LEAD-COOLED FAST REACTOR (LFR): OVERVIEW AND PERSPECTIVES

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I. INTRODUCTION

The *GIF Technology Roadmap* [1] identified the Lead-cooled Fast Reactor (LFR) as a technology with great potential to meet the small-unit electricity needs of remote sites while also offering advantages as a large system for grid-connected power stations. The LFR features a fast-neutron spectrum and a closed fuel cycle for efficient conversion of fertile uranium. It can also be used as a burner of minor actinides from spent fuel and as a burner/breeder. An important feature of the LFR is the enhanced safety that results from the choice of a relatively inert coolant. In the Roadmap, the LFR was primarily envisioned for missions in electricity and hydrogen production, and actinide management.

The application of lead technology to nuclear energy had its start in Russia in the 1970s and 80s where nuclear systems cooled by Lead-Bismuth Eutectic (LBE) were developed and deployed for submarine propulsion. More recently, attention to heavy liquid metal coolants for reactors has developed in several countries around the globe as their advantageous characteristics have gradually become recognized. This paper illustrates the technical progress achieved in the various countries.

II. LFR IN GENERATION IV

International cooperation on LFR within GIF was initiated in October 2004 and the first formal meeting of the Provisional System Steering Committee (LFR-PSSC) was held in March 2005 in Monterey, CA, USA, with participation of representatives from EURATOM, Japan, the United States and experts from the Republic of Korea. Since then, the PSSC has held regular scheduled meetings, roughly twice a year, with additional working sessions to prepare and update the draft LFR System Research Plan (SRP) [2].

The draft SRP was reviewed by the GIF Experts Group (EG) in mid-2007 and again in mid-2008. The formal PSSC meetings were supplemented by additional informal meetings with representatives of the nuclear industry, research organizations and universities involved in LFR development.

The preparation of a System Arrangement for approval by participating GIF members has been considered, but formal agreements are still pending.

The preliminary evaluation of the LFR concepts considered by the PSSC addresses their performance in the areas of sustainability, economics, safety and reliability, proliferation resistance and physical protection.

The designs that are currently proposed as candidates for international cooperation and joint development in the GIF framework are two pool-type reactors:

- the Small Secure Transportable Autonomous Reactor (SSTAR); and
- the European Lead-cooled System (ELSY).

Key design data of SSTAR and ELSY are presented in Table I.

Parameter/system	SSTAR	ELSY
Power (MWe)	19.8	600
Conversion Ratio	~1	~1
Thermal efficiency (%)	44	42
Primary coolant	Lead	Lead
Primary coolant circulation (at power)	Natural	Forced
Primary coolant circulation for DHR	Natural	Natural
Core inlet temperature (°C)	420	400
Core outlet temp. (°C)	567	480
Fuel	Nitrides	MOX, (Nitrides)
Fuel cladding material	Si-Enhanced F/M Stainless Steel	T91 (aluminized)
Peak cladding temp. (°C)	650	550
Fuel pin diameter (mm)	25	10.5
Active core Height/ equivalent diameter (m)	0.976/1.22	0.9/4.32
Primary pumps	-	N° 8, mechanical, integrated in the SG
Working fluid	Supercritical CO_2 at 20MPa, 552°C	Water-superheated steam at 18 MPa, 450°C
Primary/secondary heat transfer system	N°4 Pb-to- CO_2 HXs	N°8 Pb-to-H_2O SGs
Safety grade DHR	Reactor Vessel Air Cooling System + Multiple Direct Reactor Cooling Systems	Reactor Vessel Air Cooling System + Four Direct Reactor Cooling Systems + Four Secondary Loops Systems

TABLE I: Key Design data of GIF LFR concepts

III. SSTAR

The current reference design for the SSTAR [3] in the United States is a 20 MWe natural circulation reactor concept with a small shippable reactor vessel (Figure 1).

