Proceedings of GLOBAL 2011
Makuhari, Japan, Dec. 11-16, 2011
Paper No. 392485

Design Concepts of LFRs and Related Studies in CRINES of Tokyo Tech.

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ABSTRACT: This paper gives an overview of our feasibility studies on lead alloy-cooled fast reactor (LFR) performed since the early 1990s. It includes the main features of LFR, three innovative LFR concepts proposed: LSPR, PBWFR and LFR with minimum release of radioactive waste outside, neutronics study, safety analysis, material compatibility tests, polonium tests, and thermal-hydraulic tests in Center for Research into Innovative Nuclear Energy Systems (CRINES), Tokyo Institute of Technology (Tokyo Tech.). It is expected that this will be a guide for future studies in this field.

KEYWORDS: *Lead, Lead-bismuth Eutectic, Fast Reactor, Reactor Core, Corrosion, Welding, Polonium, Diffusion, Droplet, Direct Contact, Boiling, Two-phase Flow, Vapor Explosion*

I. INTRODUCTION

Lead-cooled fast reactor (LFR) is one of the six identified and selected nuclear energy systems for further development in Generation IV International Forum (GIF). The coolant of LFR is lead or lead-bismuth eutectic (LBE, 45%Pb-55%Bi). Recently, lead-cooled systems have been selected because lead resource is abundant and meet the requirements of sustainability. However, the main features of LBE-cooled system are the same as those of the lead-cooled systems. Thus, we deal with both of them as LFR in the present paper.

The design concept of small LFR was proposed in Tokyo Institute of Technology (Tokyo Tech.) in the early 1990s [1-3], and related feasibility studies have been performed so far. The research activity has been continued in Center for Research into Innovative Nuclear Energy Systems (CRINES), Tokyo. Tech. The GIF Memorandum of Understanding (MOU) for collaboration on LFR between CRINES and European Union started in 2010, and Rosatom (Russia) joined it in 2011.

The present paper provides the overview of the innovative design concepts of LFRs that have been proposed and the related studies conducted in CRINES. This is one of the activities of CRINES for the international collaboration on LFR in GIF.

II. MAIN FEATURES OF LFR CONCEPT

The most advantageous features of the LFR concept are the inherently and passively safe characteristics. They are summarized below.

(i) LFR shows better performances for neutron economy, burnup reactivity swing and void coefficient than sodium-cooled fast reactor (SFR) because of its large scattering cross section and heavy nuclide mass.
(ii) Lead and LBE coolants have much higher boiling temperatures of 1737˚C K and 1670˚C, respectively, than sodium with 882˚C, which eliminate the potential dangers of coolant boiling in transient.
(iii) Lead and LBE coolants are inert with water and air, which eliminate the potential dangers of coolant-induced fire.
(iv) Lead and LBE coolants have the lowest stored potential energy among water, sodium and lead alloy coolants, which causes neither release of chemical and mechanical energy nor loss of coolant in core due to vaporization [4]
(v) The use of broad fuel element lattices, or larger core equivalent hydraulic diameter, provide higher level of natural circulation compared to SFR. In case of SFR, a high fuel volume fraction is employed for higher conversion ratio and a compact core design.
(vi) As lead alloy coolants are heavy, in case of the core disruptive accident (CDA), the re-critical accident can be avoided more easily than SFR due to the lifting and dispersion of fuel pellets in the heavy metal coolant.

The additional advantages of LFR except for safety are as follows:

(i) The LBE shows large neutron confinement effects resulted from large scattering cross section. It can make the core size small.
(ii) The LBE shows also large shielding effects for neutrons and gamma-rays. It can reduce the thickness of reflector and shielding.
(iii) LBE does not produce so much gamma-ray emitters. Then, the dose-rate around the primary loop of LFR is expected much lower than SFR where ^{24}Na with half-life of 15 h emits high-energy gamma-rays.
(iv) Lead alloy coolant can be circulated at desired flow rates by the lift force of gas bubbles in the coolant because of

heavy metal.

