Status of Generation-IV Lead Fast Reactor Activities

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Abstract. Since 2012 the Lead-cooled Fast Reactor provisional System Steering Committee (LFR-pSSC) of the Generation IV International Forum (GIF) has developed a number of top level strategic activities with the aim to assist and support development of Lead-cooled Fast Reactor technology in member countries and entities. The current full members of the GIF-LFR-pSSC (i.e., signatories of the GIF LFR Memorandum of Understanding /MoU/) are: EURATOM, JAPAN, the RUSSIAN FEDERATION, and the REPUBLIC OF KOREA. The pSSC also benefits from the active participation of its observers: the UNITED STATES and the PEOPLE'S REPUBLIC OF CHINA. The paper highlights some of the main collaborative achievements of LFR-pSSC, including the development of the LFR System Research Plan, the LFR White Paper on Safety, the LFR System Safety Assessment paper as well as the LFR Safety Design Criteria paper. The paper then presents the status of the development of LFRs in the GIF member countries and entities. The collaboration among partners of the GIF-LFR-pSSC has proven its effectiveness in assisting the development of LFRs through an open, interactive and collegial environment, developing important synergies and exchange of both technical and strategic information.

Key Words: Generation IV, Lead-cooled Fast Reactor (LFR), Generation IV International Forum (GIF)

1. Introduction

Among the promising reactor technologies considered by the Generation IV international Forum (GIF), the Lead-cooled Fast Reactor (LFR) was identified as a technology with great potential to meet needs for both remote sites and central power stations, fulfilling the four main goals of GIF in terms of sustainability, safety, economics and proliferation resistance. In the Generation IV technology evaluations [1], the LFR system was top-ranked in sustainability because a closed fuel cycle can be more easily achieved, and in proliferation resistance and physical protection. It was also assessed as good in safety and economics. Safety was considered to be enhanced by the choice of a relatively inert coolant. The paper highlights the main recent collaborative achievements of the LFR-pSSC, including the development of the LFR System Research Plan, the LFR White Paper on Safety, the LFR System Safety Assessment paper as well as the LFR Safety Design Criteria paper. The paper then presents the status of the development of LFRs in the GIF member countries and entities. The collaboration among partners of the GIF-LFR-pSSC has proven its effectiveness in assisting the development of LFRs.

2. Main Characteristics of the Reference Systems of GIF-LFR Activities

The LFR concepts identified by GIF include three reference systems. The options considered are a large system rated at 600 MWe (ELFR EU), intended for central station power generation, a 300 MWe system of intermediate size (BREST-300 Russia), and a small transportable system of 10-100 MWe size (SSTAR US) that features a very long core life (*Figure 1*). The expected secondary cycle efficiency of each of the LFR reference systems is at or above 42%. It can be noted that the reference concepts for GIF-LFR systems cover the full range of power levels, including small, intermediate and large sizes. Important synergies exist among the different reference systems so that a coordination of the efforts carried out by participating countries has been one of the key points of LFR development.

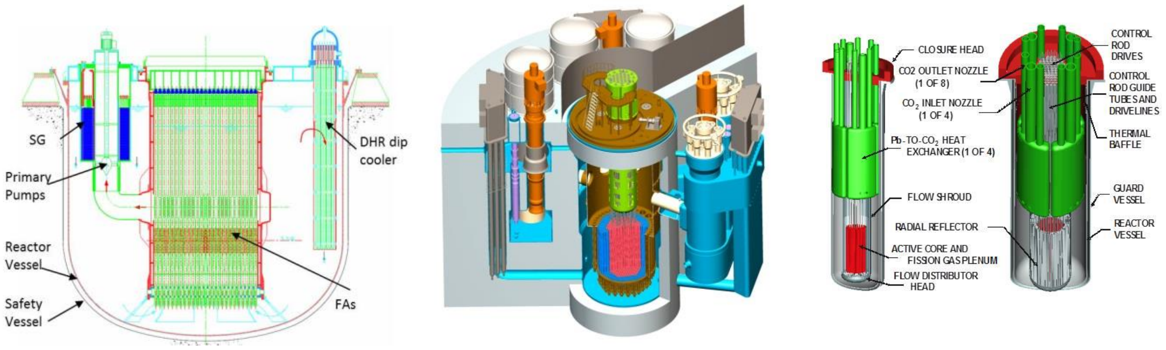

Figure 1 - reference systems of GIF-LFR: ELFR, BREST and SSTAR

The typical design parameters of the GIF-LFR systems are briefly summarized in Table I.

