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

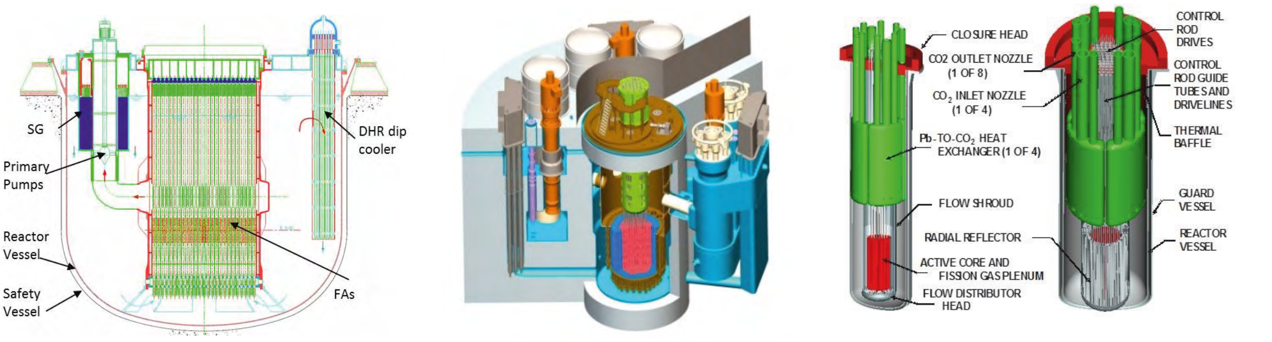
Parameters	ELFR	BREST	SSTAR
Core power (MWt)	1 500	700	45
Electrical power (MWe)	600	300	20
Primary system type	Pool	Pool	Pool
Core inlet T (°C)	400	420	420
Core outlet T (°C)	480	540	567
Secondary cycle	Superheated steam	Superheated steam	Supercritical CO_2
Net efficiency (%)	42	42	44
Turbine inlet 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

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.

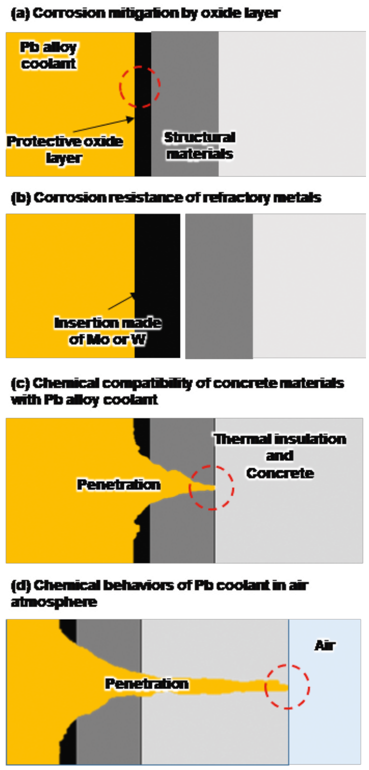


Figure LFR-4: Corrosion issues in various situations

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 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.

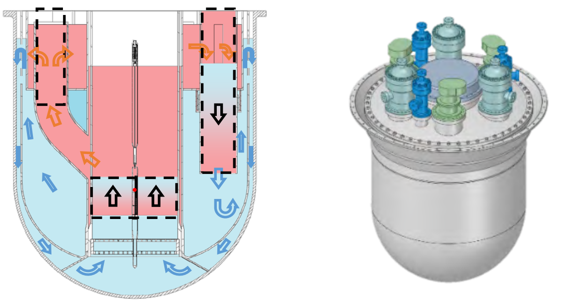
Main activities in Euratom

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;

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).

- EUR 156 million for the operating expenses of MINERVA for the period of 2027-2038.

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.

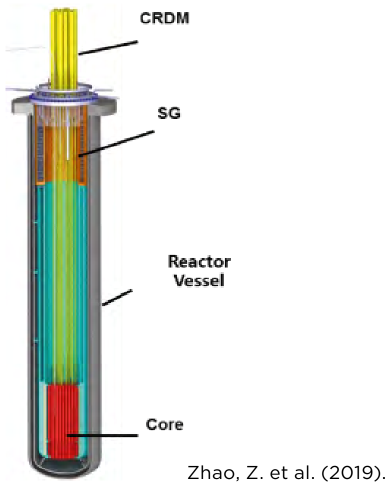
Additional university-related design activities include past work at the University of California on the Encapsulated Nuclear Heat Source (ENHS) and more recently in several projects sponsored by the US Department of Energy under Nuclear Energy University Project (NEUP) funding. These include the following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

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.

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Chair of the LFR SSC, with contributions from LFR members