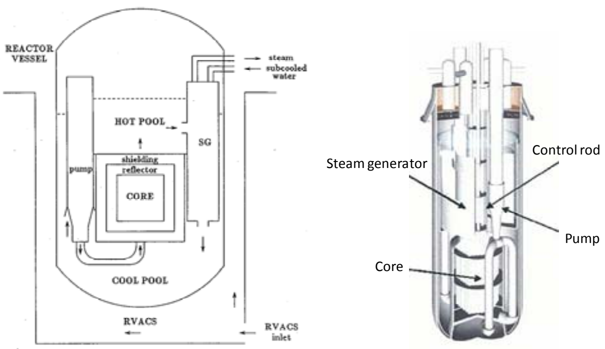
Research and development have been performed mainly to overcome the following drawbacks of LFR:

(i) A large amount of alpha-ray emitter, ^{210}Po, is produced in LBE from neutron irradiation of Bi in LFR.
(ii) Lead alloys are corrosive to structural materials, particularly austenitic stainless steels that have high content of Ni.
(iii) Lead alloys are very heavy, which restricts the size of LFR (middle size) for seismic measure, and the coolant velocity (Max. 2m/s) for erosion measure.
(iv) The operation temperature is limited for lead coolant with high melting temperature 327˚C.
(v) Bi resource is not abundant.

III. DESIGN CONCEPTS OF LFRS

1. Concept of LSPR

As a small reactor with long life core, the concept of LSPR (LBE-cooled long-life Safe Simple Small Portable Proliferation resistant Reactor) was proposed [5-8] (Figs. 1 through 3, Table 1). The features of LFR meet all of the requirements of the LSPR concept best among various types of reactors including SFR.



(a) Reactor vessel (b) Bird's eye view of LSPR
Fig. 1 Design concept of LSPR [5-8]

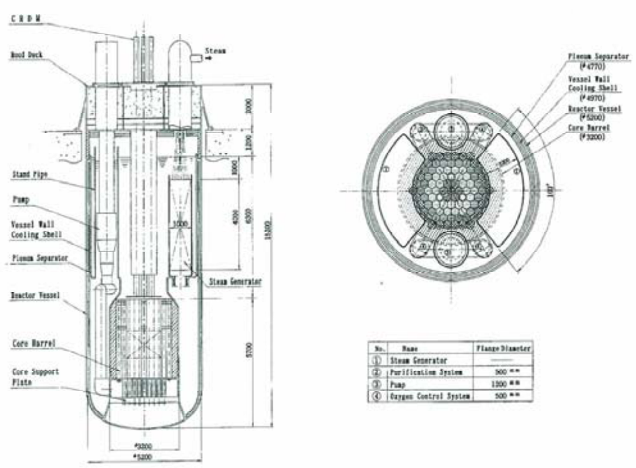


Fig. 2 Design details of LSPR [5-8]

Small reactors will be constructed in factories of the nuclear energy park, transported to the site, and set up. The reactor vessel is sealed without being opened at the site for refueling, which is excellent for proliferation resistance. At the end of the reactor life, it is replaced by a new one. The old one is shipped to the nuclear energy park. There is no radioactive waste left at the site. In other words, the site is free from the waste problems. This concept on the system of the nuclear energy park and small reactors has been continued by COE-INES (currently, CRINES), and has been reflected to the dual track approach based on the development of ELSY and SSTAR. LSPR corresponds to SSTAR, although the former is cooled by forced circulation of LBE and the latter is cooled by natural circulation of lead.