TABLE I - Key design parameters of the GIF LFR concepts

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

2.1.Lead Fast Reactor Research and Development 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. The preliminary evaluation of the concepts included in the plan covers their performance in the areas of sustainability, economics, safety and reliability, proliferation resistance and physical protection. Given the R&D needs for fuel, materials, and corrosion-erosion control, the LFR system is expected to require a two-step industrial deployment: reactors operating at relatively modest primary coolant temperatures and power densities by 2030; and higher-performance reactors by 2040. Following the reformulation of GIF-LFR-pSSC in 2012, the SRP was completely revised, and a final draft was prepared by the pSSC and sent to the GIF Expert Group for review. Comments to the SRP were received from one industrial partner (Westinghouse Electric Corporation, WEC, US). The SRP is expected to be issued in 2017.

2.2.Main Activities of LFR-pSSC and Recent Outcomes

The activities of the LFR-pSSC during 2016 centered on top level reports for GIF. After the issue of the LFR White Paper on Safety in collaboration with the GIF Risk and Safety Working Group (RSWG) in 2014 [2], the pSSC was very active on the following main lines:

<u>LFR Safety Design Criteria (SDC)</u>: development of the LFR SDC used the previously-developed SFR SDC report as a starting point. However, it was later realized that the IAEA SSR-2/1 (on which SFR SDC was based) did not require many of the features identified for the SFR to be adapted for the LFR (note that IAEA SSR2/1 refers substantially to LWR technology). At the end of 2016, the LFR pSSC received comments on its draft SDC from French GIF members and from the EURATOM ARCADIA project partners. The LFR-SDC is presently under review to address these comments and suggestions for improvements.

<u>LFR System Safety Assessment</u>: in 2014 the RSWG asked SSC chairs to develop a report on their systems to analyze them systematically, assess the safety level and identify further safety-related R&D needs. The LFR assessment report was prepared by the LFR pSSC and sent to the RSWG for comments at the end of September 2015. The RSWG provided comments in November, and the final version will be issued shortly by RSWG collecting contributions from all the six SSCs.

<u>LFR Safety Design Guidelines (SDG)</u>: The LFR-pSSC received from the RSWG the SFR Safety Design Guidelines on Safety Approach and Design Conditions in October 2016. The report will be used as a basis for the development of the LFR-SDG report.

<u>LFR-pSSC comments to the IRSN report on the Safety of Generation IV reactors [3]</u>: The LFR-pSSC sincerely appreciated the technically comprehensive review of LFR safety aspects provided by the IRSN. However, the Committee also felt that the results of recently-concluded as well as ongoing R&D efforts were possibly not considered by IRSN when drawing some of the conclusions. The comments of the pSSC are expected to form the basis for further discussions and possible update of the IRSN report.

<u>Cooperation Agreement EURATOM-ROSATOM</u>: Following the signature in May 2014 of a Cooperation Agreement (CooA) between the BREST and LEADER projects, by NIKIET (on behalf of ROSATOM) and Ansaldo (on behalf of the LEADER consortium), a first cooperation meeting was organized in Genova on December 9-11, 2015. A second meeting took place in Moscow on October 3-4, 2016. During the meetings, presentations were made covering both the BREST and ALFRED designs and safety features as well as many specific aspects related to thermal hydraulics, fuel design, cooling etc.

3. Main Activities of GIF-LFR pSSC Member Countries

In the following the main achievements of the entities collaborating in GIF under the LFR-MoU are briefly highlighted.

