore wind will produce more than enough electricity to power every home in the country by 2030, based on current electricity usage, boosting the government's previous 30 GW target to 40 GW.

These commitments are the first stage outlined as part of the Prime Minister's ten-point plan for a green industrial revolution, which was set out fully in 2020. The plan includes ambitious targets and major investment into industries, innovation and infrastructure that will accelerate the UK path to net zero emissions by 2050, and also includes plans for deep decarbonization, covering domestic and process heat and hydrogen production.

The government recognizes the potential for the United Kingdom to become a world leader in developing the next generation of nuclear technologies. The Nuclear Sector Deal (NSD) signalled a significant step up in ambitions and the pace of policy initiatives towards advanced nuclear technologies.

Small and advanced reactors have the potential to deliver the cost reductions outlined in the NSD through technology and production innovations, while creating high-skilled jobs and helping the United Kingdom meet clean growth targets. To help enable the development of small reactors, the government has outlined a new framework in the NSD, designed to encourage industry to bring technically and commercially viable small reactor propositions to a vibrant UK marketplace.

COP26: The delayed COP26 United Nations (UN) Climate Change Conference will now take place between 1-12 November 2021 in Glasgow. This decision was taken by the COP Bureau of the United Nations Framework Convention on Climate Change (UNFCCC), with the United Kingdom and its Italian partners. This rescheduling will ensure that all parties can focus on the issues to be discussed at this vital conference and allow more time for the necessary preparations to take place. The COP26 President and Secretary of State for the Department of Business, Energy and Industrial Strategy, Alok Sharma, has said:

> "While we rightly focus on fighting the immediate crisis of the Coronavirus, we must not lose sight of the huge challenges of climate change. With the new dates for COP26 now agreed we are working with our international partners on an ambitious roadmap for global climate action between now and November 2021. The steps we take to rebuild our economies will have a profound impact on our societies' future sustainability, resilience and well-being and COP26 can be a moment where the world unites behind a clean resilient recovery.... Everyone will need to raise their ambitions to tackle climate change and the expertise of the Friends of COP will be important in helping boost climate action across the globe."

UK Nuclear Innovation Programme:

Advanced manufacturing and materials: A series of contracts have been placed under the Advanced Manufacturing and Material Programme since 2017 with the overall aim of introducing new manufacturing methods and reducing costs in the construction of new nuclear reactors with a focus on SMRs and advanced modular reactors (AMRs). Government is investing GBP 5 million with companies under phase 2 of this program, which was launched in 2020 and with work now underway.

Advanced Fuel Cycle Programme (AFCP): The AFCP is progressing well with both the Programme Board and Strategy and Technical Board meeting regularly. Potential new products are being developed to enable commercial opportunities to be realized in the fuel area of the program; recycle capability is being further underpinned through cutting edge experimental activities. The program has seen advancements in a number of areas from coated-particle fuels and advanced technology fuels to recycling of used nuclear fuel.

AMRs: The UK government believes that advanced nuclear technologies have the potential to support a secure, affordable decarbonized energy system, alongside other low-carbon generation sources. The AMR Feasibility and Development (F&D) project aims to fund applied R&D to progress AMR technologies towards commercial deployment. Based on a competitive selection process, three participants have been successful and been awarded around GBP 10 million each to progress to phase 2. These are: 1) Tokamak Energy Ltd (SME), Oxfordshire; 2) U-Battery, Buckinghamshire; and 3) Westinghouse Electric Company, Lancashire. In addition, nuclear regulators have developed their capability and capacity for assessing AMRs and cultivating a regulatory environment that encourages the development of a domestic AMR supply chain. The government is also investing a further GBP 5 million to nuclear regulators in an effort to help build their advanced nuclear capabilities and capacity when assessing advanced nuclear technologies (ANTs).

Energy white paper: An Energy White Paper (https://www.gov.uk/government/publications/energy-white-paper-powering-our-net-zero-future) was published in 2020, discussing, among other things, new nuclear financing and the question of siting for small reactors.

UK SMR and low-cost nuclear energy: The UK SMR Programme has the potential to support two of the UK government's top priorities: high value manufacturing and engineering has real potential to boost the UK economy, while low-carbon technologies will be crucial for a successful transition to net zero emissions by 2050.

The Rolls Royce led UKSMR Consortium are looking to commercially deploy a fleet of small modular PWRs from early 2030. Progress in phase 1 of the match funded "Low-Cost Nuclear Challenge" is progressing well, with the intention of phase 1 to increase technical and commercial certainty of the program, which will enable robust entry into the next phase of the Low-Cost Nuclear Challenge program, subject to business case approval.

The program aims to develop a SMR designed and manufactured in the United Kingdom, and which is capable of producing cost-effective electricity. An initial GBP 36 million joint public and private investment will enable the consortium to further develop their design. This is part of a greater bid into the Industrial Strategy Challenge Fund worth around GBP 500 million (a joint investment with the private sector), subject to future approvals and a final decision on public investment. The consortium believes that a UKSMR program can support up to 40 000 jobs at its peak with each SMR capable of powering 750 000 homes.

Nuclear Innovation Research Advisory Board (NIRAB) report and UK Nuclear Landscape review: NIRAB was reconvened in 2018 to provide independent expert advice to government. The NIRAB *Annual Report 2020 – Achieving Net Zero: The Role of Nuclear Energy in Decarbonisation* was published in June 2020. NIRAB believes it is time to move forward towards demonstration of both SMR and AMR systems with appropriate underpinning R&D programs to support the decarbonization of the UK economy. The 2020 UK Nuclear Landscape review was completed and issued in March 2020. This report is published on a three-yearly cycle, tracking the development of the UK's nuclear R&D capability and capacity since the publication of the 2013 Beddington Review by the House of Lords.

Progress on Hinkley Point C: The project has made significant progress in the early stages of construction. In June 2020, EDF Energy confirmed that the 49 000-tonne concrete base for the second reactor at Hinkley Point C had been completed on time. In September 2020, it confirmed that the second reactor at Hinkley Point C had passed a major milestone with the lifting of the first part of the massive steel containment liner, just nine months after the same lift for the first reactor. Construction of the 170-tonne "liner cup" was 30% quicker than the identical part on unit 1. Work at the Somerset site has been continuing during the COVID-19 lockdown with social distancing measures in place to protect workers. EDF Energy's Delivery Director for Hinkley C, Nigel Cann, said: "We've been really able to learn lessons and be much more efficient the second time round." The site should be completed by 2025.

Updates on membership in GIF committees: The United Kingdom is continuing to present project proposals for engagement with the GIF SFR and VHTR project arrangements and working groups and task forces, and will be seeking formal agreement from the other partners to join these arrangements as soon as possible. This progress was reported at the System Steering Committee (SSC) WG–TF reporting meetings on 21-22 October 2020.

United Kingdom COVID-19 nuclear update: The United Kingdom is supporting efforts to ensure that the COVID-19 pandemic has a minimal impact on nuclear operations, as reported at the 64th IAEA General Conference. This includes guidance on social distancing and personal hygiene measures, tracking the health of workers, prioritization of tasks, identification of essential staff, provision of physical protection of equipment, temperature screening, contact tracing from the start of the pandemic, as well as proactive increased communication with staff, external stakeholders and the host communities during this unprecedented time. Nuclear electricity production in the United Kingdom has continued uninterrupted during this period.

Updates from the UK Nuclear Industry Council (NIC) and Nuclear Industry Association (NIA): The NIA produced the report, *Forty by '50: The Nuclear Roadmap*, for the industry-government NIC. It was released ahead of the annual progress update from the government's Committee on Climate Change. According to the report, an ambitious program based on existing and new technologies could provide up to 40% of clean power by 2050 and drive deeper decarbonization. It could eventually bring as many as 300 000 jobs and GBP 33 billion of added annual economic value. In the United Kingdom, nuclear energy currently contributes 40% of the UK's clean electricity, but demand is expected to quadruple from the replacement of fossil fuels and a boom in the electric vehicles and heating sectors.

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 the 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 following summary briefly highlights some of the more recent activities that the Office of Nuclear Energy supports.

The versatile test reactor (VTR): The VTR was formally launched in February 2019 as a part of efforts to modernize the nuclear R&D user facility infrastructure in the United States. The VTR will provide a leading-edge capability for accelerated testing and qualification of advanced fuels and materials. The VTR is proposed to be a 300 MWth sodium-cooled, fast spectrum reactor capable of testing advanced nuclear fuels and materials for the next generation of nuclear reactors.

On 11 September 2020, the DOE approved Critical Decision 1 for the VTR project. Critical Decision 1, known as "Approve Alternative Selection and Cost Range," is the second step in the formal process that the DOE uses to review and manage research infrastructure projects. As part of Critical Decision 1, federal committees reviewed the conceptual design, schedule, and cost range, and analyzed potential alternatives. The VTR project will move to the engineering design phase as soon as Congress appropriates funding. The DOE has requested USD 295 million for fiscal year 2021 for the VTR project.

Hydrogen production: The DOE continues to evaluate and demonstrate integrated energy systems that competitively produce electricity and non-electric products, such as hydrogen production, to optimize revenue generation by NPPs. Two NPP projects were awarded during the fiscal year 2019. The first award was to Exelon Corporation, through the Office of Energy Efficiency and Renewable Energy (EERE), to install a 1 MW proton exchange membrane (PEM) electrolyzer, storage and controls at one of their sites for on-site hydrogen needs. The second award was made to Energy Harbor, Xcel and Arizona Public Service (APS) to install a similar PEM electrolyzer at Energy Harbor's Davis Besse Nuclear Power Station for both on-site and off-site uses. The project will also develop technical and economic assessments for hydrogen generation at an Xcel and APS site.

In the fiscal year 2020, a total of USD 21 million was identified for hydrogen demonstrations in support of the existing fleet of nuclear reactors, with USD 11 million from the Office of Nuclear Energy (NE) and USD 10 million from the Office of Energy Efficiency and Renewable Energy (EERE). The funding was made available through NE's Industry Funding Opportunity Announcement. On 8 October 2020, the DOE announced that Xcel Energy and FuelCell Energy Inc. were awarded the funds to advance flexible operation of light water reactors with integrated hydrogen production systems.

Advanced Reactor Demonstration Programme (ARDP): As a part of its mission, the NE supports the development of advanced reactor designs and capabilities over a continuum of technology maturity levels. The DOE currently supports R&D activities for a variety of advanced reactor technologies that are expected to improve on the safety, security, economics and/or environmental impacts of current nuclear power plant designs. The DOE undertakes these activities in support of the Administration's objectives to maintain the nation's technological leadership position in the global nuclear industry and ensure national energy security. As part of the fiscal year 2020 Further Consolidation Appropriations Act, (H.R. 1865), Congress has provided funding for the NE to address advanced reactor development through the ARDP.

The NE released a funding opportunity announcement (FOA) that is comprised of three separate pathways. The ARDP has a goal of focusing DOE and non-federal resources (through cost-sharing agreements with industry) on the actual construction of real demonstration reactors that are safe and affordable to build in the near to midterm.

The ARDP has identified two separate pathways to meet this goal: 1) Advanced Reactor Demonstration awards, which support two reactor designs to be operational in five to seven years; and, 2) the Risk Reduction for Future Demonstration awards, which supports two to five additional diverse advanced reactor designs that have a commercialization horizon that is approximately five years longer than the Advanced Reactor Demonstration awards. A third pathway, identified in H.R. 1865, Advanced Reactor Concepts-20, will support development of at least two new public-private partnership awards focused on advancing reactor designs towards the demonstration phase; these have a commercialization horizon that is approximately five years longer than the Risk Reduction for Future Demonstration awards.

On 13 October 2020, the Secretary of Energy announced that the DOE is awarding TerraPower LLC (Bellevue, Washington) and X-energy (Rockville, Maryland) USD 80 million each in initial funding to build two advanced nuclear reactors that can be operational within seven years. TerraPower will demonstrate the Natrium reactor, an SFR that leverages decades of development and design undertaken by TerraPower and its partner, GE-Hitachi. X-energy will deliver a commercial four-unit nuclear power plant based on its Xe-100 reactor design. The Xe-100 is a high-temperature gas-cooled reactor that is ideally suited to provide flexible electricity output, as well as process heat for a wide range of industrial heat applications, such as desalination and hydrogen production.

The Advanced Reactor Demonstration awards are cost-sharing partnerships with industry that will deliver two first-of-a-kind advanced reactors to be licensed for commercial operation. The department will invest a total of USD 3.2 billion over seven years, subject to the availability of future appropriations, with industry partners providing matching funds. Additionally, awards for Risk Reduction for Future Demonstration and Advanced Reactor Concept-20 (ARC-20) projects are expected to be announced in December 2020.

The National Reactor Innovation Center (NRIC): The NRIC, which was authorized by Congress in 2018 and established by the DOE in 2019, is working on behalf of NE and in partnership with the US advanced reactor development community to define NE's role of innovative nuclear reactor technologies in the clean energy economy. The NRIC strives to empower innovators with access to facilities, sites, materials and expertise to demonstrate reactors and support the demonstration of cost-cutting technologies.

4 System reports

Gas-cooled fast reactor

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 with 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 and typically 800-850°C, is an added value of the GFR technology.

Design objectives

High-outlet temperature (850°C) for high thermal efficiency and hydrogen production, and a direct cycle for compactness, are key reference objectives. Unit power will be considered in the range of 200 MWe (modularity), up to larger 1 500 MWe. Generation IV (Gen-IV) objectives for construction time and costs are therefore to be considered.

The objective of high fuel burn-up, together with actinide recycling, results in spent-fuel characteristics (isotopic composition) that are unattractive for handling. High burn-up is the final objective.

Consensus has been reached in the project to minimize feedstock usage with a self-sustaining cycle, which requires only depleted or reprocessed uranium feed. This would call for a self-generating core with a breeding gain near zero. So as not to penalize the long-term deployment of GFRs, and based on considerations regarding both the foreseen, available plutonium stockpiles (mainly derived from water reactors' irradiated fuel) and time for GFR fleet development, it is recommended that the initial Pu inventory in the GFR core not be much higher than 15 tonnes per GWe.

Reference concept

The reference concept for the GFR is a 2 400 MWth plant having a breakeven core, operating with a core outlet temperature of 850°C that would enable an indirect, combined gas-steam cycle to be driven via three intermediate heat exchangers. The high core outlet temperature places onerous demands on the capability of the fuel to operate continuously with the high-power density necessary for good neutron economics in a fast reactor core. The core is made up of an assembly of hexagonal fuel elements, each consisting of ceramic-clad, mixed-carbide-fuelled pins contained within a ceramic hextube. The favoured material for the pin clad and hextubes at the moment is silicon carbide fibre reinforced silicon carbide (SiCf/SiC). The entire primary circuit with three loops is contained within a secondary pressure boundary, the guard containment. The produced heat is converted into electricity in the indirect combined cycle, with three gas turbines and one steam turbine. The cycle efficiency is approximately 48%. A heat exchanger transfers the heat from the primary helium coolant to a secondary gas cycle containing a helium-nitrogen mixture, which in turn drives a closed cycle gas turbine. The waste heat from the gas turbine exhaust is used to raise steam in a steam generator, which is then used to drive a steam turbine. Such a combined cycle is common practice in natural gas-fired power plants and so it represents an established technology, with the only difference in the case of the GFR being the use of a closed cycle gas turbine.

The ALLEGRO gas-cooled fast reactor demonstrator project

The objectives of ALLEGRO are to demonstrate the viability and to qualify specific GFR technologies such as fuel, fuel elements, helium-related technologies and specific safety systems, in particular the decay heat removal function. It will also demonstrate 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 reactor.

The original design of ALLEGRO consists of two helium primary circuits, three decay heat removal (DHR) loops integrated into 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 serve not only as a demonstration reactor, hosting GFR technological experiments, but also as a test pad to:

- use the high-temperature coolant of the reactor in a heat exchanger to generate process heat for industrial applications;
- carry out research in a research facility which – thanks to the fast neutron spectrum – makes it attractive for fuel and materials development;
- test some of the special devices or other research work.

The 75 MWth reactor shall be operated with two different cores: the starting core, with uranium oxide (UOX) or mixed oxide (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, enabling operation of ALLEGRO at the 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 over the long term. In addition to lifetime extensions of their nuclear units, each country has decided to build new units in the future.

Four nuclear research institutes and companies in the Visegrad-Four region (ÚJV Řež, a.s., Czech Republic, MTA EK, Hungary, NCBJ, Poland, VUJE, a.s., Slovak Republic) have decided to start joint preparations aiming at the construction and operation of the ALLEGRO demonstrator for the Gen-IV gas-cooled fast reactor (GFR) concept, based on a memorandum of understanding signed in 2010. The French Alternative Energies and Atomic Energy Commission (CEA), as the promoter of the GFR concept since 2000, supports these joint preparations, bringing its knowledge and its experience to building and operating experimental reactors, and in particular fast reactors.

In order to study safety and design issues, as well as medium- and long-term governance and financial issues, in July 2013 the four aforementioned organizations created a legal entity, the V4G4 Centre of Excellence, which performed the preparatory work needed to launch the ALLEGRO Project. The V4G4 Centre of Excellence is also in charge of international representation for this project. As a result of the preparatory work, it was revealed that during earlier work certain safety and design issues remained unsolved and for several aspects a new ALLEGRO design had to be elaborated. In 2015, therefore, when the ALLEGRO Project was launched, a detailed technical program was established with a new time schedule.

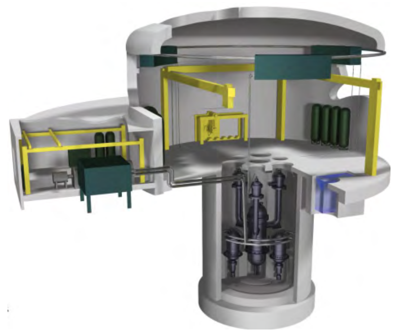
Fuel cycle and fuel

Fuel development efforts must be conducted in close relation with reactor design efforts so that both the fuel meets core design requirements and 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 high-temperature reactor [HTR] up to 1 600°C, and to be confirmed through design and safety studies for the GFR);
- resist fast neutron-induced damage, to provide excellent confinement of the fission products;
- accommodate increased heavy metal content.

Figure GFR-1. ALLEGRO Systems



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

The candidate fuel types already identified are:

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

A significant amount of knowledge is available on MOX fuel, but more 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 inconsistent. These materials need to be adapted in order to cope with the different loads (e.g. thermal gradients, interaction fuel-barrier, dynamic loads), which means that their composition and microstructure need specific developments. The main goal of high-temperature experiments is to investigate the behavior of 15-15 titanium (15-15Ti) 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 the load-bearing capacity of cladding after high-temperature treatments. The cladding microstructure will be examined by scanning electron microscopy (SEM) and metallography.

The development of a qualification procedure for start-up fuel will include specification of the steps for MOX/UOX fuel with 15-15Ti 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 FUROM code with fast reactor fuel properties and models in order to simulate fuel behavior for the ALLEGRO start-up core. Validation of the code should be based on sodium-cooled fast reactor fuel histories.

Testing of SiC claddings in high-temperature helium will be carried out to track potential changes. Mechanical testing and the examination of the microstructure with SEM and metallography is planned with the samples after high-temperature treatment.

The ion-irradiation effect on SiC composites will be investigated in order to evaluate the importance of the significant volume change observed for hydrogen (Hi)-Nicalon type-S fibre and C fibre coating. High-dose ion irradiation will be carried out with various temperature ranges, including

GFR operating temperatures for SiC composites. The high-dose irradiation effect on SiC composites will be examined.

The investigation of high-temperature oxidation behavior of SiC composites is important for severe accident studies. 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 the understanding of oxidation behavior.

The following topics will be analyzed in the short term:

Design of the ALLEGRO reactor core:

- UOX core feasibility study using ERANOS, MCNP, SERPENT;
- 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 the reactor's 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 the 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.

The SafeG project

The SafeG project has received funding from the Euratom Horizon 2020 program NFRP-2019-2020-06, under grant agreement no. 945041. The global objective of the SafeG project is to further develop GFR technology and strengthen its safety. The project will support the development of nuclear, low-CO_2 electricity and the industrial process heat generation technology through the following main objectives:

- to strengthen the safety of the GFR demonstrator ALLEGRO;
- to review the GFR reference options in materials and technologies;
- to adapt GFR safety to changing needs in electricity production worldwide, with increased and decentralized portions of nuclear electricity, by studying various fuel cycles and their suitability from safety and proliferation resistance points of view;
- to bring in students and young professionals, boosting interest in GFR research;
- to deepen the collaboration with international, non-EU research teams, and relevant European and international bodies.

The main task of the project is to respond to the safety issues of the GFR concept and to introduce the key safety systems of the ALLEGRO reactor. An important part of the design is to acquire new experimental data using recent research from experimental devices and special computational programs to carry out safety analyzes and the study of relevant physical phenomena. The SafeG project takes into account the most urgent questions and open issues concerning the GFR technology and the ALLEGRO demonstrator. To answer these questions, the SafeG project is divided into six technical work packages and one co-ordination work package.

The ambitions of the SafeG project can be divided into four tasks:

1) Completing the ALLEGRO demonstrator safety concept:

- core optimization from the neutronic, thermo-hydraulic and thermo-mechanic points of view;
- design of diversified reactor control and reactor shutdown systems;
- passive decay heat removal strategy completed with the design of fully passive systems for the decay heat removal tested on the experimental helium loop.

2) Upgrading the ALLEGRO demonstrator design and GFR concept through innovative materials and technologies, such as fuel cladding based on SiC composition, and construction materials capable of withstanding the extreme temperatures used for the primary system and safety-related systems.

3) Linking national research activities and creating an integrated platform that aims to share knowledge, and results achieved, as well as to co-ordinate activities, and spread new ideas and findings throughout the scientific society worldwide.

4) Expanding cooperation between Europe and Japan on GFR research through the sharing of knowledge about advanced high-temperature resistant materials for fuel rod claddings and other primary system components.

Branislav Hatala
Chair of the GFR SSC, with contributions from GFR members

Lead-cooled fast reactor

The Generation IV International Forum (GIF) has identified the lead-cooled fast reactor (LFR) as a technology with great potential to meet the needs of both remote sites and central power stations, fulfilling the four main goals of GIF. In the technology evaluations of the *Generation IV Technology Roadmap* (2002), and its update in 2014, the LFR system was ranked at the top in terms of sustainability (i.e. a closed fuel cycle can be easily achieved), and in proliferation resistance and physical protection. It was also assessed as good in relation to safety and economics. Safety was considered to be enhanced by the choice of a relatively inert coolant. This section highlights the main collaborative achievements of the LFR-provisional System Steering Committee (pSSC) to date. It also presents the status of the development of LFRs in GIF member countries and entities.

Main characteristics of the system

LFR concepts include three reference systems: 1) a large system rated at 600 MWe (e.g. the European lead fast reactor [ELFR EU]), intended for central station power generation; 2) a 300 MWe system of intermediate size (e.g. BREST-OD-300, Russia); and 3) a small, transportable system of 10-100 MWe size (e.g. the small secure transportable autonomous reactor [SSTAR], United States) that features a very long core life (see Figure LFR-1). The expected secondary cycle efficiency of each LFR system is at or above 42%. GIF-LFR systems thus cover the full range of power levels: small, intermediate and large sizes. Important synergies exist among the different reference systems, with the co-ordination of the efforts carried out by participating countries one of the key elements of LFR development. The typical design parameters of GIF-LFR systems are briefly summarized in Table LFR-1.

R&D objectives

The LFR System Research Plan (SRP) developed within GIF is based on the use of molten lead as the reference coolant and lead-bismuth eutectic (LBE) as the back-up option. Given the R&D needs for fuel, materials and corrosion-erosion control, the LFR system is expected to require a two-step industrial deployment: in a first step, reactors operating at relatively modest primary coolant temperatures and power densities would be deployed by 2030; and higher performance reactors by 2040. Following the reformulation of the GIF-LFR-pSSC in 2012, the SRP has been completely revised. The report is presently intended for internal use by the LFR-pSSC, but it will ultimately be used as a guideline for the definition of project arrangements once the decision of a transition from the present memorandum of understanding (MoU) status to a system arrangement organization is engaged.

Table LFR-1: Key design parameters of the GIF-LFR concepts

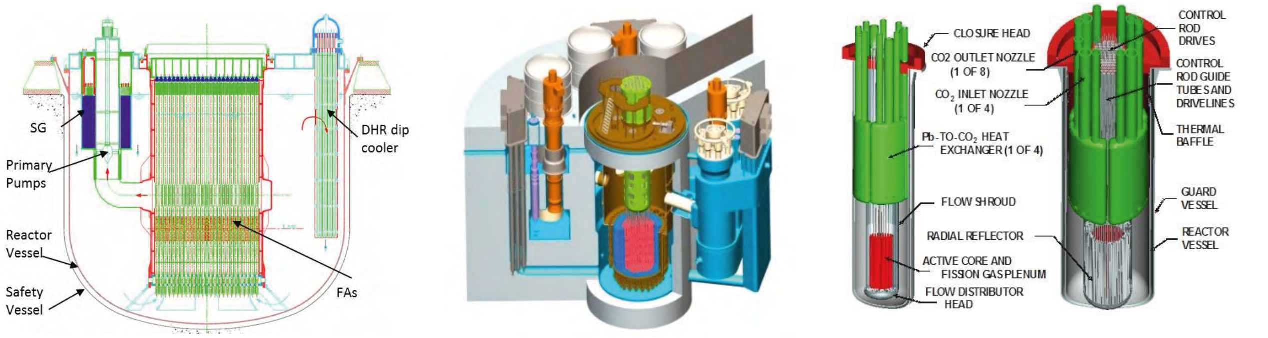
Parameters	ELFR	BREST	SSTAR
Core power (MWt)	1 500	700	45
Electrical power (MWe)	600	300	20
Primary system type	Pool	Pool	Pool
Core inlet T (°C)	400	420	420
Core outlet T (°C)	480	540	567
Secondary cycle	Superheated steam	Superheated steam	Supercritical CO_2
Net efficiency (%)	42	42	44
Turbine inlet pressure (bar)	180	180	200
Feed temperature (ºC)	335	340	402
Turbine inlet temperature (ºC)	450	505	553

Main activities and outcomes

The collaborative activities of the LFR-pSSC over the last eight years were centred on top level reports for GIF. After the issuance of the "LFR White Paper on Safety" in collaboration with the GIF Risk and Safety Working Group (RSWG) in 2014, the pSSC has been very active on the following main lines of work:

- *LFR system safety assessment* – The RSWG asked SSC chairs to develop a report on their systems in order to analyze them systematically, assess the safety level and identify further safety-related R&D needs. The LFR assessment report was prepared in collaboration with the RSWG and was published in June 2020. It is presently available.[1]

Figure LFR-1. GIF-LFR reference systems: ELFR, BREST and SSTAR



Alemberti, A. et al. (2018).

1. www.gen-4.org/gif/upload/docs/application/pdf/2020-06/gif_lfr_ssa_june_2020_2020-06-09_17-26-41_202.pdf.

Figure LFR-2(a). Experimental investigation of the dynamics of fuel assembly mock-ups

Figure LFR-2(b). Testing of loading and unloading of fuel assembly mock-ups

- "LFR Proliferation Resistance and Physical Protection (PRPP) White Paper" – In 2018, the PRPP Working Group realized the need for a substantial revision of the PRPP white paper for the six GIF systems. The modifications related to the LFR paper were mainly related to the addition of the BREST system (Russia) and refinements to information available on the SSTAR (US) and ELFR (Euratom) systems. The paper has been developed with the PRPPWG, and the final version was sent to the Experts Group for final approval at the end of 2020. A public issue on the GIF website is expected in 2021.
- *LFR safety design criteria (SDC)* – Development of the LFR-SDC was based on the previously-developed SFR SDC report. It was later realized, however, that the IAEA SSR-2/1 (the reference document for SFR SDC development) did not require many of the features identified for the SFR to be adapted for the LFR (note that the IAEA SSR-2/1 refers substantially to LWR technology). After a first set of comments, received at the end of 2016, the LFR-pSSC updated the report following the IAEA revision of SSR-2/1 and the document was re-circulated for comments within the RSWG. The final comments from the RSWG were received in December 2020, and the report is presently expected to receive Experts Group approval 1st semester 2021.

The LFR-pSSC has also been working actively with the GIF Task Force on R&D Infrastructure and has contributed to the questionnaire provided by the Advanced Manufacturing and Materials Engineering (AMME) Task Force. These activities led to participation in the February 2020 workshop organized at the NEA in Paris.

Interaction between LFR-pSSC and the Working Group on Safety of Advanced Reactors (WGSAR) started through the participation of LFR representatives in the October 2020 meeting of the WGSAR. LFR-SDC and LFR-pSSC activities were presented, and it was agreed to transmit the LFR-SDC report to the WGSAR.

Main activities in Russia

The innovative fast reactor with lead coolant, BREST-OD-300, is being developed as a pilot demonstration prototype of basic commercial reactors with a closed nuclear fuel cycle for the future nuclear power industry.

The lead coolant was chosen on the basis of the favourable characteristics of its properties, namely: 1) in combination with dense (U-Pu)N fuel, it allows for complete breeding of fissile materials in core, maintaining a constant small reactivity margin and thus preventing any prompt-neutron excursion with an uncontrolled power increase (equipment failures or personnel errors); 2) it enables the possibility to avoid the void reactivity effect due to the high boiling point and high density of lead; 3) it prevents coolant losses from the circuit in the postulated event of vessel damage because of the high melting/solidification points of the coolant and the use of an integral layout of the reactor; 4) it provides high heat capacity of the coolant circuit, which decreases the probability of fuel damage; 5) it capitalizes on its high density and albedo properties for flattening the fuel assembly (FA) power distribution; and 6) it facilitates larger time lags of the transient processes in the circuit, which makes it possible to lower the requirements for the safety systems' rate of response.

Mixed uranium-plutonium nitride fuel is used in the core design, and low-swelling ferritic-martensitic steel is used as the fuel cladding. Fuel elements are placed in shroud-less hexagonal fuel assemblies. Currently, the technology of dense nitride fuel is implemented on pilot production lines. These technological processes are being improved, and industrial fuel production is being created for the fabrication of fuel for the BREST-OD-300 reactor. For the initial stage of BREST-OD-300 operation, a reduced value of the maximum fuel burn-up is planned – 6% heavy atoms (h.a.); then, a gradual justified transition to the design target values of burn-up up to 9-10% h.a. is envisaged. The performance of the nitride fuel is confirmed by the results of radiation tests in the BN-600 power reactor and BOR-60 research reactor. In total, more than 1 000 fuel elements were irradiated. For one experimental FA with fuel elements of the BREST type, burn-up of more than 9% h.a. and a damage dose of more than 100 dpa were achieved. All semi-finished products from EP823 steel were put into production, and all properties have been obtained that ensure the operability of the fuel elements up to 6% h.a. burn-up: short-term, long-term, under irradiation, and in a lead coolant medium. FA mock-ups (all types) and reflector blocks were produced in industrial conditions, and manufacturing technology is fully developed. All the necessary experimental studies were carried out for these mock-ups: spills in water and lead, vibration tests, and tests for bending stiffness and strength. The loading and unloading of FA mock-ups from the core were experimentally tested, as shown in Figure LFR-2.

The main objective of the reactor vessel, when performing safety functions, is to exclude the loss of coolant. Estimated probability of coolant leakage from the reactor circuit is about 9·10-10 1/year. With this event, only a partial loss of the coolant is possible (non-critical, acceptable); at the same time, the primary circuit does not break, and the possibility of natural circulation of the coolant in the circuit remains. A wide range of experimental work was carried out on the metal-concrete vessel – on various concrete samples, mock-ups of the vessel itself and on its elements. Properties of high-temperature concrete were obtained experimentally at temperatures of 400-700°C, and under irradiation. The chemical inertness of the lead coolant in relation to concrete was shown. Sufficient knowledge has been collected to start manufacturing the reactor vessel of the BREST-OD-300 reactor.

The steam generator consists of monometallic tubes, corrosion resistant in water and lead, with no welds along the entire length. The steam generator has twisted heat exchange parts. To date, a comprehensive justification of the elements and processes occurring in the steam generator has been carried out. It can be noted in particular that the absence of induced failure in case of one tube rupture was experimentally demonstrated. Experiments have shown that neighbouring tubes are not damaged, which is a very important achievement for safety. Another important point that has been confirmed through calculations alone, but will be tested in an experiment at the Fast Critical Facility (BFS), is that with the postulated passage of steam bubbles through the core (i.e. when the tubes of the steam generator break) there is no burst of positive reactivity. The value of the void-vapour reactivity effect is close to zero. It should be noted that because of the presence of a free surface level in the reactor vessel, the probability of steam entering the core during depressurization of the steam generator tubes is extremely low.

To justify the pumps, a wide range of studies were carried out. At the initial stages, the flow parts were optimized, and the shapes of the impeller blades were selected. By means of calculations and experiments on scale models, the head characteristics of the main circulation pump (MCP) were obtained. Positive results were obtained on the life tests of the bearing (justification of the life of 100%), and full-scale bench MCP testing in lead is being created, see Figure LFR-3. As for other equipment and systems, the prototype of the control and protection system (CPS) actuator


Adamov, Ye.O. et al. (2020).

Figure LFR-3: Testing of the friction pair of the MCP lower radial bearing as part of the model block

passed acceptance and life tests; endurance testing of coolant quality system components is being conducted; and automated monitoring and control system has been developed. Prototypes of equipment are undergoing final testing, and thus they are ready for implementation in the reactor during construction. A large set of experimental studies were carried out concerning the assessment of the yield of fission and activation products from a lead coolant. This knowledge is important when performing a radiation safety analysis for various temperature levels, typical of normal operation (500°C) and accidents with significant lead heating (680°C). Based on the data obtained in the experiment, the requirements are determined for the composition of the initial lead for the primary coolant. In the course of the optimization performed, the composition of impurities was minimized, while maintaining an acceptable cost of lead.

