

## Modeling for U-Zr Annular Metallic Fuel in SFR using COMSOL Multi-physics

Ju-Seong Kim\*, Heung Soo Lee, Cheol Min Lee, June-Hyung Kim  
Korea Atomic Energy Research Institute, 989-111 Daedeok-daero, Yuseong-gu, Daejeon 34057, Republic of Korea  
\*Corresponding author: juskim@kaeri.re.kr

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### 1. Introduction

U-Zr metallic fuel, a primary candidate for sodium-cooled fast reactors (SFRs), offers distinct advantages including high thermal conductivity and high fissile material density. However, it exhibits significant swelling as irradiation progresses. To accommodate this behavior, conventional solid metallic fuels are designed with a relatively low smear density (approximately 75%), utilizing sodium as a thermal bond to enhance heat transfer between the fuel and cladding.

Despite its benefits, the bond sodium can pose safety and handling challenges if it is released during the management of spent nuclear fuel. Consequently, several annular metallic fuel rods were irradiated in advanced test reactor to understand the fuel behavior without bond sodium. In this study, a performance evaluation model for annular metallic fuel was developed using COMSOL Multiphysics, a commercial finite element analysis (FEA) code, to analyze the mechanical interactions between the fuel and cladding.

### 2. Methods and Results

Comprehensive fuel performance evaluation requires integrating mechanical and thermal material properties with heat transfer models between the fuel rod and coolant. These elements were implemented in the COMSOL environment, with most material properties adopted from the Metal Fuel Handbook. The implemented framework incorporates models from both the handbook and the BISON fuel performance code..

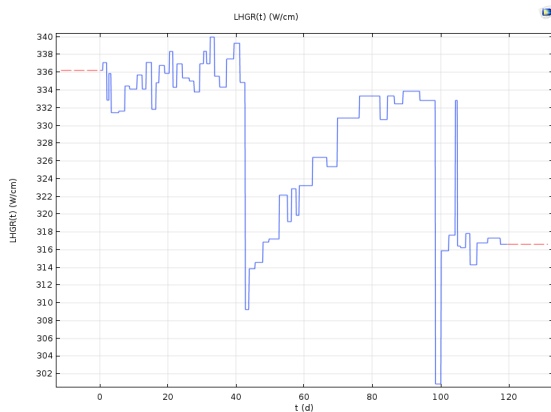


Fig. 1. Irradiation history of AFC-3A-R4 fuel rod

The fuel performance analysis can be conducted with power history and boundary conditions of the fuel. For this study, the AFC-3A-R4 fuel rod irradiated for approximately 120 days at an average linear power of 322 W/cm was selected as the reference. Fig.1 shows the linear power history of AFC-3A-R4.

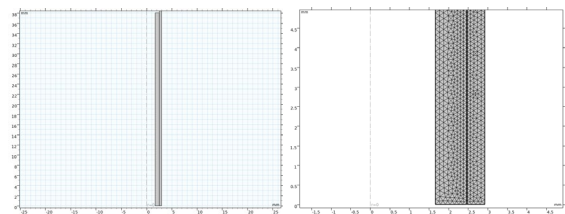


Fig. 2. Model geometry and mesh configuration

Fig. 2 shows the mesh configuration and geometrical model of AFC-3A-R4. The inner and outer radii of the fuel are 1.65mm and 2.465 mm, respectively. A cylindrical (R-Z) structural model was constructed under the assumption of axial symmetry. While axial power variance is possible, this study assumed uniform axial distributions of power and temperature to simplify the initial evaluation. The mesh configuration for the annular fuel was optimized for physical accuracy, with a maximum element size of 1 mm, totaling 11,488 elements and 1,766 boundary elements.

### 3. Results and discussions

In annular fuel, the gap is filled with Helium at the beginning of reactor operation but as the fission gases are released the gap conductance is changed with the burnup.

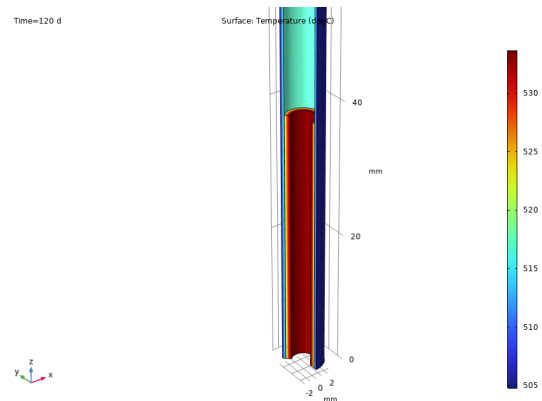


Fig. 3. Temperature distribution of AFC-3A-R4

Fig. 3 shows the temperature distribution of the annular fuel. In this work, the gap conductance was between 4000-6000 W/m<sup>2</sup>K and sodium coolant temperature was set at 455°C during the operation.