3.1. Russian Federation

BREST-OD-300, an innovative inherent-safe fast reactor, is being developed as a pilot and demonstration prototype for future nuclear power with a closed nuclear cycle. The lead coolant was chosen on the basis of the favourable characteristics of its properties, namely: 1) in combination with (U-Pu)N fuel, it allows for complete breeding of fissile materials in the reactor core, maintaining a constant small reactivity margin thus preventing an uncontrolled power increase because of equipment failures or personnel errors; 2) 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 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 for high heat capacity of the coolant circuit which decreases a possibility of fuel damage; 5) it capitalizes on its high density and albedo properties for flattening the Fuel Assembly (FA) power distribution; 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 to ensure complete breeding of fuel in the core and a constant small reactivity margin preventing any prompt-neutron excursion during reactor operation. A low-swelling ferrite-martensitic steel is used as the fuel cladding. Radiation tests of fuel elements are being conducted in the BN-600 power reactor and in the BOR-60 research reactor. At the present time, eight FAs are being irradiated in the BN-600 reactor, and the fuel elements of a previously-withdrawn FA are being subjected to post-irradiation studies. Seven FAs with nitride fuel elements are being irradiated in the BOR-60 research reactor. In the design of the reactor core items, novelty was coupled with reference solutions. The FA has a shroudless hexagonal design. Such a solution eliminates the possibility of fuel melting due to FA flow area blockage; even in the event of the flow area at the inlet of a 7-FA group being blocked, the safe operation limits in terms of the fuel cladding temperature are not exceeded. Another positive point is a 30% reduction in the metal content of the shroudless FA as compared to the shrouded option. Technologically, the adopted design is based on the experience gained when fabricating the FAs for VVER reactors.

Figure 2 - Full-scale FA mock-up

To justify the FA design serviceability, full-scale mock-ups (*Figure 2*) were manufactured and subjected to mechanical, hydraulic and vibration tests in air and water environments. A 37-rod fuel bundle mockup was used in the liquid-metal experiments to refine the heat transfer coefficients. Thus, a large quantity of data was obtained, which allows for validation of the codes intended for thermal-hydraulic calculations of the reactor core. To confirm the corrosion resistance of the FA elements in the lead coolant, tests using small-scale fuel-free mockups of the FAs at different temperatures were conducted. Absence of data on the physical experiments with nitride fuel led to the necessity of carrying out an experiment using the BFS critical facility. In the simulation lead, plutonium, and uranium nitride were used. Based on the results of the new experiments and the data obtained from the previous critical experiments, the calculation codes were validated for neutronic calculations. The results of the calculations carried out using the validated software tools show the possibility to achieve a small reactivity margin during the reactor operation and provision of a practically stable power density field during the duration of the fuel lifetime. An integral layout is used in the reactor facility to avoid coolant losses. The reactor vessel (RV) material is multilayer metal concrete; the lead coolant and the main components of the primary circuit are located in the reactor vessel (*Figure 3*). A wide range of calculations and experimental studies were required to confirm the serviceability of such a vessel type, which is novel for the nuclear power industry. The experimental justification is based on investigations and testing of the small- and full-scale components. Using the developed full-scale mockup of the vessel bottom a capability to ensure the required temperature of the building structures has been demonstrated, and joint thermal movements of the components have been determined. Using the developed full-scale mockup of the central part of the vessel (*Figure 4*), heating-up modes have been optimized, and the gas emission parameters have been determined. The properties of concrete have been determined for the anticipated operating temperatures and irradiation levels as well. With respect to metals, corrosion resistance experiments in the lead coolant environment have been conducted. To verify safety, the region of lead-concrete chemical interaction needs to be determined. The depth of lead penetration was experimentally determined to be no more than 0.5 mm without chemical interaction.

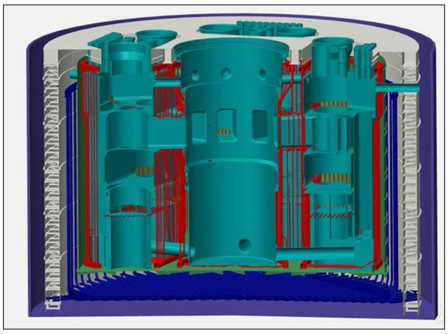

Figure 3 – Vessel of BREST *Figure 4 – RV Mockup* *Figure 5- Tube rupture exp.*

The structural analysis of the vessel was performed using newly developed techniques. The analysis took into account the actual geometrical and physical-mechanical properties of the vessel components and complex three-dimensional contact interaction between them, the non-linear concrete properties and the formation of cracks in it. The analytical justification showed that the adopted vessel design ensures the probability of formation of a leak with partial coolant loss of no more than 9.7×10-10 1/year.

