Transactions of the Korean Nuclear Society Virtual Spring Meeting
July 9-10, 2020

Metallic Fuel Performance Evaluation for Micro Lead cooled Fast Reactor

Ji Won Mun, Hyung Jin Kim, Faris. B. Sweidan, Ho Jin Ryu *
Department of Nuclear & Quantum Engineering, Korea Advanced Institute of Science and Technology, 291 Daehakro, Yuseong, 34141, Republic of Korea
**Corresponding author: hojinryu@kaist.ac.kr*

1. Introduction

The lead-cooled fast reactor (LFR) is considered as one of the most promising new generation fast nuclear reactors and they are actively being studied. Three reference systems were adopted by LFR-provisional System Steering Committee (pSSC) that include ELFR, ALFRED (EU), BREST-OD-300 (Russia) and SSTAR (USA) [1]. SCK-CEN(Belgium) developed FEMALE for MOX Fuel performance evaluation especially for LFR, and ITU (Germany) developed and validated TRANSURANUS code to apply to the ALFRED reactor [2,3].

Non-refueling ultra-long life LFR is currently being developed in Korea as a nuclear propellant system for marine ships or icebreakers [4]. Unlike conventional LFR systems, U-10Zr metallic nuclear fuel is considered as a fuel candidate. The reason is that U-10Zr metallic fuel has better breeding performance to UO_2 and has strong advantages in the licensing prospect than Nitride fuel. Therefore, the development of a new fuel performance analysis code for metallic fuel in LFR is essential.

In this study, a modified version of the fuel performance code FRAPCON-LFR, adapted for metallic fuel, stainless steel cladding materials and Lead-Bismuth eutectic coolant, was utilized. Basic fuel radial temperature calculation and verification with heat equation calculation was carried out.

2. Methods and Results

2.1 Simulation conditions and modified calculation module

Table I: Fuel rod and core design for LFR core in this study

Design Factor	*Design Value*
Fuel U/Zr composition (wt%)	*90.0/10.0*
Fuel rod outer diameter / Cladding thickness(mm)	*11.68/0.87*
Cladding material	*SS316L*
<Pb-bond fuel>	
Fuel Slug Diameter(mm)	*10.02*
Bond material	*Pb*
Bond thickness(mm)	*0.783*
Smear density(%TD)	*75.0*
<Annular fuel>	
Central hole radius(mm)	*5.84*
Smear density(%TD)	*75.0*
Core thermal power(MWt)	*60*
Average linear heat generation rate(kW/m)	*18.0*
Coolant Pb/Bi composition (wt%)	*44.5/55.5*
Coolant inlet/outlet temperature(℃)	*250.0/350.0*
Mass flux of coolant(kg/m²·s)	*5534.76*

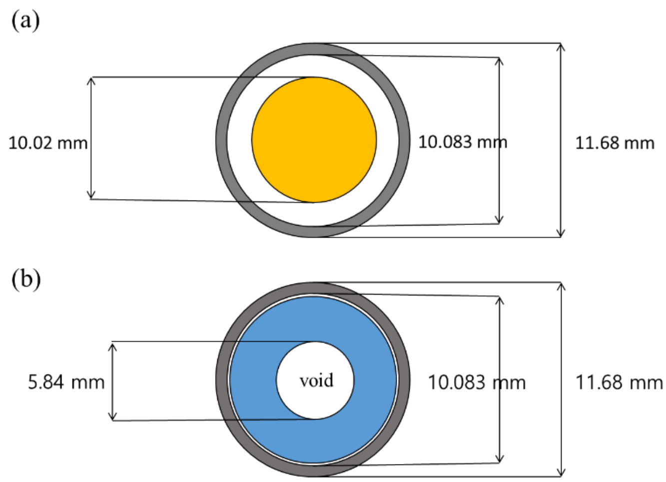

Fig. 1. Schematic cross section image of fuel rod (a) Pb-bond fuel (b) annular fuel

FRAPCON-4.0 is a LWR-based performance analysis code so mainly 3 parts of the module were modified for LFR: fuel, coolant and cladding. Thermal conductivity, swelling, heat capacity, thermal expansion, transition temperature and modulus changes were made in the fuel module. Swelling was implemented by applying an empirical correlation that comes from ALUFS(Alloyed Fuel Unified Simulator) code that simulates U-10Zr fuel in EBR-II [5].

Properties of water coolant were substituted by Pb-Bi eutectic coolant. For simplicity, coolant is assumed to have a constant temperature distribution of 250 °C to 350 °C along the axial direction.

Similarly in the cladding module, thermal conductivity, heat capacity, thermal expansion, transition temperature and modulus were modified. However, cladding irradiation growth was not applied in this calculation. The design parameters of core and fuel rod design are summarized in Table I.