The Pb coolant is contained inside a reactor vessel surrounded by a guard vessel. Lead is chosen as the coolant rather than LBE to drastically reduce the amount of alpha-emitting ^{210}Po isotope formed in the coolant relative to LBE, and to eliminate dependency upon bismuth which might be a limited or expensive resource. The Pb flows upward through the core and a chimney above the core formed by a cylindrical shroud. The vessel has a height-to-diameter ratio large enough to facilitate natural circulation heat removal at all power levels up to and exceeding 100% of nominal. The coolant flows through openings near the top of the shroud and enters four modular Pb-to-CO_2 heat exchangers located in the annulus between the reactor vessel and the cylindrical shroud. Inside each heat exchanger, the Pb flows downwards over the exterior of tubes which contain upward-flowing CO_2. The CO_2 enters each heat exchanger through a top entry nozzle which delivers the CO_2 to a lower plenum region. From this lower plenum, the CO_2 enters each of the vertical tubes and flows upward to an upper plenum. The hot CO_2 then exits the heat exchanger through two smaller diameter top entry nozzles. Meanwhile, the Pb exits the heat exchangers and flows downward through the annular downcomer to enter the flow openings in the flow distributor head beneath the core.

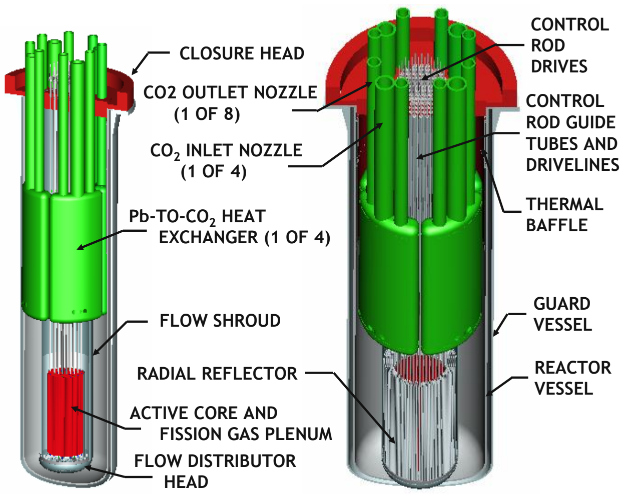

Figure 1: Small Secure Transportable Autonomous Reactor (SSTAR).

Specific features of the lead coolant, the nitride fuel containing transuranic elements, the fast spectrum core, and the small size combine to promote a unique approach to achieve proliferation resistance, while also enabling fissile self-sufficiency, autonomous load following, simplicity of operation, reliability, transportability, as well as a high degree of passive safety.