The integral layout with a steam generator (SG) located in the reactor vessel imposes a high responsibility on the developers, designers and experimentalists involved in the confirmation of serviceability and safety of the SG. Therefore, a thorough justification of the steam generator components and the processes taking place in the steam generator has been planned and is being carried out. In the course of the SG experimental justification several mockups had been developed, which were used to verify (check) the parameters, which were identified in the detailed design. To determine the thermal-hydraulic characteristics including the impact of centrifugal acceleration on the thermal-hydraulic stability, an 18-tube model was developed. From the results of the 18-tube model tests, the heat transfer coefficients and hydraulic characteristics in the steam-water and lead circuits were obtained, as well as the temperature distribution in the lead circuit. Thermal hydraulic stability was demonstrated in the investigated ranges. To determine the steam generator life, thermal cyclic strength tests of the unit for securing the tubes between the tube sheets were carried out. The degree of reliability of the “tube-tube sheet” joints was determined for superheated steam removal and feed-water supply chambers in the SG modules, and the fulfilment of the thermal cyclic strength conditions were confirmed for the heat exchange tubes and the points where they are welded to the tube sheet. Tribological tests of the “tube-spacer grid” contact points in the lead coolant environment were performed. As a result, experimental data was obtained on the wear of the friction-coupled components of the specimens in the characteristic range of stresses and movements within the contact areas. A complex three-dimensional analytic justification of the steam generator serviceability was carried out, which included thermal hydraulic calculations, strength calculations for all operating conditions, vibration strength calculations, seismic effect, aircraft crash and air shock wave calculations, and other design analyses. To verify the vibration calculations, a mockup of the steam generator with actual geometric parameters is being developed. Because of a high density of lead, it was necessary to analyse the possibility of a secondary failure of the steam generator tubes if one of the tubes breaks. The dependent failure and the subsequent ingress of steam into the coolant may in turn affect the circulation in the circuit and consequently impair the thermal condition of the fuel elements. Based on a series of conducted experiments (*Figure 5*), it was demonstrated that it is impossible for a single SG tube rupture to develop into a multiple tube rupture (dependent rupture exclusion).

The reactor main coolant pump (MCP) is intended to establish the lead coolant head and provide for its circulation in the circuit. To confirm its serviceability, several mockups of the pump set have been developed, as well as the test sections to check their performance: (i) a medium-scale test section operating with liquid lead and a MCP mockup have been developed; (ii) the flow characteristics of the lead coolant flow path have been obtained in the order of 80% of the required regions (test bench limitations); (iii) the serviceability of a hydrostatic bearing unit has been demonstrated in the conditions of the medium-scale test bench (over 300 start-up-shutdown sequences); (iv) the energy performance of the flow part in water has been optimized; and (v) the required flow, head and positive suction head have been obtained. Other main and ancillary components are being justified at small- and medium- scale test benches; the properties of structural materials in the operating temperature ranges and rated operating conditions, including irradiation, are being obtained. The main (largest) components developed for the BREST reactor facility have been justified through the experiments and calculations and are now being prepared for prototype testing.

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

The developed detailed design of the BREST-OD-300 reactor facility justified using the small and medium scale test benches and test sections, as well as the validated software tools, met the key parameters specified and the licensing procedure has been started.

3.2. Japan

Theoretical studies of fast reactors using lead-bismuth eutectic as a coolant have been perfomed 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 concept. The concepts of the Breed and Burn reactor and the CANDLE buring reactor were studied mainly at the Tokyo Institute of Technology. One of the important issues in the CANDLE burning reactor concept is to maintain the integrity of the fuel elements in very high burnup conditions. The research shows the possiblity to solve the problem by the introduction of a melt and refining process based on metallic fuel. The study also considered the use of plutonium from LWR spent fuel for the start-up core in a CANDLE reactor to achieve effective utilization of the plutonium.

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 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. The design of a solid electrolyte type oxygen sensor was improved to provide better response in a high temperature lead-bismuth coolant system.

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.