The safety analysis has shown that under the most conservative scenario of inserting the full reactivity margin, the maximum fuel temperature will reach 1 640°C, and fuel cladding 1 260°C (for a few seconds). There is no fuel melting, and the lead coolant does not boil. The implementation of such a scenario is feasible with a probability of 2.9·10-9 1/year. For another conservative scenario, complete blackout of the power unit with failure of mechanical shutdown systems (ATWS), the level of the attainable fuel-element cladding temperature is lower than in the first scenario and does not exceed 903°C. Long-term cooling is carried out using a passive emergency cooling system of the reactor, with natural circulation of lead in the primary circuit. For both scenarios, the main requirement has been met - there is no need to protect the population.

The BREST-OD-300 reactor is being created as one of the most important components of the pilot demonstration power complex operating in a closed fuel cycle, together with modules for fabrication, re-fabrication and reprocessing of spent fuel. In addition to operation (power generation), the most important task is the implementation of the R&D program at the reactor. Various studies and life tests are planned to be carried out on components, equipment, and irradiation experiments in a lead coolant and in a fast neutron spectrum. This will form an essential scientific basis for research. The BREST-OD-300-unit design received a positive conclusion from the Glavgosexpertiza, and licensing by Rostekhnadzor is being completed. In 2020, an examination was undertaken by the Russian Academy of Sciences, which gave a positive conclusion and recommended the construction of the power unit, confirming that the design corresponds to the modern level of science and technology, as well as to scientific ideas about the problems of existing nuclear energy and ways to solve them.

Main activities in Japan

Theoretical studies of fast reactors using lead-bismuth eutectic as a coolant have been performed in Japan since the beginning of LFR activities. One of the advantages of lead or lead-bismuth coolant is the better neutron economy in the core due to the hard neutron spectrum and the small neutron leakage. These features make it easy to realize the once-through fuel cycle, fast reactor concepts. The concepts of the breed-and-burn reactors and CANDLE burning reactors with lead-bismuth coolant have been studied at the Tokyo Institute of Technology (TIT). One of the important issues related to these concepts is maintaining the integrity of fuel elements in very high burn-up conditions. Research has confirmed the possibility to solve the problem through the introduction of the melt-refining process, based on metallic fuel. The study also considered the use of plutonium from LWR spent fuel for the start-up core to achieve effective use of plutonium. A new fuel shuffling scheme was proposed as the output of the studies. It has proven that it is possible to achieve a stationary wave equilibrium condition by implementing a fuel shuffling scheme concept.

Chemical compatibility of lead (Pb) and Pb alloys with various materials in different situations is being studied at the Tokyo Institute of Technology (TIT). Figure LFR-4 shows these different situations where chemical compatibility presents important issues to be addressed. The structural materials that exhibit corrosion resistance are essential to expand the operating life and to improve the reliability of Pb coolant systems. Excellent corrosion resistance of ferritic iron-chromium-aluminium (FeCrAl) alloys (Kanthal® APMT and FeCrAlZr-ODS) in liquid Pb and Pb alloys was confirmed. α-Al2O3 formed on the FeCrAl alloys from the pre-oxidation treatment in air atmosphere at 1 273K for 10 hours.

This oxide layer functions as a protective layer, which can significantly improve the thermo-dynamic stability and the chemical compatibility of the alloys. The metallurgical analysis with scanning transmission electron microscope (STEM) on the

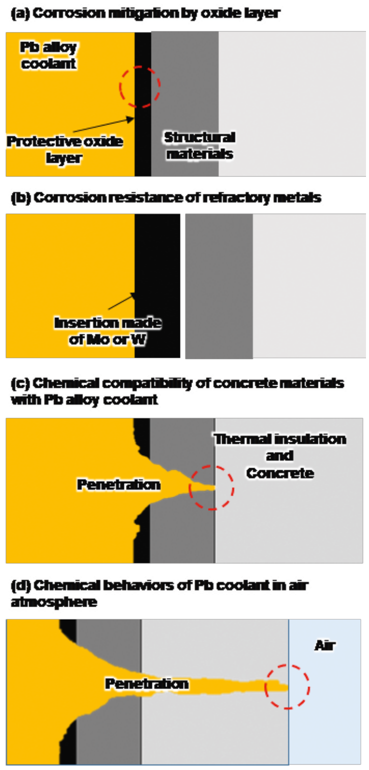


Figure LFR-4: Corrosion issues in various situations

protective oxide layer after the immersion in liquid Pb alloy was performed in the collaborative project for the development of FeCrAl zirconium (Zr)-oxide dispersion-strengthened alloys. Experimental studies on the mass transfer of metal and non-metal impurities in a lead-bismuth coolant system have been performed. The diffusion behaviors of metal impurities such as Fe and nickel (Ni) in lead-bismuth were investigated by means of long capillary experiment and molecular dynamic (MD) simulation. The diffusion coefficients of these elements were newly obtained for various temperatures. Refractory metals such as molybdenum (Mo) and tungsten (W) are also corrosion resistant in liquid Pb alloys. Therefore, the insertion and the lamination of plates made of the refractory metals are proposed to suppress the corrosion of structural materials as shown in Figure LFR-4 (b).

Concrete materials must work as an important barrier, which suppresses the Pb coolant leakage and the loss-of-coolant accident, especially for the pool-type Pb-based reactors, as shown in Figure LFR-4 (c). The chemical compatibility of some cement and concrete materials having various water/cement (W/C) ratios is being investigated by means of their immersion in liquid Pb alloys. Corrosion-resistant concrete materials are also going to be developed.

The thermo-dynamic behavior of liquid Pb alloys in the air atmosphere, shown in Figure LFR-4 (d) were investigated by means of the static oxidation experiments for Pb alloys with various chemical compositions. The results of the static oxidation tests for Pb-bismuth (Bi) alloys indicate that the chemical reactivity of Pb and Pb alloys in air at high temperatures was quite mild. In the oxidation procedure of the Pb alloys, Pb was depleted from the alloys due to the preferential formation of lead oxide (PbO) in air at 773K. Bi was not involved in this oxidation procedure. Pb-Bi oxide and Bi2O3 were formed only after the enrichment of Bi in the alloys due to Pb depletion.

The chemical control of liquid Pb alloy coolant was improved with high-performance solid electrolyte oxygen sensors, which can provide a better response in high-temperature conditions. The excellent performance of the sensor with shorter stabilization time is achieved by reducing the gas volume in the reference compartment of the oxygen sensor.

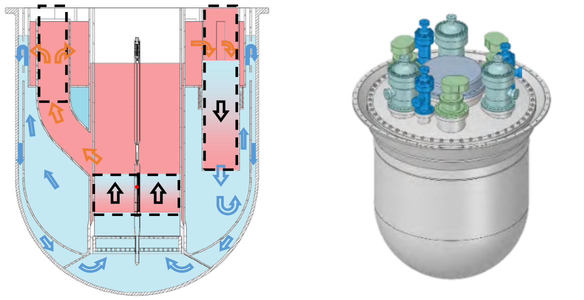
Main activities in Euratom

The main activities in Europe related to liquid metal technologies are centred on two main projects: 1) the development of the Multi-purpose hYbrid Research Reactor for High-tech Applications (MYRRHA) research infrastructure, which is being carried out by SCK-CEN in Mol (Belgium) and is aiming at the demonstration of an accelerator-driven system (ADS) technology and supporting the development of Gen-IV systems; and 2) preliminary activities for the construction of an LFR demonstrator in Romania, or the ALFRED project. These two projects are supported through dedicated Euratom initiatives.

Concerning the development of MYRRHA, the project roadmap for the implementation of Lead Bismuth Eutectic (LBE) technology for an Accelerator Driven System (ADS) was defined at the end of 2018. In September 2018, the Belgium federal government also decided to allocate EUR 558 million to the implementation of MYRRHA during the period 2019-2038 as follows:

- EUR 287 million for phase 1: building of MINERVA (linear accelerator up to 100 MeV, 4 mA + Proton Target Facility [PTF]) during the period of 2019-2026;
- EUR 115 million for phases 2 and 3: phase 2 involves the design and R&D of the second section of accelerator up to 600 MeV, while phase 3 involves further design and licensing activities related to the LBE-cooled sub-critical reactor, both to be carried out in the period of 2019-2026;
- EUR 156 million for the operating expenses of MINERVA for the period of 2027-2038.

Figure LFR-5: ALFRED primary system flow-path configuration (left) and external view (right)



Frignani, M., Alemberti, A., and Tarantino, M. (2019).   Alemberti, A., et al. (2020).

The MYRRHA project is currently being implemented, and is also supported by numerous Euratom-funded collaborative projects.

Regarding the ALFRED project, the main development activities are conducted by Ansaldo Energia (Italy), the National Agency for New Technologies, Energy and Sustainable Economic Development (ENEA, Italy) and the Institute for Nuclear Research (RATEN ICN, Romania), which are the signatories of the Fostering ALfred CONstruction (FALCON) Consortium Agreement. The FALCON Consortium Agreement was renewed at the end of 2018 for an additional phase of activities. One of the main aims of the consortium is to involve a number of additional European partners in the ALFRED project, through the signature of memoranda of agreement (MOA) expanding throughout Europe as much as possible the interest in the development of lead technology. By the end of 2020, the FALCON Consortium enlarged the community and extended the ALFRED project, with the signature of several MOAs with partners willing to provide in-kind support to technical activities related to ALFRED development.

An important event took place in June 2019 in Pitesti (Romania), where the European Commission (EC) co-organized the Fission Safety (FISA) 2019 and EURADWASTE'19 conferences with the Ministry of Research and Innovation of Romania and RATEN ICN, under the auspices of the Romanian presidency of the EU and in collaboration with the IAEA. The conference gathered 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 (FPs). In that framework, a side workshop organized by the FALCON Consortium on ALFRED infrastructure attracted a very large number of participants, stimulating discussions on the status of heavy liquid metal technology R&D activities and the roadmap for the LFR demonstrator in Europe.

The FALCON Consortium took important steps during the period of 2018-2020. First, a main step of the design review was completed, and a new system configuration was defined, consisting of three steam generators (SGs) (using benefits from the new configuration it was designed single-wall bayonet tubes), three dedicated dip coolers for the second decay heat removal (DHR) system, and three primary pumps (PP). The definition of the placement of other dedicated systems and components on the reactor roof is presently under way. Additional design changes have been carried out in the primary system configuration, involving an improved definition of hot and cold pools and a special arrangement of the primary flow path to completely eliminate the thermal stratification on the vessel for both forced and natural circulation conditions. The new configuration and its main characteristics are presented in Figure LFR-5.

The DHR-1 system consists of isolation condensers connected to steam generators (three units) and equipped with an anti-freezing system, which is being investigated in the PIACE Euratom collaborative project (cf. below). A similar system is being used for the DHR-2 system connected to a dip cooler, which uses double wall bayonet tubes.

In 2019, the FALCON Consortium also took an important decision regarding ALFRED operation and licensing. Namely, it was decided to approach both operation and licensing using a stepwise approach to better face the known limits concerning materials corrosion and consequent qualification in a representative environment.

The idea is to follow a staged approach, characterized in principle by a constant primary mass flow and increasing power levels, which results in an increase of the maximum lead coolant temperature as follows:

- 1st stage: low temperature
  - proven technology, proven materials, oxygen control, low temperatures;
  - hot FA for in-core qualification of dedicated coating for cladding;
- 2nd stage: medium temperature
  - need for FA replacement, same SGs and PPs;
  - hot FA for in-core qualification at higher temperatures;
- 3rd stage: high temperature
  - replacement of main components for improved performances;
  - representative of first-of-a-kind (FOAK) conditions for the LFR deployment.

In this way, each stage is consequently used to qualify (through the hot fuel assembly conditions) 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(s), the licensing process is expected to be a continuous process able to bring the technological solutions to the higher temperatures needed for industrial deployment. Table LFR-3 below provides the main parameters of the envisaged staged approach.

Table LFR-3: ALFRED staged approach – Main parameters

Normal operation – full power	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

For the interested reader, further information on the ALFRED design and staged approach can be found in the papers presented at the FISA 2019 Conference.

During the year 2019, the Romanian government also awarded RATEN ICN (Romanian research laboratory) a funding of EUR 2.5 million for a project dedicated to "Preparatory activities for ALFRED infrastructure development in Romania". The project ended successfully in November 2020, with a final workshop organized by RATEN ICN.

RATEN ICN also responded to a call for proposals from the Romanian government with a project called "ALFRED - Step 1, experimental research support infrastructure: ATHENA (lead pool-type experimental facility) and ChemLab (lead chemistry laboratory)". The project proposal was awarded funding in June 2020, and the competitive bids for design and construction of related facilities are to be published in 2021 for a total budget of about EUR 20 million.

Finally, with regard to Euratom R&D projects, the main collaborative projects already in place related to LFR technology and Gen-IV fuels are: 1) GEMMA, dedicated to material R&D and qualification for Gen-IV LFRs; 2) M4F, covering material R&D for Gen-IV and fusion applications; 3) INSPYRE, dedicated to fuel R&D for fast reactors; and 4) the LFR SMR INERI project, which involves the European Commission Joint Research Centre (JRC) and the US Department of Energy. This Euratom project portfolio has recently been complemented by three new projects: PIACE (started in 2019), as well as PATRICIA and PASCAL (both of which commenced in 2020). The PIACE project is dedicated to demonstrating the prevention of lead freezing in LFRs through passive safety provisions. The project had its kick-off meeting at the ENEA research laboratory (Brasimone) and is presently underway, with some experimental results expected to be available in 2021. The PATRICIA project provides further supporting R&D for the implementation of MYRRHA and related pre-licensing efforts, while the PASCAL project involves R&D on selected safety aspects for heavy liquid metal systems, specifically focusing on the extension of experimental evidence to demonstrate the increased resilience of MYRRHA and ALFRED to severe accidents. Lastly, the SESAME Euratom collaborative project was concluded in 2019 with the final workshop and release of a book dedicated to the thermal-hydraulic aspects of liquid metals.

Main activities in Korea

The Korean government joined the GIF-LFR pSSC by signing the MoU at NEA in November 2015. LFR R&D progress has been made mainly through university programs during the past 20 years, since the first study in 1996 at Seoul National University (SNU). Since 2019, the primary momentum of LFR development has been transferred to the Ulsan National Institute of Science and Technology (UNIST). The Korean LFR Programme, however, remains unchanged with two main objectives:

- a new electricity generation and hydrogen production unit development requirement to match the needs of economically competitive distributed power and hydrogen sources for both developed countries and developing nations that need massive and inexpensive electric power with an adequate margin against worst case scenarios encompassing internal and external events;

- a technology development requirement for sustainable power generation using energy produced during nuclear waste transmutation.

To meet the first goal, the Korean government has been funding international collaborative R&D to further upgrade the ubiquitous, rugged, accident-forgiving, non-proliferating, and ultra-lasting sustainer (URANUS) design into a micro-reactor design called MicroURANUS, which can be applied to maritime applications and has a 40-year lifespan without refuelling. A pre-conceptual design has been completed with all the top-tier design requirements met, by following GIF-LFR methodologies including LFR-SDC. Results using PIRT analysis (Phenomena Identification and Ranking Table) were reviewed by the LFR-pSSC members through a video conference.

For the second goal, since 1996 the Korean first LFR-based burner, the proliferation-resistant environment-friendly accident-tolerant continual-energy economical reactor (PEACER), has been transmuting long-lived waste in spent nuclear fuel into short-lived low-intermediate level waste. In 2008, the Korean Ministry of Science and Technology selected the sodium-cooled fast reactor (SFR) as the technology for long-lived waste transmutation. Since then, LFR R&D for transmutation in Korea has turned its direction towards an accelerator-driven Th-based transmutation system designated as the thorium optimized radioisotope incineration arena (TORIA), with the leadership of Sungkyunkwan University and Seoul National University, as well as UNIST.

Main activities in the United States

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

With regard to design concepts, of particular relevance is the past development of the small, secure transportable autonomous reactor (SSTAR), carried out by Argonne National Laboratory (ANL), Lawrence Livermore National Laboratory (LLNL) and other organizations over an extended period of time. SSTAR is a SMR that can supply 20 MWe/45 MWth with a reactor system that is transportable. Some notable features include reliance on natural circulation for both operational and shutdown heat removal; a very long core life (15-30 years) with cassette refuelling; and an innovative supercritical CO_2 (S-CO_2) Brayton cycle power conversion system. Although work on SSTAR is no longer active, SSTAR continues to be represented as one of the reference designs of the GIF-LFR pSSC.

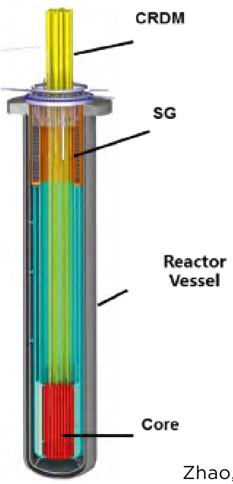
Additional university-related design activities include past work at the University of California on the Encapsulated Nuclear Heat Source (ENHS) and more recently in several projects sponsored by the US Department of Energy under Nuclear Energy University Project (NEUP) funding. These include the following ongoing efforts:

- An effort led by the Massachusetts Institute of Technology (MIT) in the area of corrosion/irradiation testing in lead and lead-bismuth eutectic. The project seeks to investigate the “Radiation Decelerated Corrosion Hypothesis”, relying on simultaneous exposure tests (rather than separate long-term corrosion and neutron irradiation), followed by microstructural characterization, mechanical testing and comparison to enable rapid down selection of potential alloy candidates and directly assess how irradiation affects corrosion.
- An effort at the University of Pittsburgh to develop a versatile liquid, lead testing facility and test material corrosion behavior and ultrasound imaging technology in liquid lead.

In the industrial sector, ongoing LFR reactor initiatives include the continuing initiative of the Westinghouse Corporation to develop a new advanced LFR system (Westinghouse-LFR) and the efforts of Hydromine, Inc. to continue development of the 200 MWe LFR identified as LFR-AS-200 (i.e. amphora shaped), as well as several micro-reactor spin-off concepts identified as the LFR-TX series (where T refers to transportable, and X is a variable identifying power options ranging from 5 to 60 MWe). It should be noted that Westinghouse is engaged with several universities and national laboratories to pursue technology developments related to the LFR, including an experimental investigation of radioisotope retention capability of liquid lead, as well as efforts to use the versatile test reactor for LFR-related investigations. Additionally, Westinghouse is currently engaged in the phase 2 effort of the UK Government’s Department for Business, Energy and Industrial Strategy’s (BEIS) Advanced Modular Reactor (AMR) Feasibility and Development project to demonstrate LFR components and accelerate the development of HT materials, advanced manufacturing technologies and modular construction strategies for the LFR.

Main activities in China

The Chinese government has provided continuous national support to develop lead-based reactor technology since 1986, by the Chinese Academy of Sciences (CAS), the Ministry of Science



Zhao, Z. et al. (2019).

Figure LFR-6. Overall view of the CLEAR-M reactor


IAEA LMFNS database

Figure LFR-7. Lead-based engineering validation reactor, CLEAR-S

and Technology (MOST), the National Science Foundation (NSF), the 13th Five-Year plan, etc. After more than 30 years of research on lead-based reactors, the China LEAd-based reactor (CLEAR), proposed by the Institute of Nuclear Energy Safety Technology (INEST)/FDS team, was selected as the reference reactor for the ADS project, as well as for the technology development of the Gen-IV lead-cooled fast reactor. Activities related to the CLEAR reactor design, reactor safety assessment, design and analysis software development and the lead-bismuth experimental loop, as well as R&D on key technologies and components, are being carried out.

The CLEAR-M project, with a typical concept of a 10 MW-grade CLEAR-M10, aiming at the construction of a small modular energy supply system, has been launched (see Figure LFR-6). The main purpose of the project is to provide electricity as a flexible power system for wide application, such as island, remote districts or industrial parks. In addition, two small LFR projects have been supported by MOST to explore innovative LFR concept designs.

For the ADS system, several concepts and related technologies are under assessment. For example, the detailed conceptual design of CLEAR-I with the final goal of minor-actinide (MA) transmutation, which has a dual operation capability of subcritical and critical modes, has been completed. An innovative ADS concept system, such as the advanced external neutron source-driven, travelling-wave reactor, CLEAR-A, was proposed for energy production. The CiADS project, conducted in collaboration with the CAS and other industrial organizations, to build a 10 MWth subcritical experimental LBE-cooled reactor coupled with an accelerator was approved, and the preliminary engineering design is underway.

In order to support the CLEAR projects, as well as to validate and test the key components and integrated operating technology of the lead-based reactor, a multi-functional lead-bismuth experiment loop platform (i.e. KYLIN-II) was built and has operated for more than 30 000 h. Various tests have been conducted, including corrosion tests, LBE thermal-hydraulic experiments and components prototype proof tests. In addition, three integrated test facilities have been built and have started commissioning since 2017, including the lead-based engineering validation reactor CLEAR-S (See Figure LFR-6), the lead-based zero power critical/subcritical reactor CLEAR-0, coupled with the HINEG neutron generator for reactor nuclear design validation, as well as the lead-based virtual reactor, CLEAR-V. A loss-of-flow benchmarking test, based on the pool-type CLEAR-S facility, is being prepared.

In recent years, other organizations have started paying more attention to LFR development. For example, the China General Nuclear Power Group (CGN), China National Nuclear Corporation (CNNC), State Power Investment Corporation (SPIC) and several universities such as Xi'an Jiaotong University (XJUT), and the University of Sciences and Technology of China (USTC) have been carrying out LFR conceptual design and related R&D, including materials tests, thermal-hydraulic analysis and safety analysis. INEST was appointed by MOST as the leading organization to co-ordinate the participation of domestic organizations in GIF activities. A domestic LFR joint working group will therefore be established.

To promote the engineering and commercial application of China lead-based reactor projects, 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 the INEST/FDS team, were established and supported by over 100 companies, and the construction of a related industrial park has begun.

Alessandro Alemberti
Chair of the LFR SSC, with contributions from LFR members

Molten salt reactor

Main characteristics of the system

Molten salt reactor (MSR) concepts have been studied since the early 1950s, but with only one test reactor operated at the Oak Ridge National Laboratory (ORNL, United States) in the 1960s. For the past 15 years, there has been a renewal of interest in this 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 alone, the concept is called a fluoride salt-cooled high-temperature reactor (FHR). Today, in the GIF pSSC MSR, most, if not to say all, the studied concepts are actual MSRs with liquid fuel.

The 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 trans-uranium (TRU) burners, with a fluoride or a fluoride carrier salt. An illustration of the most studied concept is provided in Figure MSR-1 below.

Depending on the fuel cycle, MSRs can re-use fissile and fertile materials from LWRs, or they can use uranium, or burn 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). MSRs are operated under low pressure, slightly above atmospheric pressure. They can be deployed as large power reactors or as small modular reactors (SMRs). Their deployment is today limited by technological challenges, such as high temperatures, structural materials, and corrosion.

The MSR pSSC today includes seven full members (Australia, Canada, Euratom, France, Russia, Switzerland and the United States) and three observers (China, Japan and Korea) and is moving towards a system arrangement. The mission of the MSR pSSC is to support the 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. Mastering of the technically challenging MSR technology will require concerted, long-term international R&D efforts, namely:

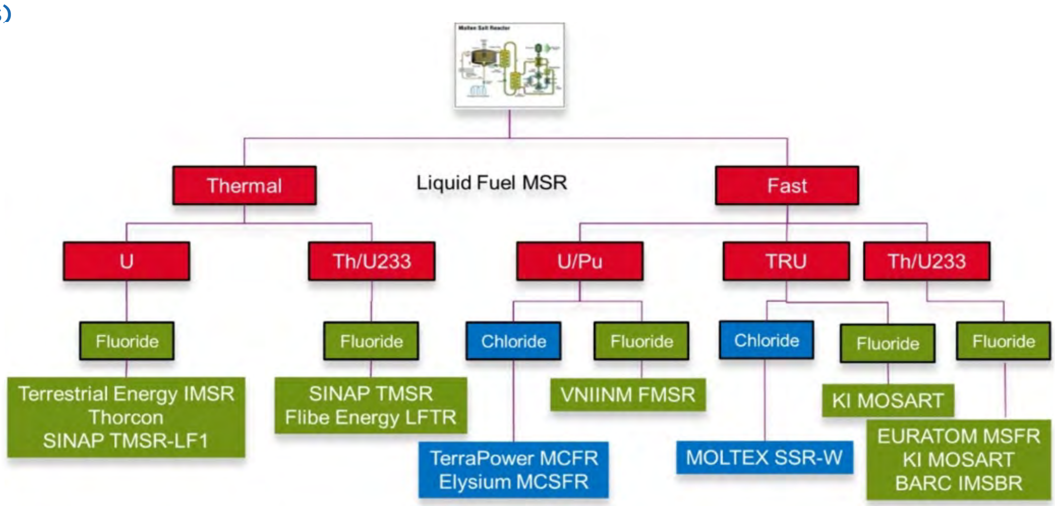
- the study of salt chemical and thermo-dynamic properties;
- for the system design, development of advanced neutronic and thermal-hydraulic coupling models;
- the study of materials compatibility with molten salt;
- salt Redox control technologies to master corrosion of the primary fuel circuit and other components;
- development of efficient techniques for the extraction of gaseous fission products from the coolant through He bubbling;
- for salt reprocessing, reductive extraction tests (actinide-lanthanide separation);
- development of a safety approach dedicated to liquid-fuelled reactors.

Main activities and outcomes

MSR pSSC activity

In 2019, the key activity was the preparation of the system arrangements (SAs) with the definition of three potential projects arrangements, which would allow the community to contribute broadly. These PAs are therefore quite transversal, and not concept-dependent, but they can support the development of any concept (see Figure MSR-2).

Figure MSR-1. The most studied MSR concepts, with key players (research & technology organization or vendors)



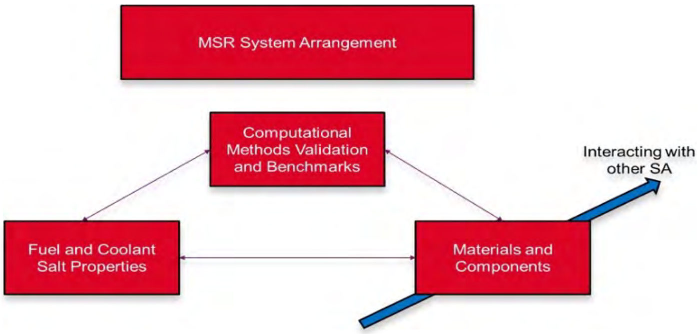


Figure MSR-2. Foreseen structure of the MSR SA, including three PAs

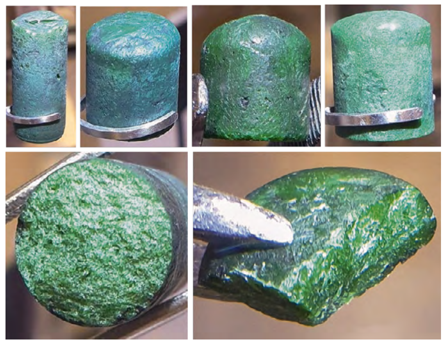


Figure MSR-3. Fuel salt ingots fabricated at JRC Karlsruhe for the SALIENT-03 irradiation experiment

They address the salt behavior, materials properties and system integration. The SA should enter into force in 2021.

Euratom

European SAMOSAFER project:

On 1 October 2019, the new Severe Accident Modeling and Safety Assessment for Fluid-fuel Energy Reactors (SAMOSAFER) project started with the aim of developing new simulation models and tools, and designing new safety barriers for the MSR. The goal of this new project is to develop and demonstrate new safety barriers for more controlled behavior of MSRs in severe accidents, based on new simulation models and tools validated with experiments. The overall 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 MSRs to ensure that they comply with the latest and future safety standards. SAMOSAFER is co-ordinated by TU Delft and will run until 2023.

In 2020, SAMOSAFER partners focused, inter alia, on the continuation of the design of the MSR, the distribution of radionuclides in the fuel treatment unit, the risk identification (list of post-irradiation examinations [PIEs]) in the Fuel Treatment µUnit (FTU) as input for the safety analysis using the failure modes and effects analysis (FMEA) method, the development of new algorithms and the design and construction of experimental setups for validation of these, as well as the generation of physico-chemical data of various molten salts to extend the JRC Karlsruhe database.

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 the NRG and JRC Karlsruhe laboratories. Follow-up irradiations (Salient-03), containing five fuel salt samples encapsulated in nickel-based alloys, are under preparation.

JRC Karlsruhe:

Among experimental studies on basic thermo-chemical properties, JRC is extensively involved in the synthesis and fabrication of the fluoride fuel salt for the planned irradiation experiment, SALIENT-03, in the high-flux reactor (HFR) at Petten via this major collaboration with NRG. Four batches of fuel salts were synthesized, having the following compositions (mol. %): 757LiF-18.7ThF4-6.0UF4-0.3PuF3 (28.02 g), 757LiF-18.7ThF4-5.7UF4-0.3UF3-0.3PuF3 (15.02 g), 757LiF-18.6ThF4-6.0UF4-0.4CrF3-0.3PuF3 (7.01 g) and 75LiF-23.0ThF4-2.0UF4-0.1UF3 (50.12 g). The first three salts will be irradiated, while the latter will serve for out-of-pile electro-chemical tests at NRG. The end members, 7LiF, ThF4, UF4 and PuF3, were synthesized and their purity verified using methods developed and published previously by JRC Karlsruhe. The method for synthesis of UF3, based on reduction of UF4 through gaseous hydrogen at 800°C, was developed specifically for the project. The fuels for irradiation were prepared in a form of solid ingots of the quenched salts exactly fitting into the irradiation capsules. The salt mixtures were melted in liners made of the same materials as the irradiation capsules (Hastelloy-N and GH3535) under a flow of pure HF gas, which was found necessary for further purification. All obtained ingots have the required shape and mass and an excellent purity, as proven by the combination of XRD, DSC, ICP-MS and oxygen analysis through Knudsen effusion mass spectrometry (KEMS) methods. The selected ingots are shown in Figure MSR-3, including a cross-section demonstrating sufficient homogeneity and purity.

With the increasing demand for reliable measurements of density for MSR fuels, significant effort has been expended in 2020 to design and test the novel densitometer at JRC Karlsruhe. The selected method of measurement is based on the Archimedean buoyancy effect, and entire set up was designed such that it fits the current glove boxes that are kept under protective argon atmosphere, and which are licensed to handle nuclear materials. Furthermore, the spherical bob that is immersed into the molten salt during measurement is made of nickel to avoid corrosions at high temperatures. The first high-temperature measurements were undertaken using the LiCl-KCl eutectic salt, and were successfully tested up to 650°C. The results obtained are shown in Figure MSR-4 below, which indicates high reliability of the method. The experimental set up is designed such that it provides data for both fluoride and chloride based MSR fuels. The figure also shows the roadmap of the densitometer development, indicating very rapid development. In early 2021, further tests are planned with a few more fluoride and chloride inactive salt mixtures, with successive installation of the final design in a hot glove box for measurement of actinide containing fuels.

Further development of the Joint Research Centre Molten Salt Database (JRCMSD) continued with the addition of BeF2-ZrF4, KF-ThF4 binary systems. With increasing worldwide demand for the use of chloride salts as fuels for certain MSR designs, the database is being extended to relevant chloride systems.

Research Centre Řež:

In addition to the activities carried out within the SAMOSAFER project, the Research Centre Řež, together with other Czech companies, has also continued the national MSR technology development program. The program is focused on the theoretical and experimental development of selected areas of MSR technology. In 2020, the main task of the program was to fully finalize all preparatory phases and stages for the measurement of neutronic characteristics of the molten fluoride salt, FLIBE, in the working temperature range of MSRs using the method of "hot inserted FLIBE zone" in the LR-0 experimental reactor. For this purpose, the core of the LR-0 reactor was completely reconstructed so that a set of the "hot inserted FLIBE zones" could be placed at its centre. Before the end of 2020, a set of inserted zones was completed, the location in the LR-0 reactor was verified and all so-called "cold" non-active experiments were successfully completed. The inserted zone occupies the space of seven fuel assemblies in the middle of the LR-0 reactor core. Hot experiments-measurements of the neutronic characteristics of the FLIBE melt in the range of MSR operating temperatures (550-750°C) will start in the first half of 2021.

In addition to hot inserted zone experiments, the R&D program in the area of materials research, development and verification of components and equipment for fluoride melt media, and the study of electro-chemical separation methods, along with methods of fused salt volatilization suitable for online MSR liquid fuel reprocessing technology, also continued.

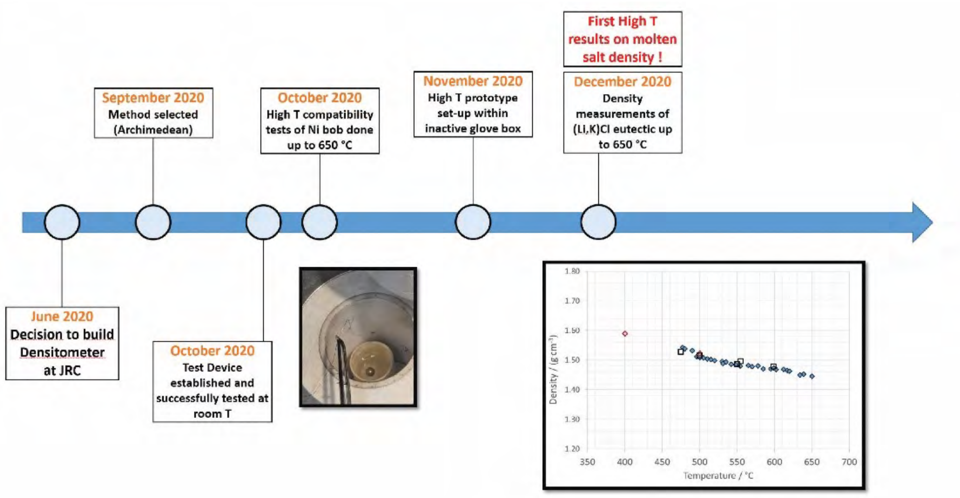
France

Since the beginning of 2020, the CEA has been carrying out its a research program oriented around defining a sketch of a molten salt reactor. Three options are being considered, all in a fast spectrum and in molten chloride: isogenerator, Pu burner, and minor-actinide transmutor.