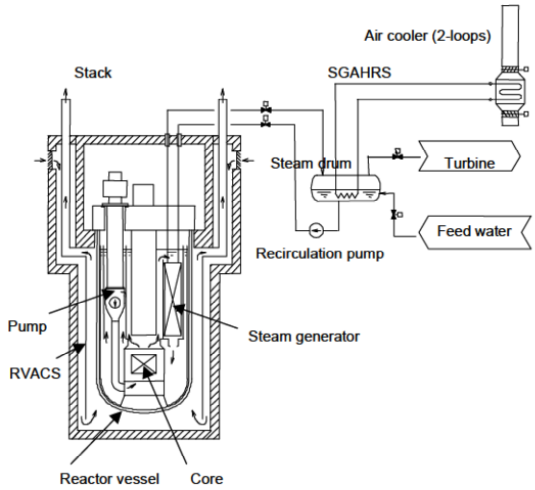


Fig. 3 Steam turbine and decay heat removal systems of LSPR [5-8]

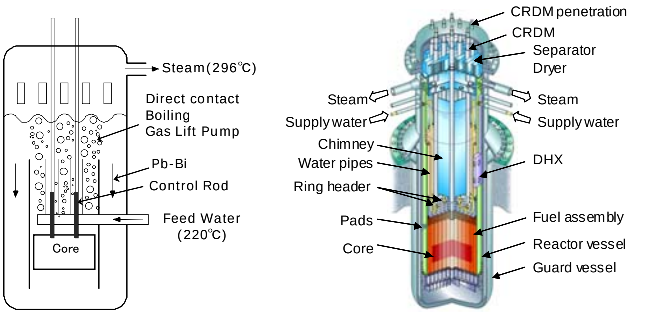
Table 1 Major parameters of LSPR [5-8] and PBWFR [10]

	LSPR	PBWFR
Power, Thermal/Electric (MW)	150 / 53	450 /150
Thermal efficiency (%)	35	33
Core, Diameter/Height (m)	1.652 / 1.08	2.78 / 0.75
Fuel pin diameter (mm)	10	12
P/D, Inner core/Outer core	1.12 / 1.18	1.3 / 1.3
Linear power density (W/cm)	51.9 (Av.)	363 (Max.)
Pump, Type/Unit number	Mechanical / 2	Gas lift / 1
Temperature, Inlet /Outlet (˚C)	360 / 510	310 /460
Coolant flow rate (t/h)	12,300	73,970
SG, Type/Unit number	Serpentine tube / 2	Direct contact / 1
Temperature, Feed water/Steam (˚C)	210 / 280	220 / 296
Steam pressure (MPa)	6.47	7.0
RV, Diameter/Height (m)	5.2 / 15.2	4.69 / 19.8
Refueling interval (y)	12	10

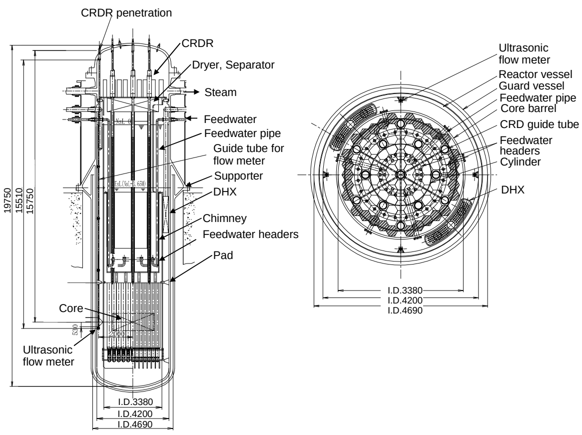
2. Concept of PBWFR

Lead alloy is corrosive and causes serious erosion to structural materials in certain conditions. To avoid the corrosion and erosion problem, the components that contact lead alloy should be eliminated as much as possible. Particularly, the concern is corrosion on tube surfaces of steam generators (SG) exposed to high temperature coolant, and erosion on the surfaces of impellers of primary pumps exposed to high velocity flow.

Thus, the feasibility of the elimination of the SGs and the primary pumps by direct injection of a feed water into hot LBE above the core has been studied. The injected feed water boils in a chimney and steam bubbles go up with buoyancy force. The bubble motion serves as a driving force of coolant circulation in the use of the heavy coolant. This design concept of LFR is called PBWFR (Pb-Bi-cooled direct contact boiling Water Fast Reactor) [9, 10] (Figs. 4 through 6, Table 1).