Post-irradiation examinations [1, 2] of AFC-3A-R4 revealed that the inner movement of the annular fuel caused the original inner hole to shrink as burnup increased. However, without considering fuel creep, this inward displacement would not occur. The U-Zr fuel creep model used in this study is secondary creep model from the BISON modeling [3].

$$\dot{\epsilon} = 5000 \times (1 + 7.9p + 470p^2)\sigma \exp\left(\frac{-52000}{RT}\right) + 6(1 - p^{0.67})^{-0.8}\sigma^{4.5} \exp\left(\frac{-52000}{RT}\right) + 7.7 \times 10^{-23}\sigma\dot{F}$$

Where, R is 1.987 cal/mole-K, p is fraction of pores,  $\sigma$  is stress MPa,  $\dot{F}$  is fission rate *fission/cm<sup>3</sup>sec* .

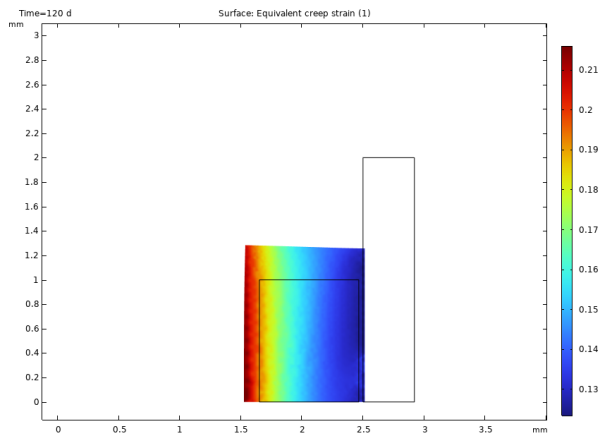


Fig. 4. Radial displacement of AFC-3A-R4 at 120 days of operation.

Fig. 4 shows the radial displacement of the AFC-3A-R4 after 120 days of operation due to the creep of the U-Zr fuel.

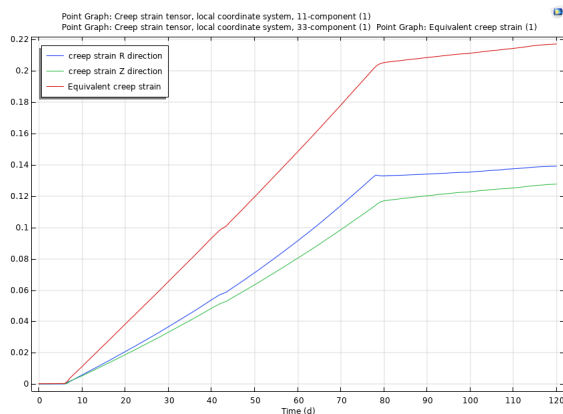


Fig. 5. Creep strains of fuel in R, Z direction and equivalent at the inner lower section

In total deformation, creep-induced strain accounted for 41% (0.14), followed by swelling (38%), with the

remainder attributed to elastic deformation and thermal expansion. At 120 days of operation, the equivalent creep strain at the inner surface of the lower fuel region reached 0.21. Plastic deformation is considered in mechanical deformation and the results indicated that for annular metallic fuel, radial displacement exhibits negative values (inward movement), while the corresponding strain remains positive. The radial strain peaked at the inner surface, whereas the axial strain was nearly uniform across both the inner and outer surfaces. Early-stage irradiation showed an increase in creep deformation due to mechanical interactions, which gradually stabilized as irradiation proceeded. The report from Idaho National Laboratory also identifies creep as the dominant mechanism with a maximum strain of 0.52, whereas the current study obtained a lower equivalent strain of 0.21. This discrepancy suggests the need for further refinement of the coupling models between swelling and creep of fuel. Furthermore, the module does not yet incorporate a constituent redistribution model, which significantly impacts the thermal and mechanical properties of the fuel. Future work will focus on integrating these aspects.

#### 4. Conclusions

In this study, a specialized performance evaluation model for annular metallic fuel was successfully developed using COMSOL Multiphysics. By integrating mechanical and thermal models from the Metal Fuel Handbook and the BISON codes, the simulation model can provide a detailed analysis of the deformation behavior of AFC-3A-R4 fuel rods and the other fuel rods.

#### Acknowledgement

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