Studies cover the reactor system (i.e. neutronics, materials, components) and the associated fuel cycle (i.e. salt behavior, corrosion, salt polishing). Multi-physics and chemistry modelling and simulation are also part of the scope. This program, involving the three research institutes of the CEA (IRESNE at Cadarache, ISEC at Marcoule and ISAS at Saclay[1]), is carried out in collaboration with the CNRS (Grenoble, Orsay), with the support of Orano. JRC Karlsruhe is also contributing.

In 2020, work focused on the definition of the plutonium burner option with the study of different

Figure MSR-4. Roadmap of densitometer development at JRC for measurement of MSR fuels



Note: the graph first shows the successful results on the (li,K)Cl eutectic salt – blue: data obtained during heating; red: data obtained during successive cooling; black: data obtained during repeated measurement.

1. IRESNE : Institut de recherche sur les systèmes nucléaires pour la production d'énergie bas carbone; ISEC : Institut pour les Sciences et Technologies pour une Économie Circulaire des Énergies bas carbone, ISAS : Institut des Sciences Appliquées et de la Simulation pour les énergies bas carbone.

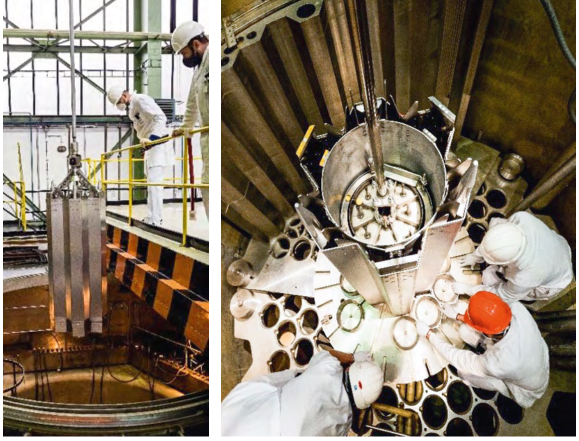
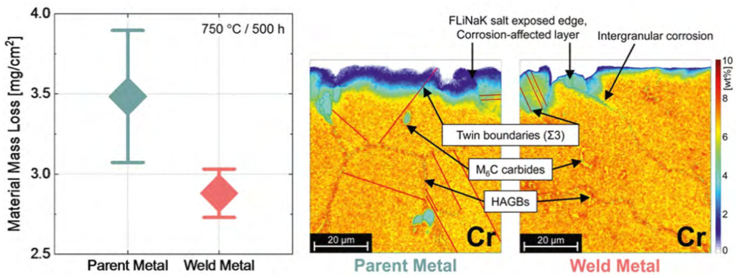


Figure MSR-5. Test insertion of the complete FLIBE insertion zone assembly into LR-0 and verification its location in the centre of the reactor core



Source: Danon, A. E. et al. (2020)

Figure MSR-6. Corrosion rate and microstructural differences in a GH3535 weldment a corrosion tested in FLiNaK molten salt at 750°C

concepts for the reactor, and for different salt composition. In parallel, a tool calculating the evolution of the composition of the salt under irradiation was built (MOSARELA). This tool will contribute to the definition of the reactor operation conditions and the salt treatment strategy.

Australia

The molten salt technology is becoming increasingly important in a wide range of low-carbon energy production and storage systems. Successful deployment requires the development and qualification of materials and components capable of withstanding the challenging operation conditions. Australia's Nuclear Science and Technology Organisation (ANSTO) continues to collaborate with GIF partners to understand corrosion in FLiNaK of candidate stainless steels and nickel-based alloys, as well as how advanced manufacturing techniques may be used to decrease their time to deployment in advanced reactors.

Highlights include a recent investigation of the corrosion performance of the welded Ni-Mo-Cr (GH3535) alloy, where it was shown that the weld, while having a very similar composition, had a superior corrosion resistance to the parent metal. The difference is attributed to the differences in microstructure and in particular, a significantly lower density of high-angle grain boundaries (HAGBs) in the weld metal and the large M6C carbides present in the parent metal.

Russia

During the year 2020, Rosatom continued to provide support to preliminary design development for: 1) test 10-megawatt thermal (MWt) Lithium, Beryllium, Actinides/Fluorine MSR with homogeneous core; and 2) its 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 long-lived actinide loadings, drain-out, shut down, etc.

Two main objectives of the MSR project for the period up to year 2024 include:

- development and demonstration of key technological solutions for a MSR with circulating fuel for the transmutation of long-lived actinides;
- development of a preliminary design for the test MSR and required materials to obtain a license for its placement.

During the year 2020, the main R&D efforts were focused on the following issues:

- optimization of neutron and the thermal-hydraulic characteristics of the core and fuel circuit;
- development of analytical methods to measure the impurities in the fuel salt and intermediate coolant;
- development of an advanced high nickel alloy with enhanced corrosion and radiation resistance properties for the fuel circuit;
- construction of experimental units for materials tests with fuel and coolant salts in laboratory and reactor conditions.

United States

The US government continues to foster US MSR industry development through a number of cost-sharing R&D programs. The US Department of Energy (DOE) in particular is supporting both university and national laboratory activities at a limited scale to overcome the remaining technical hurdles to MSR deployment.

In 2020, MIT received a DOE award to build an in-reactor molten salt test loop.[2] This facility will provide researchers with an understanding of by-products in an MSR and test instrumentation. It will also serve as a prototype for other university loop studies and DOE test reactors. The research will use both a non-irradiated and irradiated flowing salt loop to examine the behavior of fission by-products, especially ones that do not stay dissolved in the salt. These by-products in particular will deposit themselves on surfaces

2. https://nrl.mit.edu/announcements/2020/nrl-receives-four-doe-project-awards.

within the loop, or separate from the liquid salt as gaseous by-products. Understanding how these by-products affect the loops will give valuable insights into the MSR design. Natura Resources LLC also granted Abilene Christian University (ACU) USD 21.5 million[3] over the next three years as part of a USD 30.5 million effort to design and license a research reactor in collaboration with three major universities: Georgia Institute of Technology, Texas A&M University, and The University of Texas at Austin. Launched in spring 2019, the consortium's goal is to design, license and commission the first university-based molten salt research reactor, which ACU will host and own. The deal represents the largest sponsored research agreement in the university's history. Recent national laboratory activities related to MSRs include developing a molten salt thermal properties database based on molten salt thermo-physical and thermo-chemical property measurement and engineering evaluation of off-gas system technology.

In parallel, the Nuclear Regulatory Commission (NRC) is making progress on the process of modernizing its licensing requirements to better reflect the safety characteristics of advanced reactors. MSR features and phenomena are being incorporated into accident progression evaluation tools. Progress in being made in developing methodologies for qualifying liquid fuel salts, as well as non-power reactor review guidance.

Vendors such as Kairos Power and Terrestrial Energy USA have also filed multiple topical reports and white papers to the NRC in 2020 in order to support the licensing process of their MSR concepts.

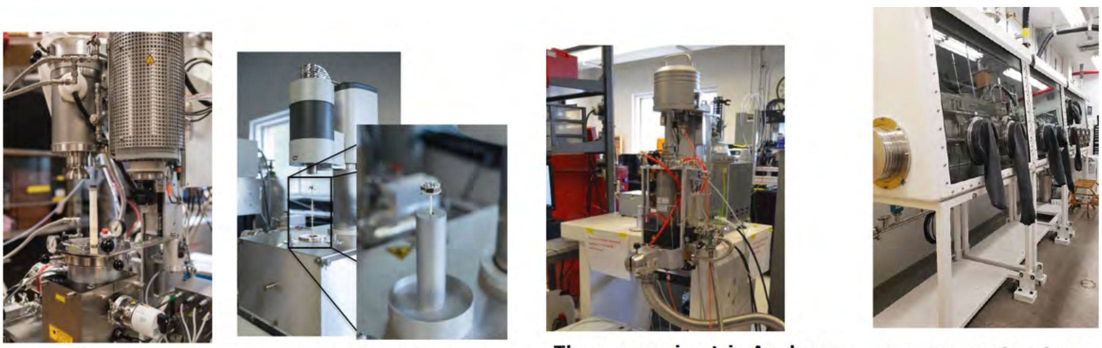
Canada

In 2020, Canadian Nuclear Laboratories (CNL) continued to develop expertise and capabilities in support of SMRs. The CNL executed multiple projects for SMR vendors under a new cost-sharing R&D program called the Canadian Nuclear Research Initiative (CNRI). The CNRI program was established by the CNL to accelerate the deployment of SMRs in Canada, enabling research and development and connecting the SMR industry with 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 expertise so as to help advance the commercialization of SMR technologies. Among the first to take part in this new program, three MSR vendors worked with the CNL on a diverse program of work, including electro-chemical separation methods, tritium management, reactor physics, thermal-hydraulics and safeguards studies.

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

- development of actinide molten salt fuel synthesis using no gaseous reagents;
- 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 the corrosion of structural materials;
- modelling and simulation of MSR designs, including evaluation of the codes for an 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.

Figure MSR-7. Sample encapsulation and measurement technique development at CNL



Laser Flash (LFA)
Thermal diffusivity

Differential Scanning Calorimeter (DSC)
- Liquidus/solidus temperatures
- Specific heat capacity
- Phase diagrams

Thermogravimetric Analyser (TGA/STA)
Mass change with temperature to verify thermal stability of molten salt mixtures

Dry, Inert Gloveboxes
Commissioned and authorized to be used for preparation of salt fuels, including plutonium-bearing salts

3. https://techtransfercentral.com/2020/08/04/abilene-christian-u-with-21-million-from-natura-resources-to-build-research-reactor-as-part-of-consortium/.

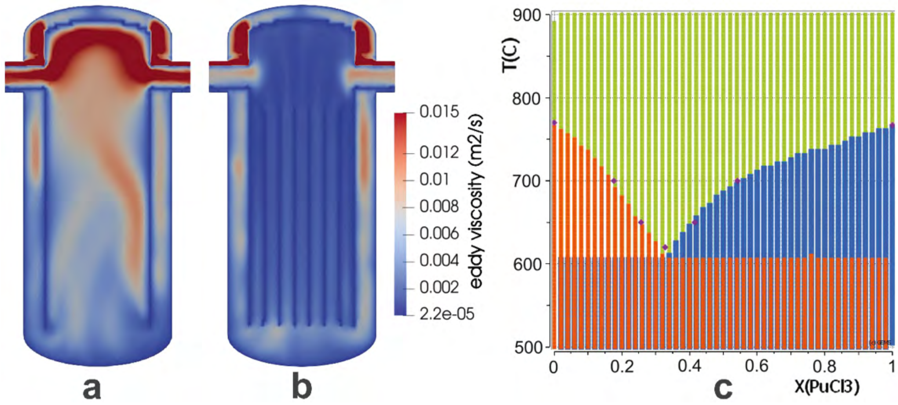


Figure MSR-8. Conceptual layout of the MCFR core vessel a) without and b) with baffles. Phase diagram of the ThCl-PuCl3 system c).

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

Terrestrial Energy: Terrestrial Energy Inc. (TEI) is a privately funded reactor vendor developing the integral MSR (IMSR). A 195 MWe, graphite-moderated design employing standard assay low-enriched uranium (LEU) for near- term deployment in the late 2020's. TEI is sourcing R&D services worldwide in order to develop its IMSR concept. In 2020, NRG was contracted for a major multi-year irradiation program in the Petten reactor of multiple commercial graphite grades suitable for MSR use (sub-micron pore diameter). The first specimens were introduced in the reactors in October 2020 with similar protocols to the INNOGRAPH study for gas-cooled graphite candidates. Argonne National Laboratory will provide services for various fuel salt property verifications with the high standards needed for the regulator process.

In terms of commercialization milestones achieved in 2020, the IMSR concept has been short-listed by Ontario Power Generation as one of three technologies developed for potential commercialization, after extensive commercial due diligence. TEI has also been awarded CAD 20 million from the Canadian federal government's Strategic Innovation Fund.

Switzerland

Swiss MSR research continued in 2020 at the Paul Scherrer Institute (PSI) with a major aim of monitoring the technology, education of new experts, and development of knowledge and simulation capabilities in: fuel cycle, system behavior, and thermo-dynamics areas of molten salt research. A big part of PSI activities represent contributions to the EU Horizon 2020 project, SAMOSAFER, and therefore these belong to the EU progress report.

In the area of fuel cycle assessment, the PSI continued to develop a dedicated benchmark for the respective simulation tools with partners from the SAMOSAFER project. The breed-and-burn fuel cycle in the molten chloride fast reactor (MCFR) was further assessed, and the Serpent 2-based procedure, EQL0D, was applied to several additional fuel cycle configurations. The in-depth knowledge of this reactor and fuel cycle type, together with past Swiss research in this area, was used for preparation of an MCFR chapter for Elsevier's *Encyclopedia of Nuclear Energy* (to be published in 2021). For the same encyclopedia, a series of three chapters dedicated to self-sustained breeding in advanced reactors was prepared based on the extensive knowledge of actinide behavior during irradiation in numerous MSR concepts and other advanced reactors.

The system behavior study with Open-FOAM based solver, ATARI, continued in 2020 covering three different aspects: 1) simulation of the SAMOSAFER project reference concept MSFR; 2) assessment of freezing phenomena in printed circuit heat exchangers; and 3) conceptual design of an MCFR core with tube-in-tube and baffles options. The impact of baffles on eddy viscosity is illustrated in Figure MSR-8 a and b.

The thermo-dynamics simulation of molten salts continued with the GEMS TM code, focusing on further refinement of the database and major fluoride and chloride salt components. The adjusted database was applied to phase diagram calculations (see Figure MSR-8 c) and to the cGEMS code for the estimation of evaporation behavior.

China

In 2020, the Shanghai Institute of Applied Physics and the Chinese Academy of Sciences (SINAP-CAS) have been steadily promoting the related work of the thorium molten salt reactor (TMSR). This 2 MWth molten salt test reactor (TMSR-LF1) was approved with a construction license. Construction of the plant structure for the experimental reactor was started and completed in 2020, and the equipment was delivered for installation. The application for an operation license (including the final safety analysis report [FSAR] and other relevant attachments) was submitted, and the first stage review was completed. Key equipment has entered the final stage of manufacturing and will be delivered to the project site in succession. At present, installation of the main equipment has started.

A number of experiments have been completed on the scaled experimental device (TMSR-SF0), including: the thermal-hydraulic performance experiment of key equipment and the steady-state and transient characteristic experiment on the salt system. The experimental program will continue in 2021.

The conceptual design of the flowsheet for TMSR fuels has been finished, and validation of the flowsheet for thorium fuel reprocessing is in progress. Fundamental studies on the structure and reaction of actinide and fission product fluorides in molten salt have been carried out.

Significant progress has been achieved on MSR material research. It was proven that GH3535 alloys maintain good creep properties in FLiNaK molten salts. Experiments on molten salt erosion of nuclear graphite at elevated temperatures were carried out, which provided data support for the further application of nuclear graphite in MSRs. In addition, the neutron-radiation-induced defect evolution of nickel-based alloy has been studied using the newly developed rate theory method.

Japan

In Japan, the International Thorium Molten Salt Forum (ITMSF) was established in 2008 for the basic study of MSR technology, such as conceptual designs and safety analysis for MSR-FUJI. The ITMSF has been an observer in the GIF-MSR System Steering Committee from the beginning of the committee. In 2010, Thorium Tech Solution Inc. (TTS) was established for the business application of the MSR-FUJI. In addition to these activities, several universities have been carrying out basic studies in the recent decade.

The Japanese government began supporting the development of MSR technology in 2019, and continued to do so until the beginning of 2021. Three MSR companies were selected: two (the Thermal Transient Test Facility for Structures [TTS] and MOSTECH) are promoting MSRs with fluoride salt moderated by graphite, and one is promoting a fast spectrum MSR with chloride salt, on which universities (Tokyo Institute of Technology [TIT], Fukui, Doshisha) and the Central Research Institute of Electric Power Industry (CRIEPI) are working together.

Figure MSR-9. FLiNaK molten salt loop at the National Institute for Fusion Science

MOSTECH is planning to construct a molten salt loop at Kyushu University and is also preparing freeze valve tests for a fusion blanket loop system of molten salt, together with Kyushu University and the University of Electro Communications (UEC), as shown below. This loop system (Oroshhi-2: described by A. Sagara et al. in Fusion Science and Technology in 2015), was built in the National Institute for Fusion Science (NIFS) using FLiNaK.

Stéphane Bourg
Chair of the MSR SSC, with contributions from MSR members

Supercritical water reactor

Main characteristics of the system

The supercritical water-cooled reactor (SCWR) is a high-temperature, high-pressure, water-cooled reactor that operates above the thermo-dynamic critical point (374°C, 22.1 megapascals [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 with China, and a pressure-tube concept proposed by Canada. Apart from the specifics of the core design, these concepts have many similar features. The R&D needs for each reactor type are therefore common, enabling collaborative research to be pursued.

The main advantage of the SCWR is improved economics because of the high thermo-dynamic efficiency and the potential for plant simplification. Improvements in the areas of safety, sustainability, and proliferation resistance and physical protection are also possible, and are being pursued by considering several design options using thermal and fast spectra, including the use of advanced fuel cycles.

There are currently three Project Management Boards (PMBs) within the SCWR system: for system integration and assessment (provisional), materials and chemistry (MC), and thermal-hydraulics and safety (THS). The extension of the project arrangements to thermal-hydraulics and safety, as well as to MC, with project plans covering 2021-2025, are in progress and have been discussed during the steering committee and Project Management Board meetings held in early December 2020.

R&D objectives

The following critical-path R&D projects have been identified in the SCWR system research plan (SRP):

- System integration and assessment: definition of a reference design, based on the pressure tube and pressure-vessel concepts, which meets Gen-IV requirements (sustainability, economics, safe and reliable performance, proliferation resistance).
- Thermal-hydraulics and safety: gaps exist in the heat transfer and critical flow databases for the SCWR. Data at prototypical SCWR conditions are needed to validate 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.
- MC: 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 on an understanding of water radiolysis.

Main activities and outcomes

System integration and assessment

Because of the COVID-19 pandemic, and priority work on the THS and MC PMBs, no GIF SCWR activities were undertaken in this field in 2020.

The Joint European Canadian Chinese Development of Small Modular Reactor Technology (ECC-SMART) project was launched in September 2020. ECC-SMART is a collaborative project covering most GIF SCWR research fields. ECC-SMART is oriented towards assessing the feasibility and identification of safety features of an intrinsically and passively safe SMR cooled by supercritical water (SCW-SMR). The project takes into account specific knowledge gaps related to the future licensing process and the implementation of this technology. The main objectives of the project are to define the design requirements for the future SCW-SMR technology, to develop a pre-licensing study and guidelines for demonstration of safety in the further development stages of the SCW-SMR concept, including the methodologies and tools to be used, and to identify key obstacles for future SMR licensing and a strategy for this process.

To reach these objectives, specific technical know-ledge gaps were defined and will be assessed to better achieve future licensing and implementation of the SCW-SMR technology, particularly in terms of the behavior and irradiation of materials in the SCW environment, and validation of the codes and design of the reactor core, evaluated through simulations and experimentally validated.

The ECC-SMART project consortium consists of the EU, and Canadian and Chinese partners, who are making use of trans-continental synergy and knowledge developed separately by each partner, as well as under the GIF umbrella. The project consortium and scope were created according to joint research activities at the IAEA and at GIF, and as much data as possible will be taken from projects already performed. ECC-SMART brings together the best scientific teams working in the field of SCWRs, using the best facilities and methods worldwide, to fulfil the common vision of building an SCW-SMR in the near future.

For China, two projects supported by the Ministry of Science and Technology of China (MOST) were started in 2020. One is the GIF SCWR THS and the other is the GIF SCWR MC. The main goals of the two projects are to improve the China CSR1000 design and finish the international review before the end of 2022 so as to compile an expanded database based on previous research results. A kick-off meeting was held in 2020 in the Nuclear power Institute of China (NPIC), Chengdu, China. Five Chinese institutes participated in the two projects, including the NPIC, Shanghai Jiaotong University (SJTU), Xi'an Jiaotong University (XJTU), the China Institute of Atomic Energy (CIAE) and the University of Science and Technology Beijing (USTB). Two virtual meetings were held online in March 2020 and September 2020 as a result of the COVID-19 situation.

Thermal-hydraulics and safety

Euratom activities

The ECC-SMART project was launched in September 2020, and it comprises several different work packages (WP). WP 3 in particular focuses on thermo-hydraulics and safety analyses, and the current task is to define a common design that is to be analyzed.

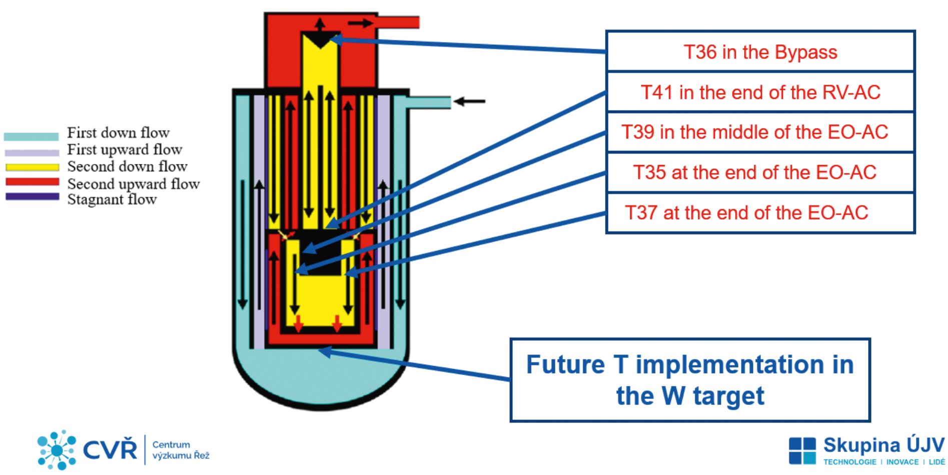
In Hungary, two institutions are working in close collaboration on THS research for SCWRs: the Centre for Energy Research (EK) and the Budapest University of Technology and Economics (BME). The EK continued its experimental activities (not only on heat transfer) in collaboration with the Department of Energy Engineering (DEE), the Department of Chemical and Environmental Process Engineering (DCEPE) and the Institute of Nuclear Techniques (NTI) of the BME. Two experimental works have been proposed in 2020 and may be elaborated further in the near future. The BME DEE has been working on theoretical research of water chemistry and thermal-hydraulic issues related to SCWs during the 2020 year. The BME NTI continued its numerical and theoretical research on the thermal-hydraulics of SCWs.

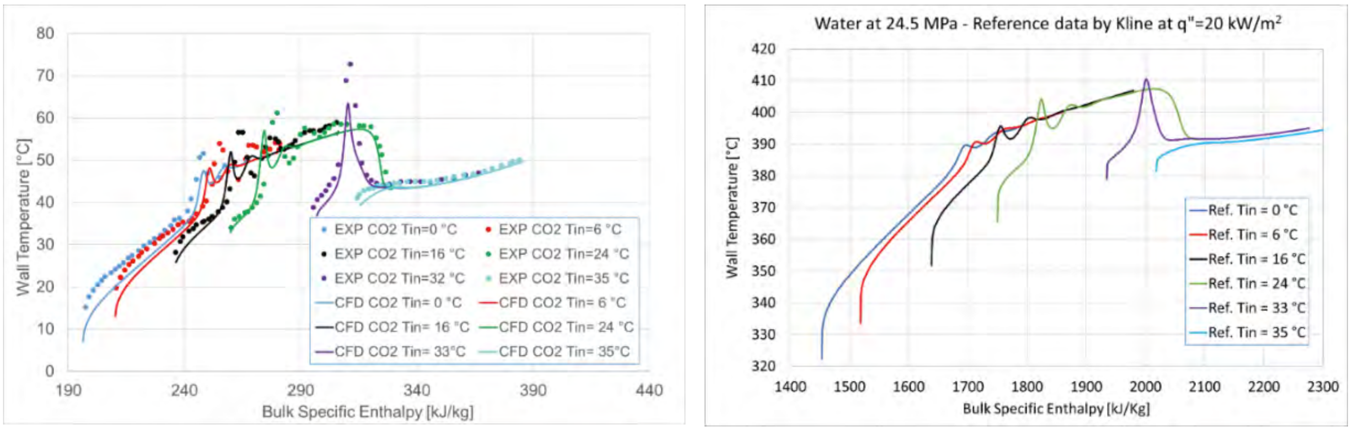
The main activity of the Research Centre Řež focused on the second licensing phase in order to insert the supercritical water loop (SCWL) in the LVR-15 reactor (Musa et al., 2020). This advanced facility is monitored by the Czech Republic State Office for Nuclear Safety (SONS). New analyses were performed with the goal of providing boundary conditions for the assessment of stress and strain calculations. The analyses were performed in ATHLET 3.1 code. After the initial revision of the flow regimes, all scenarios were reconsidered as a result of the lowering of the SCWL operational pressure from 25 MPa to 24 MPa. In addition, a new criterion for a new SCRAM signal will be implemented. The design will be improved by inserting a thermocouple in one of the two tungsten targets (T SCRAM around 550°C). This thermocouple could provide additional information on the heat transfer during transient phase.

Additional activities focused on simulating the SCWR thermo-hydraulic behavior. ATHLET code was benchmarked with the experimental data from the out-of-pile configuration. Among the postulated scenarios, an abnormal sequence (labelled A2 – loss of power in the loop) was analyzed. This scenario is similar to the postulated in-pile A2. The analyzed correlation in this phase were performed by Gupta et al. and Mokry et al.

In the past years, the University of Pisa in Italy has been addressing by a RANS model CO_2 data, water data and other data produced by several researchers, making use of an algebraic heat flux model (AHFM) developed in the STAR-CCM+ code, on the basis of the Lien et al. (1996) model. The RANS model was assessed and improved on a variety of experimental data, obtaining good results in comparison with experimental data. Based on these results and on data provided through direct numerical simulation (DNS) studies by a group at the University of Sheffield, in 2020 the subject of a fluid-to-fluid similarity theory for heat transfer at supercritical pressure (already proposed in past years) was further developed. Very good results were obtained as a result of these new steps, making it possible to confirm a sound rationale for assessing the scalability of results obtained with different fluids. The results of this work, performed at a distance during the current pandemic, have been published in three papers produced in 2020 and 2021 (Pucciarelli et al., 2020; Pucciarelli and Ambrosini, 2020; Kassem et al., 2021). Figure SCWR-2 provides comparisons between the original experimental data by Kline

Figure SCWR-1. Active channel thermocouples map





Note: as predicted by the similarity theory (data by Kline: CO_2, 8.35 MPa, 4.6 mm ID, q"=20 kW/m2, G=300 kg/m2s, upward flow, different inlet temperatures).

Figure SCWR-2. Comparison between experimental and data (right) and corresponding similar water trends (left)

obtained with CO_2, with the corresponding trends predicted in fluid similarity in the case in which water would be used instead, providing a clear idea of corresponding trends that can guide experimentalists in planning their experiments with simulant fluids.

SCWR research at the University of Sheffield (USFD) focuses on high-fidelity numerical simulations using DNS to produce high-quality reliable data in order to complement physical experiments. These tend to be for lower Reynolds numbers, but are able to produce detailed information and data to help understand the physics and the development of practical engineering models. The USFD has developed a versatile DNS code, CHAPSim, which has now been selected by the UK Collaborative Computational Project for Nuclear Thermal Hydraulics – supporting next generation civil nuclear reactors (CCP NTH) (sponsored by EPSRC EP/T026685/1 2020-2025) to be developed as a UK NTH community code. Over 2020, USFD work has included: 1) implementing conjugate heat transfer in CHPASim and carrying out some preliminary simulations; 2) carrying out simulations of flows in a horizontal orientation; and 3) investigating the development of a unified approach to explain the mechanisms of flow laminarization and heat transfer deterioration in a heated vertical flow of supercritical fluid. The USFD has also started working on simulations of SuperCritical fluid flows over rough/corroded surfaces in the context of the ECC-SMART project.

Activities in China

Two supercritical water thermal-hydraulics benchmarks were released in November 2020 in China. One is on the 2X2 bundle SC-water tests performed by the NPIC and the XJTU several years ago. The other is on parallel channel instability experiments of supercritical water. The structure layouts are shown in Figure SCWR-3. The 2X2 rod bundle tests are used to validate the computational fluid dynamics (CFD) tools (e.g. CFX, Fluent, Star CCM+, Open-FOAM) and subchannel tools (e.g. ATHAS, SC-COBRA). The parallel channel instability experiments were used to validate the system analysis tools, such as SC-TRAN, RELAP5 and APROS. The first-round comparisons are planned for 2021.

Experiments on heat transfer of supercritical water in a single subchannel with grid spacer were successfully carried out. More reference characteristics about the influence mechanism of the grid spacer can be observed through this experiment in Figure SCWR-4. The section structure of the test part is clearly shown in in Figure SCWR-3. The test section is a triangular-shaped sub-channel with a standard grid spacer of an 8 mm core rod diameter, 1.4 pitch ratio, 990 mm total length and 2.5 mm thickness. The grid spacer was located 550 mm from the inlet. The influence of pressure, flow rate, and heat flux on the standard grid spacer within the framework of sub-channel flow heat

Figure SCWR-3. The structure layout of 2X2 rod bundles (left: from NPIC; right: from XJTU)

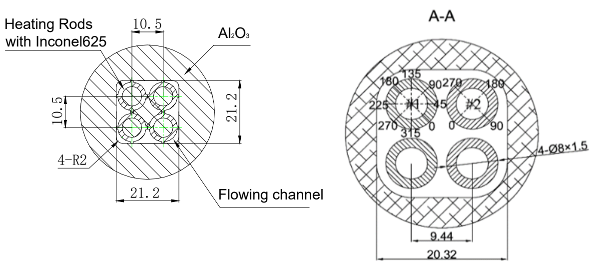
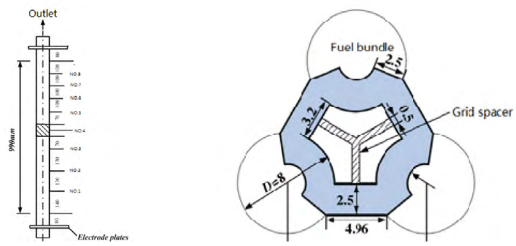


Figure SCWR-4. The structure layout of the sub-channel test section



transfer characteristics of supercritical water were studied. Figure SCWR-3 shows the distribution curve of the heat transfer coefficient, changed with the enthalpy in several working conditions. With the comparison of the five curves, the grid spacer greatly influenced the heat transfer characteristic of the entire flow, especially downstream. For the downstream flow direction, the fluid disturbance was reduced because of the decreasing blockage area. The circulation area abruptly widened in comparison to the grid spacer.

A new thermal amplification system was constructed to improve the accuracy of the calculated heat transfer coefficient of a supercritical fluid near its pseudo-critical point with the influence of the parameters of supercritical water cooling via downward flow inside a tube accompanied by pool boiling outside of the tube (McLellan et al., 2021). The results of this study will be helpful in understanding the effects of pressure, fluid temperature, mass flux and heat flux on the characteristics of supercritical downward cooling heat transfer. The test section is shown in Figure SCWR-6. A type 316 stainless steel circular tube was used as the test section. Its inner diameter and wall thickness were 20 mm and 2.5 mm, respectively. Five-sixths of the test section was directly immersed in the water of test pool. Figure SCWR-6 presents the variations in the wall temperature, heat flux and heat transfer coefficient with respect to the fluid temperature. The wall temperature gradually increases with fluid temperature below and above the pseudo-critical point. At a pseudo-critical point (384.9°C), a sharp jump appears in the wall temperature, in which it increases from 253.5°C to 317.4°C with an increase in fluid temperature from 375.7°C to 383°C. The heat flux increases with fluid temperature below the pseudo-critical point, and is almost maintained at a constant value above the pseudo-critical point, but decreases sharply near the pseudo-critical point itself. The heat transfer coefficient increases and then decreases with fluid temperature, reaching its maximum at 14 kW·m−2·K at the pseudo-critical point.