The chemical behaviors of the lead based coolants in the air ingress accident of fast reactors were investigated by means of the thermodynamic considerations and the static oxidation experiments for Pb alloys with various chemical compositions. The results of the static oxidation tests for Pb-Bi alloys indicate that Pb was depleted from the alloy due to the preferential formation of PbO in air at 773K. Bi was not involved in this oxidation procedure. Lead bismuth oxide and Bi_2O_3 were formed after the enrichment of Bi in the alloys due to the Pb depletion.

3.3. Republic of Korea

The Government of the Republic of Korea has joined the GIF-LFR pSSC by signing the MoU at OECD-NEA in November 2015. LFR R&D progress has been made mainly by university programs during the past twenty years, since the first study in 1996 at Seoul National University (SNU). The Korean LFR Program has two main objectives:

- a technology development requirement for sustainable power generation using energy produced during nuclear waste transmutation;
- a new electricity generation unit development requirement to match the needs of economically competitive distributed power sources for both developed countries and developing nations that need massive and inexpensive electric power with an adequate margin against worst case scenarios encompassing internal and external events.

To meet the first goal, the Korean first LFR-based burner PEACER (Proliferation-resistant Environment-friendly Accident-tolerant Continual-energy Economical Reactor) has been developed to transmute long-lived wastes in spent nuclear fuel into short-lived low-intermediate level wastes, since 1996. In 2008, the Korean Ministry of Science and Technology selected the sodium 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 System (ADS) Th-based transmutation system designated as TORIA (Thorium Optimized Radioisotope Incineration Arena) with the leadership of the Nuclear Transmutation Energy Research Centre (NUTRECK) of Korea at SNU. For the second goal Korea has also started to develop PASCAR (Proliferation-resistant, Accident-tolerant, Self-supported, Capsular and Assured Reactor) for 20-year operation without on-site refueling. Recently the Korean Government has been funding an international collaborative R&D to further develop PASCAR into an improved design called URANUS (Ubiquitous, Rugged, Accident-forgiving, Nonproliferating, and Ultra-lasting Sustainer).

3.4. EURATOM

Following the signature of the FALCON (Fostering ALfred CONstruction) Consortium Agreement in December 2013 by Ansaldo, ENEA (Italy) and ICN (Romania) the consortium was enlarged by the addition of the CV Řež laboratory (Czech Republic) in December 2014. The consortium successfully involved a number of additional European partners through the signature of a number of Memorandum of Agreements (MoA) expanding throughout Europe as much as possible the interest in the development of lead technology. In 2016, the main activities related to the ALFRED design development included: (i) development of a new conceptual design configuration for the primary side; (ii) evaluation of options for steam generators (SGs), including bayonet double wall as well as helical SGs configurations; (iii) evaluation of different options for primary pumps; (iv) integration of a new Decay Heat Removal (DHR) system in the primary pool; (v) optimization studies of core and Fuel Assemblies; and (vi) development of a new anti-freezing system for DHRs. A testing facility of the system is expected to start construction in 2017 following a grant from the Italian government. Relevant material activities were performed in 2016. Studies to further optimize the composition of double-stabilized DS4 low-swelling austenitic steel were performed. In the frame of experimental qualification of corrosion protection barriers, several coating techniques were developed and tested. FeCrAl alloys were deposited by the physical vapor deposition (PVD) method on AISI 316L, AISI 304, AISI 441 and P91 steels. Good adhesion but non-uniform thickness were observed. The chemical vapor deposition (CVD) technique, suitable for complex shapes, was used as well to coat P91 and 15-15 Ti by the same alloys as by PVD. Namely, the pack cementation and diffusion coating processes were employed. Although promising, these processes, due to their high temperatures, induce modifications of the substrate microstructure. Two other different techniques were adopted: thermal spay (high velocity oxygen fuel /HVOF/) of FeCrAlY alloy and laser ablation (pulsed laser deposition /PLD/) of alumina. The former assures absence of porosities and good adhesion but needs further mechanical grinding to reduce the excessive roughness. On the other hand, PLD produces excellent coating from any point of view. Corrosion tests were performed showing excellent resistance by all the barriers except for the observation of local defects in PVD. PLD specimens were exposed to heavy ion irradiation to verify their damage resistance up to 450 dpa. Progressive crystallization of the amorphous phase was observed, but no delamination or cracking was found, even at the highest levels of irradiation damage. In the area of core design, refueling studies of the ALFRED reactor were made. The conceptual design of an experimental facility to test operating procedures for FA handling and the reliability of the fuel-handling machine was drafted. The core design was optimized to avoid overheating of corner pins. An improvement of core shielding by additional dummy elements was studied.