(a) Concept of PBWFR (b) Bird's eye view of PBWFR
Fig. 4 Design concept of PBWFR [9,10]



(c) Design details
Fig. 5 Design concept of PBWFR [10]

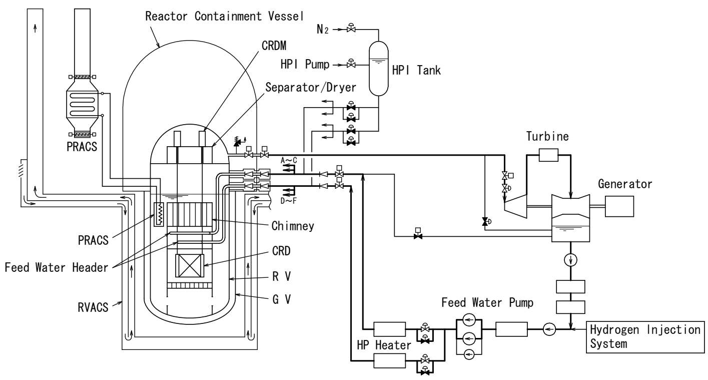


Fig. 6 PBWFR plant flow diagram with decay heat removal system [10]

3. LFR for Confinement of MA and LLFP

For the reduction of release of radioactive wastes (RW) as much as possible, the possibility of critical equilibrium reactor was studied [11], where the reactor confines all of the self-produced transuranium (TRU) and long-lived fission products (LLFP) in the core. The features of LFR are suitable for the reactor. The confinement consists of three cases (Fig. 7) depending on technical development as follows:

(a) Case 1: Natural uranium is supplied to the reactor, and minor actinides (MA) , FP and a part of plutonium are released from the reactor. The toxicity of the released waste is higher than that of the supplied uranium. The accumulation of RW in repository is $2.3x10^7$ t/GWt.
(b) Case 2: Natural uranium is supplied, TRU including plutonium and MA is confined, and FP is released. The toxicity of the released waste is lower than that of supplied U. The accumulation of RW in repository is $1.4x10^5$ t/GWt.
(c) Case 3: Natural uranium is supplied, TRU and LLFP are confined, and all FPs except ^{79}Se, ^{93}Zr, ^{99}Tc, ^{107}Pd, ^{126}Sn, ^{129}I and ^{135}Cs are released. The toxicity of the released wastes is lower than that of supplied U. The accumulation of RW in repository is 1.0 t/GWt.

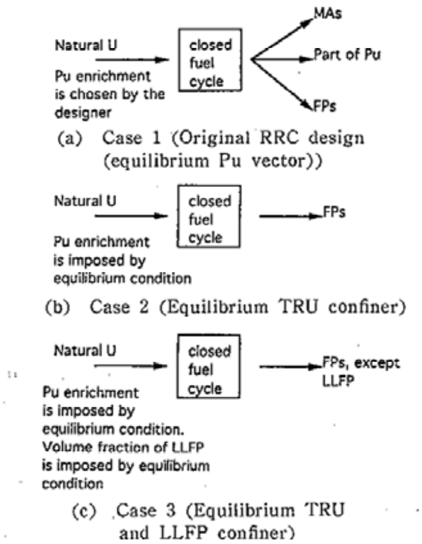


Fig. 7 Three phases of waste confinement in LFR [11]

IV. RELATED STUDIES

1. Neutronics Study

The neutronic performance of LFR and SFR was compared with each other for the small size long life fast reactor shown in Fig. 8 [2]. In general, the coexistence of low excess reactivity less than β_{eff} , low reactivity swing and negative coolant void coefficient during burn-up period should be pursued for safety core design. Low excess reactivity is necessary to avoid the transient over power (TOP) accident. The reactivity swing lower than or equal to 0.1%Δk could be achieved in both of LFR and SFR.

