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Preliminary Safety Analysis Results of Lead Cooled Fast Reactor Design using MARS Code

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1. Introduction

As the demands for safety, economics, and the sustainability of nuclear power plants increase, the development of the 4th generation reactor attracts lots of attention. The lead-cooled fast reactor (LFR) concept is considered as one of the most promising candidates for the 4th generation nuclear reactor [1].

The molten lead-bismuth eutectic (LBE) or pure lead coolants shows enhanced features such as low-pressure operation, inert chemical characteristics, very good thermodynamic, and neutronic properties [2].

The various countries including Europe (ELFR, ALFRED), United States (SSTAR), Russia (RESTD-OD-300), Korea (URAUS) are developing the LFR technologies to utilize the unique features of lead/LBE coolant [3].

However, the key-thermal hydraulic phenomena during the reactor transient and the reasonable safety acceptance criteria for the LFR design are not clearly revealed due to the lack of the reactor operating experience [4].

This paper presents the simplified thermal-hydraulic system design for the LFR with 300 MWth thermal power. The preliminary safety analysis (unprotected loss of offsite power and unprotected transient overpower) was conducted on the current LFR design with the MARS-LBE code to evaluate the safety features.

2. UNIST-LFR System Design

The UNIST-LFR is a lead-bismuth eutectic (LBE) cooled fast reactor with a thermal power of 300 MWth. The LBE coolant is pumped into the core via the electromagnetic pump located at the lower part of the downcomer. The core inlet and outlet temperature are set as 300 and 450 ºC respectively to ensure the structural integrity during both normal and transient operation. The operating pressure of the UNIST-LFR is set as 1[atm] and the guard vessel is located outside the reactor vessel to minimize the risk of the loss of coolant accident. The natural circulation flow path is formed inside the primary system to increase the passive safety features and to minimize the dependency of offsite power.

The reactor core consists of 144 fuel assemblies, 174 reflector assemblies, and 7 control rod assemblies. The hexagonal fuel assembly geometry is adopted to reduce the pin pitch to decrease the neutron moderation intensity. The uranium nitride is selected as fuel material and Ti 15-15 alloy is selected as cladding material to increase the corrosion resistance against the LBE coolant. No gap distance between fuel pellets and cladding is considered in the current study. Detailed information such as composition and geometry of each assembly can be found in Table I.

Table I: Problem Description

Parameters	Fuel	Reflector	Control
Pin material	UN	ZrO_2/MgO	B_4C
Cladding material	Ti 15-15	Ti 15-15	Ti 15-15
Pin diameter [cm]	1.088	1.600	2.800
Cladding thickness [cm]	0.060	0.060	0.200
Pin pitch [cm]	1.230	1.676	5.950
Pin pitch to diameter ratio	1.130	-	-
Number of pins per assembly	169	91	7
Duct thickness [cm]	0.4	0.4	0.4
Assembly pitch [cm]	17.0	17.0	17.0

2.1 Thermal Hydraulic System Design

The pool-type reactor vessel is chosen to reduce the natural circulation flow resistance during both normal and transient operation. The equivalent core diameter is 3.5 m and the diameter of the reactor vessel is set as 7.0 m. The thermal center difference is set as 6.29 m to guarantee 50% of thermal power can be removed by the natural circulation flow inside the primary system. The detailed geometric information of the primary system can be found in Fig. 1.

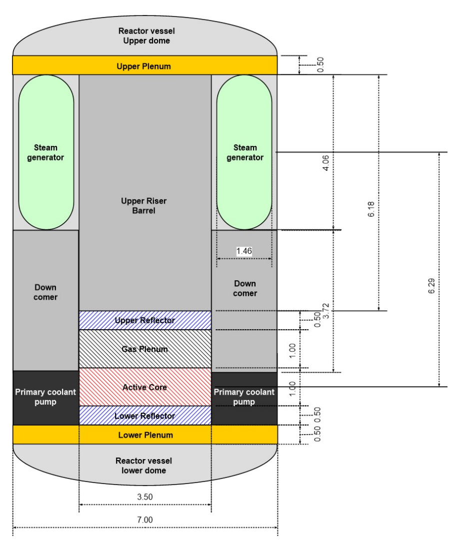

Fig. 1. Schematics on the UNIST-LFR primary system design.

The water-steam based Rankine cycle is chosen for the balance of the plant. Six steam generator (SG) modules with 4.05 m height are located inside the upper side of the downcomer. Six SG modules are divided into two trains and the separated main feedwater line and the main steam line are connected to each train.