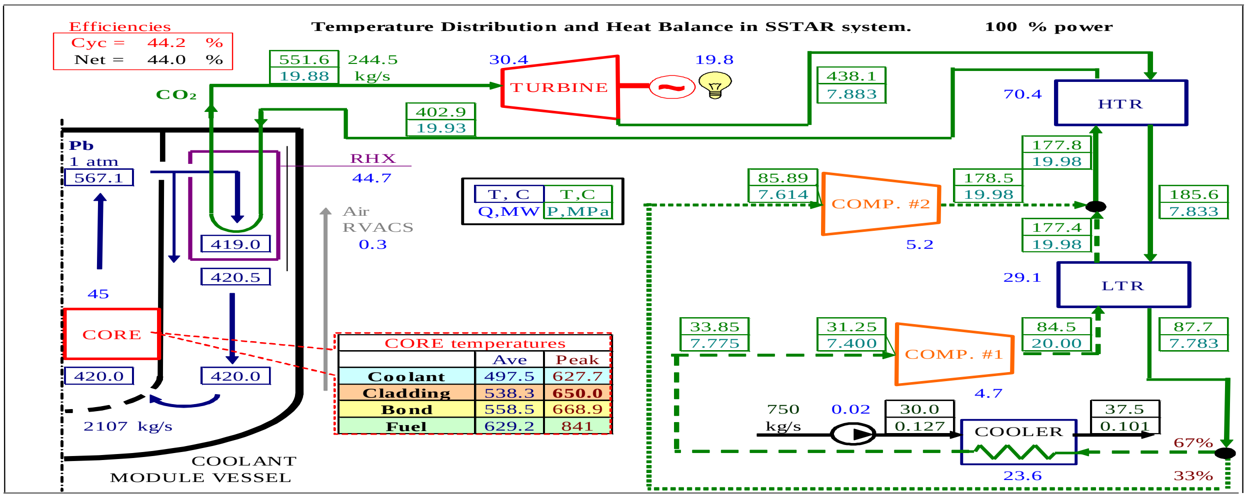

Figure 2: SSTAR pre-conceptual design concept and operating parameters with S-CO_2 Brayton-cycle energy converte

Conversion of the core thermal power into electricity at a high plant efficiency of 44% is accomplished utilizing a supercritical carbon dioxide Brayton cycle power converter (Figure 2).

The SSTAR preconceptual design assumes that a number of advanced technologies will be successfully developed including yet to be developed and code-qualified advanced cladding and structural materials enabling service in Pb for 15 to 30 years at peak cladding temperatures up to about 650°C at a core outlet temperature of 570°C, qualified transuranic nitride fuel meeting fuel performance requirements, whole-core cassette refuelling, and in-service inspection approaches for components immersed in Pb coolant. In turn, the SSTAR concept provides a driver for the development of the advanced technologies. If SSTAR were to be developed for near-term deployment, then the operating system temperatures would likely be reduced (*e.g.*, 480°C, as in the ELSY design) to enable the use of existing codified materials and an existing fuel type such as metallic fuel may have to be qualified and utilized. In the US, an initial scoping investigation has been carried out into the viability of a near-term deployable LFR technology pilot plant/demonstration test reactor (demo) operating at low temperatures enabling the use of existing materials such as T91 ferritic/martensitic steel or Type 316 stainless steel shown in numerous worldwide tests conducted during the past decade to have corrosion resistance to lead alloys at temperatures up to ~ 550°C with active oxygen control. Neutronic and system thermal hydraulic analyses indicate that a 100 MWth lead-cooled metallic-fueled demo with forced flow and a 480°C core outlet temperature supporting the development of both the ELSY and SSTAR LFRs may be a viable concept power converter.

IV. ELSY

The ELSY reference design is a 600 MWe reactor cooled by pure lead [4]. ELSY has been under development since September 2006, and is funded by the Sixth Framework Programme of Euratom. The ELSY project is being conducted by a large consortium of European organizations to demonstrate the possibility of designing a competitive and safe fast critical reactor using simple engineered features, while fully complying with Generation IV goals, including that of minor actinide burning capability.

The use of a compact and simple primary circuit with the additional objective that all internal components be removable, are among the reactor features intended to assure competitive electric energy generation and long-term investment protection. Simplicity is expected to reduce both the capital cost and the construction time; these are also supported by the compactness of the reactor building (*i.e.*, reduced footprint and height). The reduced footprint would be possible due to the elimination of the intermediate cooling system, with a reduced elevation resulting from the design approach of reduced-height components.

One of the main objectives of ELSY from the beginning of the activity has been the identification of innovative solutions to reduce the primary system volume and the complexity of the reactor internals. The result is that most components are unconventional (Figure 3).

A newly designed steam generator (SG), whose volume is about half that of a comparable helical-tube SG, is characterized by a spiral-wound tube bundle. The inlet and outlet ends of each tube are connected to the feed water header and steam header, respectively, both arranged above the reactor roof. An axial-flow primary pump, located inside the inner shell of the SG, provides the head required to force the coolant to enter from the bottom of the SG and to flow in a radial direction. This scheme is almost equivalent to a pure counter-current scheme, because the water circulates in the tube from the outer spirals towards the inner spiral, while the primary coolant flows in the radial direction from the inside to the outside of the SG.

This ensures that the coolant will flow over the SG bundles even in the event of reduction in the primary coolant level in case of leakage from the reactor vessel. As a by-product, the SG unit can be positioned at a higher level in the downcomer and the Reactor Vessel (RV) shortened, accordingly.