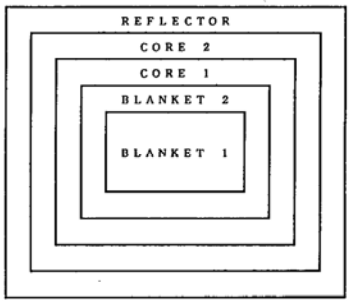
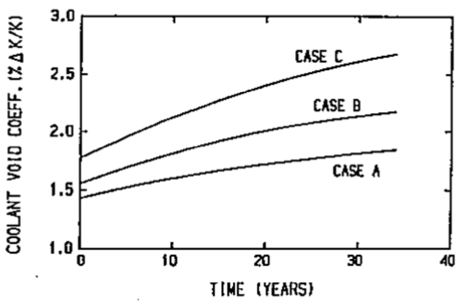


Fig. 8 Configuration of core [2]

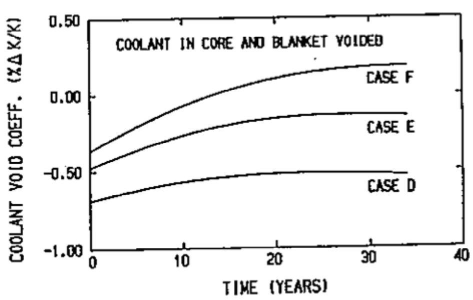
Table 2 Result of design study, where Pu enrichment is 8.5% (inner), 9.25% (outer) in SFR; 8.0% (inner), 9.0% (outer) in LFR [2]

Coolant		Power (MWt)	Core	Reflector	Coolant void coefficient	
			Radius (cm)	Width (cm)	BOL	EOL
SFR	A	50	68.75	40.0	1.438	1.814
SFR	G	50	71.25	22.5	1.194	1.580
LFR(LBE)	D	50	64.3	17.5	-	-
LFR(LBE)	H	50	63.5	22.5	-0.335	-0.131
SFR	B	75	72.0	40.0	-	-
LFR(LBE)	E	75	70.0	17.5	-	-
SFR	C	100	79.0	40.0	-	-
LFR(LBE)	F	100	76.0	17.5	-	-
SFR	I	900	146.5	40.0	-	-

However, the coolant void coefficient could not be reduced to negative in SFR (Table 2, Fig. 9). In case of SFR, the employment of pancake or tall slender core or the insertion of absorber material can reduce the coolant void coefficient negative, but the reactivity swing becomes larger. The mechanism of positive coolant void coefficient, particularly in a large SFR, is the hardening of neutron spectrum which increase the neutron production to absorption ratio. On the other hand, LBE has lower moderating power and higher transport cross section than sodium. Therefore, in case of LFR, the neutron leakage in voiding is more dominant to the void coefficient than the decrease of neutron absorption due to the neutron spectrum hardening.



(a) SFR



(b) LFR (LBE)

Fig. 9 Comparison of coolant void coefficient between SFR and LFR [2]

Neutronic performance was compared among LFRs with metallic and nitride fuels and lead and LBE coolant (Fig. 10, Table 3) [12,13]. In order to minimize the excess reactivity and coolant void coefficient, fertile material was charged in the central core, and the most active component was located in the outer core so that neutron leakage was higher in coolant dilatation or voiding.

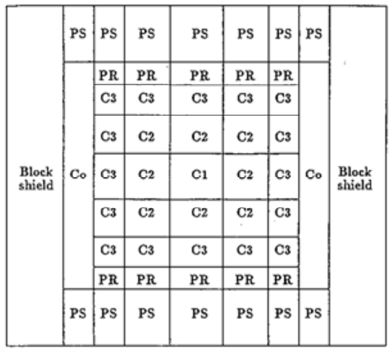


Fig. 10 Core configuration model, where C1-C3: Core, C0: Coolant, PS: Shield (B_4C), PR: Reflector (SS) [12,13]

Table 3 Reactor design parameters [12,13]

Reactor power (MWt)	150
Life time (yr)	12
Shielding material	B_4C
Fuel	U-Pu-10%Zr metallic or UN-PuN nitride
Reactivity swing (%$\Delta k/k$)	< 0.1
Void reactivity coefficient	Negative over the whole life time
Peak burnup (% heavy metal)	9

The result showed that the metallic fuel gives lower coolant void coefficient than the nitride fuel, and LBE coolant gives lower coefficient than lead coolant (Fig. 11).