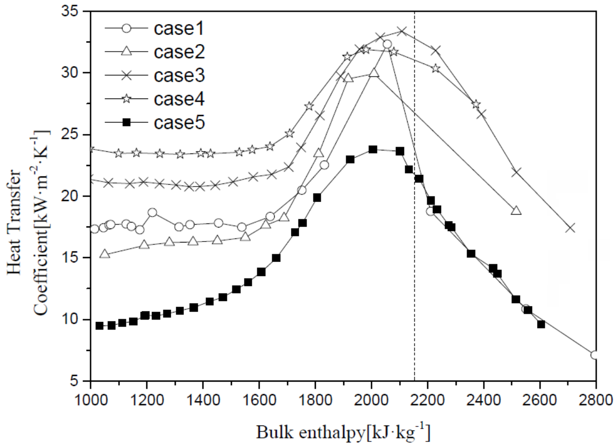
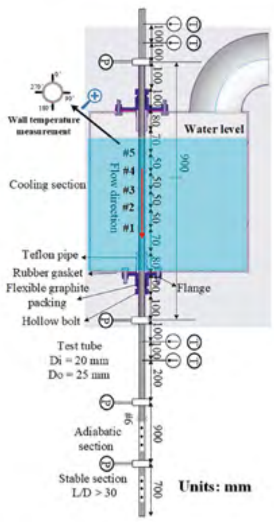


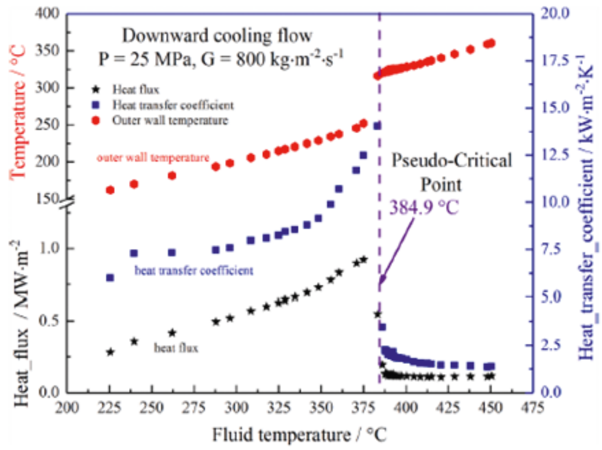
Figure SCWR-5. The distribution curve of heat transfer coefficient under supercritical conditions

Activities in Canada

In Canada, a preliminary SCW-SMR concept has been established from the reference Canadian SCWR concept. The overall core diameter is about 3 meters, which has resulted in a slender reactor core, inefficient from both neutronics and thermal hydraulics points of view. The selection and optimization of the SCW-SMR fuel assembly uses knowledge gained from the development of Canadian fuel bundles. It considers geometrical features (such as heated perimeter, flow area, sub-channel size) to minimize the maximum cladding temperature (MCT). From a thermal-hydraulics and reactor physics coupled analysis, the study focused on computing the reactor power distribution and the maximum channel power. Based on these results, a thermal-hydraulics analysis using the subchannel code ASSERT-PV was performed. The analysis focused on the maximum cladding temperature. Several concepts have been proposed, however, based on the results, and two concepts are being further investigated, namely: 1) THE CANFLEX-20 fuel bundle; and 2) the 64-element concept used for the Canadian SCWR concept. The assessment of these fuel bundles is ongoing and the results are still preliminary. Moreover, a proper and complete development requires linking to operational and

Figure SCWR-6. The downward flow test section (left) and the general behaviors (right)





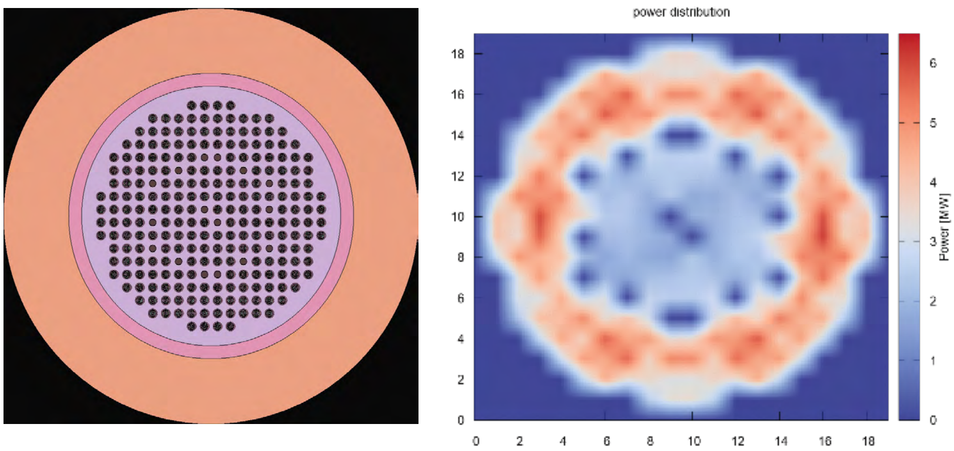


Figure SCWR-7. (a) Preliminary Canadian supercritical water-cooled SMR (core layout), (b) channel power heat map (in MW) of the preliminary Canadian supercritical water-cooled SMR

safety constraints and requirements, such as higher burn-up, maximum linear power, and higher-reliability or reduced operational and maintenance costs. These analyses are ongoing.

The CNL gravity assisted loop uses a heat pipe to remove heat from a pool of water. The experimental loop was originally designed to be used to remove heat from the spent-fuel pools. A study to examine the detailed behavior of this gravity assisted loop was recently undertaken. The loop was verified and tested to ensure it is fully operational, and a series of tests were conducted to provide experimental data that can be compared to a computer model. The objective was to further analyze the loop and to dimension components that could be used to design and assess a passive cooling system that could be included in the Canadian SCW-SMR. The system was modelled using CATHENA and RELAP5-3D codes. The investigation revealed unexpected effects in the steam piping between the evaporator and condenser due to the use of a Coriolis mass flow metre, which indirectly limited heat transfer. Additionally, the presence of non-condensable gases in the system further reduced heat transfer. The mathematical models were refined to include these factors, resulting in good agreement with experimental data. Further, it was demonstrated that changes to the loop, such as increasing the diameter of the condensate return line and relocating the mass flow metre are needed. Simulation results also provide guidance for the next phase of investigation, including the addition of a steam-to-air heat exchanger.

Materials and chemistry

Euratom activities

A European Union funded project called Mitigating Environmentally-Assisted Cracking Through Optimisation of Surface Condition (MEACTOS) is ongoing to study the effect of surface finishing on the corrosion resistance of selected alloys (A182 and stainless steel 316 L). In this project, one task involves SCW being used as an aggressive environment because of the higher test temperatures. Several laboratories from the EU are involved in these SCW activities (e.g. JRC, Valtion Teknillinen Tutkimuskeskus [VTT] and CIEMAT).

In the context of the ECC-SMART project, work package 2 is focusing on materials testing with more than 400 person months and the participation of 11 laboratories, among which 8 European laboratories/companies[1], one from Canada (CNL) and two from China (University of Science and Technology Beijing [USTB] and Shanghai Jiao Tong University [SJTU]). In this work package, the corrosion behavior in SCW of two of the most promising materials for cladding applications (i.e. 800 H and 310 S) will be studied.

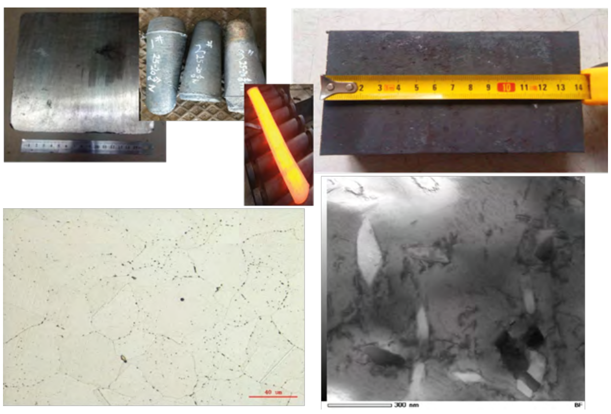
JRC Petten, in collaboration with VSCHT, has published a paper in the Corrosion Science Journal entitled: "In-situ electro-chemical impedance measurement of corroding stainless steel in high subcritical and supercritical water". This work summarized the main findings from a study on changes in the physico-chemical properties of SCW with pressure and temperature. This work could be considered the starting point of the tests to be carried out in the ECC-SMART project on this topic.

CIEMAT finished a preliminary work in collaboration with the Research Centre Řež. Selected, in-situ tensile tests with a nickel-based alloy 690 pretested in SCW were performed in this work. Both laboratories plan to continue these tests throughout 2021, and for this reason CIEMAT has designed and machined new specimens that fit into the scanning electron microscope available at the Research Centre Řež Pilsen, where the previous tests were performed. It is expected that this new configuration of specimens will allow a more in-depth study of the role of microstructural defects in the corrosion behavior of Ni base alloys. Preliminary results from these tests were presented at the Electric Power Research Institute (EPRI) annual meeting on alloy 690 (Tampa, December 2019).

1. CIEMAT, Research Centre Řež, JRC, Regia Autonoma Tehnologii Pentru Energia Nucleara [RATEN], Slovak University of Technology in Bratislava (STU), VTT, the University of Prague (VSCHT), ENEN (European Nuclear Education Network)

Research activities at the Research Centre Řež are linked to previous years, focusing on two more exposures in the SCWL, commissioning and test operation of the ultracritical water loop (UCWL) and publishing of data from exposures in 2018-2020. Effects of supercritical water on the corrosion behavior was studied on perspective materials for nuclear power plants, such as alloy 800H, AISI 310S, AISI 321SS (08Cr18Ni10Ti), T505 (T91), Inconel 718, NIMONIC 91 and NITRONIC 60. Two exposures in the SCWL were carried out at 395°C, 25 MPa, each for 1 000 hours. The supercritical medium was deoxygenated water with pH 5.5-6.8, conductivity 0.77-1.88 µS/cm, Fe <92 µg/l and TOC 886 to 251 µg/l. Commissioning of the UCWL took place during the year 2020 and the test operation at 600°C, 25 MPa, with 500 hours running since November 2020. The effects of SCW were evaluated by weight changes, scanning electron microscopy (SEM) with chemical analysis detectors (BSE, EDS, EBSD) in combination with X-ray diffraction. In addition, the special analysis to detect the very thin surface layer by focused ion beam (FIB) was used on alloy 800H. In this case, no thin surface layer was observed. Only randomly distributed spinels were observed: trevorite ($NiFe_2O_4$, a = 8.347 Å), magnetite ($FeFe_2O_4$, a = 8.397 Å) and chromite ($FeCr_2O_4$, a = 8.376 Å) crystals with dimensions up to 1 µm on 800H and only magnetite crystals on 08Cr18Ni10Ti after three exposures – 1 700 hours. The density of all crystals increased slightly after the second and third exposure. Simultaneously, the surface of Inconel was irregularly covered by oxide crystals about 1 µm, identified as hercynite and chromite $Fe(Al,Cr)_2O_4$ after one exposure – 1 000 hours. Other materials will be analyzed in 2021. Selected materials, such as 800H, AISI 321SS and T91, were exposed several times, for 2 700 hours (four exposures) in total. The commissioning of new facilities such as in-pile and out-of-pile autoclaves (volume 137 ml, 600°C/25 MPa and volume 850 ml, 700°C/30 MPa) are in progress. The effect of radiation on microstructure stability and corrosion resistance of candidate materials exposed to SCW will thus be possible to assess.

Figure SCWR-8. AFA steels developed at the University of Science and Technology Beijing (China)



Activities in China

The SJTU is studying the corrosion behavior of alloy 800H, austenitic stainless steel 310S and alumina forming austenitic (AFA) alloys in SCW. Moreover, particular interest is being taken in the effects of variables such as temperature, plastic deformation, water chemistry and surface finishing in the corrosion behavior of these alloys, and for this reason the SJTU is performing tests to study the effects of these variables on general corrosion and stress corrosion cracking processes in SCW. In addition, the SJTU is leading the third international round robin on corrosion behavior of candidate alloys for the SCWR. In this case, the international group will focus their efforts on the study of stress corrosion cracking processes in the SCW. Part of these activities will be complementary to the ECC-SMART project.

The USTB is designing and fabricating new grade materials suitable for fuel-cladding application under a high-temperature SCWR environment. These materials will be co-evaluated by colleagues at SJTU and the NPIC to determine the candidate materials for further round robin tests. The composition design of new grade materials is mainly based on 310SS, including oxide dispersion-strengthened (ODS) steel type fabricated through the powder metallurgy technique and AFA steel type through a traditional melting and casting method. Both types of new grade materials show promising high-temperature strength and SCW corrosion resistance. ODS steels show much better microstructure stability at high temperature, while AFA steels are attractive in terms of engineering and manufacturing. AFA steels show good performances in the aspect of high-temperature strength as compared with similar traditional steels because of the formation of strengthening phases of NbC, Laves and B2-NiAl, which are more stable than M23C6 in traditional steels. It can be expected that the AFA steels will also show better corrosion resistance in SCW environments as a result of the formation of alumina surface oxide, which is dense, thin and stable, as approved by exposure tests in SCW. Figure SCWR-8 shows the pictures of ingots, microstructure and precipitates of a fabricated AFA steel.

Activities in Canada

The CNL has studied the corrosion behavior in SCW of alloy 625 and alloy 800H. Both have showed an excellent strength and ductility after welding. In addition, Cr-coated Zr-2.5Nb, Zr-1.2Cr-0.1Fe, Ti and Ti-6Al-4V met performance criteria in short-term tests. Moreover, they have developed a schedule for proton and heavy ion (Cr3+) irradiation up to 5-15 dpa in top 20-30 µm, followed by micro-mechanical testing. It is expected that the irradiation will start during the year 2021.

Engineering, structural and core metallic nuclear components of NPPs must handle thermal loading; otherwise, localized hotspots resulting from changes in geometry or heat transfer fouling could develop within the column and degrade the performance of components during operation and over time, such as that of fuel-cladding tubes. To further progress in this area, a project was established at the CNL with the objective of determining the thermal properties data (thermal conductivity) of candidate cladding tube material of small modular SCWRs, and to assist thermal-hydraulics calculations.

Chromium coated Zr-2.5% Nb material, such as a cladding material, showed a better resistance to corrosion in supercritical water conditions. In the open literature, empirical or semi-empirical models are available to assess the thermal conductivity of zirconium based fuel-cladding materials, but there is limited or no data on thermal conductivity of Cr-coated and/or oxided zirconium based fuel-cladding materials is lacking or not available to support thermal hydraulics modelling. Thermal conductivity measurements using the laser flash method were performed on the as-received material (Zr-2.5% Nb) at different test temperatures. The specimens are disc shaped with 12.16 mm in diameter, and an average thickness of 1.364 mm was cut from Zr-2.5% Nb. At each temperature, measurements were repeated five times to obtain an average. In the current study, thermal diffusivity and conductivity of baseline material is determined.

The CNL has an ongoing R&D program to support the development of a scaled-down 300 MWe version of the Canadian supercritical water reactor (SCWR) concept. The 300 MWe and 170-channel reactor core concept uses LEU fuel and features a maximum cladding temperature of 500°C (McLellan et al., 2021). There are challenges to using zirconium alloys at temperatures exceeding 400°C. Zirconium alloys such as Zr-2 and Zr-4 typically experience high corrosion rates, and they are known to experience hydrogen embrittlement from aggressive hydrogen pickup during corrosion.

Two materials from the previous experimental campaign – Zr-1.2Cr-0.1Fe (R60804) and Zr-2.5Nb (R60901) –were also used in the campaign described here. A nominal thickness of 5 to 10 µm of chromium coating was tested for about 150 hours of exposure time in oxygenated SCW. The oxidizing environment was chosen to simulate water radiolysis in the SCWR core. In addition, the corrosion behavior of candidate materials in an alkaline environment using LiOH solution was also evaluated.

Microstructural analyses, including scanning electron microscopy (SEM) and energy-dispersive X-ray spectroscopy (EDX) were performed to observe the effects of the microstructure of the base alloys on the observed chromium coating. The results from short-term autoclave oxidation at supercritical water conditions show that when the coating reaches approximately 10 µm thickness, the grain orientation of base Zr- and Ti-based alloys does not affect the morphology of the chromium coating. Moreover, weight gain measurements indicate a significant improvement in corrosion resistance of coated coupons compared to the as-received alloys, for Zr-2.5Nb and Zr-1.2Cr-0.1Fe. Long exposure experiments are ongoing.

Yanping Huang
Chair of the SCWR SSC, with contributions from SCWR members

Sodium-cooled fast reactor

Main characteristics of the system

The primary mission of the sodium-cooled fast reactor (SFR) is the effective management of high-level waste and uranium resources. If innovations to reduce capital cost and improve efficiency can be realized, the Gen-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 closed fuel cycle significantly improves the use 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 the 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 used, and a large margin to coolant boiling at low pressure can be maintained. The reactor unit can be arranged in a pool layout or a compact loop layout. The typical design parameters of the SFR concept being developed in the framework of the Gen-IV system arrangement (SA) are summarized in Table SFR-1. Plant sizes ranging from small modular systems to large monolithic reactors are being considered.

Table SFR-1. Typical design parameters for the Gen-IV SFR

Reactor parameters	Reference value
Outlet temperature	500-550°C
Pressure	~1 atmosphere
Power rating	30-5 000 MWt (10-2 000 MWe)
Fuel	Oxide, metal alloy, and others
Cladding	Ferritic-martensitic, ODS, and 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. For example, 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 United States, as well as 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 the general classes of SFR design concepts have been identified: loop configuration, pool configuration and SMRs. Furthermore, within this structure several design tracks that vary in size, key features (e.g. fuel type) and safety approaches have been identified with pre-conceptual design contributions by Gen-IV SFR members: CFR1200 (China), the Japanese sodium-cooled fast reactor (JSFR, Japan), Korea advanced liquid metal reactor (KALIMER, Korea), ESFR (Euratom), BN-1200 (Russia) and AFR-100 (United States). Gen-IV SFR design tracks incorporate significant technology innovations to reduce SFR capital costs through a combination of configuration simplicity, advanced fuels and materials and refined safety systems. They are thus used to guide and assess Gen-IV SFR R&D collaborations.

Status of cooperation

The system arrangement for Gen-IV international R&D collaboration on 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 the United Kingdom was welcomed to the system arrangement in 2019. The present signatories are: the French Alternative Energies and Atomic Energy Commission (CEA), France; the Department of Energy, United States; the Joint Research Centre, Euratom; the Japan Atomic Energy Agency, Japan; the Ministry of Science and Information and Communication and Technology (ICT), Korea; the China National Nuclear Corporation, China; Rosatom, Russia; the Department for Business, Energy and Industrial Strategy, the United Kingdom.

Based on international R&D plans, Gen-IV SFR research activities are arranged by SFR signatories into four technical projects: system integration and assessment (SIA), safety and operations (SO), advanced fuel (AF) and component design and balance of plant (CD&BOP).

R&D objectives

SFR designs rely heavily on technologies already developed and demonstrated for sodium-cooled reactors, and for the 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: 300 years on smaller test reactors and 100 years on demonstration or prototype reactors. Significant SFR research and development programs have been conducted in

France, Japan, India,[1] Russia, the United States and the United Kingdom. The only SFR power reactors in operation are the BN-600 and the BN-800 (both in Russia). Currently operating test reactors include the BOR-60 (Russia) and CEFR (China). The Joyo (Japan) test reactor is in the licensing process for restart. New SFR test reactors, the MBIR (Russia) and VTR (United States), are expected in the next decade. In addition, SFR technology R&D programs are being pursued by all SFR GIF members.

A major benefit of the maturity of the SFR technology is that the majority of the remaining R&D needs are related to performance rather than the viability of the system. Accordingly, Gen-IV collaborative R&D focuses on a variety of design innovations for actinide management, improved SFR economics, the development of recycle fuels, in-service inspection and repair (ISI&R) and verification of favourable safety performance.

The system integration and assessment (SIA) project: through a systematic review of the technical projects and relevant contributions on design options and performance, the SIA project will help define and refine requirements for Gen-IV SFR concept R&D. The SFR system options are assessed with respect to Gen-IV goals and objectives. Results from the R&D projects will be evaluated and integrated to ensure consistency.

The safety and operation (SO) project: the SO project is arranged into three work packages (WPs): 1) WP SO1 “methods, models and codes” for safety technology and evaluation; 2) WP SO_2 “experimental programs and operational experience”, including the operation, maintenance and testing experience in facilities and SFRs (e.g. Monju, Joyo, Phenix, BN-600, BN-800 and CEFR); and 3) WP SO3 “studies of innovative design and safety systems” related to safety technology, such as inherent safety features and passive systems.

The advanced fuel (AF) project: the AF project aims at developing and demonstrating minor-actinide-bearing (MA-bearing) high burn-up fuel for SFRs. The R&D activities of the AF project include fuel fabrication, fuel irradiation and core materials (cladding materials) development. The advanced fuel concepts include both non-MA-bearing driver fuels (reactor start-up) and MA-bearing fuels as driver fuels and targets (dedicated to transmutation). The fuels considered are oxide, metal, nitride and carbide. Currently, cladding/wrapper materials under consideration include austenitic and ferritic/martensitic steels, but the aim is to transition in the longer term to other advanced alloys, such as ODS steels.

The component design and balance-of-plant (CD&BOP) project: this project includes the development of advanced energy conversion systems (ECS) to improve thermal efficiency and reduce secondary system capital costs. The project

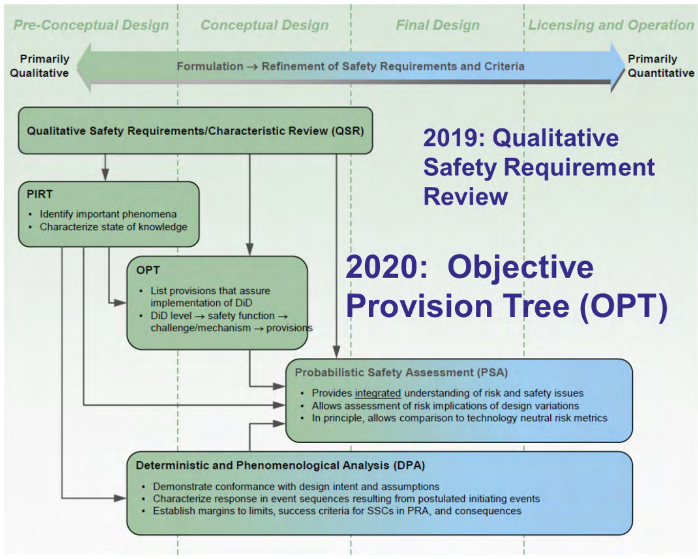


Figure SFR-1. Assessment of safety requirements for the European sodium fast reactor

also includes R&D on advances in sodium ISI&R technologies, small sodium leak consequences and new sodium testing capabilities. The main activities in ECS include: 1) development of advanced, high-reliability steam generators and related instrumentation; and 2) the development of advanced energy conversion systems (ECS) based on a Brayton cycle with supercritical carbon dioxide or nitrogen as the working fluid.

Main activities and outcomes

SIA project

The China Institute of Atomic Energy (CIAE) contributed a study that evaluates the main heat transfer parameters of the CFR1200 design. 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.

The CEA has developed the “CADOR” core concept, which reduces the volume power and adds some moderators in order to eliminate severe accidents scenarios induced by unprotected events. As a result of safety analysis studies, the CADOR core has demonstrated good natural behavior in the case of unprotected transients through improvements inDoppler feedback. However, the volume of the core becomes larger than the classical core design (ASTRID CFV core).

The JRC conducted on an assessment of safety requirements for the European sodium fast reactor (ESFR) using Integrated safety assessment methodology (ISAM) tools. The Horizon 2020 European Sodium Fast Reactor Safety Measures

1. India is not a member of GIF.

Assessment and Research Tools (ESFR-SMART) project, launched in 2017, aims at enhancing further the safety of Gen-IV SFRs, and in particular the commercial-size ESFR in accordance with the European Sustainable Nuclear Industrial Initiative (ESNII) roadmap. Within the project, Euratom applied the ISAM tools developed by GIF RSWG to the ESFR in order to assess safety requirements. This contribution is multi-annual, and 2019 deliverables dealt with the application of the ISAM qualitative safety feature review (QSR). In 2020, Euratom released deliverables on the application of the ISAM objective provision tree (OPT) to ESFR-SMART studies.

The JAEA has been reconsidering advantages gained from previous innovative technologies because significant changes were required in the design of the JSFR after the Fukushima Daiichi NPP accident. The R&D load, the risk of each innovative technology and updated lists of innovative technologies for loop-type reactors were reviewed. This review is based on development easiness and preparation of design standards. Additionally, the total mass and safety of the design are being considered.

KAERI is developing a steam generator concept to minimize sodium-water reaction. A copper bounded steam generator (CBSG) was selected as an alternative steam generator concept. Triple isolation wall structure with steel tube/copper matrix/steel tube layered geometry and eliminated welds were introduced to achieve a very low probability of sodium-water reaction, and the heat exchanger modules were designed by sizing and CFD analysis. Manufacturing tests of a small-scale module through hot isostatic pressing (HIP) diffusion bonding, tension and contact thermal resistance tests of HIP bonding materials, and thermal fluid visualization tests and structural analyses for CBSG were performed.

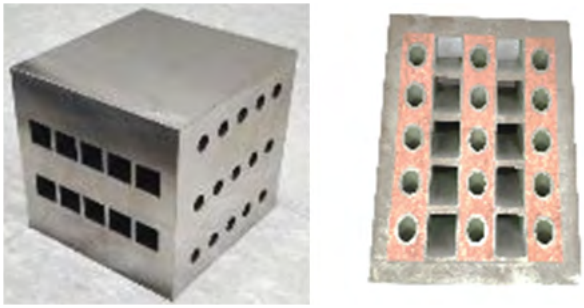


Figure SFR-2. Manufacture of a small-scale copper bonded steam generator module using the hot isostatic pressing process

Safety and operations project

On the topic of the safety and operation (SO) project, the common project that consists of two benchmark analyses (the EBR-II test and Phenix dissymmetric tests) started from the last quarter of 2019. The first phase of the benchmark analysis is a "blind phase", which will take two years to complete. Argonne National Laboratory (ANL), the JAEA and KAERI completed the blind phase of the EBR-II test study at the end of 2020.

WP SO 1: Methods, models and codes

The CEA has analyzed debris bed cooling after a severe accident that followed an unprotected loss of flow (ULOF). In this analysis, one in-vessel DHX was considered and one ex-vessel decay heat removal system. Two types of models have been used for CFD calculations: the laminar model and the K-Ω SST turbulence model. The CEA has demonstrated the calculation results on core catcher temperatures and hot pool temperatures. In terms of the core catcher, the temperature of the insulator (i.e. a few centimetres of ZrO_2) was less than 1 000°C. That value was verified at less than the fusion temperature of ZrO_2; 2 750°C during the simulations. The temperature of the debris, taking into account the conservative assumptions performed, were less than 1 500°C. This result provides a wide margin for a non-desired scenario disruption.

The CEA also presented detailed calculations for flow patterns in the lower plenum, which requires a full 3D-calculation approach. The heat removal trend up until 35 000 seconds was calculated through CFD, modified by the porous media

Figure SFR-3. Local debris and insulator temperatures

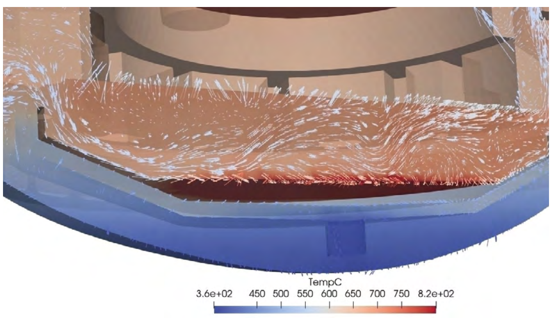
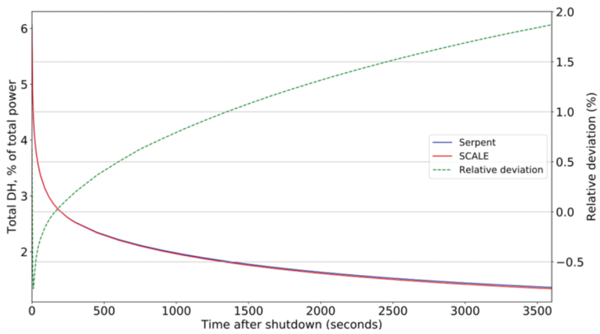


Figure SFR-4. Total decay heat normalized to nominal power



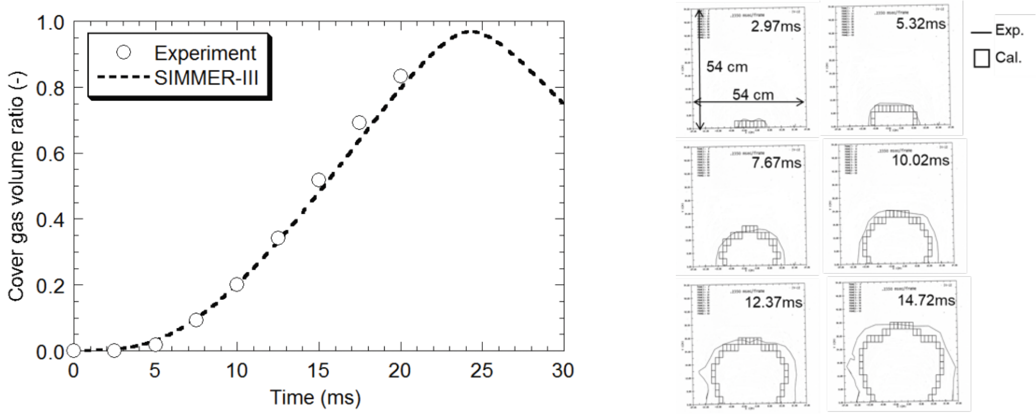


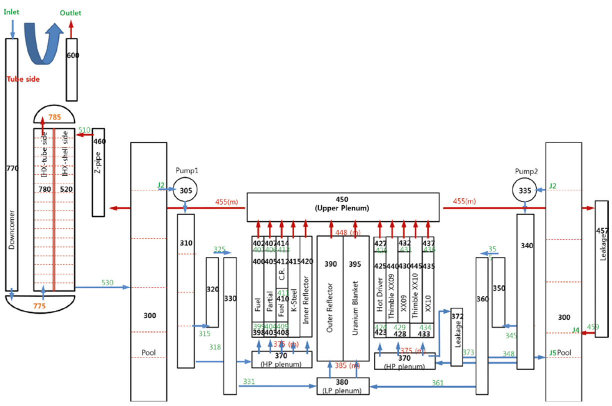
Figure SFR-5. Examples of SIMMER-III analysis results for the OMEGA test

model. After 35 000 seconds of transient, the power balance between heat decay and power removed had not yet been reached. The power gap decreases suggest a convergence after a few days.

Euratom has assessed safety parameters related to the end of cycle (EOC) loading of the ESFR-SMART core, including full core and local effect analysis. The EOC state of the core presents the most limiting case for safety analysis. The deliverable discusses estimation safety parameters, such as control rod insertion S-curve, sodium void reactivity, thermal expansion, Doppler constant and void worth. In addition, sensitivity and uncertainty analyses of different safety parameters related to nuclear data have been performed. The spatial and time-dependent decay heat characteristics normalized to nominal power were also estimated.

The JAEA has been developing an advanced computer code, SIMMER-III/IV, for the analysis of a core disruptive accident (CDA). For the validation of the SIMMER-III code, the JAEA presented a comparison of analysis results for material expansion dynamics, with experimental results. The JAEA selected two experiments for the validation study: the VECTORS test carried out by JAEA to focus on the phenomena of multi-phase flow in structure, and the OMEGA test undertaken by Purdue University to focus on the phenomena of huge vapour bubble expansion dynamics. SIMMER-III successfully simulated both the VECTORS test and the OMEGA test, and has proven to be practical and useful for SFR severe accident analysis.

Figure SFR-6. MARS-LMR modelling of the EBR-II BOP test



KAERI performed a preliminary benchmark analysis of EBR-II BOP-301/302R tests using MARS-LMR. Typical behaviors and significant increases in the core inlet temperature in unprotected loss of heat sink (ULOHS) were investigated in the EBR-II BOP-301/302R. Similar trends in the BOP301 and 302R results were observed and compared with ANL calculations.

The Institute of Physics and Power Engineering (IPPE, Rosatom) continued to develop the 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, and up to the ventilation system) has been developed and implemented into the code. This model has been integrated into the primary circuit 3D thermo-hydraulic model to simulate the transport of gaseous fission products from disintegrated fuel pins to the ventilation system, as well as potential releases into the environment.

The IPPE has performed integral analysis of the consequences of severe accidents in the BN-1200. The following codes have been used: 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). The IPPE has performed preliminary experiments with thermite compositions to obtain the melt of stainless steel with high temperatures. This technique will be used in a facility (which is currently being designed) to simulate transport of melted core in SFR conditions.

WP SO 2: Experimental programs and operational experiences

The CIAE conducted experimental research and code development for a heat transfer analysis of the China experimental fast reactor (CEFR) damaged spent-fuel assemblies in a closed space. The experiment simulated the spent-fuel assemblies during transportation, and the heat transfer characteristics were investigated.

Euratom contributed to the standard procedure for the sodium loop operation and measurement treatment. The deliverable presents the results of the selection of review elements for sodium technology. Aspects considered include procedures used for testing prototypical components at large facilities, procedures for calibration of sensors and signal treatment for the measuring system, methodologies for treatments of the measurement for subsequent use as input data for codes and the conservation of a facility in appropriate conditions enable restart in safe conditions.

WP SO 3: Studies of innovative design and safety systems

The JAEA attempted to identify accident sequences against severe accidents, referring to the rule of new regulations for LWRs. The JAEA carried out an internal event probabilistic risk assessment (PRA) in order to identify the accident sequences to be evaluated. Measures against the accident sequences (anticipated transient without scram [ATWS] and loss of heat removal system [LOHRS]) were studied and developed (see Figure SFR-7). Regarding external events, earthquakes and tsunamis were studied as the most prioritized initiating events. The accidents initiated with these events can be categorized into the accident category identified through internal event PRA (i.e. protected loss of heat sink [PLOHS]). By evaluating identified accident sequences, the JAEA can confirm that most of the SA (i.e. core damage and control valve failure) can be prevented.

The CEA has been studying the design of a small modular fast reactor. It has been considering constraints such as the plutonium content and maximal linear heat rate. The CEA has shown that criticality, which is an issue for small cores, could be achieved by adjusting the number and design of the assemblies. Further studies are planned (in 2021) to investigate other dimensioning parameters for a small modular sodium-cooled fast reactor (SMSFR).