2.2 Fuel radial temperature with Helium gap

In the case of metallic fuel, pellet-cladding mechanical interaction (PCMI) due to swelling is a major nuclear fuel failure mechanism, and the smeared density is low to about 75% to prevent this phenomenon. First, fuel temperature for metallic fuel with the helium gap case was calculated by FRAPCON-KAIST and validated by heat equation calculation. The fuel centerline temperature was calculated at about 2351 K and it leads to fuel melt at the first time step. No further calculations was made since then.

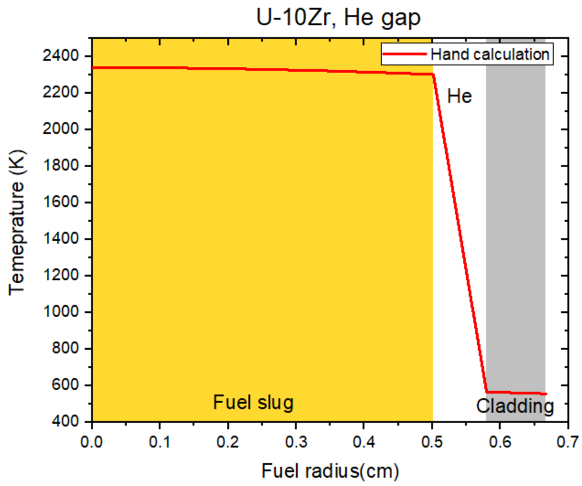

Fig. 2. Radial temperature calculation results by hand calculation with the helium gap

Hand calculation was conducted to cross-check the results of FRAPCON-KAIST and Fig. 2 shows that the fuel centerline temperature, similar to the previous results, is calculated to be around 2300 K. Therefore, without any bonding materials between fuel slug and cladding, a meltdown of nuclear fuel is expected.

2.3 Fuel radial temperature with Pb bond

In section 2.2, it is obvious that thermal bonding material is needed to compensate for the reduction in thermal conductivity due to wide gaps. Usually, Na bonding is utilized for metallic fuels in Sodium cooled Fast Reactor (SFR). However, in the situation where the fuel-coolant interaction occurs, Na bonding and Bi in the coolant can interact. In this case, the Na- Bi reaction is exothermic so there are possibilities of local heating. Also, when the sodium is included inside the LBE, solid intermetallics will form and small channel space between the cladding can block the flow of the coolant [6]. In this sense, lead bonding was suggested in previous research that utilizes metallic fuel in LFR core [7].

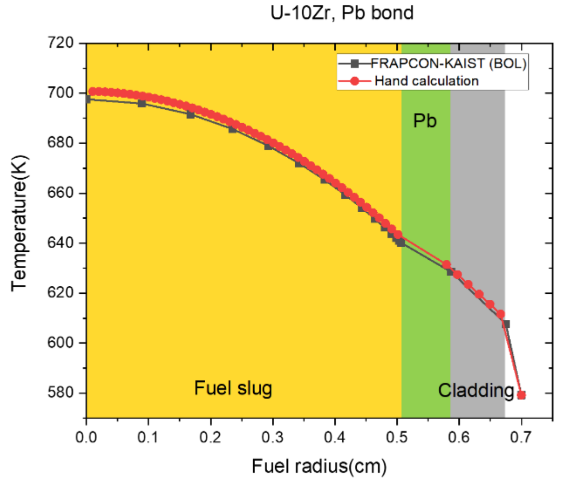

Fig. 3. Radial temperature calculation results with lead bond

Fig. 3 shows the radial temperature distribution when molten lead bonding was included. In both FRAPCON-LFR and calculation cases, the bulk coolant temperature was similar at about 573K. It was shown that the calculation results were almost identical except for the error when transferred from the coolant to cladding.

However in lead bonded fuel, high temperature corrosion of cladding by lead should be considered. Also, Fig. 4 shows that lead bond can be limited due to the formation of intermetallics PbU [7].

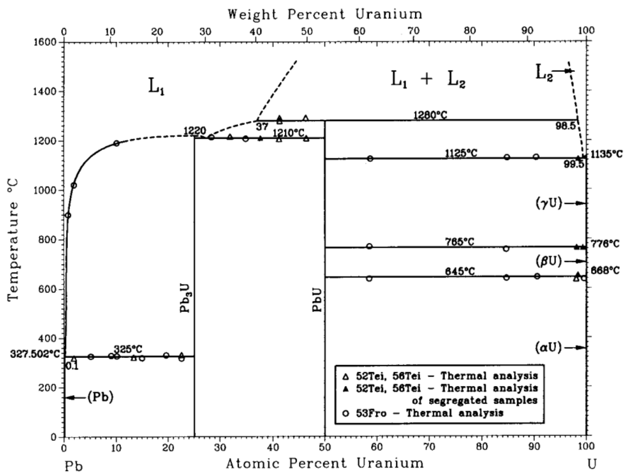

Fig. 4. Phase diagram of U-Pb system [8]

2.4 Annular fuel radial temperature distribution

Annular fuel or mechanically bonded fuel concept was proposed for advanced fuel for liquid-metal cooled fast breeder reactor because it has not only lower axial swelling, but also very large improvements of the neutron economy of fast reactors than conventional nuclear fuel [9, 10]. Normally the fuel and cladding are mechanically bonded by Zr-liner at the inner wall of the cladding but the liner is not considered in this calculation.