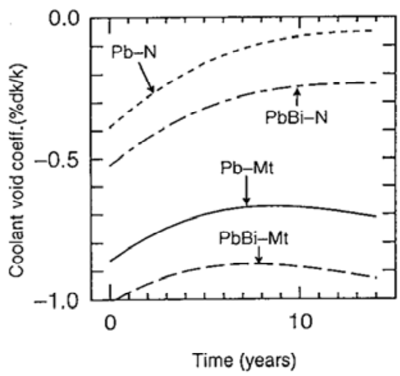


Fig. 11 Comparison of coolant void coefficient among metallic and nitride fuels and lead and LBE coolant [12,13]

The other comparison of neutronic performance between LFR and SFR is as follows:

(i) As the reflector effect of LBE is higher than sodium, the reflector width is narrower in LFR than in SFR, and the size of LFR is smaller than that of SFR for the same power level (Table 2).
(ii) The temperature coefficient generally is lower in LFR than in SFR.
(iii) Conversion ratio is nearly the same between SFR and LFR [2].

As an alternative coolant, the use of ^{208}Pb was proposed

[14]. Since ^{208}Pb has a smaller capture and inelastic-scattering cross section than the other natural lead isotopes, the use of ^{208}Pb makes it possible to reduce neutron capture by coolant and to make neutron spectrum harder. As a result, the core dimensions and pressure drop can be reduced, which is good for the transportable-compact reactor for isolated areas. In the neutronic study, it was confirmed that the core volume could be reduced by less than ~20%, and when natural Pb coolant was changed to ^{208}Pb, the effective multiplication factor increased from 0.984 to 1.006.

2. Safety Analysis

Neutronic and thermal-hydraulic analyses were performed for the safety evaluation of LSPR [15,16]. The effects of differences of fuel and coolant on the safety characteristics were studied. As a result, all of the LFRs could survive the UTOP (complete withdrawal of all control rods), ULOF (loss of primary loop pumping power without scram), simultaneous UTOP and ULOF and simultaneous UTOP, ULOF and ULOHS (loss of secondary loop pumping power without scram) accidents without the help of an operator or active devices. The safety was confirmed because the fuel and cladding temperatures at hot spot that reached were 700-800ºC at the events [15,16]. Corrosion characteristics of materials were investigated for the transient high temperature events as mentioned later.

Doppler effect was large and dominant for the nitride fuel core, but relatively small and less important for the metallic fuel core. The radial expansion gave the largest contribution and was similar for both metallic and nitride fuel cores. The effects of fuels and coolants on reactivity coefficients were as follows:

(i) The nitride fuel gave a more negative Doppler coefficient than the metallic fuel owing to a softer spectrum caused by moderation by the nitrogen nucleus.
(ii) The metallic fuel gave more negative coolant density and radial expansion coefficients, and axial expansion coefficient than the nitride fuel owing to a larger leakage component and the larger thermal expansion coefficient, respectively.
(iii) LBE coolant gave more negative Doppler and coolant density coefficients than the lead coolant, although fuel axial expansion and core radial expansion coefficients were similar. The differences were attributed to the difference in enrichment and the scattering cross section of coolants.

In case of PBWFR, there is no concern of LOF (loss of flow) accident due to primary pump trip because of no pumps. However, the feed-water pump trip will initiate LOF event, where high pressure water will be injected into the reactor vessel for a slowdown of the flow coast down of feed-water. The safety performance at the ULOF-ULOHS event (unprotected loss of flow and heat sink without scram) was evaluated [17]. The result showed that the fuel temperatures were kept lower than the safety limits. The cladding temperature at hot spot reached 792ºC in the simultaneous ULOHS and ULOF event in PBWFR.