All reactor internal structures are removable and in particular the SG Unit can be withdrawn by radial and vertical displacements to disengage the unit from the reactor cover plate.

The core consists of an array of 162 open fuel assemblies (FAs) of square pitch surrounded by reflector-assemblies, a configuration that presents reduced risk of coolant flow blockage. An alternative solution with closed hexagonal FAs has also been retained as a fall-back option. The core is self sufficient in plutonium and can burn its own generated minor actinides with a content at equilibrium of about 1% heavy metal.

The upper part of the FA is peculiar to this novel ELSY design, because it extends well above the fixed reactor cover, and the fuel elements, the weight of which is supported by buoyancy in lead, are fixed at their upper end in the cold gas space, well above the molten lead surface. This avoids the classical problem of a core support grid immersed in the coolant which would require a tricky procedure for In-Service Inspection (ISI) in the molten lead.

FA heads are directly accessible for handling using a simple handling machine that operates in the cover gas at ambient temperature, under full visibility.

Considering the high temperature and other characteristics of the molten lead environment, any approach that foresees the use of in-vessel refuelling equipment, would represent a tremendous R&D effort and substantial associated technical risk, especially because of the need to develop reliable bearings operating in lead, an unknown technology at present. For these reasons the adopted design approach represents a real breakthrough.

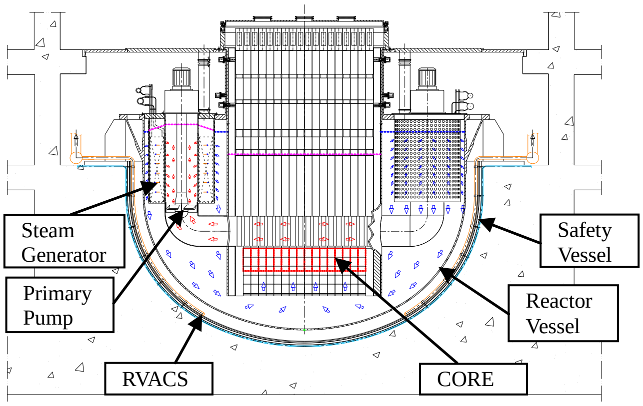

Figure 3: ELSY Primary system configuration.

The installation of SGs inside the reactor vessel is another major challenge of a LFR design that has been resolved by the selected approach. Particular challenges related to the operation of in-vessel SGs include the need for:

- a sensitive and reliable leak detection system;

- a highly reliable depressurization and isolation system.

Careful attention has been also given to the issue of mitigating the consequences of the Steam Generator Tube Rupture (SGTR) accident to reduce the risk of pressurization of the primary boundary; to this end, innovative provisions have been conceived which make the primary system more tolerant of the SGTR event.

The first provision is the elimination of the risk of failure of the water and steam collectors inside the primary boundary by installing them outside the reactor vessel. This approach aims to eliminate by design a potential initiator of a severe accident of low probability but potentially catastrophic consequences.

The second provision is the installation on each tube of a check valve close to the steam header and of a venturi nozzle close to the feed water header.

The third provision aims at ensuring that the flow of any feedwater-steam-primary coolant mixture be re-directed upwards inside the SG, reducing by design the risk of propagation of large pressure waves across the reactor vessel.

This occurs because the inner pressure surge itself promptly causes the closure of the normal radial coolant flow path. The redundant, diverse Decay Heat Removal (DHR) system is provided with (i) steam condensers on the steam loops, (ii) direct reactor cooling loops with innovative lead-water dip coolers using storage water at ambient pressure and (iii) a Reactor Vessel Air Cooling System (RVACS).

General seismic behaviour is strongly improved by the embodied technical solutions, in particular the short-height vessel and the 2D antiseismic supports above the reactor building. Additional loads under investigation are lead sloshing resulting from seismic motion or as a result of a SGTR accident. An extensive safety analysis is also ongoing to address accidents representative of design basis conditions and of design extended conditions.

A preliminary plant Layout showing the main buildings is presented in Figure 4.