Advanced fuels project

The AF project consists of three work packages (WPs): WP2.1 "SFR non-minor-actinide (MA)-bearing driver fuel evaluation, optimization and demonstration"; WP2.2 "MA-bearing transmutation fuel evaluation, optimization and demonstration"; and WP2.3 "high-burn-up fuel evaluation, optimization and demonstration".

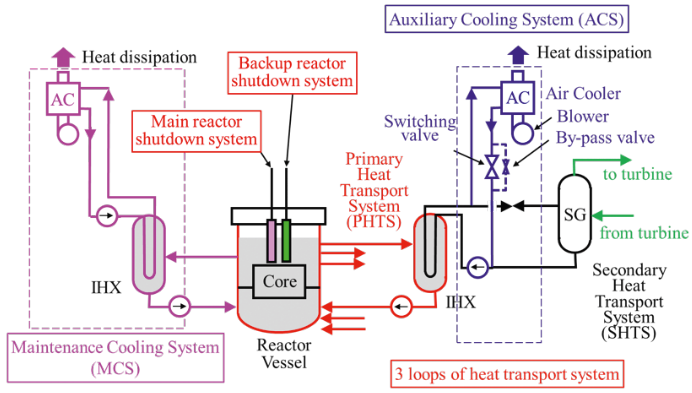


Figure SFR-7. MARS-LMR Modelling of the EBR-II BOP test

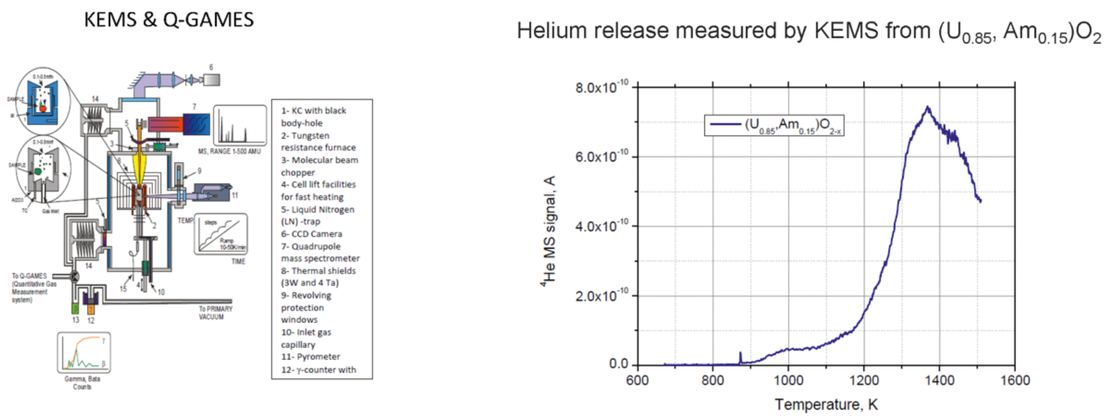
WP2.1: SFR non-MA-bearing driver fuel evaluation, optimization and demonstration

The CIAE is preparing to undertake some irradiation tests. It has finished the fabrication of dummy irradiation assemblies and out-of-pile hydraulic tests (hydraulic characteristic experiments). In 2020, more out-of-pile hydraulic and mechanical tests were conducted to ensure the future safety of in-pile irradiation assemblies.

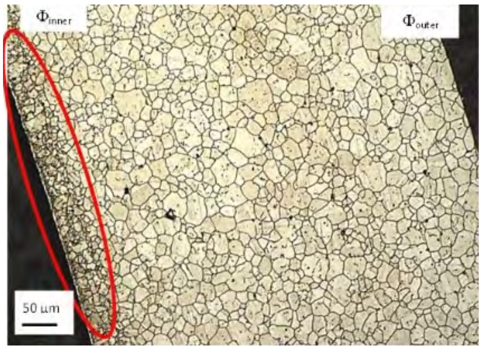
The CEA characterized a PAVIX-8 axially heterogeneous pin irradiated in the Phenix SFR at intermediate linear heat rate (LHR) to extend the validation basis of the GERMINAL V2 fuel performance code. Compared to high LHR irradiated fuels, major differences resulting from the lower fuel operating temperature have been observed. The GERMINAL V2 code underestimated the fuel swelling because this code does not consider gaseous swelling. The implementation of a new fuel swelling model in the GERMINAL code is underway.

The JAEA developed a plutonium and uranium mixed oxide powder adhesion prevention technology, applying nanoparticle coating on the acrylic panels

Figure SFR-8. Helium transport and release behavior in $(U,Am)O_2$



Cladding tubes made of 15-15Ti AIM1 steel



Grain size reduction on inner side of cladding tubes due to nitrogen contamination

Figure SFR-9. Characterization of the 15-15Ti AIM1 cladding tubes

of the glove box to minimize retention of nuclear fuel materials in glove box components and curtail the external exposure dose.

Rosatom manufactured three experimental nitride fuel assemblies for the BN-600 reactor.

WP2.2: MA-bearing transmutation fuel evaluation, optimization and demonstration

Euratom has also contributed to the work of MA-bearing oxide fuel performance evaluation. $(U,Am)O_2$ mixed dioxides are promising candidate fuels for the transmutation of americium (Am) in fast reactors in the heterogeneous recycling concept. One of the major differences in the irradiation performance of these fuels, compared to conventional MOX or uranium dioxide (UO_2), is their large He production, which can have a significant impact on safety-related phenomena, such as fuel swelling and pressure build-up inside the fuel pin. However, knowledge of its behavior in fuel is limited. Therefore, separate effect tests were performed on He generated in situ by alpha decay in $(U,Am)O_2$, and He introduced by ion-beam bombardment in $(U,La)O_2$ simulant materials. He transport and release mechanisms were then investigated by Knudsen Cell effusion mass spectrometry for both sample types, with complementary experiments on their microstructure evolution by transmission electron microscopy (TEM).

The JAEA is developing a simplified pelletizing process for MA-bearing MOX fuel fabrication. As part of this project, the JAEA evaluated performance of the granulation system in actual scale with simulated powder, which is composed of modernized a wet granulator, sizing machine, dryer and other auxiliary equipment.

KAERI developed metal fuel for the prototype Gen-IV sodium-cooled fast reactor (PGSFR). The fuel assembly was designed to satisfy requirements for the core performance and safety. The PGSFR fuel assembly consists of the handling socket, upper/lower reflector, hexagonal duct, fuel rods and nose piece. The structural characteristics and design features of PGSFR fuel assembly and its components have been described.

KAERI also analyzed the interaction between casting parts and U-10 wt.% Zr alloy containing rare-earth (RE) elements through the sessile drop test. Candidates for alternative crucibles and moulds was demonstrated using a casting of the U-Zr-RE alloy. Interaction behaviors and defects of the casting parts were evaluated after casting.

Rosatom developed technological processes for the manufacture of americium-burning elements within the framework of the "heterogeneous" scenario of the nuclear fuel cycle closing (NFC). Rosatom manufactured an experimental batch of mixed nitride of uranium and plutonium (MNUP) fuel samples through the method of high-voltage electric pulse consolidation and control of their characteristics.

WP 2.3: High-burn-up fuel evaluation, optimization and demonstration

The CIAE will conduct a program to do some CN-1515 and CN-FMS material irradiation tests in the CEFR in the coming years. The R&D and fabrication of CN-1515 and CN-FMS has been completed. The design of the irradiation rig and the fabrication of irradiation assemblies has also been finished. The mechanical properties of CN-1515 and CN-FMS have been tested and have been used to evaluate the irradiation assemblies.

The CEA has characterized the cladding tubes of 15-15Ti AIM1 from two different fabrication routes. The AIM1 is a titanium (Ti)-stabilized austenitic stainless steel treated as the reference cladding material for Phenix and ASTRID SFRs. In the results of the glow discharge mass spectrometry (GDMS), nitrogen contamination is observed on the inner surface of the cladding tubes of one of the two batches. The CEA carried out tensile property measurements on the two batches. Based on these results and past experience with AIM1 cladding irradiated in Phenix, the CEA concluded that a good behavior in pile can be foreseen for these AIM1 cladding tubes.

The JAEA carried out high- and ultra-high-temperature creep rupture tests, internally pressurized creep rupture and ring rupture tests, and temperature-transient-to-burst tests of 9Cr-ODS steel claddings. For comparison, the transient burst strength of 11Cr-ferritic/martensitic steel (PNC-FMS) cladding was also evaluated. The obtained data was used to investigate the applicability of the life fraction rule to rupture life prediction of 9Cr-ODS steel and PNC-FMS claddings in various load-time temperature histories.

KAERI developed technology for a barrier cladding tube to suppress fuel-cladding chemical interaction (FCCI) for the use of MA-bearing metal fuel. Cr plating was applied at the inner surface of the cladding tube to achieve 20 µm thickness of Cr at the 500 mm length of HT9 cladding. Optimization of Cr plating to enhance layer property, such as pulse plating and surface treatment through nitriding process, has been reported.

Rosatom developed and manufactured BN-600 irradiation assemblies for testing fuel elements up to extreme parameters.

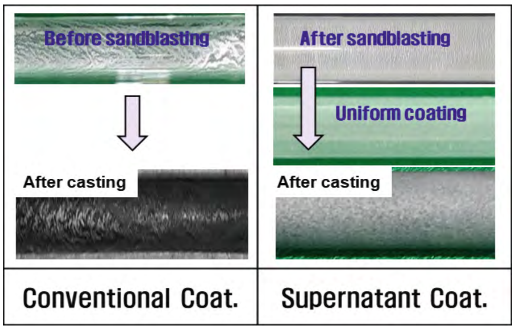


Figure SFR-10. Development of coating technology for the development of reusable mould

Component design and balance-of-plant project

ISI&R technologies

The CEA has studied the capability of the leaky Llmb waves in view of inspection from the outside of the main vessel. In 2020, the CEA conducted this experiment using devices that consisted of several austenitic steel plates immersed in water, an ultrasonic emitter and receiver. The experimental results with one plate were compared to simulation results.

KAERI demonstrated the performance of the plate-type ultrasonic waveguide sensor in a sodium environment. KAERI fabricated under-sodium waveguide sensors and then conducted several under-sodium tests for viewing and ranging performance verification.

The JAEA performed the imaging test under-sodium viewer for medium distance in a sodium environment, and a performance test for long distance in actual plant configuration.

Supercritical CO_2 Brayton cycle

The CEA investigated the development of sensors to examine this heat exchanger. For this, eddy current probes were developed after defining the

Figure SFR-11. Under-sodium performance demonstration of the ultrasonic waveguide sensor

New sodium test facility *Inspection software for under-sodium viewing and ranging* *Prototype under-sodium viewing sensors*

Figure SFR-12. Imaging experiments in sodium environment

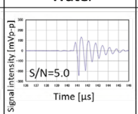
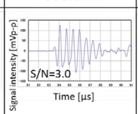
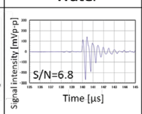
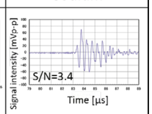
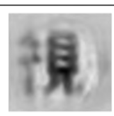
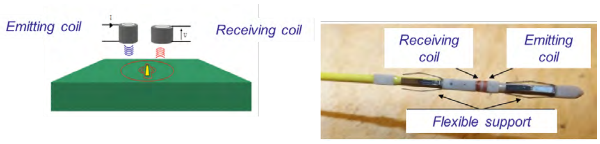
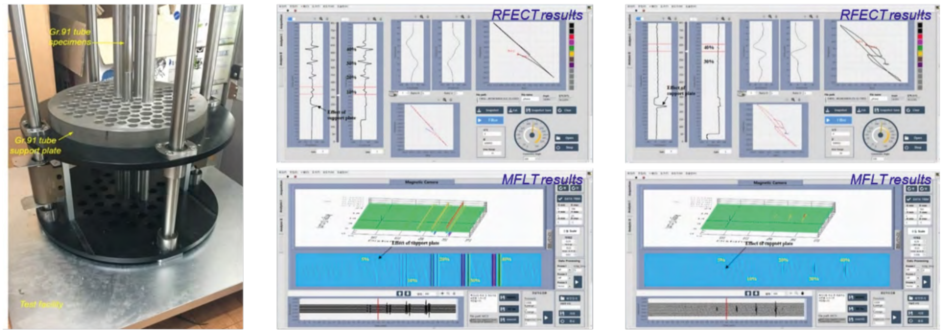
	Type-A		Type-B	
	Water	Sodium	Water	Sodium
Wave profiles				
Regenerated images				

Figure SFR-13. Eddy current technique for NDT within small channels





Installation of Gr.91 support plate

Test results for short circumferential grooves with Gr.91 support plate

Test results for circumferential notches with Gr.91 support plate

Figure SFR-14. Sodium-water reaction in a semi-open volume

Figure SFR-15. Inspection system for a single-walled tube of a Rankine-type steam generator

specifications. A very small (3 mm by 3 mm, as an order of magnitude) probe was designed and manufactured.

The ANL prepared a report summarizing what has been learnt about sodium-CO_2 interactions and sodium-CO_2 reaction products based mainly upon data and results from the SNAKE sodium-CO_2 interaction experiments carried out at ANL.

Sodium leakages and consequences

No specific activity was conducted this year in this work package.

Steam generators

The CEA started a new activity in the field of the sodium-water reaction. It studied the sodium-water reaction (SWR) in specific conditions, such as open or semi-open volume (see Figure SFR-14). For this, after a review of the existing knowledge, the CEA defined and performed SWR in a dedicated facility (MININANET). The first experimental campaign results were then presented.

The JAEA performed the investigation on the applicability of mechanistic sodium-water reaction analysis code for steam generator performance and safety evaluation.

KAERI continued its performance demonstration of the upgraded prototype combined steam generator tube inspection system. More specifically, KAERI investigated the effects of the tube support plate and neighbouring tubes on measured remote field eddy current testing (RFECT) and magnetic flux leakage signals.

Sodium operation technology and new sodium testing facilities

KAERI finished constructing the sodium thermal-hydraulic integral effect test facility (STELLA-2), and completed its shake-down test and start-up operation with liquid sodium. A couple of sets of sodium integral effect test databases were collected and were used for some computational verification and validation (V&V) codes. KAERI continued the study of specific sodium technologies to get some useful measurements of process variables,

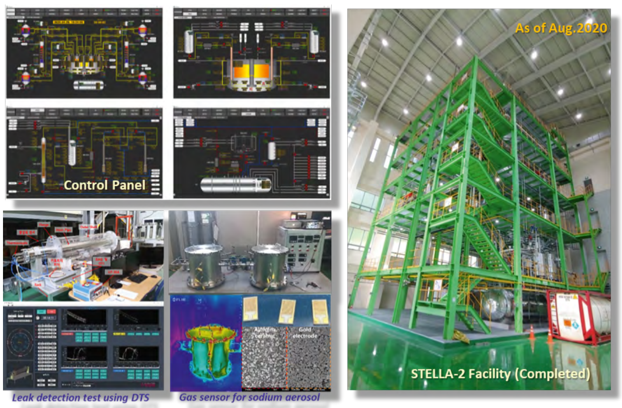


Figure SFR-16. Progress of the sodium integral effect test (STELLA-2) and development of advanced instrumentation techniques for liquid sodium applications

as well as for better operation of large-scale sodium facilities covering sodium feeding, draining, accident prevention, high-temperature operation and measurements.

The DOE conducted the design and construction of an intermediate-scale sodium test facility for the purpose of testing systems and components (mechanisms) in prototypical sodium environments. This facility consists of four experimental test vessels (of two sizes). Sodium has been fed to these test vessels from a main loop. The test vessels were designed to provide an independent testing environment, if isolated from the main loop. In addition, the test vessels were designed to allow for independent draining to the main dump tank without impacting the sodium environments in the other test vessels.

Frédéric Serre
Chair of the SFR SSC, with contributions from SFR members

Very-high-temperature reactor

High-temperature gas-cooled reactors (HTRs or HTGRs) are helium-cooled graphite-moderated nuclear fission reactors using fully ceramic fuels. They are characterized by inherent safety features, excellent fission product retention in the fuel, and high-temperature operation suitable for the delivery of industrial process heat, and in particular for hydrogen production. Typical coolant outlet temperatures range between 750°C and 850°C, thus enabling power conversion efficiencies up to 48%. The very-high-temperature reactor (VHTR) is understood to be a longer-term evolution of the HTR, targeting even higher efficiency and more versatile use by further increasing the helium outlet temperature to 950°C or even higher. Above 950°C, however, and such reactors will require the use of new structural materials.

VHTRs can be built with power outputs that are typical of SMRs. They are primarily dedicated to the cogeneration of electricity and process heat (combined heat and power [CHP]), for example for hydrogen production. The initial driver for VHTR development in GIF was thermo-chemical hydrogen production with the sulphur-iodine cycle requiring a core outlet temperature of approximately 950°C. Further market research across GIF signatories has shown that there is also a very large near-term market for process steam of approximately 550°C, achievable with lower temperature HTR designs. R&D in GIF has therefore shifted to cover both lower and higher temperature versions of this reactor type.

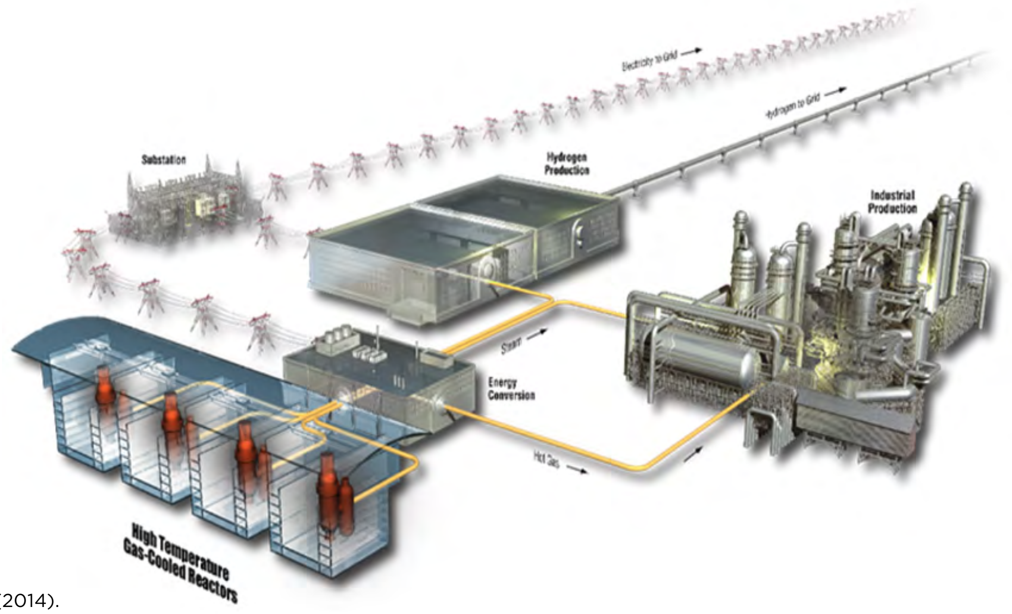
Cogeneration of heat and power makes HTRs and VHTRs attractive heat sources for big industrial complexes, such as chemical plants, to substitute large amounts of process heat at different temperatures, which are today produced by fossil fuels. Depending on the coolant outlet temperature, such reactors can be employed to produce hydrogen from heat and water by using thermo-chemical, electro-chemical or hybrid processes with largely reduced CO_2 emissions. Typical HTR coolant outlet temperatures range from below 750 to 850°C, thus enabling power conversion efficiencies up to 48% in pure power generation and even much higher in the combined heat and power (CHP) mode.

HTRs and VHTRs can be operated with a once-through LEU (<20% 235U) fuel cycle and with a closed fuel cycle (improved sustainability). This reactor type was identified quite early as particularly suitable for the Th-U fuel cycle, while potential symbiotic fuel cycles with other reactor types (especially light water and fast reactors) are also an option.

The operational temperatures of HTRs and VHTRs can be adapted to specific end-user needs. Thermal reactor power is limited by the requirement for fully passive heat removal in accident conditions. The different core pressure drops, which govern the capacity for passive heat removal, translates to <250 MWth for pebble bed reactors and <625 MWth for hexagonal block type reactors. The actual reactor power can be flexibly adapted to local requirements, for example the electricity/heat ratio of an industrial site. The power density is low and the thermal inertia of the core is high thus granting walk-away safety in accident conditions. The potential for high fuel burn-up (150-200 GWd/tHM), high efficiency, high market potential, low operational and maintenance costs, as well as modular construction, all constitute advantages favouring commercial deployment.

The basic technology has been established in former high-temperature gas-cooled reactor plants, starting with the NEA DRAGON project, which led to the development of coated-particle fuel and demonstrated the safety features of HTRs, including a final core heat-up experiment. Later, the United States Peach Bottom and Fort Saint-Vrain plants were built, as well as the German AVR and THTR prototypes, which produced high-quality steam up to 550°C. After resolving some initial problems, the technology has advanced through near- and medium-term projects led by several plant vendors and national laboratories, such as HTR-PM (China), PBMR (South Africa), the gas turbine high-temperature reactor 300 for cogeneration (GTHTR-300C, Japan), Antares project (France), the Nuclear Hydrogen Production Project (NHDD, Korea), the gas turbine modular helium reactor (GT-MHR, US and Russia) and the next generation nuclear plant (NGNP, US). Experimental reactors such as the HTTR (Japan, 30 MWth) and HTR-10 (China, 10 MWth) support technology development, including CHP, hydrogen production and other nuclear heat applications.

The VHTR can be designed with either a pebble bed or a prismatic block core. Despite these differences, however, all VHTR concepts show extensive commonalities, allowing for a joint R&D approach. The standard fuel is based on UO_2 tri-structural isotropic (TRISO) coated particles (UO_2 kernel, buffer/iPyC/SiC/oPyC coatings) embedded in a graphite matrix, which is then formed either into pebbles (tennis ball size spheres) or into compacts (thumb-size rodlets). This fuel form exhibits a demonstrated long-term temperature tolerance of 1 600°C in accident situations. This safety performance may be further enhanced, for example through the use of a uranium-carbon-oxygen fuel kernel, a ZrC coating instead of SiC, or the replacement of the graphite matrix material with SiC. The fuel cycle will first be a once-through, very high burn-up, low enriched uranium fuel cycle. Solutions to adequately manage the back-end of the fuel cycle are under investigation and potential operation with a closed fuel cycle will be prepared by specific head-end processes to enable the use of existing reprocessing techniques. Power conversion options include indirect Rankine cycles or direct or indirect Brayton cycles. Near-term concepts will be developed using existing materials, whereas more



Gougar, H., (2014).

Figure VHTR-1. Artist's view of a 4-module VHTR poly-generation plant

advanced concepts will require the development, qualification and coding of new materials and manufacturing methods.

High core outlet temperatures enable high efficiencies for power conversion and hydrogen production, as well as high steam qualities (superheated or supercritical). Hydrogen production methods include high-temperature electrolysis and thermo-chemical cycles, such as the sulphur-iodine process, hybrid cycles or steam methane reforming. The transfer of heat to a user facility over a distance of several kilometres can be achieved with steam, gases, certain molten salts or liquid metals. The use of nuclear CHP with HTRs has a very large potential for the reduction of fossil fuel use and of noxious emissions, and is the prime motivation for the signatories of the VHTR system. The increased use of nuclear energy for powering industrial processes and for large-scale bulk hydrogen is a strong motivation for VHTR development and enables the integration of nuclear power with renewable energy sources in hybrid energy systems (see Figure VHTR-1).

Status of cooperation

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 system arrangement. South Africa formally acceded to the GIF framework agreement in 2008, but announced in December 2011 that it no longer intended 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 SA was subsequently signed by Australia (December 2017) and the United Kingdom (January 2019). In 2020, the VHTR System Steering Committee updated its work plan of the high-level system R&D for the development of the VHTR in support of national or international VHTR demonstrator projects and enhanced performance capability in the long term.

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 (PA) 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 VHTR materials project plan was completed in 2012, the materials PA continued through 2019 under its first amendment, which added China as a signatory. On 27 April 2020, the second amendment of the PA became effective. It incorporated a new project plan for technical activities and planned contributions from 2018-2022, and added Australia as an additional signatory. It also extended the term of the PA through April 2030. Contributions to the new PP for 2018-2022 were developed by all previous signatories (China, the European Union, France, Japan, Korea, Switzerland, and United States), as well as Australia, the newest member. In 2020, the United Kingdom expressed an interest in joining the PA, and Canada also expressed an interest in joining the PA once again, after its earlier withdrawal. Both countries presented their capabilities and potential contributions for VHTR materials to the Materials Project Management Board (PMB), which invited both of these countries to prepare formal planned contributions to the PP. These planned contributions are expected to be completed in 2021. If accepted, Canada and the United Kingdom will be invited to join the third amendment to the PA. It is also anticipated that the current PP will be extended through 2024, with augmented contributions from all existing signatories.

The hydrogen production PA became effective on 19 March 2008, with implementing agents from Canada, Euratom, France, Japan, Korea and the United States. In 2020, the forthcoming five-year project plan was prepared to incorporate contributions from China and updated contributions from other countries, under the consensus of the PMB. The amendment of the hydrogen production PA to welcome China's Institute of Nuclear and New Energy Technology (INET) as a member of the PMB is expected in 2021.

The computational methods validation and benchmarks (CMVB) PA remained provisional in 2020. After the draft PP was approved by the VHTR System Steering Committee (SSC), the draft PA was confirmed by each signatory. The signatories are now ready to pursue the signature of the PA expected in 2021.

R&D objectives

While VHTR development is mainly driven by the achievement of very high temperatures, 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 these challenging R&D goals.

The VHTR system research plan is intended to cover the needs of the viability and performance phases of the development plan described in the *GIF R&D Outlook for Generation IV Nuclear Energy Systems: 2018 Update*. From the six projects outlined in the SRP, three are effective and one is provisional, as discussed below. Today, most of the performed activities are licensing-relevant:

- Fuel and fuel cycle (FFC) investigations are focusing on the performance of TRISO coated particles (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-oxycarbide 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 (MAs) in support of a closed cycle.
- Materials (MAT) development and qualification, design codes and standards, as well as manufacturing methodologies, are essential for VHTR system development. Primary challenges for VHTR structural materials are irradiation-induced and/or time-dependent failure and microstructural instability in operating environments. For core coolant outlet temperatures up to 950°C, it is envisaged 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. Multi-scale modelling is needed to support improved design methods. In addition to high-temperature heat exchangers, additional attention is being paid to metal performance in steam generators, which reflects the current interest in steam-based process applications at somewhat lower core outlet temperature of 750°C to 850°C. Structural materials are considered in three categories: graphite for core structures, or for fuel matrix; very/medium-high-temperature metals; and ceramics and composites. A materials handbook has been developed and is being used to store and manage VHTR data, facilitate international R&D co-ordination, and support modelling to predict damage and lifetime assessment.
- For hydrogen production, 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: 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 are being assessed through integrated test loops, from laboratory scale through pilot and demonstration scale, and include component development such as advanced process heat exchangers. Hydrogen process coupling technology with the nuclear reactor is also being investigated, and design-associated risk analysis is being 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 view of making them compatible with other Gen-IV nuclear reactor systems.
- CMVB in the areas of thermal-hydraulics, thermal-mechanics, core physics and chemical transport, are major activities. They are needed for the assessment of reactor performance in normal, upset and accident conditions and for licensing. Code validation needs to be carried out through benchmark tests and code-to-code comparison, from basic phenomena to integrated experiments, supported by HTTR, HTR-10 and

HTR-PM tests or by past HT reactor data (AVR, THTR and Fort Saint-Vrain). Computational methods will also facilitate the elimination of unnecessary design conservatisms and improve construction cost estimates.

Even though they may not be implemented, the development of components needs to be addressed for key reactor systems (core structures, absorber rods, core barrel, pressure vessel, etc.) and for the energy conversion or coupling processes (e.g. 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 simulating normal and off-normal events. The project on components should address development needs that are partly common to the GFR, so that common R&D can be envisaged for specific requirements.

SIA is necessary to guide R&D so as to meet the needs of different VHTR baseline concepts and new applications (cogeneration and hydrogen production). Near and medium-term projects should provide information on their designs to identify the potential for further technology and economic improvements. This topic is directly addressed by the SSC.

Milestones

In the near term, lower-temperature demonstration projects (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 HT alloys, qualification of new graphite types and the development of composite ceramic materials. Lower-temperature versions of HTRs (from 700°C to 950°C) will enter the demonstration phase, based on HTR-PM experience in China, which is scheduled to reach the operation stage in 2021. A future higher temperature version (1 000°C and above) will require more research.

Main activities and outcomes

Fuel and fuel cycle project

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

TRISO coated particles need to be qualified for relevant service conditions. Furthermore, its standard design (UO_2 kernels surrounded by successive layers of porous graphite, dense pyrocarbon (PyC), silicon carbide (SiC) and then PyC) could evolve through the use of a uranium oxycarbide (UCO) kernels or a zirconium carbide (ZrC) coating for enhanced burn-up capability, minimized fission product release, and increased resistance to core heat-up accidents. Fuel characterization work, post-irradiation examinations (PIE), safety testing and fission product release evaluations, as well as the measurement of chemical and thermo-mechanical material properties in representative conditions, will feed a fuel materials database. Further development of physical models enables assessment of in-pile fuel behavior under normal and off-normal conditions.

The back-end of the fuel cycle 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 envisaged, the potential for the 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.

Irradiation and PIE

Irradiation and PIE includes activities on fuel irradiation testing, PIE facility and equipment development and post-irradiation examination of fuel specimens, with activity currently taking place in China and the United States.

PIE on the AGR-2 fuel (including both UCO and UO_2 TRISO particles) has continued in the United States and is near completion, with a final report expected in 2021. This work includes extensive destructive examination of fuel compacts and particles. Up to this time, 12 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 migration of fission products in the layer.

The US AGR-5/6/7 irradiation of UCO TRISO fuel into the advanced test reactor was completed in July 2020 after achieving approximately 360 effective full power days in the reactor and a peak fuel burn-up of ~15% FIMA. This experiment is both the final fuel qualification irradiation and a separate HT fuel performance margin test (peak temperatures of ~1 500°C) and contains approximately 570 000 fuel particles in 194 fuel compacts. PIE is expected to begin around April 2021. Development of PIE capabilities with the newly established INET hot cells continues, after installation of an Irradiated Microsphere Gamma Analyzer (IMGA) apparatus and pebble deconsolidation equipment was completed. This equipment will be used to perform destructive examination on irradiated fuel pebbles.

Safety

A fuel pebble of HTR-10 production, which was irradiated previously in the HFR-EU1 experiment in HFR Petten, was heated in the KÜFA facility at JRC Karlsruhe to evaluate fission product release at elevated temperatures. The specimen was held at temperatures of 1 620°C, 1 700°C, and 1 800°C for 150 hours at each temperature. Release of Kr-85, Cs-134, and Cs-137 were measured. A new KÜFA furnace system similar to the one deployed in Karlsruhe has been installed in INET hot cells (see Figure VHTR-2) Hot testing of the system has been delayed by the COVID-19 pandemic, but is planned for 2021 using low-burn-up fuel pebbles discharged from HTR-10.

The United States is continuing to perform 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. The PIE of these materials thus helps to understand fission product transport in fuel matrix and graphite materials, and will be used to refine fission product transport models that are critical for reactor safety analyses. Work in 2020 focused on destructive examination and heating tests of the AGR-3/4 fuel compacts. Heating tests have been performed on a total of seven irradiated AGR-3/4 compacts in which the fuel is heated at temperatures between 1 100°C and 1 600°C while fission product release is monitored. In three of these tests, the fuel specimens were re-irradiated in the neutron radiograph reactor (NRAD) to generate short-lived iodine-131 (I-131) and xenon-133 (Xe-133) prior to the heating tests. These tests provide data on the release of short-lived fission products (including I-131) that can be significant contributors to off-site dose during reactor accidents. Accurately quantifying the release from the kernels requires measurements of the fission product inventory in the compact matrix. In order to avoid the exposed kernels that lie roughly along the compact axial centerline, the fuel compacts are deconsolidated by rotating the compact during the process to remove successive layers of particles (~1 mm-thick layers).

The United States also continues with the development of a dedicated furnace to heat irradiated TRISO fuel specimens up to 1 600°C in oxidizing atmospheres. The system will be used to test oxidation behavior of fuel and fuel materials in air/He and moisture/He gas mixtures, while monitoring online the release of fission products and reaction products. The system is expected to be deployed in 2021.

Post-irradiation heating tests of loose TRISO particles in helium for very long durations (up to 1 500 h) under a range of temperatures (1 150°C to 1 600°C) are being performed in the United States with the goal of quantifying the release of certain fission products through intact particle coatings. This study focuses in particular on silver (Ag) and europium (Eu), as these have been observed to be released in significant (Ag) or modest (Eu) amounts depending on the fuel irradiation temperature. The majority of the experimental matrix has been completed, with final tests to be performed in 2021.

Enhanced and advanced fuel fabrication

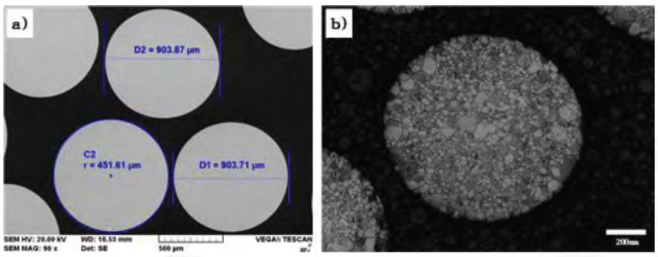
Work to develop advanced TRISO fuels, in some cases involving varying particle designs or new coating materials, is in progress in several member countries. In Korea, KAERI has been working to develop fabrication methods for UO_2 kernels with diameters significantly larger than the conventional 500 µm used in most UO_2 TRISO particles. With some process development, kernels with diameters of ~900 µm and good sphericity and density have been fabricated (see Figure VHTR-3). These kernels are envisaged for use as accident-tolerant fuels for light water reactors. KAERI is planning to refine the particle coating process to accommodate these larger kernels. In addition, this work has involved computational studies to optimize the particle dimensions. Future work is planned on developing a composite double layer ZrC/SiC coating for TRISO, which may result in improved performance.