3. Material Compatibility Tests

Lead alloy coolants are generally corrosive to steels, particularly to austenitic stainless steels that contain Ni. Therefore, the compatibility of materials is one of the key issues in the feasibility study of LFRs. The compatibility depends mainly on coolant temperatures and oxygen concentration in the coolant. The core outlet coolant temperature of coolant ranges in 450-550°C in most of the LFRs. Cladding tube at hot spot reaches 650°C in some designs.

Thus, for the candidate structural materials in the hot leg, the corrosion test of existing steels were conducted using a LBE flow loop with the test section temperature of 550°C under well controlled oxygen concentration (5.0×10^{-7} wt%) [18, 19]. It is noted that the oxygen concentrations (3.7×10^{-8} wt%) was under-estimated in [19]. The result showed better corrosion resistance for high Cr feritic-martensitic steels (STBA26, SUS405, SUS430, HCM12, HCM12A) than for low Cr steel (SCM420) and austenitic stainless steel (SS316) (e.g., Figs. 18). However, under lower oxygen concentration of 7.5×10^{-8} wt.% [18], erosion damage took place on the surfaces of not only low Cr steels (SCM420) but also high Cr steels (F82H, STBA26, HCM12) [20]. It indicated the importance of adequate control and measurement of oxygen concentration in LBE for protection of steels from serious corrosion/erosion.

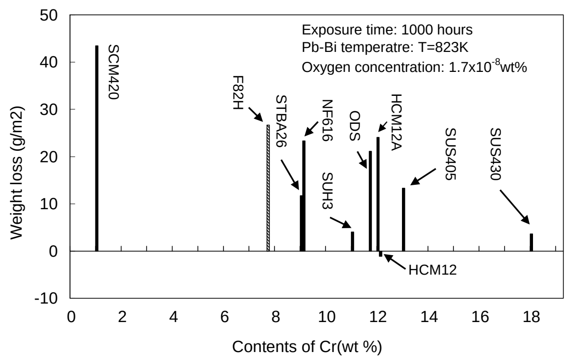
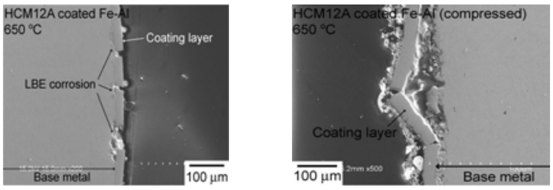


Fig. 18 Weight loss of steels in corrosion test at 550°C for 1,000 h [19]

For searching cladding materials corrosion resistant in more corrosive lead alloys at higher temperature, Si- and Al-rich steels were tested in a flowing LBE at 550°C. As a result, corrosion resistance was much better compared to the high Cr steels without Si and Al [21].

To have good corrosion resistance without changing base metal, surface heat treated and Al-alloying steels were tested in a flowing LBE at 550°C [22]. Furthermore, Al-Fe alloy coated steels using the sputtering technique were tested, and good corrosion resistance was obtained at 700°C [23,24]. The integrity of the Al-Fe alloy coating layer was investigated in LBE under bending stress conditions [25,26]. Although the base metal was not corroded, the coating layer was damaged by the stress (Fig.19). It may be necessary to improve the performance of the coating layer. Corrosion tests for welded steels was conducted, and it was found that tthe welded region was more corroded than the other surface [27,28]. Under transient conditions up to 800°C, the Al-Fe alloy coating layer could protect the base metal [29].



(a) Tensile stress (b) Compression stress

Fig. 19 HCM12A with Fe-Al alloy coating after corrosion test at 650 °C under loading for 240 h.

It was also found that refractory metals (Mo, W) and ceramics (SiC, Si_3N_4) were corrosion resistant at the temperature up to 700°C [23,30]. Various other corrosion behaviors were investigated, e. g., under the injection of steam and hydrogen mixture [31], precipitation at low temperature region (400°C) of a temperature gradient loop [32], and tube rupture with liquid metal corrosion under repeated heating and cooling [33].