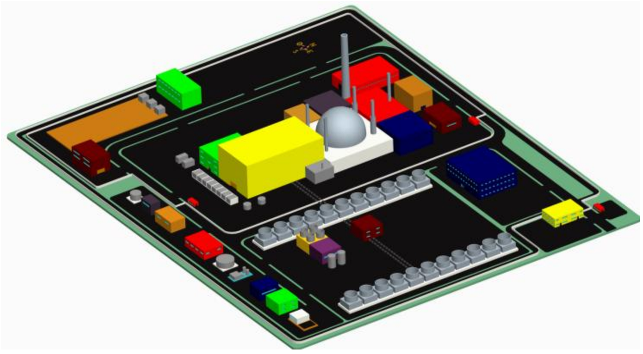

Figure 4: ELSY Preliminary plant layout

V. OTHER RELEVANT ACTIVITIES ON LFR

In addition to the ongoing activities in Europe (ELSY) and the USA (SSTAR), it is important also to recognize the ongoing LFR efforts elsewhere.

Research activities in Japan concentrate on the heat transfer performance of LBE in the intermediate loop; two phase flow characteristics of LBE in water and steam; the gas and steam lift performance of LBE; LBE-water direct contact boiling mechanism; the corrosion characteristics and corrosion behaviour of the reactor coolant; the structural and cladding materials; oxygen control with steam injection into LBE; and Polonium behaviour in the coolant system.

The LFR program is strongly promoted in the Center for Research into Innovative Nuclear Energy Systems in Tokyo Institute of Technology. It covers wide areas of lead and LBE coolant studies such as nuclear reactor design studies, cross section measurements, thermal hydraulics experiment especially for steam lift pump, [5] static and dynamic corrosion test, [6] and polonium behavior experiments. [7] The design studies include several kinds of CANDLE reactors [8] and the LBE Cooled Direct Contact Boiling Water Fast Reactor (PBWFR) with electric power of 150 MW. [9]

Two systems are developed in the Republic of Korea, the proliferation-resistant, environment - friendly, accident-tolerant, continual and economical reactor (PEACER) [10] and the BORIS [11]. In the

Russian Federation, two systems are considered: the SVBR-75/100, a LBE-cooled modular fast reactor having a power range of 75 to 100 MWe [12], and the BREST lead-cooled fast reactor concept and the associated fuel cycle. [13]

VI. CONCLUSION

The draft SRP for the Lead-Cooled Fast Reactor has pure lead as the reference coolant and the LBE as a fall-back option. The basic approach recommended in the draft SRP portrays the dual track viability research program with convergence to a single, combined Technology Pilot Plant (TPP) leading to eventual deployment of both types of systems.

The approach adopted aims at addressing the research priorities of each participant party, while developing an integrated and coordinated research program to achieve common objectives and avoid duplication of effort.

Following the successful operation of the TPP around the year 2020,[1] a prototype independent development effort is expected for the central station LFR and the SSTAR.

The design of the industrial prototypes of the central station LFR and of the SSTAR should be planned in such a way as to start construction as soon as beginning of the TPP operation at full power has given assurance of the viability of this new technology.

[1] This is consistent with the European Sustainable Nuclear Energy Technology Platform: "Whilst the SFR remains the reference technology, two alternative technologies for fast reactors, namely the gas-cooled fast reactor (GFR) and the lead-cooled fast reactor (LFR) also need to be assessed at European level. After selection of an alternative technology, an experimental reactor in the range of 50-100 MWth will be needed to gain experience feedback by 2020 on this innovative technology".

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Nomenclature

DHR	Decay Heat Removal
ELSY	European Lead-cooled System
FA	Fuel Assembly
GIF	Generation IV International Forum
LBE	Lead-Bismuth Eutectic
LFR	Lead-Cooled Fast Reactor
ISI	In-Service Inspection
PSSC	Provisional System Steering Committee
RV	Reactor Vessel
RVACS	Reactor Vessel Air Cooling System
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SRP	System Research Plan
TPP	Technology Pilot Plant

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