In China, work has been progressing to develop processing methods for UCO kernels, which may offer enhanced fuel performance compared to the conventional UO_2 fuel to be used in the first loading of the HTR-PM reactor.

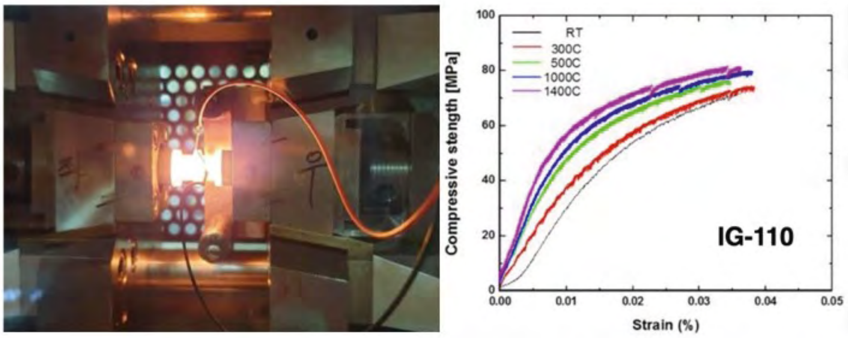
Figure VHTR-2. KÜFA furnace installed in INET hot cells

Liu, B., (2020).

Figure VHTR-3. Example of large-diameter UO_2 kernels fabricated at KAERI



Mean diameter(µm)	Sphericity	Aspect ratio	Density(g/cm³)
907.5	0.948	0.947	10.78



Source: Xu, A. et al. (2021).

Figure VHTR-4. Very high-temperature compression testing of graphite

Other activities

While there has been little VHTR fuel work taking place in the European Union in recent years, representatives from the National Centre for Nuclear Research (NCBJ) in Poland attended the September 2020 FFC PMB meeting as observers and discussed plans for TRISO fuel development in Poland. The Polish representative has been appointed as a Euratom member to the VHTR PMB. A representative from the United Kingdom also attended the FFC PMB meeting and provided a presentation on TRISO capabilities. It is expected that the United Kingdom will become a new signatory to the FFC PMB in the near future.

Materials project

As part of the new PA, a thorough review was made of all high-level deliverables (HLDs). Additionally, by the end of 2019, over 450 technical reports and over 30 000 materials test records, including contributions from all signatories, had been uploaded into the Gen-IV Materials Handbook (i.e. the database used to share materials information within the PMB). This reflects the outstanding technical output of the membership, which has now been shared to support system design and codes and standards development.

In 2020, research activities continued to focus on near- and medium-term project needs (i.e. graphite and HT metallic alloys), with limited activities related to ceramics and composites.

Additional characterization and analysis of selected baseline data, and its inherent scatter of candidate grades of graphite, were performed by multiple members. Mechanical, physical and fracture property behavior was examined for numerous grades. Graphite irradiations, PIE and other analyses 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 to graphite. Tests on the use of boron coatings to minimize the impact of oxidation on graphite core components were conducted. Examination and validation of the multi-axial loading response of graphite from dimensional changes and seismic events, using large-scale experiments on graphite blocks, continued. An example of very-high-temperature testing of graphite mechanical properties is shown in Figure VHTR-4.

Data to support graphite model development was generated in the areas of microstructural evolution, irradiation damage mechanisms and creep. Support was provided for both the American Society for Testing and Materials (ASTM International) and American Society of Mechanical Engineers (ASME) codes and standards required for the use of nuclear graphite, which continue to be updated and improved.

Examination of HT alloys provided valuable information for their use in a heat exchanger and steam generator. These studies included an evaluation of the existing database and its extension through ageing, creep, creep-fatigue and creep crack growth rate testing to 950°C for alloys 800H and 617. Welding studies on 617, 800H, and dissimilar welds of T22 to 800H were performed. Examination of enhanced diffusion bonding techniques for construction of compact heat exchangers (CHEs) showed promising results, and extensive modelling and testing of CHEs are laying the groundwork for their qualification in VHTRs. Testing to qualify new metallic materials (alloy 709, high entropy alloys, oxide dispersion-strengthened [ODS] alloys) for construction of high-temperature nuclear components was pursued.

A new thrust to develop and qualify advanced manufacturing methods for nuclear components (laser fusion, consolidation of metal powders, direct deposition, etc.) was extensively investigated by several signatories. Additionally, new approaches to the synthesis of novel HT structural materials were explored.

Advanced characterization techniques are being used to evaluate the impact of irradiation effects on HT structural materials. Figure VHTR-5 illustrates the novel use of nano-scale tensile specimens fabricated in situ to assess the degradation of tensile properties in ODS alloy (MA957) specimens exposed to ion-beam irradiation. The micro-scale region of maximum damage in ion-beam-irradiated samples can be evaluated using such nano-scale specimens.

For the near/medium term, metallic alloys are considered as the main option for control rods and internals in VHTRs, which target outlet 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. Work continued to examine the thermo-mechanical properties of SiC and SiC-SiC composites, including irradiation-creep effects and the oxidation in carbon- carbon (C-C) composites. Studies to evaluate radiation damage and examine the fracture behavior of C-C composites have begun, as were methods for direct 3D printing of SiC and SiC-SiC composites. The results of this work are being actively incorporated into developing testing standards and design codes for composite materials, and to examine irradiation effects on ceramic composites.

Hydrogen production project

Canada has continued its efforts to demonstrate an integrated copper-chlorine (Cu-Cl) cycle for the production of 50 NL/h hydrogen. Continuing advancements in the four steps of the cycle has also been carried out during 2020. The equipment being assembled for integrating the whole cycle are shown in Figure VHTR-6.

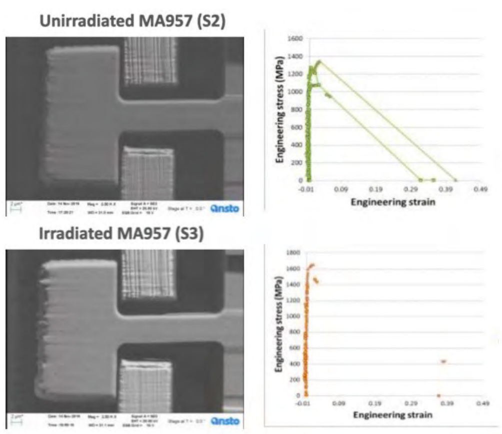
A detailed flowsheet analysis has also been carried out, and the ancillary components required for rendering the process a closed cycle were defined. Optimization of the process with respect to operating expenses (OPEX) and capital expenses (CAPEX) is being undertaken to fully understand their impact on the cost of hydrogen. As expected, the thermal and electrical demand of the process significantly affects the OPEX. A balance between the CAPEX and OPEX has been found to be necessary to arrive at a reasonable cost for the hydrogen produced. Further refinements of the analysis are being carried out with a more detailed look at large-scale hydrogen production.

Chinese efforts with regard to nuclear hydrogen production in the past years have focused on the development of the components of both the sulphur-iodine (S-I) and hybrid sulphur (HyS) processes. In the development of the components of the S-I process, two reactors, which intend to use the heat from the HTGR (the sulphuric-acid [H2SO4] decomposer and hydriodic acid [HI] decomposer), were designed and constructed. The SA decomposer was designed as a shell-and-tube heat exchanger, with the bayonet type silica carbide (SiC) tube as the reaction zone for SA decomposition, as well as the pressure boundary, and the integrity of SiC component has been verified with 100 hours of testing. In addition, the lifetime test of the catalyst for the SA decomposition reaction has been conducted for more than 700 hours and will be continued. The prototype SA decomposer is being manufactured and will be completed in several months. The HI decomposer, composed of an evaporator and an adiabatic reactor, is also being produced. At the same time, a high-temperature helium loop (>900°C, 100 kW) was designed to provide heat for the performance test of those components of the S-I process (see figure VHTR-7).

In the development of the HyS process, efforts have gone into the development of the sulphur dioxide (SO_2)-depolarized electrolyzer (SDE) stack, as well as the auxiliary facility for the test and operation of the SDE stack, particularly under enhanced pressure. A SDE stack with an H2 production rate of 100 NL/h has been developed and tested. Currently, the scaling-up of the stack is in progress.

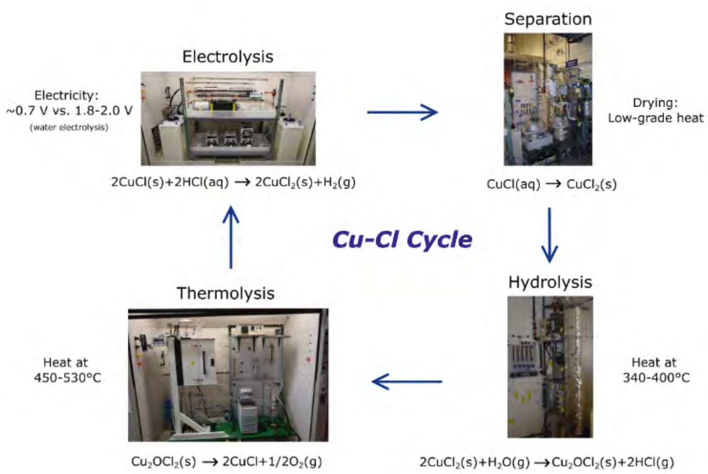
With the announcement of an ambitious recovery plan of EUR 7 billion over a ten-year period,

Figure VHTR-5. Use of in-situ nano-scale tensile specimens to evaluate irradiation effects in ion-beam exposed samples of ODS alloy MA957



Xu, A. et al. (2021).

Figure VHTR-6. Equipment used for the four steps of the Cu-Cl Cycle integration in Canada



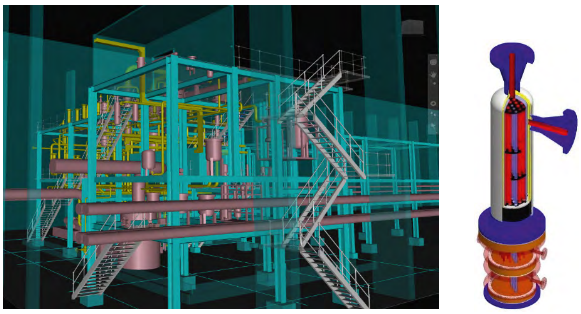
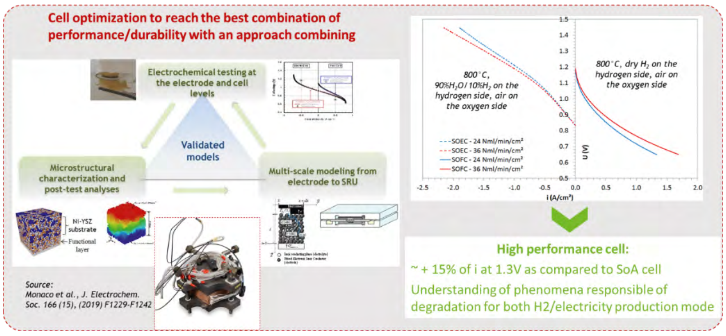


Figure VHTR-7. Sulphuric-acid and hydriodic-acid decomposition facilities and the high-temperature helium loop



Monaco, F. et al. (2019).

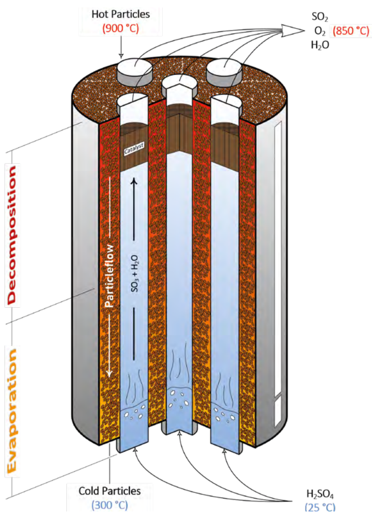
Figure VHTR-8. CEA activities to develop high-temperature steam electrolysis

hydrogen is a pillar of the energy transition in France and a market priority. Aligned with the EU Green Deal, the plan proposes financial incentives to foster clean H2 in industry (e.g. refinery and steel) and the transport sector (heavy or intensive duty vehicle). The electrolysis system, in particular high-temperature steam electrolysis (HTSE), is going to play a major role in producing clean hydrogen, thanks to the mix of nuclear energy and renewable energy. CEA developments have passed the first generation of cell and stack, and the manufacturing step at industrial scale is in progress with a new public-private partnership for developing a pilot line and producing high-power modules of stacks. The CEA is now developing a second generation, higher performance and durability cell by combining numerical and experimental approaches at different scales from raw material to the single cell. Through modelling and characterization of the microstructure using the European Synchrotron Radiation Facility (ESRF), it has been possible to predict the performances of the cell by incorporating a mass transportation model. The CEA has also identified possible impacts of the electrical polarity in addition to the high temperature on the aggregation of the Ni phase on the electrode catalyst. This can explain the higher degradation of the performance of the cell in electrolysis mode compared to performance in fuel cell mode. Thanks to the recovery plan, research on the second generation of cell and stack has been moving forward at good speed with significant progress. Figure VHTR-8 shows a pictorial view of the overall activities at CEA in the development of the HTSE technology.

In the area of sulphur-based hydrogen production process development, different reactor concepts were developed and evaluated in the European PEGASUS project to demonstrate the feasibility of sulphuric-acid decomposition with high-temperature heat absorbed by particles. In the final design, an indirect contact approach, in a strictly counter-current moving bed heat exchanger (MBHE), was chosen. A proof of concept (POC) sulphuric-acid splitting/decomposition prototype driven by hot bauxite particles was developed and designed. The laboratory-scale test reactor is a novel counter-current flow shell-and-tube heat exchanger with particles on the shell side and sulphuric acid on the tube side, and with mass flow rates of 10 and 2 kg/h, respectively. A one-dimensional heat transfer model was developed based on correlations of the flowing fluid boiling heat transfer coefficient and particle bed heat transfer coefficient for sizing the shell-and-tube heat exchanger. A detailed study was carried out in order to choose suitable materials, particularly in the sulphuric-acid inlet and evaporation section. A new concept of an electrically heated, continuously operated particle heating system was designed and developed to provide the splitting reactor with hot particles. Different cases were studied using a finite element method analysis to qualify the particle heater and examine its thermo-mechanical stability.

A kinetic study of the sulphur-trioxide decomposition in the particle heated laboratory reactor for the EU research project PEGASUS was also carried out. The reactor (see Figure VHTR-9) was developed for the use of hot ceramic particles to evaporate sulphuric acid and dissociate into sulphur trioxide (SO_3) and water, and then further decompose SO_3 into SO_2 and oxygen in a sulphur-based thermo-chemical energy-storage cycle (TCES) for concentrated solar power (CSP) plants. The kinetic study is separated into two parts. First a literature search was conducted on available

Figure VHTR-9. Particles driven Sulphuric-acid splitting reactor



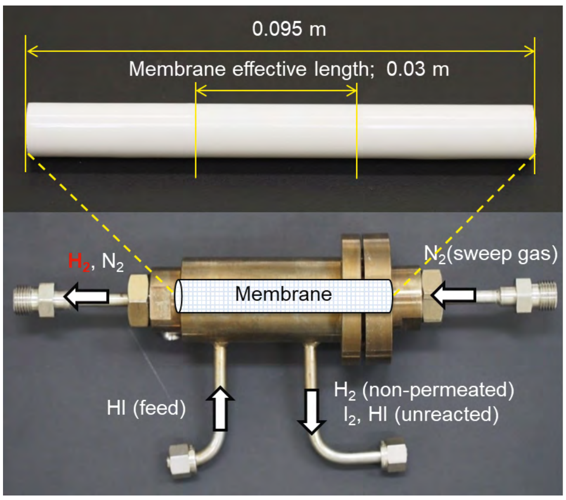
Thanda, V. K. et al. (2020).

kinetic models that describe the sulphur-trioxide decomposition, and these were evaluated. The kinetic models were found to be strongly limited with respect to their application range and several influencing factors such as active catalyst surface area, catalyst loading, and operating pressure – however, equilibrium effects were not considered by the kinetic models. Then, the kinetic models were implemented in a discretized model of the SO_3 decomposition section of the particle heated laboratory reactor. The simulation results were used to evaluate the reactor design, to find favourable operating conditions for the upcoming experiments and to develop the evaluation method for the experiments. The influence of parameters such as gas temperature at the catalyst inlet, the temperature difference between the particles and the gas, and the mass flow in the decomposer on the SO_3 decomposition were varied and the conversions calculated. With the simulation results, initial test plans for the upcoming experiments with a H_2SO_4 decomposition reactor were developed. Furthermore, the evaluation method for the experiments was defined based on the model equations for the decomposer. The evaluation method was implemented in software modules to prepare the interface of the evaluation method to the overall measurement and control software.

The JAEA has continued working on essential R&D tasks of the S-I process to verify the integrity of components made of practical structural materials and the stability of hydrogen production operation in harsh working conditions. For stable hydrogen production, technical issues for instrumental improvements (stable pumping of HI-I2-H2O solution, prevention of leakage, prevention of I2 precipitation) were resolved. In parallel, the JAEA has also focused on the development of membranes and a separation materials database, including selection development and performance assessments of separation techniques and materials. If hydrogen can be separated effectively from product gases such as HI and I2 without phase change, the thermal efficiency of the total IS process would increase and the cost of hydrogen would decrease. Therefore, the successful development of a hydrogen-separation membrane for the hydrogen iodine (HI) molecule decomposition is significant. The objective is to test the membrane and to investigate separation performance in HI decomposition. Silica ceramic membranes were selected for their thermal and chemical stability, thickness control capability, access to high permeation flux and high selectivity, as well as their feasibility of controlling the porosity structure. The JAEA succeeded in the preparation of silica ceramic membranes and the demonstration of a lab-scale catalytic membrane reactor for HI decomposition, as shown in Figure VHTR-10.

The Korean government released two roadmaps in 2019: 1) "Hydrogen Economy Roadmap" to drive a new growth engine and turn Korea into a society fuelled by eco-friendly energy; and 2) "Hydrogen Technology Development Roadmap" for technology development across ministries to support the implementation of the hydrogen economy by enhancing domestic technological competitiveness in the hydrogen energy sector. The establishment of these roadmaps has provided impetus to activities on hydrogen production. KAERI has conducted simulations on coupling various hydrogen production processes to a 350 MWth HTGR. Hydrogen production processes include steam methane reforming, HTSE and the S-I process. KAERI has been planning a new project related to nuclear hydrogen production, focused on the integration of HTSE and a high-temperature system, and the development of an analysis of the coupling of the reactor and HTSE system. KAERI is considering use of the available helium loop facility for the integral test, which operates at 600 kWe and 950°C (see Figure VHTR-11).

Figure VHTR-10. A lab-scale catalytic membrane reactor for HI decomposition

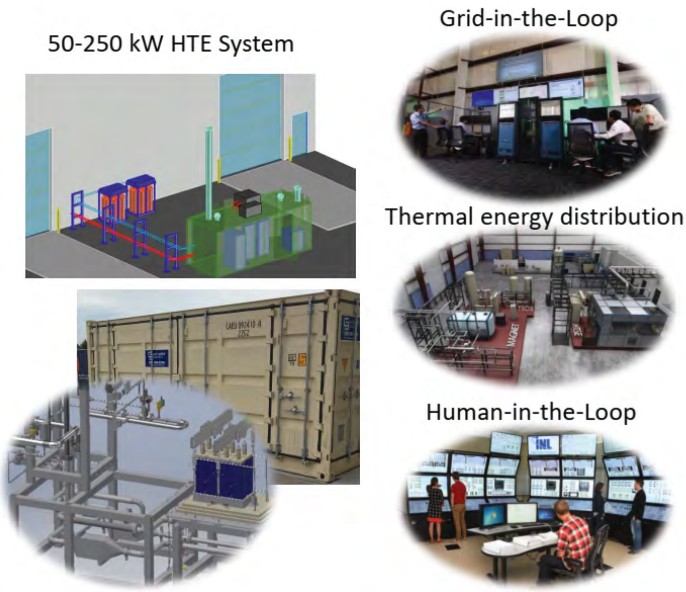


Myagmarjav, O. et al. (2019).

Figure VHTR-11. Component scale helium gas loop

The United States' efforts have continued on the development and application of HTSE for hydrogen production in the context of being a dynamic and flexible part of an integrated nuclear energy system. Advanced reactors and renewable energy sources will provide heat and electricity for this integrated system, supporting the production of hydrogen and transport fuel, the electric grid, industrial needs, clean water production and new chemical processes. Advanced nuclear reactor systems under consideration would range from micro reactors (1 to 20 MW) and small modular reactors (20 to 300 MW) to full-size reactors (300 to 1 000 MW).

The objectives with respect to the HTSE have been to verify operation of solid oxide electrolysis cell (SOEC) stacks from US suppliers, qualify them for use in nuclear hydrogen demonstrations and benchmark stack performance under laboratory environment for industrial applications. In this effort, a 25 kWe HTSE test facility was commissioned with some 1 000 initial tests on a 5 kWe stack. Remote supervisory control of stack operation, including multiple voltage-current sweeps, has been conducted. There is also a plan being developed for the demonstration of a 250 kWe integrated HTSE system (see Figure VHTR-12).



Boardman, R. (2020).

Figure VHTR-12. 250 kWe integrated HTSE system development

Computational methods validation and benchmarks

The computational methods validation and benchmarks (CMVB) project was restarted in 2014. From 2015 to 2020, a total of 11 meetings were organized by the CMVB pPMB and held in turn in different participating countries. The main activities resulting from these meetings included discussions and confirmation of the research tasks in each work package (WP), review and approval of the draft project plan, of which the final version is the indispensable annex of the project arrangement (PA), discussions on some common topics and potential test facilities that will be fundamental resources of this project, and the process and guidelines to launch the PA. To date, the PP has been approved by all SSC members. In 2020, confirmation of the PA was under processing by each signatory (China, the EU, Japan, Korea and the United States). Signatories are ready to pursue the signature for the PA.

Because of the impact of COVID-19, the originally planned 22nd CMVB pPMB meeting in the spring of 2020 (to be held in China) was postponed. Instead, a video conference was held online. Considering the status of the CMVB PA and PP, the main objective of the meeting was to review the CMVB pPMB and receive updates on the status of CMVB R&D work from each participant. The United Kingdom showed interest in joining the project. The United Kingdom signed the GIF VHTR system arrangement in 2019. UK representatives were invited as observers to discuss potential interest in collaborating within GIF, and these representatives presented the UK's VHTR CMVB activities, identifying what the United Kingdom could potentially contribute to the current project. CMVB members are now ready to sign the PA. Through pPMB meetings, past, current and new test facilities and projects have been identified, proposed and confirmed as fundamental resources for the development and assessment of codes and models covering HTR physics, TH, CFD, fission product transport, plant dynamics, etc.

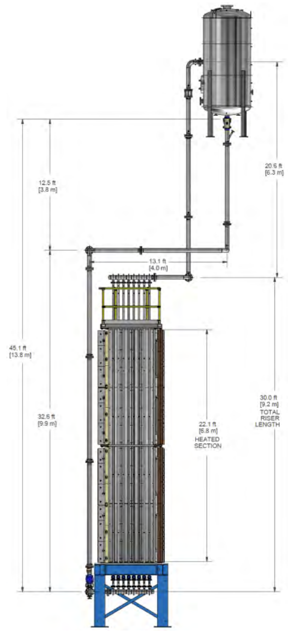
Table VHTR-1. Work Package titles of the VHTR CMVB PMB

WP No.	WP title	Lead
1	Phenomena identification and ranking table (PIRT) methodology	DOE (US)
2	Computational fluid dynamics (CFD)	INET (CHINA)
3	Reactor core physics and nuclear data	DOE (US)
4	Chemistry and transport	INET (CHINA)
5	Reactor and plant dynamics	INET (CHINA)

In China, the HTR-PM demonstration project is in its commissioning stage. In 2020, the reactor pressure vessel, the steam generator pressure vessel and the hot gas duct pressure vessel were connected in both of the two NSSS modules. The two modules underwent helium leak detection of the pressure boundary and cold tests, including strength and leakage tests. Then, the primary circuits were heated by the helium blower. On 30 December, the two modules achieved the status of 250°C and 7 MPa, under which the hot test could be performed. The first fuel loading is planned for the first half of 2021. The design of the HTR-PM600, a

600 MWe commercial plant, was pursued in 2020. The emphasis of the HTR-PM600 project in 2020 was on the feasibility study, preliminary design and preparation of the PSAR. Regarding HTR-10, the in-core temperature measurement experiment was conducted and completed after it was restarted, with the purpose of determining temperatures inside the fuel elements (see Figure VHTR-14). The experimental results have been summarized and used to support the safety review of the HTR-PM. In the Institute of Nuclear and New Energy Technology (INET), the self-reliant HTR design software package, covering the fields of reactor physics, thermal hydraulics and source term analysis, is under development and assessment. Comprehensive verification and validation (V&V) was carried out in 2020 for the in-house version of the domestic codes, using the test data or benchmark cases defined, based on the HTR-10, HTR-PM, AVR, Proteus, ASTRA, etc. The domestic codes are supposed to be used in the design verification of HTR-PM600 as the first step of their application.

In the EU, most of the current (V)HTR-related activities are taking place in the Euratom Horizon 2020 project GEMINI+, which is supporting the demonstration of an HTGR nuclear cogeneration system. The outcome will be submitted to the CMVB PMB as the GIF contribution, in addition to the (V)HTR-related projects in past Euratom Framework Programmes (and EC national projects). Some additional, national HTR-related projects will deliver contributions to the CMVB: the Polish projects GOSPOSTRATEG – HTR (2019-2022), and the NOMATEN Centre of Excellence (2019-2026).

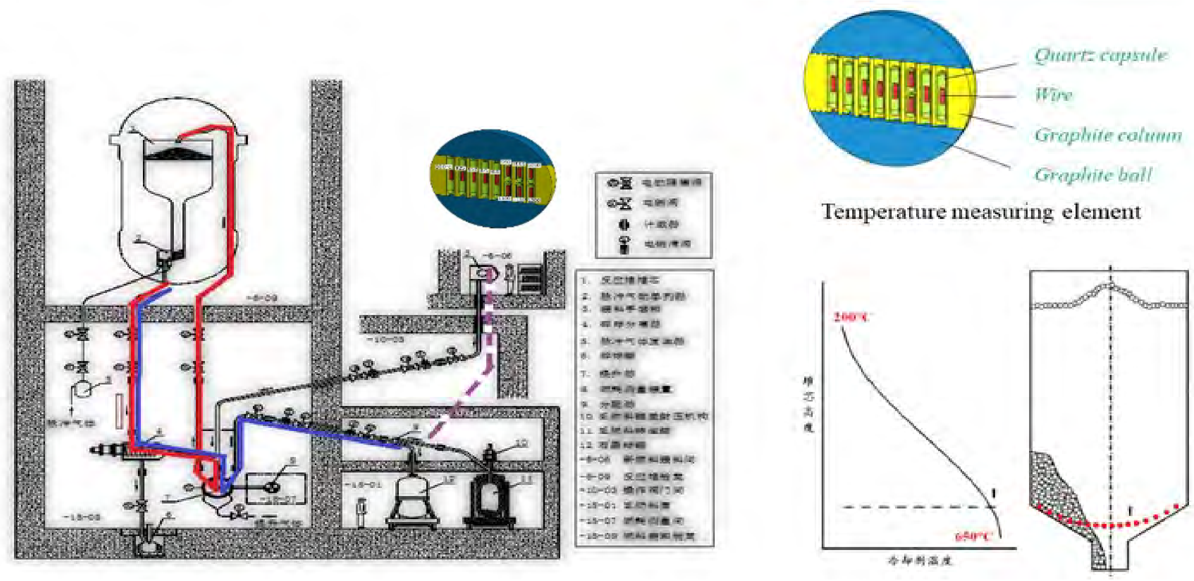


Zhang, Y. et al. (2021).

Figure VHTR-13. Schematic diagram of water-cooled natural convection shutdown heat removal test facility

The JAEA is making a strong effort to restart the HTTR. It is developing and benchmarking various models and analysis methodologies, as well as codes for reactor physics, thermal fluids, etc. JAEA R&D is expected to support planning of the CMVB co-operative activities, such as a benchmark activity using ATR irradiation data. Consequently, the JAEA defined a provisional calculation problem to verify the geometrical model for MVP code, based on an ATR critical experiment in 2020. R&D on a system analysis code based on RELAP5/MOD3 code was

Figure VHTR-14. Schematic diagram of in-core temperature measurement experiment of HTR-10[1]



Lisowski, D. et al. (2017).

1. Launch temperature measuring elements from the top of the core, then start to shuffle the fuel elements and temperature measuring elements to a certain position. After running the reactor at 3 MW for 10 hours, then shutdown the reactor and discharge the elements. If the temperature measuring element is identified by the X-ray machine, it will be removed from the discharged loop for further analyses.

performed for transient thermal-hydraulic behavior in a prismatic-type VHTR. The flow distribution analysis model and molecular diffusion model were newly developed and validated for the code to simulate key thermal-hydraulic phenomena in a prismatic-type VHTR. Based on this study, the JAEA will explore the possibility of further collaboration, such as the launch of a new benchmark task in CMVB.

The VHTR R&D program in Korea aims at improving high-temperature system key technologies in terms of the design code development and assessment, and also high-temperature materials performance verification. With regard to its VHTR program, the five-year project on the development of HT system key technologies was launched in 2020, with the aim of developing the HT system performance evaluation technology and verifying the materials performance for VHTR. Some specific CMVB related R&D activities include scale-down standard fuel block tests to validate CORONA code, cross section generation based on triangular node in DeCART2D code and simulation of the total control rod withdrawal transient for PBMR400 benchmark problems by using the neutronics and system code coupled system.

Regarding CMVB more specifically, neutronics code improvement is underway to predict power distribution precisely. Neutronics/system code coupled calculations have been updated to enhance the thermal margin. Fission product transport from fuel to containment will be assessed under normal and accident conditions. An HTSE system analysis/experiment project will be launched in 2021.

In the United States, the latest progress was from metals and TRISO fuel-related activities. Alloy 617 has been approved by the ASME for inclusion in its boiler and pressure-vessel code. This means the alloy, which was tested by Idaho National Laboratory (INL), can be used in proposed molten salt, HT, gas-cooled, or sodium reactors. It is the first new material to be added to the code in 30 years. The UCO TRISO fuel performance topical report submitted by the Electric Power Research Institute (EPRI) is being reviewed and approved by the regulator (i.e. NRC). In addition, modelling of the HT test facility (HTTF) with RELAP5-3D was performed, and the two-phase testing at water-cooled Natural Convection Shutdown Heat Removal Test Facility (NSTF) was ongoing.

Michael A. Fütterer
Chair of the VHTR SSC, with contributions from VHTR members

5 Methodology working groups

Economic Modelling Working Group

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

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

In 2007, the EMWG published cost estimating guidelines and the Excel-based software package, G4ECONS v2.0, for the calculation of two figures of merit: the levelized cost of energy and the total investment cost, and to assess Gen-IV systems against GIF economic goals. These EMWG tools were made available to the public through the GIF Technical Secretariat, which has resulted in several publications that demonstrate the EMWG methodology for the economic assessments of Gen-III and Gen-IV systems, as well for 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 Programme (HEEP), and the results have been published in peer-reviewed publications. Lessons learnt from the benchmarking exercise and from the feedback of users has informed the refinement of the G4ECONS tool. The EMWG released the new version, G4ECONS v3.0, with an improved user interface, in October 2018.

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

Since October 2016, the EMWG has been working collaboratively with the Generation IV International Forum's Senior Industry Advisory Panel (SIAP) to investigate challenges and opportunities for deployment of Gen-IV systems in electricity markets with a significant penetration of renewable energy resources and to produce a position paper for the Policy Group. An abridged version of the EMWG position paper on the impact of increasing shares of renewables on the deployment prospects of Gen-IV systems was presented at the 4th GIF Symposium (2018) and an executive summary was posted on the GIF website (www.gen-4.org/gif/jcms/c_117863/2018-gif-symposium-proceedings). The study found that Gen-IV systems need to be more flexible compared to current reactors for deployment in low-carbon energy systems, and such requirements are already being proposed by the utilities. Large-scale energy-storage and cogeneration applications, for example, would allow flexible dispatch while ensuring high-capacity utilization. Nuclear-renewable hybrid energy systems with adequate energy-storage and cogeneration applications could, in this way, meet flexible demands from the grid while operating power generators at full capacity to ensure overall economically viable operation. However, such flexibility considerations impose additional requirements on the R&D of Gen-IV systems.

Activities in 2020

In 2020, the EMWG identified two priorities for its 2021 work program:

Advanced nuclear technology cost reduction strategies and a systematic economic review: The EMWG will evaluate nuclear cost reduction strategies based on past/current lessons learnt, along with assessments of readiness levels for the technologies and the potential for cost reduction. Key areas for nuclear cost reduction and enabling technologies will be researched under design, and construction/production, as well as project management. EMWG members will research specific strategies and technologies (e.g. functional containment, advanced concrete, machine learning) to assess cost reduction potential, applicability to Gen-IV technologies and technology readiness, as well as to identify further RD&D that may be needed to advance the strategy. This activity will develop a GIF systematic economic review process, where cost reduction strategies will be shared among GIF members (via the ETWG) and used for training and publication purposes. Results and the methodology developed can inform the design and selection of future cost reduction demonstration projects. Information and updates on cost reduction strategies and the study outcomes will be posted on an online repository ("Nukipedia").