The reliability of measured oxygen concentration in lead alloys is important in the corrosion tests. The reliability was not good when steels contacted with lead alloys, and it has not been improved yet. The reliability was good when lead alloys contacted only with ceramic materials [34,35]. The control technique of oxygen potential was developed for Ar-steam-hydrogen injection method and PbO particle-dissolution/precipitation method [36,37].

The corrosion phenomena were simulated using a the molecular dynamic method [38]. In addition, the interaction of Fe crystal with LBE atoms was simulated using the first-principles molecular dynamics method [39]. The diffusion coefficients of metal elements in LBE was measured using the capillary method [40]. Mutual diffusion of molten LBE and sodium was also simulated by means of the molecular dynamics method [41].

4. Polonium Tests

Another important issue specific to the LFR is the measure of ^{210}Po produced by neutron capture of ^{209}Bi. ^{210}Po is a radioactive nuclide emitting alpha-rays with the energy of 5.3MeV. LBE and steam are contaminated by the ^{210}Po. For the safety of the LFRs, some experiments were conducted to investigate the Po contamination, to develop the Po removal technique [42-45] and the unfolding method for determination of Po distribution [46], and to investigate the behavior of Po evaporation and adhesion [47].

5. Thermal-hydraulic Tests

It is also an issue to suppress carry-over of LBE droplets from free surface of LBE into steam flow system in PBWFR (Fig. 20). The removal performances of droplets in Chevron type steam dryers [48] and in an electrostatic precipitator [49, 50] were investigated. The electrostatic precipitator had high removal efficiency.

In order to simulate the LBE flow in the chimney above the core in PBWFR, LBE-water direct contact boiling flow loop was set-up and operated at the pressure of 7 MPa [51]. At first, LBE single-phase natural circulation phenomena was realized using the loop and good agreement of experimental flow rates with theoretical ones was obtained [52]. Then, LBE-water direct contact boiling two-phase flow in the chimney was experimentally simulated well under the operating condition of PBWFR. Good agreement of experimental results with theoretical ones was obtained for flow rate and heat transfer [53, 54]. Multi-dimensional thermal-hydraulic behavior of LBE-water direct contact boiling flow in the chimney was also simulated numerically [54-57]. Bubble and heat transfer behaviors were clarified.

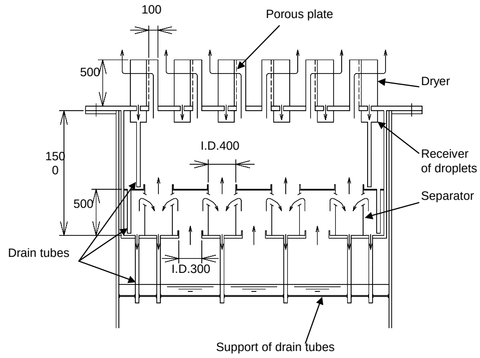


Fig. 20 Separators and dryers above free surface in PBWFR

By simulating the pipe break accident of SG in LFRs, and the direct contact of feed water into LBE in PBWFR, thermal interaction of lead alloy droplet with subcooled water was investigated to see if vapor explosion occurred or not (Fig. 21) [58, 59]. Fragmentation of droplet and violent boiling took place without occurrence of the vapor explosion (Fig. 21).

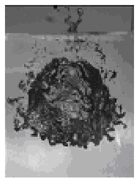


Fig. 21 Violent boiling in contact of LBE droplet at 400°C with water pool at 23°C

V. CONCLUSIONS

For the purposes of pursuing small, safe and economical reactors with a long life core and the reduction of radioactive wastes from nuclear systems, we have proposed the innovative concepts of LFR since the early 1990s, and performed their feasibility studies. Various findings have been obtained from the studies on core neutronics, reactor safety, material compatibility, polonium measure, and thermal-hydraulics. In show the status, an overview of the studies conducted so far is given in the present paper. It is expected that before the next stage of reactor demonstration in 2015-2030, this overview will be the guide for the future studies in this field.

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