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Thermal Design for 2nd Irradiation Test of SFR Nuclear Fuel

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1. Introduction

A capsule was designed for the second irradiation test of SFR nuclear fuel. Design for the first irradiation test of SFR nuclear fuel was conducted in 2009 [1], and the second test of double the burn-up is under planning. In performing a worthy irradiation test, keeping the temperature of the capsule stable during the irradiation period is essential. To predict the temperature distribution in the capsule, a temperature analysis is performed using the thermal analysis code. In this project, we attempted a thermal analysis for a stability verification of the capsule. The objectives of this study are as follows.

A. Determining the linear power for the cladding target temperature.
B. Confirmation of the nuclear fuel melting temperature and ONB (Onset of Nucleate Boiling) temperature at the sealing tube surface.
C. Accident analysis on the locked rotor event and control rod withdrawal accident.

The capsule was designed based on the results obtained through the analysis.

2. Capsule model

2.1 Modeling

The capsule will be irradiated in the OR5 test hole of HANARO at 30 MW up to a burn-up 6 at%. Fig. 1 shows the cross section of the fuel rod surrounded by a sealing tube. In this study, it was considered that only the fuel rod channel of the coolant is passed. A gap between the cladding and sealing tube was fixed at 0.06 mm and 0.07 mm. the gap filled with helium gas. The gap between the cladding and the fuel was filled with sodium. The cladding is T92, and the sealing tube is Stainless Steel 316L. The fuel is determined by U-Zr and U-Zr-5Ce. The specifications of the fuel rod are listed in Table 1. Depending on its diameter, fuel is divided into two types. Fig. 2 shows the full body of the SFR irradiation capsule. Six fuel rods are respectively loaded at the top and bottom.

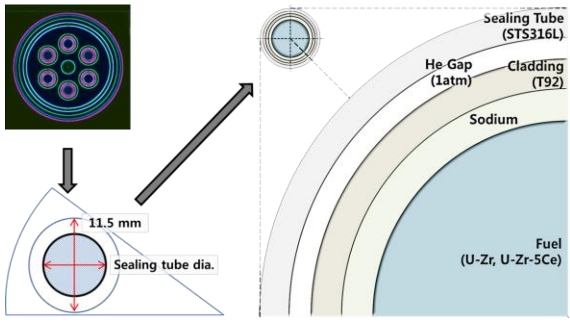

Fig. 1. The cross section of the fuel rod.

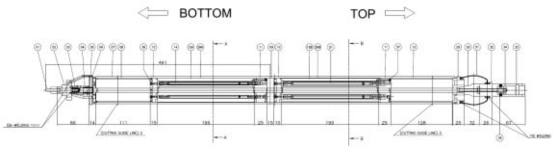

Fig. 2. The full body of SFR irradiation capsule.

Table 1: The dimensions of the fuel rod in the axial direction.

(mm)

Top	Radius in	Radius out	Diameter in	Diameter out	Thickness
Fuel		2.81		5.62	
Inner	3.00	3.50	6.00	7.00	0.50
gap	3.50	3.56	7.00	7.12	**0.06~0.07**
Outer	3.56	4.06	7.12	8.12	0.50

(mm)

Bottom	Radius in	Radius out	Diameter in	Diameter out	Thickness
Fuel		1.99		3.98	
Inner	2.25	2.75	4.50	5.50	0.50
gap	2.75	2.81	5.50	5.62	**0.06~0.07**
Outer	2.81	3.31	5.62	6.62	0.50

* The gap can be changed to 0.07 mm.
* Outer diameter of the fuel rod channel: 11.5 mm

2.2 Analysis

The temperature distribution of the model of an irradiation test capsule is estimated using the 1-dimensional heat transfer code, GENGTC [2]. The GENGTC code is calculated by considering the structure in the radius direction and the heat generation. The temperature distribution and linear power of the fuel were calculated based on the cladding temperature. Considerations of this design are as follows [3].

A. The target temperature of the cladding is 650 ℃ under a steady state.
B. Fuel center T: < melting point (1235 ℃)

C. Sealing tube surface T: < ONB temp. (125 ℃)

To simulate an accident, the temperatures were calculated after raising the linear power to 131% (the control rod withdrawal accident) and lowering the flow rate to 56% (the locked rotor event) [4]. Fig. 3 shows the out pile test system. The velocities of the fuel rod channel can be found through the out pile hydraulic test results, and the convective heat transfer coefficients are calculated using these values. The flow cross section of channel is different according to the vertical position, so the flow velocity of the fuel rod channel at the top is 10.96 m/s and at the bottom is 8.2 m/s.

