Low Linear Power Oxide Fuel Performance Evaluation for Micro Lead-cooled Fast Reactor

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1. Introduction

Oxide fuel UO_2 and mixed oxide fuel MOX are the most used nuclear fuel for fast reactors. Among the three reference lead-cooled fast reactors (LFR), ELFR is utilizing MOX fuel. Oxide nuclear fuel has low thermal conductivity, but it has advantages because of low swelling, high melting point, and good irradiation behavior.

The preliminary design of ultra-long life micro LFR is currently being studied in Korea as a nuclear propulsion system for icebreakers. The difference between this reactor and the conventional fast reactor is that it has a linear heat generation rate as low as 1/10 scale.

In this study, a fuel performance code for LFR FRAPCON-KAIST has been developed based on FRAPCON-4.0. FRAPCON-4.0 is an LWR-based performance analysis code, and thus the coolant and cladding modules were modified. Material properties of water coolant and Zircaloy cladding were substituted by lead-bismuth eutectic coolant and stainless steel 316Ti, respectively. Preliminary fuel performance evaluation was carried out by using the modified code.

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 material	*UO_2*
Cladding material	*SS316Ti*
Fill gas material	*He*
Fuel rod outer diameter / Cladding thickness(mm)	*20.0/1.0*
Fuel column length (cm)	*180*
Core thermal power(MWt)	*60*
Average linear heat generation rate(kW/m)	*6.86*
Effective full power year	*40*
Coolant Pb/Bi composition (wt%)	*44.5/55.5*
Coolant inlet/outlet temperature(℃)	*250.0/350.0*
Coolant pressure (MPa)	*0.1*

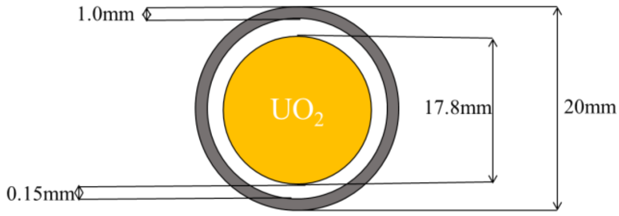

Fig. 1. Schematic cross-section image of a fuel rod

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

Similarly in the cladding module, thermal conductivity, heat capacity, thermal expansion, transition temperature, and modulus were modified. In the case of the irradiation swelling term, the property of Ti modified stainless steel 316 was implemented. The design parameters of core and fuel rod design are summarized in Table I.

2.2 Fuel performance evaluation results

The fuel radial temperature distribution at the beginning of life (BOL) with the bulk coolant temperature of 300℃ is presented in Fig. 2 . The centerline temperature was very low, approximately 800K, and the temperature gradient inside the fuel pellet was also very low, about 100K/cm. Whereas the operating temperature of the existing oxide fuel is very high and has a steep temperature gradient of about 2000℃, the fuel temperature of the micro LFR is calculated very low because the power density of the current design is very low.

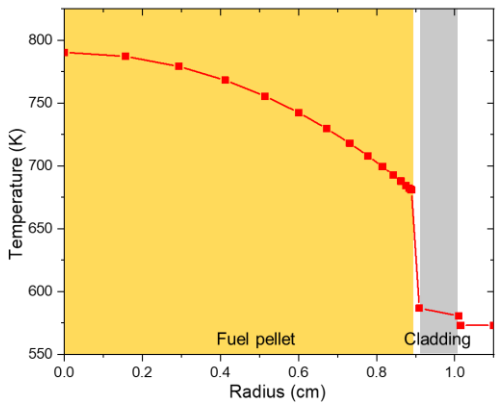

Fig. 2. Radial temperature calculation results at BOL

Initially, gap closure occurred due to fuel swelling, but after that, irradiation swelling of cladding became dominant and the gap width increased as we can see from Fig. 3(a). Therefore, in the conventional fast reactor oxide fuel pin, gap contact occurs after 30~40 GWd/tU, whereas in the current calculation, fuel-cladding contact does not occur during the whole lifetime. Also, from Fig.3 (b) as the gap size decreased, outer surface temperature of the fuel decreased as burnup increased.