With respect to MYRRHA, the Front-End Engineering Design (FEED) contract, awarded in October 2013 to a consortium formed by AREVA, ANSALDO, EMPRESARIOS AGRUPADOS and GRONTMIJ, was suspended in the beginning of 2015. The reason for the suspension is that a deep review of the primary system configuration of MYRRHA was needed. Activities have been conducted during 2016 to improve the reactor design configuration and are expected to be continued as well during 2017. In September 2016 the MYRRHA Management Team took a major decision to concentrate the activities on the development of a 100 MeV accelerator, expected to be operational by 2024. At the same time activities on reactor and upgrade of the accelerator to 600 MeV are underway in order to be able in 2024 to start procurement for both reactor and the accelerator upgrade.

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

3.5. People's Republic of China

In China, the Chinese Academy of Sciences (CAS) launched a project to develop ADS and lead-based fast reactors technology since 2011. The China Lead-based Reactor (CLEAR), proposed by the Institute of Nuclear Energy Safety Technology, was selected as the reference reactor for ADS development, as well as for the technology development of the Generation

IV lead-cooled fast reactor. The program consists of three stages with the goal of developing a 10MWth lead-based research reactor (CLEAR-I), a 100MWth lead-based engineering demonstration reactor (CLEAR-II) and a 1000MWth lead-based commercial prototype reactor (CLEAR-III). To promote the CLEAR project successfully, INEST is deeply involved in reactor design, reactor safety assessment and in design and analysis software development, testing activities using lead-bismuth experimental loops, key technologies and components R&D activities. The detailed conceptual design of CLEAR-I has been completed, and the engineering design is underway, which has subcritical and critical dual-mode operation capability for validation of the ADS transmutation system and the LFR technologies. The KYLIN series LBE experimental loops have been constructed and have operated for more than 10,000h. R&D activities on structural material corrosion experiments, oxygen control technology development, thermal-hydraulics tests and safety experiments are underway. The key components, including the control rod drive mechanism, refueling system, fuel assembly, and simulator for principle verification etc., have been fabricated and tested. In order to validate and test the key components and integrated operating technology of the lead-based reactor, the lead alloy-cooled non-nuclear reactor CLEAR-S, the lead-based zero-power nuclear reactor CLEAR-0, and the lead-based virtual reactor CLEAR-V are under realization. In addition, series of innovative concepts for different purposes are being developed to enlarge the application perspective of lead-based reactors, which are not only for ADS and fast reactor, but also for other innovative applications, such as CLEAR-SFB for spent fuel burning, CLEAR-Th for thorium utilization, CLEAR-H for hydrogen production, etc.

3.6. United States

Work on LFR concepts and technology in the U.S. has been carried out since 1997. In addition to reactor design efforts, past activities included work on lead corrosion and thermal-hydraulic testing at a number of 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 US are very limited, past and ongoing efforts at national laboratories, universities and the industrial sector demonstrate continued interest in LFR technology. With regard to design concepts, of particular relevance is the past development of the Small, Secure Transportable Autonomous Reactor (SSTAR), carried out by Argonne National Laboratory (ANL), Lawrence Livermore National Laboratory (LLNL) and other 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 refueling; and an innovative supercritical CO2 (S-CO2) Brayton cycle power conversion system. Additional university-related design activities include past work at the University of California on the Encapsulated Nuclear Heat Source (ENHS) and more recent efforts at the University of Alaska and Texas A&M University to design a Passively Operated Lead Arctic Reactor (POLAR). In the US industrial sector, ongoing LFR reactor initiatives include the Gen4 Module (G4M) by Gen4 Energy, a new LFR reactor concept identified as LFR-AS (Amphora Shaped) by Hydromine, Inc., and a recent initiative by Westinghouse Corporation to design a new advanced LFR system.

4. Conclusion

The paper briefly outlines the present activities of the LFR-pSSC in the frame of GIF. Details have been presented about the status of developments in the individual signatories of GIF-LFR-MoU. The number of activities and involvement shows that the LFR should be considered as a promising technology for future nuclear development in the world.

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