The paper will outline a systematic economic review process to:

- identify opportunities/conditions for cost reduction under the categories of design, construction/ production, and project management, emphasizing cost reductions for the balance of plant, with varying applicability to all Gen-IV technologies;
- provide a methodology to review progress in designs towards reducing costs;
- inform and provide training on cost reduction strategies for reactor designers and other stakeholders.

Advanced nuclear technology private financing: The EMWG has established a working group of financing experts to identify the changes that need to be made to international, low-carbon, sustainability principles in order to enable private sector financing of nuclear power, and particularly Gen-IV technologies.

The paper will consider:

- the enablers required to facilitate private financing of nuclear projects;
- the risks associated with nuclear projects, how such risks are mitigated and how these risks may vary (if at all) with Gen-IV technologies (adapted from existing materials);
- an assessment of international regimes on sustainable financing, including:
  - why nuclear projects are sustainable developments;
  - the environmental, social and corporate governance ("ESG") metrics (i.e. those used by investors to assess companies and projects to determine whether an institution should be invested), and what needs to change, if anything, to create a level playing field for energy technologies and to ensure that nuclear energy is assessed in line with other energy projects;
  - the various taxonomies and how nuclear power meets the "do-no-harm" principle;
  - the Green Bond Principles, and how nuclear meets these principles.

David Shropshire
Co-Chair of the EMWG

Megan Moore
Co-Chair of the EMWG

Fiona Reilly
Co-Chair of the EMWG

Education and Training Working Group

Background/terms of reference

GIF's Education and Training Working Group (ETWG) started as a task force in November 2015 and was elevated to a working group in 2020. It serves as a platform to enhance open education and training (E&T), as well as communication and networking of people and organizations in support of the Gen-IV International Forum. Several objectives of the working group focus on promoting E&T by developing the webinar series dedicated to Gen-IV systems and related cross-cutting topics, advertising these at the international level, converting all the archived GIF webinars to videos, and creating and maintaining a modern, social media platform (such as LinkedIn[1]) to exchange information and ideas on GIF R&D topics, as well as on related GIF E&T activities.

Main achievements

Numerous tools exist today designed to increase knowledge on a specific study. There is the traditional curriculum developed at universities for undergraduate and graduate students. Distance learning is also used at different universities and is undertaken via various technological media where the training course can be delivered simultaneously to students off campus. Massive open online courses (MOOCs) are free online courses aimed at unlimited participation and open access via the Internet. Because of its easy access, the ETWG decided to create a series of webinars, exploiting this modern Internet technology, so as to reach a broader audience long before the pandemic obliged organizations to more widely use this technology. To promote training in Gen-IV systems and to ensure that a knowledgeable workforce exists, GIF's ETWG therefore created and made available to the public since September 2016 a series of webinars on topics specific to advanced reactor systems. These webinars are intended to be of interest not only to students currently pursuing formal education in universities but also to those already in the workforce, who may be in need of a refresher course or a better understanding of a specific topic, but most importantly to the more general public. GIF is therefore developing and proposing world-class webinars that will also be useful for people like quality assurance officers, data validators, technicians, managers, regulators and others who may benefit from an enhanced understanding of advanced reactor concepts in their work. Forty-eight webinars have been developed thus far (See Table ETWG-1), recorded and archived, and can be found at www.gen-4.org/gif/jcms/c_82831/webinars.

Involving the junior workforce is a priority for the ETWG, and consequently the winner of the American Nuclear Society (ANS) 2019 "Pitch your PhD competition", Dr Coddy Wiggins, was invited to present a webinar in December 2020 entitled "Development of Multiple-Particle Positron Emission Particle Tracking for Flow Measurement".

As depicted in Table 1, a total of 12 webinars were presented and archived in 2020, with subjects varying from Gen-IV reactor systems and fuels to the sustainability of the fuel cycle. Each presenter is a renowned expert on the subject matter and is internationally recognized as being so.

Twelve webinars are planned in 2021 and are displayed in Table ETWG-2. The presenter scheduled for December 2021 will be the winner of the first "Pitch your Gen-IV research competition", and will be announced at the next Experts Group/Policy Group meeting scheduled for May 2021.

Table ETWG-2. GIF Webinar Series (January 2021 to December 2021)

Presenter	Title of webinar	Webinar presentation
Dr. Nathalie Chauvin, CEA France	MOX fuel for advanced reactors	January 2021
Dr David Peeler, PNNL, US	Overview of waste treatment plant, Hanford site	February 2021
Prof. Nawal Prinja, JACOBS, UK	Introducing new plant systems design (PSD) code	March 2021
Mr Etsuo Ishitsuka, JAEA Japan	Experience of HTTR licensing for Japan's new nuclear regulation	April 2021
Dr Isabella Van Rooyen, INL, US	Advanced manufacturing for Gen-IV reactors	May 2021
Dr François Baque, CEA, France	In-service inspection and repair developments for SFRs and extension to other Gen-IV systems	June 2021
Ms Jessica Lovering, Carnegie Mellon University, US Winner of the ANS 2020 Pitch your PhD competition	Evaluating changing paradigms across the nuclear industry	July 2021
Mr Vince (Alois) Chermak, INL, US	Comparing and contrasting approaches to quality assurance for nuclear applications	August 2021
Dr Julia Kyzina, IPPE, Russia	Experimental R&D in Russia to justify sodium fast reactors	September 2021
Dr John Vienna, PNNL, US	Nuclear waste management strategy for molten salt reactor systems	October 2021
Dr Jun Wang, University of Wisconsin, Madison US	Geometry design and transient simulation of a heat pipe micro reactor.	November 2021
1st Winner of the "Pitch your Gen-IV research 2021 competition"	To be determined.	December 2021

1. www.linkedin.com/groups/8416234.

Table ETWG-1. The GIF webinar series, presented and archived between 2016 and 2020

	2016 (4 webinars)	2017(12 webinars)	2018(8 webinars)	2019(12 webinars)	2020(12 webinars)
Introduction	**Atoms for peace** –John Kelly, US **Introduction to nuclear reactor design** – Claude Renault, France			**European sodium fast reactor: An Introduction** – Konstantin Mikityuk, Switzerland	
Gen-IV systems	**Sodium-cooled fast reactor** – Bob Hill, US	**Lead fast reactor** – Craig Smith, US **Gas-cooled fast reactor** – Alfredo Vassile, France **Very-high-temperature reactors** – Carl Sink, US **Supercritical water reactors (SCWR)** – Laurence Leung, Canada **Fluoride cooled-high-temperature reactors** – Per Peterson, US **Molten salt reactors** – Elsa Merle, France	**MYRRHA: An accelerator-driven system based on LFR technology** – Hamid Ait Abderrahim, Belgium **Molten salt actinide recycler & transforming system with and without Th-U support: MOSART** – Victor Ignatiev, Russia	**Lead containing mainly isotope Pb-208: New reflector for improving safety of fast neutron reactors** – Evgeny Kulikov, Russia **Gen-IV coolants quality control** – Christian Latge, France **Czech experimental programme on MSR technology development** – Jan Uhlir, Czech Republic	**GIF VHTR Hydrogen Production Project Management Board** – Sam Suppiah, Canada **Thermal hydraulics in liquid metal fast reactor** – Antoine Gerschenfeld, CEA, France **Micro reactors: A technology option for accelerated innovation** – D.V. Rao, US **Overview of small modular reactor technology development** – Frederik Reitsma, IAEA
Operational experience		**Feedback Phenix and Superphenix** – Joel Guidez, France	**Design, safety features and progress of HTR-PM** - Yujie Dong, China **ASTRID: Lessons learned** – Gilles Rodriguez, France **Advanced Lead Fast Reactor European Demonstrator (ALFRED) project** – Alessandro Alemberti, EC **Russia BN 600 & BN 800** – Ilya Pakhomov, Russia	**Safety of Gen-IV reactors** – Luca Ammirabile, EC **The ALLEGRO experimental gas-cooled fast reactor project** – Ladislav Belovsky, Czech Republic **Passive decay heat removal** – Mitchell Farmer, ANL US	**Molten salt SFR safety design criteria (SDC) and safety design guidelines (SDGs)** – Shigenobu Kubo, JAEA, Japan **Reactor safety evaluation: A US perspective** – David Holcomb, ORNL, US
Gen-IV cross-cutting topics		**Energy conversion** – Richard Stainsby, UK **Estimating costs of Gen-IV systems** – Geoffrey Rothwell, NEA	**Materials challenges for Gen-IV reactors** – Stu Maloy, US **Proliferation resistance and physical protection of Gen-IV reactor systems** – Robert Bari, US		**Maximizing clean energy integration: The role of nuclear and renewable technologies in integrated energy systems** – Shannon Bragg-Sitton, INL, US **Global potential for small and micro-reactor systems to provide electricity access** – Amy Schweikert, US **Neutrino and Gen-IV reactor systems** – Jonathan Link, US
Fuel types		**General considerations on thorium as a nuclear fuel** – Franco Michel-Sendis, NEA **Metallic fuels for SFRs** – Steven Hayes, US		**Advanced gas reactor TRISO particle fuel** – Madeline Feltus, USA	**Performance assessments for fuels and materials for advanced nuclear reactors** – Daniel LaBrier, ISU, US
Sustainability and the fuel cycle	**Closing the fuel cycle**– Myeung Seung, Korea	**Sustainability: A relevant approach for defining future nuclear fuel cycles** – Christophe Poinssot, France		**Scientific and technical problems of closed nuclear fuel in two-component nuclear energetics** – Alexander Orlov, Russia	**Comparison of 16 reactors' neutronic performance in closed Th-U and U-Pu cycles** – Jiri Krepel, PSI, Switzerland
Winners of the Pitch 2018 competition				**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 the frame of sodium-cooled fast reactors** –Fang Chen, France	**Development of multiple particle positron emission particle tracking for flow measurement** – Cody Wiggins, VCU, US

As of December 2020, attendance during the live webcasts totalled 3 990. It is worth mentioning that attendance during the calendar year (CY) 2019 was 2 179. The number of viewings of the recorded webinars in the online archive is 5 657, a strong increase when compared to attendance in CY 2020, which totalled 3 747. Total viewing over the four-year period was 9 647.

Participants in GIF webinars include representatives from organizations such as federal agencies, national laboratories, state agencies, universities, international organizations, contractors and commercial organizations. Figure ETWG-1 presents a comparison of GIF webinar attendance distribution for the 36 webinars presented at the end of CY 2019 against the 48 webinars presented in CY 2020. The figure shows an increase of viewing by international organizations (i.e. 35% of viewers were from international organizations in CY 2019, and this figure increased to 54% in CY 2020).

The increase of international participation is also reflected by an increase in the number of countries that are participating in either the live webinar presentation or watching the recorded webinars (Figure ETWG-2)

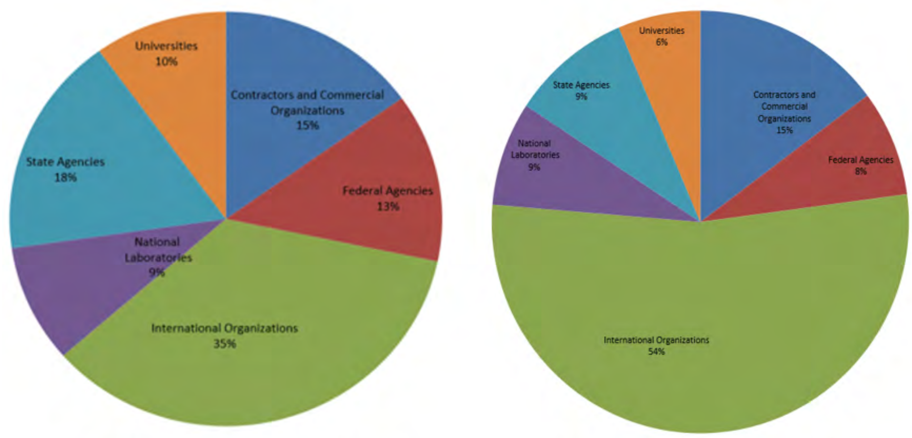
GIF continues its efforts to advertize the webinars by presenting them at different venues. A paper summarizing the ETWG's activities, entitled: "Gen-IV Education and Training Working Group Webinar Initiative" was presented at the virtual American Nuclear Society winter meeting (16-19 November 2020), paper No. 32874.

To facilitate the advertisement of the webinars, a handout containing a list of all the webinars that have been presented, as well as those proposed in future, has been created and is available during each live presentation. Participants can download the flyer as a PDF file (see Figure ETWG-3).

Looking ahead

The ETWG is planning to organize a "Pitch your Gen-IV research competition" that will be launched on 1 February 2021 (see Figure ETWG-4). The

Figure ETWG-1. Comparison of participants by organization type in 2019 and 2020



1a: GIF Webinar attendance distribution in 2019 1b: GIF Webinar attendance distribution in 2020

Figure ETWG-2. Comparison of international participation in the GIF webinar series

October 2019

Participants from **20 countries** have listened to at least one webinar

May 2020

Participants from **41 countries** have listened to at least one webinar

December 2020

Participants from **58 countries** have listened to at least one webinar

Figure ETWG-3. Handout advertising the GIF webinar series

Figure ETWG-4. Flyer announcing the "Pitch your Gen-IV research competition"

Pitch your Gen-IV research competition will be open to: a) currently enrolled PhD students and b) post-doctoral fellows and junior researchers who defended their PhD after 1 January 2019. The research must be related to GIF advanced nuclear energy systems and could be either an independent research project or one that concerns a research mentor. Participants will be asked to submit a short executive summary, and the 25 pre-selected candidates will be invited to record a three-minute video pitch of their project.

The expected schedule is as follows:

- 1 February 2021 – Executive summary submission opens
- 28 February 2021 – Executive summary submission closes
- Mid-March 2021 – 25 finalists selected
- 31 March 2021 – Video submission due
- 1 April 2021 – Popular voting begins
- 30 April 2021 – Popular voting ends
- May 2021 – Winners announced at the EG/PG meeting

Patricia Paviet
Chair of the ETWG, with contributions from ETWG members

Proliferation Resistance and Physical Protection Methodology Working Group

The Proliferation Resistance and Physical Protection Working Group (PRPPWG) was established to develop, implement and foster the use of an evaluation methodology so as to assess Gen-IV nuclear energy systems with respect to the GIF PR&PP goal, whereby:

> "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 methodology provides designers and policymakers with a technology-neutral framework and a 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&PP robustness of system concepts throughout their development cycle. The working group released the current version (revision 6) of the methodology for general distribution in 2011,[1] and Japanese and Korean translations of the methodology report have been produced for national use.

Since 2018, the main focus of the PRPPWG has been on updating the white papers on proliferation resistance and physical protection robustness of the six GIF design concepts. This is a joint effort with the System Steering Committees (SSCs) and provisional System Steering Committees (pSSCs) of the six Gen-IV technologies. The first versions of these white papers were produced in the period

Table PRPP-1. System designs considered in the white paper updates

GIF System	System options considered in the update	Design tracks considered in the update	Comments
GFR	Reference concept	2400MWt GFR ALLEGRO as a GFR demonstrtor (EU)	Other Gen- IV designs include: EM2 (GA) ALLEGRO (V4G4) High-energy neutron modular helium reactor (HEN MHR) (CEA-ANL and GA-AREVA)
LFR	Large system	ELFR, (EU))	These are the three reference design configurations discussed in the GIF-LFR System Research Plan
	Intermediate system	BREST-OD-300, (RF)	
	Small transportable	SSTAR, (US)	
MSR	Liquid-fuelled with integrated salt processing	MSFR (EU), MOSART (RF)	There is a wide variety of MSR technologies, encompassing thermal/fast spectrum reactors, solid/ fluid fuel, burner/breeder modes, Th/Pu fuel cycles, and on-site/ off-site fissile separation.
	Solid-fuelled with salt coolant	Mk1 PB-FHR (US)	
	Liquid-fuelled without integrated salt processing	IMSR (Canada)	
SCWR	Pressure vessel	HPLWR (EU) (thermal)	Most concepts are based on "familiar' technology, such as light water coolant, solid fuel assemblies, and batch refuelling. Implementation of Th and Pu fuel cycles creates additional, special nuclear materials of concern.
		Super FR (Japan)	
		Super LWR (Japan) (thermal)	
		CSR 1000 (China) (thermal)	
		Mixed spectrum (China)	
		Fast core (RF)	
	Pressure tube	Canadian SCWR (Canada) (thermal)	
SFR	Loop configuration	JSFR (Japan)	Expect key PR&PP issues to be tied to fuel handling, TRU inventory and fuel cycle options.
	Pool configuration	ESFR (EU), BN-1200 (Russia), KALIMER-600 (Korea)	
	Small modular	AFR-100 (US)	
VHTR	Prismatic fuel block	Modular HTR, Framatome (ANTARES)	SC-HTGR is a follow on of the ANTARES and the GA GT-MHR development. Expect some PR&PP differences between the prismatic block and pebble bed design.
		SC-HTGR, Framatome (US)	
		GT-MHR General Atomics (US)	
		GT-MHR OKBM (Russia)	
		GTHTR300C, JAEA (Japan)	
		NHDD, KAERI (Korea)	
	Pebble bed	Xe-100, X-Energy (US)	
		HTR-PM (China)	

1. www.gen-4.org/gif/jcms/c_40413/evaluation-methodology-for-proliferation-resistance-and-physical-protection-of-generation-iv-nuclear-energy-systems-rev-6

2008-2011. The white papers were included as individual chapters of an integrated report that was published in 2011 and is available on the GIF website.[2] The papers are being updated according to a revised, common template. The current update reflects changes in designs, new tracks that were added (see Table PRPP-1) and advancements in the designing of the six GIF systems, with enhanced intrinsic PR&PP features.

Individual white papers, after endorsement by both the PRPPWG and the responsible SSC/pSSC, will be transmitted to the EG for approval and published as GIF documents. Cross-cutting PR&PP aspects that transcend all six GIF systems are also being investigated. The plan is to complete the white paper updates in 2021. Below is a summary status of each white paper as of the end of 2020.

- LFR – paper endorsed by the PRPPWG and the LFR pSSC; EG completed review.
- SFR – paper endorsed by the PRPPWG and the SFR SSC; EG completed review.
- SCWR – paper incorporated comments from the SSC and endorsed by the PRPPWG; awaiting endorsement by the SSC.
- MSR – paper incorporated comments from the pSSC and endorsed by the PRPPWG; awaiting endorsement by the SSC.
- VHTR – PRPPWG presented a draft of the white paper at the last VHTR SSC meeting (October 2020); the final draft is being reviewed and revised by the PRPPWG before being released to the SSC for endorsement.
- GFR – early draft reviewed by the PRPPWG and the GFR SSC; a final draft is under preparation to incorporate comments from the PRPPWG and the GFR SSC.

The PRPPWG holds monthly teleconferences to report on the progress of group and member activities. In 2020, Russia appointed Mr Vladimir Artisyuk from Rosatom as a new member of the PRPPWG.

The group maintains an annually updated bibliography of official publications, of publications referring to the PR&PP methodology and of relevant issues. The latest edition, revision 7, was published in March 2020. It is available on the GIF website.[3]

The PRPPWG maintains regular exchanges with the IAEA International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) and the agency's Department of Safeguards. An IAEA representative participates regularly in the PRPPWG activities. The PRPPWG made a presentation at the 14th GIF-IAEA Interface meeting on 8 July 2020, highlighting collaboration on the INPRO PR methodology that the IAEA plans to update, as well as on emerging safeguards issues related to the deployment of small modular reactors and micro reactors. Collaboration with the RSWG was strengthened through personal exchanges at each group's meeting. PRPPWG representatives attended the 31st and 32nd meetings of the RSWG. A discussion was initiated to explore the interfaces between safety, security and safeguards.

The group held its 31st annual meeting on 9-11 December 2020 via video conference. All member countries, including, for the first time, China and South Africa, attended the meeting and delivered country reports. Representatives from the IAEA and the RSWG also participated. The meeting was dedicated to discussing the advancement of the white papers, planning of new activities – namely cross-cutting topics from the white papers – and developing the work plan for the period 2021-2022.

Giacomo Cojazzi
Co-Chair of the PRPPWG

Lap-Yan Cheng
Co-Chair of the PRPPWG

2. www.gen-4.org/gif/jcms/c_40414/proliferation-resistance-and-physical-protection-of-the-six-generation-iv-nuclear-energy-system.

3. www.gen-4.org/gif/jcms/c_101559/gif-prppwg-bibliography.

Risk and Safety Working Group

The Risk and Safety Working Group (RSWG) has been an active methodology working group since 2005, with a mission to establish a harmonized approach to, and provide assessment tools for, the risk and safety of Gen-IV systems. RSWG membership currently includes representatives from Canada, China, the European Union, France, Japan, Korea, South Africa, Russia, the United Kingdom and the United States as a forum of advanced reactor designers and regulators. The IAEA also participates as an observer. Prior to 2020 the RSWG:

- proposed a set of broad safety principles and attributes based on GIF safety and reliability goals as input to R&D plans for specific Gen-IV design tracks;
- developed a technology-neutral, comprehensive, integrated safety assessment methodology (ISAM) as a toolkit to evaluate risk and safety for all six systems based on a consistent framework, and supported its implementation for specific Gen-IV design tracks;
- established technical interfaces with the IAEA and the NEA Working Group on the Safety of Advanced Reactors (WGSAR) under the NEA Committee on Nuclear Regulatory Activities (CNRA).

Ongoing RSWG projects include the development of white papers on: 1) a pilot application of ISAM to assess its impact on the design and licensing of select, Gen-IV design tracks; and 2) preparation of system safety assessments as summaries of high-level safety design attributes and remaining R&D needs. Both of these white papers will be elaborated in close co-ordination with the respective GIF System Steering Committees (SSCs). Published white papers and system safety assessment reports can be downloaded from the GIF RSWG web page: www.gen-4.org/gif/jcms/c_9366/risk-safety. Development of safety design criteria and guidelines for specific Gen-IV systems are also an ongoing collaborative effort between the RSWG and SSCs to establish the basic requirements for design, fabrication, construction, testing, and operation of Gen-IV systems.

As a result of the ongoing COVID-19 pandemic, the RSWG held both of its 2020 semi-annual meetings in virtual format through online meetings. A major RSWG accomplishment in 2020 was the update of the *GIF Basic Safety Approach* report (www.gen-4.org/gif/jcms/c_9366/risk-safety). The updated report captures post-Fukushima recommendations and requirements to ensure a level of safety for Gen-IV systems compatible with the expectations of safety authorities. It also provides common definitions for the plant states considered in a design and their alignment with levels of defence in depth, reinforces the independence of prevention/mitigation features at different defence-in-depth levels, and clarifies the concepts of design extension conditions and practically eliminated accidents.

Efforts also continued in 2020 towards a new RSWG-WGSAR joint initiative on the development of a risk-informed approach to the selection of licensing basis events and the safety classification of systems, structures and components. Novel aspects of numerous Gen-IV systems make the identification of hazards, initiating events and event sequences a challenge, requiring a systematic approach for their design and licensing. Critical examination of these designs, their safety behavior, and all aspects of their operations is key to addressing uncertainties, mainly due to initially limited information. The proposed risk-informed approach offers a process that combines both deterministic and probabilistic input in a complementary way for a systematic search of accident scenarios, which enables a classification of the responding plant equipment based on their risk significance.

It aims to establish event sequence categories that must be included in design assessments and reviews, integrate the deterministic inputs and risk insights so as to identify axnd classify initiating events and event sequences in each category, evaluate the event sequences against the regulatory criteria based on defined frequency-consequence targets, classify the plant equipment to identify risk-significant items, and define design-basis accidents and design extension conditions.

Having completed most of its missions, in 2019, SFR Safety Design Criteria Task Force (SDC TF) members joined the RSWG to contribute to the drafting of safety design criteria for other Gen-IV systems, including the lead- and gas-cooled fast reactors, as well as very-high-temperature reactors. The RSWG will inherit the remainder of work on updating the Safety Design Guidelines Report for SFR structures, systems and components based on comments received from the IAEA and WGSAR in October 2020. IAEA and WGSAR comments will be addressed and incorporated into the next version of the report in 2021. SDC TF members also support the implementation of the proposed risk-informed approach for GIF SFR design tracks in an effort to develop best-practice guidelines for its application and to ensure its consistency with the SFR SDC and *Safety Design Guidelines* (SDGs) completed in earlier years.

The RSWG, with the support of new SDC TF members, plans a more proactive participation in the new IAEA initiative on the development of safety standards for small modular reactors (SMRs) since non-LWR SMRs have many technical similarities to GIF systems. Reflecting GIF experience in the development of safety design criteria and guidelines for specific Gen-IV systems, in 2020, the RWSG already made some important contributions to the IAEA initiative by supporting the development of a safety approach/methodology for non-LWR SMRs, assessing the applicability of design requirements in safety considerations, and making direct contributions to the IAEA's new safety document: *Towards a Technology Neutral Nuclear Safety and Regulatory Framework: Applicability of IAEA Safety Standards to SMRs.*

Tanju Sofu
Chair of the RSWG, with contributions from RSWG members

6 Task force reports

Advanced Manufacturing and Material Engineering Task Force

In addition to being deployed as base-load power in large, centralized grids, nuclear power plants are increasingly likely to complement variable energy sources in distributed, localized and sometime remote grids. For nuclear energy to compete in this new paradigm, the industry must now focus on certain key characteristics. For example, designs with built-in, enhanced safety features and designs that focus on lower upfront capital costs, as well as those with shorter construction schedules, would allow nuclear power to obtain the social license required to operate and compete on the basis of overnight capital costs, and not simply based on the levelized cost of energy (LCOE). In addition, attributes such as non-electric applications (e.g. district heating, industrial process heat, clean hydrogen and synthetic fuels) and flexibility (e.g. load following) must now be considered for nuclear energy in these emerging, low-carbon energy systems.

Generation IV reactors are particularly suited to these requirements, and the last decade has seen a substantial rise in the number of active Gen-IV reactor designers and vendors worldwide. They typically involve:

- smaller, scalable designs, both in terms of size and output;
- simpler and compact modular designs that allow factory assembly and easier transport to construction sites;
- designs that focus on lower upfront capital costs and shorter construction schedules;
- designs with built-in enhanced safety features (passive/inherently safe designs);
- standardized designs to support volume production levels and a fleet approach to deployment;
- higher outlet temperatures and steam production for industrial applications (e.g. hydrogen production, water desalination, district heating, mining and resource extraction);
- designs with load-following flexibility to enable deployment in smart grids and hybrid energy systems.

Innovation in the nuclear supply chain, particularly in the areas of advanced manufacturing and materials engineering, is necessary if these advanced reactor technologies are to be delivered on time and on budget. However, nuclear design codes 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. Furthermore, current developments in advanced manufacturing are occurring much faster than the ability of most to introduce new materials and methods into design codes, potentially stifling innovation and hampering deployment. These issues need to be addressed if advanced reactors, integrating innovative materials and components, are to be brought to the market in reasonable time frames.

The GIF Advanced Manufacturing and Materials Engineering Task Force (AMME TF) was therefore formed in order to better characterize and address these issues. As an initial step, the task force undertook a survey investigating the status of advanced manufacturing for nuclear reactor development and construction. The main outcomes of the survey can be summarized as follows:

- most advanced manufacturing methods were considered opportunities by potential end users;
- the techniques identified as having the greatest potential were cladding and surface modification techniques, welding and joining, and additive manufacturing;
- a total of 90% of respondents identified the greatest obstacle to adoption as being the approval of codes and standards organizations.

The survey also highlighted the evidence for strong support in collaborating at the international level on:

- establishing codes and achieving regulatory acceptance;
- organizing joint workshops as a means to design, initiate and promote joint activities;
- collaborating on materials and component/structural performance assessments to enable regulatory acceptance.

A well-attended Advanced Manufacturing Workshop was held at the Nuclear Energy Agency in Paris in conjunction with the R&D Infrastructure Task Force (RDTF) meeting on 18-20 February 2020. The purpose of the workshop was to identify both areas and methods where collaboration could lead to a reduction in the time to deployment of advanced manufacturing for advanced reactors. Workshop participants, from reactor vendors, nuclear supply chain firms, regulators, national laboratories and R&D providers enabled broad representation from across the nuclear industry, and the cross-functional breakout sessions were particularly engaging and successful. The output from the six breakout groups was discussed in the final session of the workshop, and final conclusions were established. The overall recommendation of the workshop was that collaborative activities should be actively encouraged in three main areas:

- Qualification
  - codes and standards development;
  - a new qualification modality (e.g. real time process qualification);
  - an increased need for component testing.
- Demonstration and deployment
  - materials property database structure and content;
  - specific component testing;
  - round robin activity, e.g. generic intermediate heat exchanger (IHX) component.
- Design and modelling
  - collect experience and experimental data (feed data-driven methods);
  - share practices for inspection and design optimization;
  - resolve modelling and simulation benchmark problems.

There was strong support from the community for the AMME TF to continue its effort and organize follow-up workshops in due course. The future direction of the task force was discussed at the 43rd Experts Group and 49th Policy Group meetings. The AMME TF was encouraged to further develop its terms of reference (TOR), consistent with the workshop's recommendations.

Although progress in 2020 was affected by COVID-19, new TOR were drafted to refine the task force's objectives, define a new task force structure and provide action plans for the next 24 months. Following the 44th Experts Group and 50th Policy Group meetings, the reviewed TOR were approved in late 2020. Consequently, the task force will expand its activities and membership, and will conduct its activities through the three sub-working groups described below.

Requirements Capture Sub-Work Group

A key outcome of the February 2020 Advanced Manufacturing Workshop was the requirement to communicate and consult with the wider community to ensure the task force's success. The initial AMME TF survey was very successful in both identifying the needs of the community and raising awareness of the activities of the task force. These tools for requirements capture will thus be regularly used to update the task forces aims and outcomes. Community engagement will be pursued through regular targeted surveys enabling the task force to monitor stakeholder requirements, provide opportunities for stakeholder involvement and report progress on its activities. The next AMME TF survey is scheduled for 2021.

Qualification, Demonstration and Deployment Sub-Work Group

New approaches and methods for qualification of manufacturing processes, materials or components are key to the timely deployment of advanced manufacturing methods and materials. Although different AMME technologies may require different approaches, there are likely to be commonalities. Therefore, the first focus of the working group will be to identify these commonalities by sharing experience across different reactor systems, AMME technologies and national qualification approaches. The second focus of this working group is to elaborate on the qualification of specific components/materials or processes by studying their real or projected demonstration and deployment. As a final goal, a roadmap and guidelines for the development and implementation of qualified nuclear advanced manufacturing will be developed.

Design and Modelling Sub-Work Group

The February 2020 workshop underlined the need to capture and share processes and methodologies for ensuring product quality, and more specifically to: 1) collect experience; 2) share practices for inspection and design optimization, and 3) develop modelling and simulation benchmarks. This working group will consider three different categories of modelling: software and modelling assisted design, best practices for inspection and design optimization and organization of modelling and simulation benchmarks. In addition to requirements and the efficient use of modelling into materials and component accelerated development, qualification process has also been identified in the roadmap to meet these requirements.

Lyndon Edwards
Chair of the AMME TF, with contributions from AMME members

Research and Development Infrastructure Task Force

R&D infrastructure

Today's research infrastructure needs, from R&D to demonstration and deployment, cover major scientific equipment, scientific collections, structured information, information and communications technology (ICT)-based infrastructures. These facilities 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. A great deal of stakeholders are involved, from ministries to researchers and industry, with an underlying and growing use of e-infrastructure. Research infrastructures present opportunities for, and yet difficulties in, interactions between basic research and industry. Public and private funding appears to always be lacking, and individual countries do not have the critical mass or the dimensions to implement large research infrastructures. There is therefore a real need to co-operate on a broad international level. Substantial research, development and demonstration (RD&D) of systems' conceptual/detailed designs are needed, as are other analyses. Refurbishment and/or construction of research infrastructure and facilities are increasingly complex and costly. By identifying the latest R&D needs and mapping infrastructure, opportunities exist 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 the fuel cycle, fuel and materials irradiation, reactor safety, dedicated loops, mock-ups and test facilities, advanced simulation and validation tools, and transnational access to infrastructures, as well as the education and training (E&T) and knowledge management (KM) of scientists and engineers. GIF members strongly support a co-ordinated revitalization of nuclear RD&D infrastructures worldwide to a level that would once again help move forward in an accelerated manner a new generation of reactors.

Background/terms of reference

Background: At the 43rd GIF Policy Group (PG) meeting held on April 2017 in Paris, France, it was decided to establish the new GIF Task Force on R&D Infrastructure (RDTF). This task force accomplished its objectives over a short duration and took maximum benefit from the results through a dedicated workshop that was held in February 2020.

Objectives: Identify essential R&D experimental facilities needed for the development, demonstration and qualification of Gen-IV components and systems, including activities to meet safety and security objectives. To this end, the task force prepared relevant presentations and papers, and engaged with the private sector through a dedicated workshop. In the second phase, the task force promotes the utilization of the experimental facilities for collaborative R&D activities among GIF partners.

Organization: In 2019, the task force gathered and compiled from the six Gen-IV System Steering and provisional System Steering Committees their respective contributions in the area of infrastructures (existing infrastructure, needs and gaps). The year 2020 was focused on the second phase of this task force: "Promote the utilization of the experimental facilities for collaborative R&D activities among the GIF partners." To this end, existing mechanisms and approaches were identified, including organizational points of contact, to obtain access to relevant R&D facilities in GIF member countries. This information will then be made accessible to GIF participants and R&D organization, for example through the GIF website and the GIF members network. This action will promote closer NEA, GIF and IAEA international cooperation initiatives to stimulate joint funding from member countries and/or enterprises, as well as mutual benefits to be capitalized.