Each train is connected to a decay heat removal system (DHRS) which consists of the isolated condenser. During transient operation, the superheated steam from the SG outlet is condensed. Then condensed water is injected into the steam generator modules to maintain the natural circulation flow inside the primary system. Each DHRS is designed to remove the 4% (12 MW_{th}) of the full power operation power.

The main thermal hydraulic parameters of the UNIST-LFR design are summarized in Table II.

Table II: The main design parameters of the UNIST-LFR.

Thermal Power [MW^{th}]	300
Primary Coolant	LBE
Active core length	1.00
Equivalent core diameter	3.50
Thermal center difference	6.29
Primary mass flow rate [kg/s]	13839.48
Core inlet/outlet temperature [ºC]	300 / 450
Circulation type	Mixed forced convection and natural convection
Primary operating pressure [bar]	1.01325
Secondary Coolant	Water/Steam
Steam Generator	6 modules (Once through type)
Thermodynamic cycle type	Rankine cycle
SG operating pressure [bar]	80
SG inlet/outlet temperature [ºC]	245.40 / 360.00

2.2 MARS Code Modeling

The current safety analysis results are based on the end of equilibrium cycle (EOC) reactor core power distribution condition. The radial and axial power distribution of the UNIST-LFR core design is shown in Fig. 2.

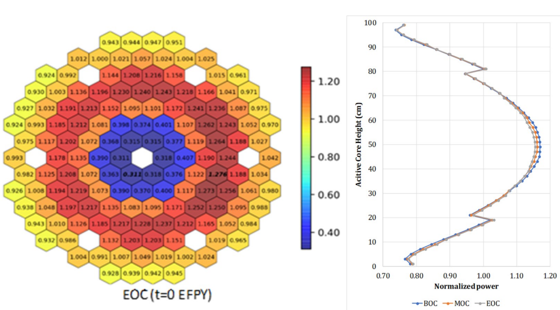

Fig. 2. The radial (left) and axial thermal power distribution of UNIST-LFR core at EOC condition.

The MARS-LBE code developed by Seoul National University was utilized in current safety analysis [5]. The MARS nodalization of the UNIST-LFR design is depicted in Fig. 3. The reactor core part is modeled into the two channels. One is average channel and the other is hot channel.

The 1.2 times ANS-79 decay heat curve was utilized to simulating the decay heat from the reactor core. The doppler and coolant density reactivity feedback is considered in current analysis.

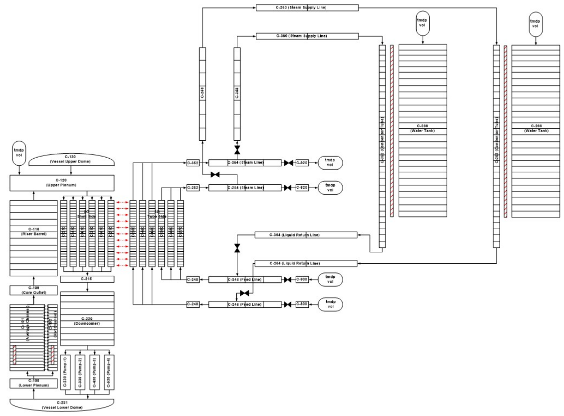

Fig. 3. MARS-LBE code nodalization for the UNIST-LFR thermal hydraulic systems.

3. Results and Discussion

3.1 Steady State Calculation Results

The steady-state calculation was conducted before the transient calculation with the MARS-LBE code to validate the thermal-hydraulic design. The calculation results are summarized in TableⅢ. The steady-state calculation results with MARS codes show good agreement with design parameters. The transient calculation was conducted based on the steady-steady calculation results.

Table III: Summary of steady state calculation results.

	Target	**Calculation**	**Error**
Reactor power [MW]	300.00	300.00	0.00%
Heat removal per SG [MW]	50.00	50.30	0.56%
Core outlet T [ºC]	450.00	468.09	4.02%
Core inlet T [ºC]	300.00	308.38	2.79%
Core mass flow rate [kg/s]	13839.48	13019.00	5.93%
SG feed inlet T [ºC]	245.40	245.40	0.00%
SG Steam outlet T [ºC]	360.00	360.23	0.06%

3.2 Transient Analysis Results

The reactor transient scenarios including unprotected loss of offsite power (ULOOP), and unprotected transient overpower (UTOP) cases were simulated in the current study. The reactor protection system consists of two types of trip signal. One is generated when the core outlet temperature is increased 20 ºC compared to the normal operating conditions and the other one is generated when the electromagnetic pump is shutdown.