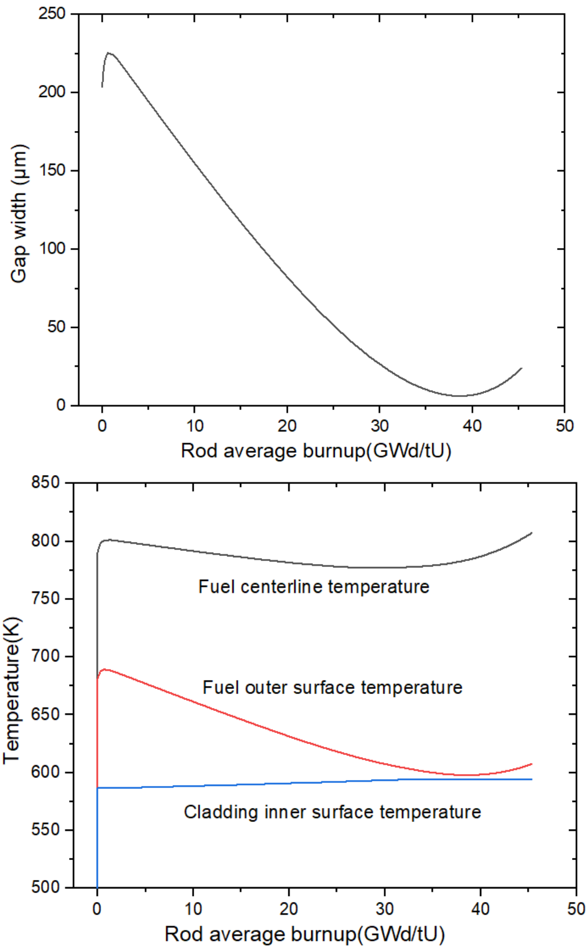

Fig. 3. (a) Gap size evolution and (b) fuel temperature profile at the middle axial region of fuel column

2.3 Fission gas release and plenum pressure

Fission gas release was calculated by the modified Forsberg-Massih model that consider low temperature behavior. According to the low temperature fission gas release model that is stated in eq (1), fission gas release fraction increase linearly until 40GWD/tU. After that, it increases more steeply[1]. As shown in Fig. 4, the fission gas release was only 0.75% at the end of life (EOL) which is negligible.

$$F = 7\times 10^{-5} BU + C \quad \text{eq (1)}$$

F = fission gas release fraction
BU = local burnup in GWd/MTU
C = 0; for BU ≤ 40 GWd/MTU
= 0.01(BU-40)/10; for burnup > 40 GWd/MTU and F ≤ 0.05

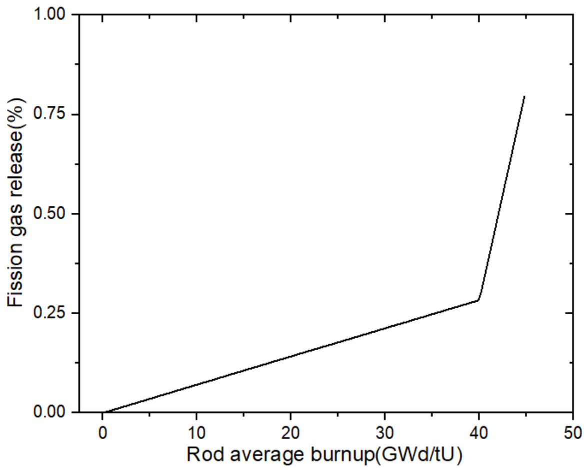

Fig. 4. Fission gas release

In the case of Phénix pins irradiated in nominal conditions, there was significant gas release (~30 – 50%) even at a very low burnup [2]. However, in the current case, since the temperature is very low, diffusion of fission gas atoms hardly occurs, so bubble formation will not occur. According to the previous studies [3], the threshold temperature for fission gas release is about 1000℃ at 40GWd/ tU burnup which follows eq. (2). Therefore, it is reasonable that the fission gas release is negligible because the temperature of the current fuel is calculated at a maximum of 1000K.

$$T_s = \frac{7460}{-3.41 + ln(10^3 B)} \quad \text{eq (2)}$$

T_s : threshold temperature in ℃ for fission gas release
B : Burnup in GWd/tU

From Fig. 5, we can see that as the burnup increases, the plenum pressure reduced, which is the opposite behavior of the conventional nuclear reactor. This is because there is almost no fission gas release, but according to Fig. 3 (a), the gap width has increased, so the free volume that can accommodate gas has increased.

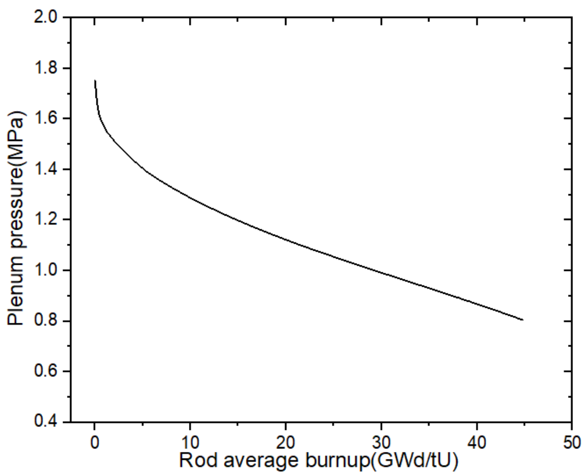

Fig. 5. Plenum pressure evolution

3. Conclusion

In this study, the LWR fuel performance analysis code FRAPCON-4.0 was modified to be applied to LFR. The material properties of water coolant, and zircaloy cladding were substituted with those of Lead-Bismuth coolant, and SS316Ti cladding, respectively.

First, it was calculated that both fuel centerline temperature and temperature gradient within the fuel were very low due to the low linear heat generation rate. Also, fuel temperature was increased with the burnup due to the gap size increment. However, it was confirmed that the maximum temperature does not exceed 1000K. Lastly, the fission gas release was very low and the plenum pressure was also kept low.

ACKNOWLEDGMENT

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