Main achievements in 2020

The completion of phase 1 and 2 of this task force led to two main actions in 2020/2021. Chronologically:

- Action no. 1: the GIF International Workshop with Nuclear Industry, which included SMR vendors and supply chain SMEs was organized successfully, with a total of 60 high-level participants, on 18-20 February 2020, at the NEA headquarters, in Boulogne-Billancourt, France. The first day and a half, the workshop was devoted to the topic of advanced manufacturing (see the AMME TF report in this chapter). The second half of the workshop was dedicated

Figure RDTF-1. GIF RDTF Workshop: "Views from the Private Sector, an Outlook for SMRs"

The discussion and questions session; from left to right: Dominique Hittner (USNC), Lou Martinez Sancho (CIO KAIROS Power), Stefano Monti (IAEA), Robin Manley (OPG), Richard Wain (Rolls Royce), Arkady Karneev (Rosatom), Frederik Vitabäck (GE-Hitachi), David Leblanc (Terrestrial Energy).

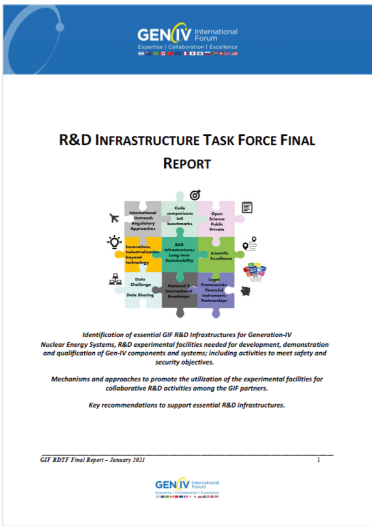


Figure RDTF-2. Cover page of the RDTF final report

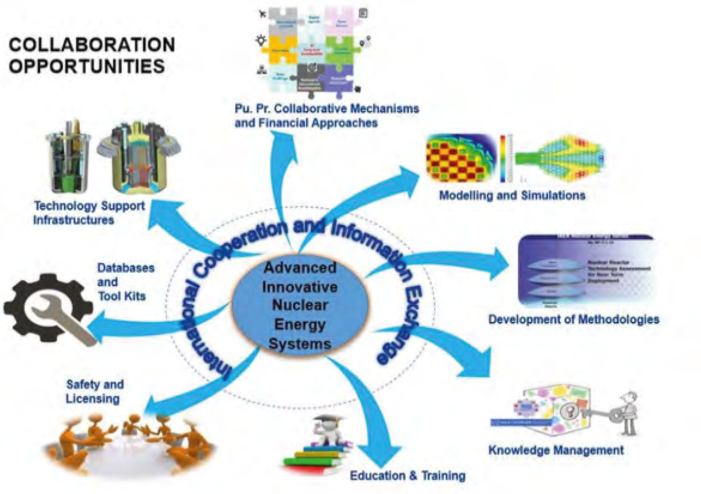


Figure RDTF-3. The virtuous wheel of international cooperation

to R&D infrastructure needs and opportunities. It included a review of RDTF efforts by system and roundtables with the private sector. The aim was: 1) to identify collaboration opportunities between private and public sectors for Gen-IV systems; 2) to ensure a networking event, gathering both GIF representatives and industry to create links; 3) to present concrete examples of collaboration between governmental organizations and industry; and 4) to gather views from the private sector on how to expand the relationship between GIF and the private sector in this field. The workshop was video recorded and can be seen on the GIF website (www.gen-4.org/gif/jcms/c_82829/workshops).

- Action no. 2: A dedicated GIF RDTF final report was presented at the GIF EG/PG meeting in Weihai (China, October 2019) for complete validation in 2020 from the Experts Group and the Policy Group. The report was finalized in early 2021 and uploaded to the GIF website. This final report is made up of 12 chapters that can be read independently, including an overview of R&D infrastructures for the six systems, along with cross-cutting R&D infrastructures, mechanisms and approaches for collaboration and key recommendations and conclusions.

The report thus provides a clear overview of R&D infrastructure for the six systems, as well as a cross-cutting approach. Chapters 10 and 11 offer an explanation of mechanisms and approaches for collaborative R&D activities (with examples in the appendix) and some recommendations to enhance or facilitate these activities.

It is stipulated in the report that because of its position within the relevant bodies of all member countries, and its close relationship with key influential nuclear institutions, GIF should play a proactive role in ensuring the optimization of available experimental platforms, as well as their sustainable use over the longer term. This can be carried out effectively by:

- promoting regular meetings to update relevant catalogues, compendium or databases of installations (i.e. at least once every two years);
- recalling that this subject is essential when preparing future international symposia and seminars;
- acting as the driving force for proposals within the framework of international initiatives, which could promote experimental infrastructures or enable the creation of new shared tools.

These two materials (i.e. the workshop record and final report) should be considered as a real springboard for enhancement of R&D facility use in future.

Conclusions (and/or next steps)

The missions assigned to this task force have been successfully fulfilled. As such, it is not considered relevant to further pursue the actions of this task force, as stipulated in the RDTF terms of reference. In accordance with the key recommendations given above, and the position of the GIF Policy Group in this regard, it will be necessary to determine how these initiatives will be articulated and should evolve in future

Taking into account the analyses and recommendations made by RDTF members, as well as private sector feedback from the workshop in February 2020, it would be preferable to seek for a new dedicated task force emerging from the conclusions of this RDTF report. Moreover, following all the recommendations set out above, it is necessary to highlight two topics largely put forward during the RDTF workshop. They are the sharing of:

- verification/validation and uncertainty quantification (VV&UQ) approaches and best practices between the different member countries;
- reflections on how to improve exchanges with regulators, at an early stage, to simplify and enable faster licensing processes for innovative systems, for example SMRs.

These two items could be the starting point for new GIF task forces that would be considered a logical continuation of the RDTF Task Force.

Roger Garbil
Chair of the RDTF, with contributions from RDTF members

7 Market and industry perspectives and the GIF Senior Industry Advisory Panel report

Market issues

Since the creation of the Generation IV International Forum (GIF) in 2000, market conditions have continued to evolve, and they continue to be a common concern among users and developers of Generation IV (Gen-IV) concepts. The role of the GIF Senior Industrial Advisory Panel (SIAP) is to understand core drivers, opportunities and constraints related to the market environment with the objective of identifying the most appropriate advice in terms of GIF activities, in collaboration with the System Steering Committee (SSC) chairs, task forces and working groups, and with the guidance of the members of GIF Policy Group (PG).

Following the conclusions of the 2015 Paris Agreement, numerous countries have initiated major endeavours to reduce CO_2 emissions related to economic activities. Efforts taking place over the past years to decarbonize the electricity sector have largely been concentrated on massive capacity additions of variable renewable energy (VRE) resources, such as wind and solar power. As recently illustrated by the International Energy Agency (IEA, 2019),[1] low-carbon electricity demand is set to increase by 2040, and the mobilization of all low-carbon technologies will be needed in order to meet climate engagements. For instance, according to the IEA "Sustainable Development Scenario", nuclear capacity should increase by 60% compared to today's levels if these engagements are to be met.[2] However, several issues continue to challenge the economic rationale of nuclear energy, ultimately slowing the development of nuclear power.

The cost of VRE resources has been steadily decreasing, enabling a higher penetration of these types' technologies in current electricity systems. This trend, combined with cheap and abundant fossil fuels – particularly in the United States – is undermining the profitability of nuclear projects in terms of the levelized cost of energy (LCOE). At the same time, and partly as a result of the long hiatus in nuclear new build since the 1990s, recent nuclear projects are experiencing difficulties in OECD member countries in relation to being delivered on time and on budget, thus increasing investor risks.

It is important to underline that the eruption 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. Moreover, decarbonization of energy systems also involves low-carbon heat generation for domestic and industrial processes, or even hydrogen production.

These issues were explored during the Vancouver GIF Workshop on Flexibility, held in May 2019. The event gathered members of the GIF Economic Modelling Working Group (EMWG), SIAP and SSCs, with the objective of assessing the flexibility of the Gen-IV systems. It was a good opportunity for SIAP to share with the GIF community the main findings of the 2018 SIAP charge[3], which has 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, as well as hydrogen production. Technologies with lower technology readiness levels have the highest potential as they face reduced constraints from the design standpoint. The different options for flexibility may allow Gen-IV systems to better adjust to more uncertain and turbulent energy markets. Integrating flexibility in Gen-IV designs could nevertheless come at a cost and should be fairly compensated through adequate market designs.

In this context, small modular reactors (SMRs) are capturing the attention of the nuclear industry since 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, which integrate by design higher modularization, standardization and factory-based construction in order to maximize economies of series (or the series effect). The different modules can then be transported and assembled on-site, leading to predictability and savings in construction times. More recently, vendors have been proposing micro modular

1. *World Energy Outlook* (IEA, 2019), available at: www.iea.org/reports/world-energy-outlook-2019.

2. www.iea.org/topics/tracking-clean-energy-progress.

3. A "Charge" address to SIAP is a technical question or a problem addressed by the Policy Group and the Experts Group to SIAP to get their feedback, vision & recommendations with their industrial point of view. Usually a maximum of one charge per year is addressed to SIAP.

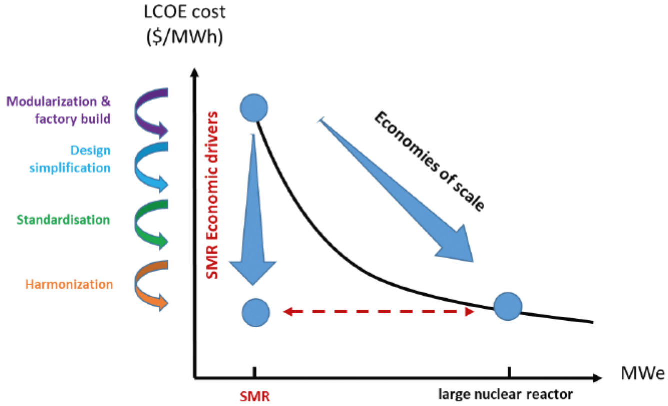
reactors (MMRs), with power outputs of lower than 10 MWe. Such reactors are capable of semi-autonomous operation and can take advantage of higher levels of transportability compared to the larger SMRs.

Among other factors, the series effect plays a central role in the economic competitiveness of SMRs. In fact, the small size of this type of reactor introduces a considerable economic penalty in terms of the LCOE (i.e. diseconomies of scale). The cumulative effect of modularization, simplification, standardization and harmonization may be driving the series effect, which is necessary to compensate for the penalty in relation to economies of scale, and could potentially improve the overall economic performance of SMRs. This effect is illustrated in Figure SIAP-1. Moreover, once established, the series effect could trigger increased confidence from financial institutions, thus leading to lower financing costs, which today represents a significant share of nuclear LCOE. Additional cost reductions could be unlocked with the introduction of new technologies.

The potential of the economic drivers of SMRs is supported by past experience in other industries (e.g. aviation). Additional empirical evidence is nevertheless needed for the SMR technology. Through this process, access to a global market that would enable the large-scale deployment of SMRs will be essential. It is worth noting that an Nth-of-a-kind cost level for an SMR can be achieved with significantly less cumulative installed power, and thus less upfront investment.

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 because of the lower overall capital outlay compared to large reactors. This lower capital outlay implies that private investors will face lower capital risk, which could make SMRs a more affordable option. It would, in turn, 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 allows revenues to be available sooner in capital projects. Shorter construction times will also work synergistically with other factors, namely serial production and smaller capital expenditure, to ultimately bring the financial cost of SMRs to a level that is much more in line with other major industrial projects.

According to the International Atomic Energy Agency (IAEA),[4] there are around 72 SMR concepts under development, with varying technology and licensing readiness levels. Among these concepts, around 50% are Gen-IV concepts, also called



Source: NEA (2020), *Unlocking Reductions in the Construction Costs of Nuclear: A Practical Guide for Stakeholders*, OECD Publishing, Paris.

Figure SIAP-1. SMR economic drivers that help compensate for diseconomies of scale

advanced small modular reactors (ASMRs). The HTR-PM, a 200 MWe model of gas-cooled high-temperature reactor, is under commissioning in China, with cold functional tests completed in unit 1 in October 2020. Even if most of these projects are endorsed, to some extent, by the governments of GIF member countries, involvement of the private sector has been increasing substantially.

Countries such as Canada, the United Kingdom and the United States have made significant progress in the development of policies and licensing frameworks to accelerate the time to market for SMRs.

In Canada, one ASMR vendor (i.e. Global First Power, with its 5 MWe HTGR MMR concept) is in the licensing process to build, own and operate a first demonstration unit at the Canadian National Laboratories (CNL) site in Chalk River by 2026. The Canadian government is investing CAD 20 million (USD 15 million) to accelerate the development of Terrestrial Energy's Integral molten salt reactor.

In the United Kingdom, phase 1 of advanced modular reactor (AMR) feasibility and development (F&D) has provided eight AMR organizations with funding of up to GBP 4 million (USD 5.5 million) to undertake a series of feasibility studies with contracts worth up to GBP 0.3 million (USD 0.4 million). Tokamak Energy, Westinghouse and U-Battery were selected to receive up to GBP 10m to undertake development activities for phase 2 of the project. Going forward, the Prime Minister's recent "Green Industrial Revolution" program announced up to GBP 385 million (USD 528 million) in an advanced nuclear fund to further invest in the next generation of nuclear technology. It includes funding of up to GBP 215 million (USD 295 million) for SMRs and up to GBP 170 million (USD 233 million) for AMRs. The

4. *Advances in Small Modular Reactor Technology Developments* (IAEA, 2020), available at: https://aris.iaea.org/Publications/SMR_Book_2020.pdf.

objective is to build a first AMR demonstrator by the early 2030s. In parallel, government will invest up to an additional GBP 40 million to develop the regulatory frameworks and support UK supply chains so as to help bring these technologies to market. Plans are also underway to open the generic design assessment process for SMR vendors in 2021 and to further the work on a comprehensive siting assessment and strategy.

In the United States, the advanced MMR design from Oklo is under review by the US Nuclear Regulatory Commission (NRC) and carries a site permit from the US Department of Energy to build a demonstration at the Idaho National Laboratory. In addition, under the "Advanced Reactor Demonstration Program", the Xe-100 and Natrium ASMRs received USD 80 million each in October 2020 to accelerate their development. Five additional SMR developers have received smaller grants in the (investment) "risk reduction" category.

Several challenges will nevertheless need to be overcome in order to achieve large-scale commercialization. The current, wide variety of ASMRs 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.

In addition to the need to reduce cost, industry is also facing the challenge of further improving safety as a result of new and demanding regulations. The IAEA's new design extension conditions require consideration of extreme events, which were previously beyond-design basis. To cope with these combined challenges, new system engineering and risk-informed, performance-based design approaches are being considered. Additional efforts will be required to revisit current licensing frameworks, which rely extensively on the experience developed with Gen-III/III+ and Gen-II light water reactors. At the same time, enabling policy frameworks and international collaboration will continue to be key factors in the timely deployment of new reactor concepts. Avoiding diversion of sensitive materials and ensuring physical protection will also be key in fostering the large-scale deployment of SMRs. In December 2019, the Canadian Nuclear Safety Commission (CNSC) and the US NRC selected Terrestrial Energy's integral molten salt reactor (IMSR) for the first joint technical review of an advanced, non-light water nuclear reactor.

The COVID-19 pandemic has set in place a new paradigm to rethink the importance of developing more sustainable and resilient energy systems, while boosting economic recovery. Given the lack of maturity of most SMR concepts, having this technology included in recovery packages around the world may not bring the near-term employment and emission reduction effects currently being sought by governments. However, these efforts could help to bring the new technologies needed to decarbonize advanced economies into existence by 2050, along with the long-term benefits associated with sustainable and resilient energy systems and the development of new industries and markets.

Senior Industry Advisory Panel report

SIAP 2020 charge and response

ASMRs are gaining recognition from both governments and the private sector. Over the years, GIF has created a significant knowledge base of Gen-IV systems and a unique global network of experts that could help accelerate the commercialization of the ASMR technology. In 2019, SIAP explored how interactions with the private sector could take place. The main outcomes that arose from this exploration include:

- due diligence in relation to ASMR concepts is needed;
- intellectual property rights could hinder collaboration, especially at high technology readiness levels;
- the need to go beyond technology-specific R&D, focusing on cross-cutting initiatives and support for research on processes and methods.

Furthermore, some ASMR designs may not be fully aligned with GIF goals, and the conditions for interaction with the private sector should be carefully defined. The GIF February 2020 workshop was a good opportunity to let the industry know that GIF is reviewing its activities and would like to create a perennial collaboration framework with the nuclear industry, and particularly with ASMR vendors.

The private sector's expectations of GIF, reflected in the 2020 SIAP Charge, cover four main areas:

- public and governmental recognition and acceptance;
- research and data structuring;
- technology acceptance and multinational pre-licensing;
- the global Gen-IV research infrastructure.

Public and governmental recognition and acceptance

If ASMR technology gains public and governmental recognition and acceptance, new policies could emerge to facilitate ASMR industrial development. GIF therefore needs to create a new narrative to explain to policymakers the role that ASMRs could play in the energy transition, and particularly in scenarios with high shares of variable renewables. To increase its policy impact, GIF could actively engage with organizations with high visibility at the policy level – for example the IEA; the Nuclear Innovation: Clean Energy Future (NICE Future) initiative; the IAEA; the Nuclear Energy Agency (NEA) and the World Nuclear Association (WNA) – and contribute to their publications. A recent example is the GIF contribution to a recent report released by the NICE Future initiative on the flexibility of nuclear systems[5]. Overall knowledge dissemination efforts should increase at the same time so as to reach out to universities and industry, attracting the necessary talent and building skills. GIF could also be involved in the knowledge management regional workshops that are organized periodically by the IAEA.

Research and data structuring

Examples already exist in relation to the consolidating and structuring of data to support the qualification of Gen-IV systems. In the United States, a topical report on tri-structural isotropic particle (TRISO) fuel has been elaborated for the US NRC following this approach. This project involved national laboratories, vendors and regulators with the Electric Power Research Institute (EPRI) working as an integrator (see Figure SIAP-2). Most of the data used for the elaboration of this topical report was already in the public domain, but additional efforts were needed in terms of formatting and making it more accessible to regulators and vendors. Similar initiatives are already underway within GIF. The *Gen-IV Materials Handbook* developed under the VHTR materials project arrangement gathers and structures research data that is then used by the American Society of Mechanical Engineers (ASME) to update codes and standards. GIF should communicate more on these examples and propose new initiatives on data structuring and qualification to the member countries. Joint projects could be set up around non-technology-specific materials, components, processes and fuels, such as graphite, liquid-fuel properties, mixed oxide (MOX) fuel or heat exchangers at high temperatures.

Technology acceptance and multinational pre-licensing

The ASMR vendor community could benefit from very early interactions with regulators in order to have an appreciation of their designs and the associated time to market. The GIF Risk and Safety Working Group (RSWG), in collaboration with the NEA Working Group on Safety of Advanced Reactors (WGSAR), could offer this type of service, based on documentation provided by interested vendors on how to comply with design guidelines and the criteria of a given system. These early safety evaluations could then be used to develop harmonized and internationally recognized design criteria and guidelines for Gen-IV systems in collaboration with the IAEA. Some vendors may nevertheless be reluctant to participate in these type of initiatives.

Opportunities in the pre-licensing arena are also important, and GIF should identify pre-licensing safety issues (e.g. licensing of verification codes, experimental data for validation, bounding severe accident cases for ASMRs) and explore the potential for developing a common pre-licensing base for ASMRs. A technology-neutral approach, which properly addresses the necessary adaptations from LWR-based regulations, could be used. Existing initiatives on licensing harmonization, such as the WNA Cooperation in Reactor Design Evaluation and Licensing Working Group (CORDEL) and the NEA Multinational Design Evaluation Programme (MDEP), may provide valuable lessons.

Global Gen-IV research infrastructure

While most of the ongoing research within GIF is focused on developing a mechanistic understanding of the Gen-IV systems, new research opportunities are emerging in relation to design and manufacturing processes and methodologies. Some areas of interest for GIF involve system engineering and risk-informed approaches, as well as the application of digital technologies and advanced manufacturing processes. Moreover, the development of technology readiness levels tailored to advanced reactors remains key to adequately assess which technologies could be adopted by Gen-IV systems.

Based on the success of the February GIF 2020 workshop, GIF should continue to interact periodically with the private sector, organizing dedicated meetings/workshops in order to identify critical research areas and priorities, and match emerging experimental and qualification needs with existing or future R&D infrastructure capabilities.

SIAP intentions for 2021

Since its inception, GIF has been focusing on and supporting the necessary R&D elements to support Gen-IV systems. The recent commercial thrust towards SMR development has awakened more interest in nuclear power in general. The Senior Industry Advisory Panel will continue to advise and support GIF so as to harness this new momentum.

It will stand ready to offer advice on how to interact with the private sector and implement some of the recommendations of the 2020 SIAP charge. SIAP will also continue to provide industrial insight for GIF activities and strengthen its collaboration with the GIF Economic Modelling Working Group (EMWG) to assess cost reduction and safety improvement opportunities arising from new design methodologies for Gen-IV concepts.

Eric Loewen
Chair of the SIAP, with contributions from SIAP members

5. www.nice-future.org/sites/default/files/document/Generation%20IV%20-%20Web%20Page.pdf.

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1. List of abbreviations and acronyms

Generation-IV specific acronyms

AF	Advanced fuel
AMME	Advanced manufacturing and materials engineering
CD&BOP	Component design and balance of plant
CMVB	Computational methods validation and benchmarking
EG	Experts Group
EMWG	Economic Modelling Working Group (GIF)
ETTF	Education and Training Task Force (GIF)
ETWG	Education and Training Working Group (GIF)
FA	Framework agreement
FFC	Fuel and fuel cycle
GIF	Generation IV International Forum
GFR	Gas-cooled fast reactor
HP	Hydrogen production
HTR	High-temperature gas-cooled reactor
ISAM	Integrated safety assessment methodology
IT	Information Technology
LFR	Lead-cooled fast reactor
MAT	Materials
MC	Materials and chemistry
MCFR	Molten chloride salt fast reactor
MSR	Molten salt reactor
MWG	Methodology working group
PA	Project arrangement
PG	Policy Group (GIF)
PMB	Project Management Board (GIF)
PP	Physical protection
PR	Proliferation resistance
PR&PP	Proliferation resistance and physical protection
PSSC	Provisional System Steering Committee (GIF)
RDTF	R&D Infrastructure Task Force (GIF)
RSWG	Risk and Safety Working Group (GIF)
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 (GIF)
SO	Safety and operation
SRP	System research plan
SSC	System Steering Committee
TF	Task force
VHTR	Very-high-temperature reactor
WGSAR	Working Group on the Safety Advanced Reactor

Technical terms, projects and facilities

AD	Accelerator-driven
ADS	Accelerator-driven system
AGR	Advanced gas-cooled reactor (US)
AFA	Alumina forming austenitic
AFCP	Advanced Fuel Cycle Programme
AFR	Advanced fast reactor
AHFM	Algebraic heat flux model
ALFRED	Advanced Lead Fast Reactor European Demonstrator
ALLEGRO	The European Gas Fast Reactor Demonstrator Project
AMR	Advanced modular reactor
ANM	ANSTO Nuclear Medicine project
ANT	Advanced Nuclear Technologies
ARDP	Advanced Reactor Demonstration Program
ARKADIA	Advanced Reactor Knowledge- and AI-based Design Integration Approach
ARTMS	Alternative Radioisotope Technologies for Medical Science
ASTRID	Advanced Sodium Technological Reactor for Industrial Demonstration
ASMR	Advanced small modular reactor
ATR	Advanced test reactor (at INL)
AVR	Arbeitsgemeinschaft Versuchsreaktor
BWR	Boiling water reactor
CAPEX	Capital expenses
CASLER	Co-operative Alliance for Small Lead-based Fast Reactor
CBSG	Copper bounded steam generator
CIIALER	China Industry Innovation Alliance of Lead-based Reactor
CEFR	China experimental fast reactor
CFD	Computational fluid dynamics
CHE	Compact heat exchangers
CHP	Combined heat and power
CLEAR	China lead-based reactor
CNRI	Canadian Nuclear Research Initiative
DHR	Decay heat removal
DNS	Direct numerical simulation
EBR	Experimental breeder reactor (US)
ECC-SMART	Joint European Canadian Chinese Small Modular Reactor Technology project
EOC	End of cycle
ESFR-SMART	European Sodium Fast Reactor Safety Measures Assessment and Research Tools
ECS	Energy conversion system
ELFR	European lead fast reactor
EPZ	Emergency planning zones

ESFR	European sodium fast reactor
E&T	Education & training
FA	Fuel assembly
FALCON	Fostering ALfred CONstruction
F&D	Feasibility and Development
FLIBE	mixture of lithium and beryllium fluoride (BeF2)
FOA	Funding opportunity announcement (US)
FHR	Fluoride salt-cooled high-temperature reactor
FOAK	First-of-a-kind
FP	Framework Programmes
FR	Fast reactor
GW	Gigawatt
HEEP	Hydrogen Economic Evaluation Programme (IAEA)
HI	Hydrogen iodine
HINEG	High intensity D-T fusion neutron Generator[2]
HIP	Hot isostatic pressing
HFR	High-flux reactor
HLD	High-level deliverable
HPR	Advanced pressurized water reactor
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
IHX	Intermediate heat exchanger
IMSR	Integral molten salt reactor
ISI&R	In-service inspection and repair
ITMSF	International Thorium Molten Salt Forum
JHR	Jules Horowitz reactor
JRC	Joint Research Centre (EU)
JSFR	Japanese sodium-cooled fast reactor
KALIMER	Korea advanced liquid metal reactor
KM	Knowledge management
LCOE	Levelized cost of energy
LEU	Low-enriched uranium
LHR	Linear heat rate
LWR	Light water reactor
LBE	Lead-bismuth eutectic
MA	Minor actinides
MINERVA	Micro nuclear energy research and verification arena
MBIR	Russian multipurpose fast neutron research reactor
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
NE	Nuclear Energy
NEANH	Non-electrical application of nuclear heat
NEUP	Nuclear Energy University Project (US)
NPP	Nuclear power plant
NSD	Nuclear Sector Deal
NSSS	Nuclear steam supply system
NSTF	Natural Convection Shutdown Heat Removal Test Facility (US)
NRAD	Neutron radiography reactor
NUWARD	French PWR SMR Project
ODS	Oxide dispersion-strengthened
OPT	Objective provision tree
OPEX	Operating expenses
PBMR	Pebble bed modular reactor
PEACER	Proliferation-resistant environment-friendly accident-tolerant continual economical reactor
PEM	Proton exchange membrane
PGSFR	Prototype Generation IV sodium-cooled fast reactor
PIE	Post-irradiation examinations
PIRT	Phenomena identification and ranking table
PP	Primary pump
PPE	Multiannual Energy Plan (France)
PRA	Probabilistic risk assessment
PRISM	Power reactor innovative small module
PRPPWG	Proliferation Resistance and Physical Protection Working Group
PV	Photovoltaic
PWR	Pressurized water reactor
QSR	Qualitative safety features review
RANS	Reynolds Analysis Navier-Stokes
RE	Rare-earth
R&D	Research and development
RD&D	Research development & demonstration
RFI	Request for information
S-CO_2	Supercritical carbon dioxide
SAMOSAFER	Severe Accident Modeling and Safety Assessment for Fluid-fuel Energy Reactors
SCW	Supercritical water
SCWL	Supercritical water loop
SDSAR	Specific design safety analysis report
SELAAD	Sodium experimental loop for advanced aerosol detection
SEM	Scanning electron microscopy
SER	Safety evaluation report
SFEAR	Support Facilities for Existing and Advanced Reactors (NEA)
SG	Steam generator
S-I	Sulphur-iodine process
SME	Small and medium enterprise
SMR	Small modular reactor
SNETP	Sustainable Nuclear Energy Technology Platform
SNF	Spent nuclear fuel
SSA	System safety assessment
SSTAR	Small, secure, transportable, autonomous reactor
STELLA	Sodium integral effect test loop for safety simulation and assessment
SWR	Sodium-water reaction
TEM	Transmission electron microscopy
THTR	Thorium high-temperature reactor
THS	Thermal-hydraulics and safety
TMSR	Thorium molten salt reactor (China)
TORIA	Thorium-optimized radioisotope incineration arena
TR	Topical reports
TRISO	Tri-structural isotropic (nuclear fuel)
TRU	Transuranic
TTS	Transient Test Facility for Structures
UCO	Uranium oxycarbide
ULOF	Unprotected loss of flow

UOX	Uranium oxyde	VTR	Versatile testing reactor (US)
V&V	Verification and validation	VVER	Russian light water power pressurized reactor model
VDR	Vendor design review	WP	Work package
VRE	Variable renewable energy		

Organizations, companies and agencies

ACU	Abilene Christian University
ANL	Argonne National Laboratory (US)
ANSTO	Australian Nuclear Science and Technology Organisation
APS	Arizona Public Service
ARC	Department of Energy Office of Advanced Reactor Concepts (US)
ASME	American Society of Mechanical Engineers
ASN	Autorité de Sûreté Nucléaire (France)
BEIS	Business, Energy and Industrial Strategy Department (UK)
CAS	Chinese Academy of Sciences
CEA	Le Commissariat à l'énergie atomique et aux énergies alternatives (France)
CGN	China General Nuclear Power Group
CIAE	China Institute of Atomic Energy
CNL	Canadian Nuclear Laboratories
CNNC	Chinese National Nuclear Corporation
CNRA	Committee on Nuclear Regulatory Authorities (NEA)
CNRS	Centre national de la recherche scientifique (France)
CNSC	Canadian Nuclear Safety Commission
DEE	Department of Energy Engineering
DOE	Department of Energy (United States)
EC	European Commission
EDF	Electricité de France
EERE	(Office of) Energy Efficiency and Renewable Energy (US)
ENEA	Italian National Agency for New Technologies, Energy and Sustainable Economic Development
EPRI	Electric Power Research Institute
ESNII	European Sustainable Nuclear Industrial Initiative
EU	European Union
IAEA	International Atomic Energy Agency
ICN	Institute of Nuclear Research (Romania)
INET	Institute of Nuclear and New Energy Technology (China)
INEST	Institute of Nuclear Energy Safety Technology (China)
INL	Idaho National Laboratory (US)
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
JNFL	Japan Nuclear Fuel Limited
JRC	Joint Research Centre (Euratom)
KAERI	Korea Atomic Energy Research Institute
LLNL	Lawrence Livermore National Laboratory (US)
MDEP	Multinational Design Evaluation Programme (NEA)
MIT	Massachusetts Institute of Technology
MOST	Ministry of Science and Technology (China)
MTA	Hungarian Academy of Sciences Centre for Energy Research
NCBJ	Narodowe Centrum Badan Jadrowych (Poland)
NEA	Nuclear Energy Agency
NIA	Nuclear Industry Association
NIC	Nuclear Industry Council
NIFS	National Institute for Fusion Science
NIRAB	Nuclear Innovation Research Advisory Board (UK)
NPIC	Nuclear Power Institute of China
NRA	Nuclear Regulation Authority (Japan)
NRC	Nuclear Regulatory Commission (US)
NRCan	Department of Natural Resources (Canada)
NRG	Dutch Nuclear Safety Research Institute
NRIC	National Reactor Innovation Center
NSSC	Nuclear Safety and Security Commission (China)
OECD	Organisation for Economic Co-operation and Development
OPG	Ontario Power Generation (Canada)
ORNL	Oak Ridge National Laboratory (US)
PSI	Paul Scherrer Institute (Switzerland)
RATEN	Regia Autonoma Tehnologii Pentru Energia Nucleara (Romania)
SINAP	Shanghai Institute of Applied Physics (China)
SJTU	Shanghai Jiaotong University (China)
SNU	Seoul National University (Korea)
SPIC	State Power Investment Corporation (China)
TEPCO	Tokyo Electric Power Company (Japan)
TIT	Tokyo Institute of Technology
UEC	University of Electro Communications
UNIST	Ulsan National Institute of Science and Technology (Korea)
USNC	Ultra Safe Nuclear Corporation
USTB	University of Science and Technology Beijing
USTC	University of Sciences and Technology of China
V4G4	Visegrad GEN-4 Centre of Excellence
VTT	Valtion Teknillinen Tutkimuskeskus (Finland)
VUJE	Slovakian engineering company
WNA	World Nuclear Association
XJUT	Xi'an Jiaotong University (China)

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2. Selection of GIF publications (2020)

General paper

GIF Newsletters nos. 1 to 6, and one Special Summer Edition, available at: www.gen-4.org/gif/jcms/c_122378/newsletters-archive

Lead-cooled fast reactor

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GIF RSWG public web page: www.gen-4.org/gif/jcms/c_9366/risk-safety

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Education and Training Working Group

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Safety Design Criteria Task Force

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Advanced Manufacturing and Material Engineering Task Force

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Research and Development Task Force

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R&D Infrastructure Task Force Final Report, www.gen-4.org/gif/jcms/c_173654/gif-rdtf-final-report-jan2021.

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 34 countries: Argentina, Australia, Austria, Belgium, Bulgaria, 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.

This thirteenth edition of the *Generation IV International Forum (GIF) Annual Report* covers 2020. In 2020, the GIF, as have all, had to adapt its way of working to the worldwide COVID-19 pandemic situation. In the face of this, all GIF members made their best efforts to produce deliverables and fulfil their objectives in an optimized manner.

In 2020 the GIF organization started its transition towards a new communications approach through a rebranding of its logo and website. This transitional phase will lead the Generation IV International Forum to a new approach in line with the current situation: more virtual meetings and exchanges; a powerful and updated GIF website to ease and simplify interactions between members; and regular communication through high standard monthly webinars and newsletters. Thus the GIF is ready to enter its third decade of existence in the particular context of a new energy paradigm and an unpredictable sanitary situation.

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