The transient calculations were performed until the 10,000 s. However, only the calculation results up to 2,000 s are presented in the current paper to analyze the initial thermal-hydraulic and neutronic behavior during the reactor transient conditions.

3.2.1 ULOOP Accident

The ULOOP accident is initiated by the loss of primary electromagnetic pump and secondary flow caused by a loss of offsite at 10 s after transient calculation. The reactor scram does not occur, and the reactor power is governed by the reactivity feedback from Doppler effect and coolant density variation. The transient calculation results are shown in Fig. 4-6.

The secondary feedwater mass flow rate is suddenly dropped at 10 s. The primary LBE coolant mass flow rate also rapidly decreases from 13019 kg/s to 6025.6 kg/s at 10 s due to the loss of the primary pump. The LBE coolant in the primary side is circulated by the buoyancy force made from the thermal center difference. As shown in Fig. 5, the coolant and fuel centerline temperature abruptly increased. The DHRS valve is open at 15.37 s when the core outlet temperature exceeds the 488.09 ºC. The negative reactivity feedback is generated due to the increased fuel temperature and reduced coolant density as described in Fig. 6.

The reactor power starts to decrease due to reactivity feedback and the maximum fuel temperature and the cladding temperature also decrease. However, the overall coolant temperature is still increase since the reduced heat removal rate by SG. The LBE coolant temperature at core inlet is kept increasing.

As shown in Fig. 4, the heat removal rate from the DHRS exceeding the reactor power around ~1033 s and overall coolant temperature starts to decrease.

The LBE coolant temperature does not exceed the boiling point (~1600 ºC) [6] during all transient calculation. The fuel center line temperature also does not exceed the melting point (~2850 ºC) [7] of the uranium nitride.

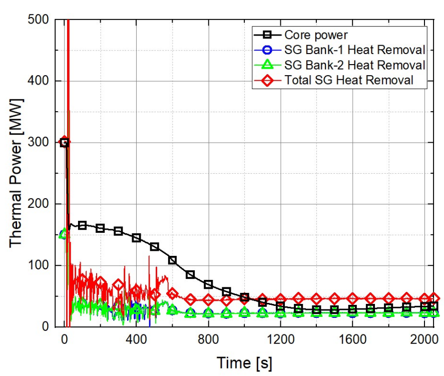

Fig. 4. The reactor power and heat removal rate behavior inside of the UNIST-LFR design under ULOOP accident.

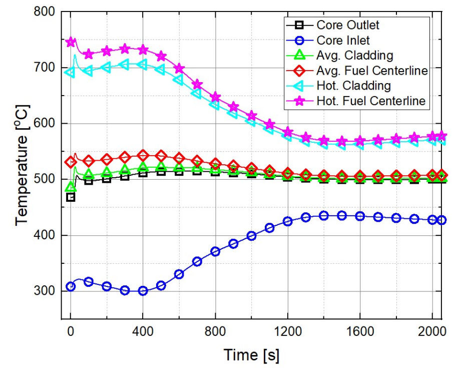

Fig. 5. The temperature behavior of LBE coolant, fuel centerline, and cladding inside of the UNIST-LFR design under ULOOP accident.

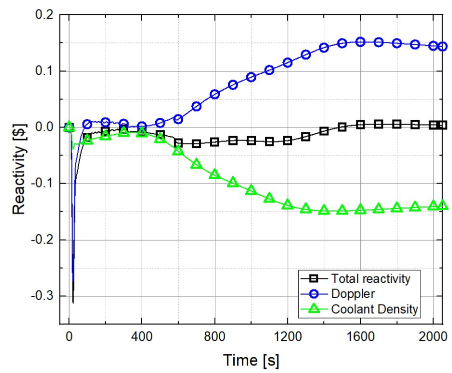

Fig. 6. The reactivity behavior inside of the UNIST-LFR design under ULOOP accident.

3.2.2 UTOP Accident

The UTOP accident is initiated by insertion of positive reactivity (0.49832 $) by withdraw of one control rod assembly at 10 s. The loss of primary electromagnetic pump was additionally assumed at 10 s. The reactor scram does not occur, and the reactor power is governed by the reactivity feedback from Doppler effect and coolant density variation. The transient calculation results are shown in Fig. 7-9.

