

Development of Low-Temperature Fuel Performance Analysis Code for Micro Ultra Long life Lead-cooled Fast Reactor

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1. Introduction

The importance of the Northern Sea Route (NSR) for future international trade and energy supply stability is rising. For ships to navigate through the NSR, nuclear reactor propelled icebreakers are a good alternative to natural gas ships because they have no greenhouse gas emissions. In the case of a Pressurized Water Reactor (PWR) that uses Low Enriched Uranium (LEU), it is inevitable to replace fuel in a maximum of 2.3 years [1]. To compensate for these shortcomings of PWR propelled ships, a micro-lead cooled fast reactor (LFR) called MicroURANUS which can be operated for 40 years without refueling is currently being studied [2].

It is difficult to apply the fuel performance codes developed for the existing LFR to this reactor due to the different operational conditions. The first difference is that MicroURANUS has a low fuel power density for long life operation, thus, nuclear fuel is operated at very low temperatures. Therefore, it is necessary to carefully consider the low-temperature characteristics of nuclear fuel. The second is the difference in cladding material. In the case of MicroURANUS, austenitic stainless steel 15-15Ti was selected for the cladding material.

In this study, a fuel performance code for LFR FRAPCON-KAIST-1.0 has been developed based on FRAPCON-4.0. Because FRAPCON-4.0 is an LWR-based performance analysis code, the coolant and cladding modules were modified. Material properties of water coolant and Zircaloy cladding were substituted by lead-bismuth eutectic coolant and austenitic stainless steel 15-15Ti, respectively. The evaluation of the thermal and mechanical performance of nuclear fuel during steady-state operation for 30 years was conducted through developed FARPCON-KAIST-1.0 which is the effective full power life of the reactor.

2. Methods and Results

2.1 Simulation conditions and modified calculation module

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

Design Factor	Design Value
Fuel material	UO ₂
Cladding material	15-15Ti
Fill gas material	He
Fuel rod outer diameter / Cladding thickness(mm)	20.0/0.95

Core thermal power(MWt)	60
Average linear heat generation rate(kW/m)	10.43
Effective full power year	30
Coolant Pb/Bi composition (wt%)	44.5/55.5
Coolant inlet/outlet temperature(°C)	250.0/350.0
Coolant pressure (MPa)	0.1

The schematic calculation structure of this code and the goal to be obtained from the code development are as shown in Figure 1. The operation condition for calculation is obtained by the existing core neutronics study that contains the linear power of nuclear fuel, coolant inlet temperature, and heat transfer coefficient of LBE coolant.

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 austenitic stainless steel 15-15Ti was implemented. The design parameters of core and fuel rod design are summarized in Table I. The final goal of this code is to evaluate the operational safety of nuclear fuel through the calculated target variables.

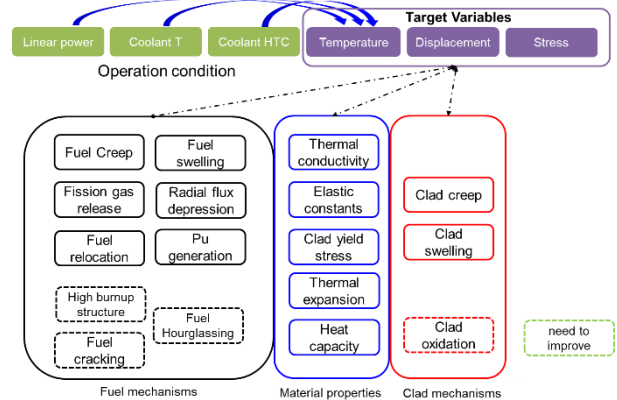


Fig. 1. Overall code structures and objectives

2.2 Thermo-mechanical behavior of the hottest fuel rod

After 30 years of effective full power operation, fuel burnup of end-of-life was about 50 GWd/tU. The reason that the burn-up of fuel is even with a long cycle of 30 years is because the power density is about 4 to 5 times lower than that of the conventional light water reactor and the loading of nuclear fuel is large. Based on the calculated fuel and cladding deformation, it was evaluated how the gap size between fuel-cladding evolved during operation. Initially, gap expansion occurred due to thermal expansion caused by the temperature rise of the cladding. Thereafter, the fuel-cladding gap is gradually decreased due to fuel swelling that increases linearly, although cladding deforms outward direction as we can see from Fig. 3 (a). Therefore, in the conventional fast reactor oxide fuel pin, gap closure occurs after 30~40 GWd/tU, whereas in the current calculation, fuel-cladding contact does not occur during the whole lifetime. Also, the centerline temperature of nuclear fuel shows an increase of about 150K during operation. This is due to the thermal conductivity degradation of UO₂ fuel due to irradiation as shown in Fig. 3 (b).

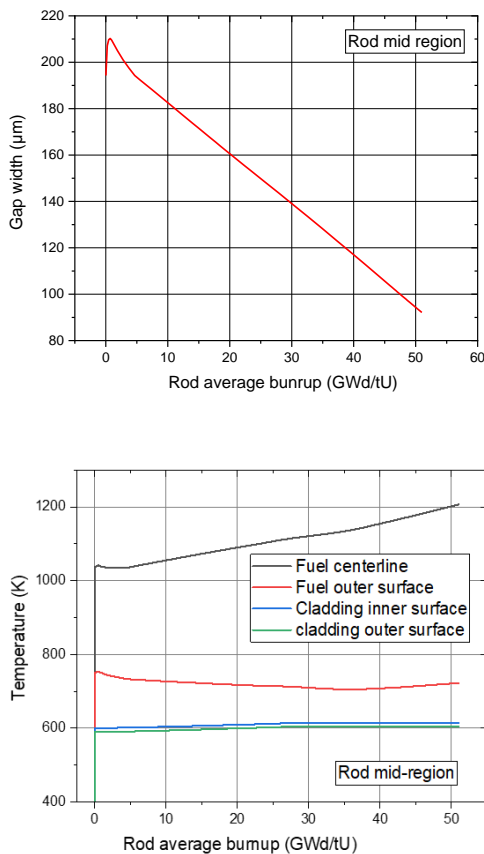


Fig. 3. (a) Gap size evolution and (b) fuel temperature profile at the middle axial region of fuel column

2.3 Fission gas release and plenum pressure

The low-temperature fission gas release model that is stated in Eq. (1) was used in this calculation [3]. As

shown in Fig. 4, the fission gas release was only 2.5% at the end of life (EOL). Since the temperature is very low, diffusion of fission gas atoms hardly occurs, and thus, bubble formation and interconnections are highly suppressed. Also, fission gas release leads to plenum pressure increment up to 2.5 MPa.

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

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

3. Conclusion

In this study, the development of the fuel performance code FRAPCON-KAIST-1.0, which is for the low temperature UO₂, austenitic stainless cladding, and LBE coolant fast reactor was conducted. Based on LWR normal operation fuel performance code FRAPCON-4.0, the material properties and models were modified to adopt the low-temperature and fast spectrum characteristics of the current reactor. Through the developed code, full core life fuel performance analysis of MicroURANUS was conducted to evaluate fuel performance and safety in terms of thermal stability and mechanical integrity.

In the case of MicroURANUS, the maximum fuel temperature is evaluated as 1225 K. Therefore low fuel temperature provides a large safety margin for fuel melting as well as low pressure build-up fission gas release. Maximum fractional fission gas release does not exceed 1.7% which can reduce rod internal pressure build-up and allow high initial He pressurization to reduce fuel temperature effectively. Also, the results show that the fuel temperature is sufficiently low to allow for safe operation without gap closure during the entire core life.

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