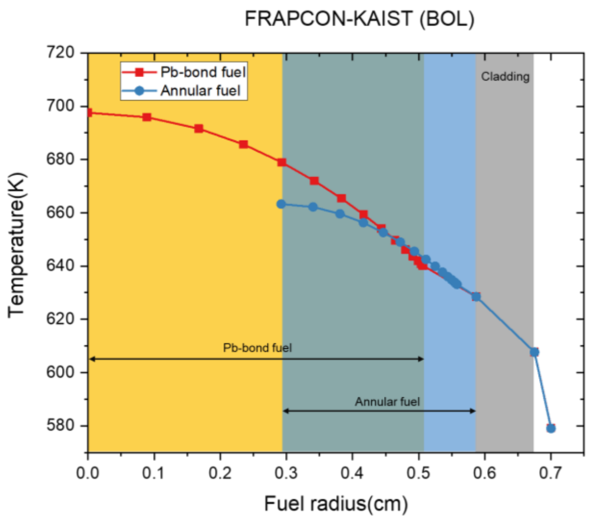

Fig. 5. Radial temperature calculation comparison results

Fig.5 shows a radial temperature comparison with the molten-lead bond fuel and annular fuel that contains a central hole for the same axial region and the same bulk coolant temperature. In the case of annular fuel, the fuel temperature can be kept lower without any bond material. The annular fuel has a lower peak temperature, resulting in a larger thermal margin considering fuel melting. Because the fuel outer surface and the cladding inner surface are in contact, the low thermal conductivity of the conventional fuel gap can be overcome by introducing the annular fuel design.

3. Conclusion

In this study, the LWR fuel performance analysis code FRAPCON-4.0 was modified to apply to LFR. The property-related values of the nuclear fuel, coolant, and cladding in the existing code were substituted by those of U-10Zr metallic fuel, Lead-Bismuth coolant, and SS316L cladding, respectively.

First, it was calculated how the fuel temperature would be distributed when the gap was filled with helium. The fuel core temperature was increased up to about 2300 K, which was higher than the fuel melting temperature. It was confirmed that the gap must be filled with a material with high thermal conductivity, and the centerline temperature of the fuel can be effectively reduced by applying a lead bond. In order to verify the calculation by modified code, the calculation of the heat equation was conducted in both cases and they agree well each other. Lastly, the temperature of an annular fuel design that contains a central hole was calculated and the peak temperature was lower than that of lead-bonded fuel.

ACKNOWLEDGMENT

This work was supported by the National Research Foundation of Korea (NRF, No. 2019M2D1A1067210) grant funded by the Korea government (MSIT)

REFERENCES

[1] A. Alemberti, M.L. Frogheri, S. Hermsmeyer, L. Ammirabile, V. Smirnov, M. Takahashi, C.F. Smith, Y. Wu, I.S. Hwang, Lead-cooled Fast Reactor (LFR) Risk and Safety Assessment White Paper, GENIV Int. Forum, p.17, 2014

[2] D. Rozzia, A. Del Nevo, M. Tarantino, N. Forgione, Preliminary discussion on LFR fuel pin design: Current status, fuel modeling and open issues, Int. Conf. Nucl. Eng. Proceedings, ICONE. 1, p. 467–473, 2012

[3] L. Luzzi, A. Cammi, V. Di Marcello, S. Lorenzi, D. Pizzocri, P. Van Uffelen, Application of the TRANSURANUS code for the fuel pin design process of the ALFRED reactor, Nucl. Eng. Des. 2014.

[4] J. Choe, E. Joeng, D. Lee, CORE DESIGN FOR LONG-LIFE SMALL MODULAR LFR, 2019.

[5] M. Kinoshita, H. Saito, T. Yokoo, Analytical study on deformation and fission gas behavior of metallic fast reactor fuel, 230, p. 129–139, 1996.

[6] OECD/NEA, Thermodynamic relationships and heavy liquid metal interaction with other coolants, in: Handb. Lead-Bismuth Eutectic Alloy Lead Prop. , Mater. Compat. , Therm. Hydraul. Technol., 2015.

[7] S. Choi, J.H. Cho, M.H. Bae, J. Lim, D. Puspitarini, J.H. Jeun, H.G. Joo, I.S. Hwang, PASCAR: Long burning small modular reactor based on natural circulation, in: Nucl. Eng. Des., 2011.

[8] R.I. Sheldon, E.M. Foltyn, D.E. Peterson, The Pb-U (Lead-Uranium) system, Bull. Alloy Phase Diagrams., 1987.

[9] S. Qvist, Safety and core design of large liquid-metal cooled fast breeder reactors, 2013.

[10] D. Hartanto, C. Kim, Y. Kim, A comparative physics study for an innovative sodium-cooled fast reactor (iSFR), in: Int. J. Energy Res., 2018.