As shown in Fig. 9, the positive reactivity of 0.49832 $ is inserted to reactor core within 2 s. The reactor power is abruptly increased and reach the maximum value of 442.92 MW_{th} at ~12 s. The primary LBE coolant mass flow rate suddenly decreases from 13019 kg/s to 6025.6 kg/s at 10 s due to the loss of the primary pump. The LBE coolant in the primary side is circulated by the buoyancy force made from the thermal center difference.

As shown in Fig. 8, the coolant and fuel centerline temperature rapidly increased. The DHRS valve is open at 10.02 s when the core outlet temperature exceeds the 488.09 ºC. The strong negative reactivity feedback is generated due to the increased fuel temperature and reduced coolant density.

The reactor power starts to decrease due to reactivity feedback and the maximum fuel temperature and the cladding temperature also decrease. However, the overall coolant temperature is still increase since the reduced heat removal rate by SG. The LBE coolant temperature at core inlet is kept increasing.

As shown in Fig. 7, the heat removal rate from the DHRS exceeding the reactor power around ~950 s and overall coolant temperature starts to decrease.

The LBE coolant temperature does not exceed the boiling point (~1600 ºC) during all transient calculation. The fuel center line temperature also does not exceed the melting point (~2850 ºC) of the uranium nitride.

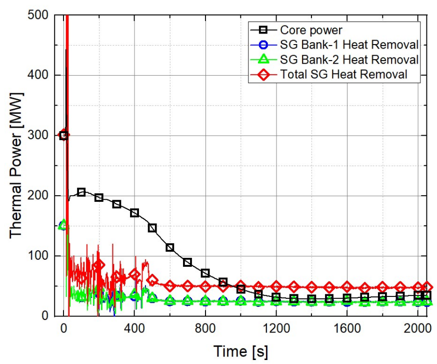

Fig. 7. The reactor power and heat removal rate behavior inside of the UNIST-LFR design under UTOP accident.

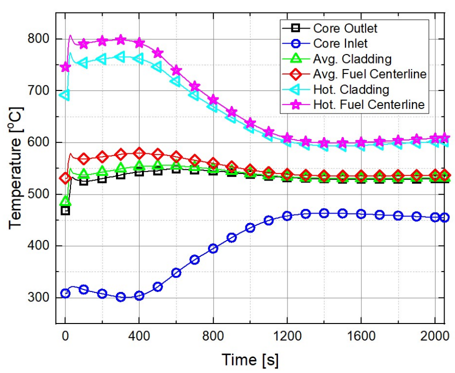

Fig. 8. The temperature behavior of LBE coolant, fuel centerline, and cladding inside of the UNIST-LFR design under UTOP accident.

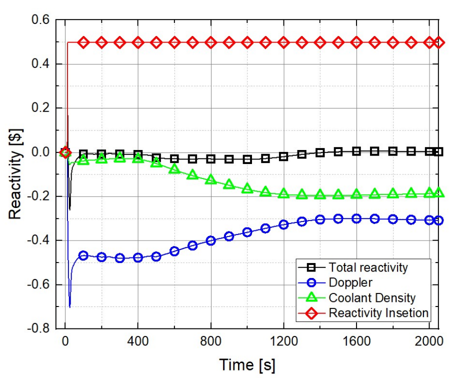

Fig. 9. The reactivity behavior inside of the UNIST-LFR design under UTOP accident.

4. Conclusion

The thermal-hydraulic system design of the 300 MWth lead-bismuth eutectic alloy cooled fast reactor (UNIST-LFR) was conducted based on the reactor core design suggested by the UNIST research group.

The preliminary safety analysis was conducted with MARS-LBE code to evaluate the safety features of the UNIST-LFR design. The two unprotected transient scenarios including ULOOP and UTOP was simulated.

The analysis results reveal that the reactor power is reduced to decay heat level by the reactivity feedback from the Doppler and coolant temperature variation. Also, the coolant and fuel centerline temperature does not exceed the boiling point of coolant and the melting point due to the actuation of DHRS.

As further work, the primary and secondary system design will be further refined. The additional decay heat removal system will be made to increase the redundancy features of the UNIST-LFR design. The reasonable safety acceptance criteria for UNIST-LFR design will be made to qualitative and quantitively evaluate the safety feature of the reactor design.

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