Fig 3. The out-pile test system.

The heat transfer coefficient at the outer surface of the sealing tube is 2.08 to 4.45 W/cm^2.℃. Under a locked rotor event, it is 2.06 to 3.28 W/cm^2.℃. The temperature of the cooling water in the reactor in-core is about 40 ℃. Details related to the modifications of the cladding are not considered.

3. Results

Tables 2 and 3 shows the temperature analysis results of the fuel rod channel. The center temperatures of the nuclear fuel were confirmed to not exceed the melting point including under an accident situation. In a locked rotor event (flow rate 56%), the surface temperatures of the sealing tube exceeded the ONB temperature at the bottom position. The linear power in a 0.06 mm gap is 400 to 634 W/m, in the transient analysis is 525 to 831 W/m, in a 0.07 mm gap is 371 to 546 W/m, and in the transient analysis is 456 to 716 W/m. We could see that the linear power in the 0.06 mm gap case is higher than in the 0.07 mm gap case. The thermal distribution satisfies the temperature stability in the 0.07 mm gap case. However, the temperature distribution of the first irradiation test capsule is lower than the calculated value, and a high linear power reduces the period of irradiation.

Table 2: The temperature distribution and linear power calculated using the GENGTC code (helium gap 0.06 mm).

A. Standard (Target temperature 650 ℃)

Fuel Type (Position)	Temperature (℃)			Linear power (W/cm)
	Fuel center	Inner cladding	Tube surface	
U-Zr (Top)	948.9	649.9	92.7	596
U-Zr (Bottom)	958.3	650.5	102.8	400
U-Zr-5Ce (Top)	953.4	649.7	92.7	634
U-Zr-5Ce (Bottom)	960.5	649.5	102.7	426

Convection heat transfer coefficient
: 4.45 (3.28) W/cm^2.℃

B. Flow rate 56%

Fuel Type (Position)	Temperature (℃)			Linear power (W/cm)
	Fuel center	Inner cladding	Tube surface	
U-Zr (Top)	973.1	674.3	123.7	596
U-Zr (Bottom)	985.7	678.1	140.0	400
U-Zr-5Ce (Top)	977.4	673.8	123.7	634
U-Zr-5Ce (Bottom)	988.2	677.4	139.8	426

Convection heat transfer coefficient
: 2.80 (2.06) W/cm^2.℃

C. Power 131%

Fuel Type (Position)	Temperature (℃)			Linear power (W/cm)
	Fuel center	Inner cladding	Tube surface	
U-Zr (Top)	1151.2	762.4	108.9	781
U-Zr (Bottom)	1170.7	770.1	122.2	525
U-Zr-5Ce (Top)	1157.9	762.2	108.9	831
U-Zr-5Ce (Bottom)	1174.3	769.1	122.0	558

Convection heat transfer coefficient
: 4.45 (3.28) W/cm^2.℃

Table 3: The temperature distribution and linear power calculated using the GENGTC code (helium gap 0.07 mm).

A. Standard (Target temperature 650 ℃)

Fuel Type (Position)	Temperature (℃)			Linear power (W/cm)
	Fuel center	Inner cladding	Tube surface	
U-Zr (Top)	908.7	650.3	85.3	514
U-Zr (Bottom)	918.2	650.1	94.6	348
U-Zr-5Ce (Top)	911.9	649.8	85.2	546
U-Zr-5Ce (Bottom)	921.3	650.0	94.5	371

Convection heat transfer coefficient
: 4.46 (3.28) W/cm^2.℃

B. Flow rate 56%

Fuel Type (Position)	Temperature (℃)			Linear power (W/cm)
	Fuel center	Inner cladding	Tube surface	
U-Zr (Top)	928.8	670.7	112.1	514
U-Zr (Bottom)	941.3	673.2	126.4	348
U-Zr-5Ce (Top)	931.8	669.8	112.0	546
U-Zr-5Ce (Bottom)	943.8	672.7	126.3	371

Convection heat transfer coefficient
: 2.80 (2.07) W/cm^2.℃

C. Power 131%

Fuel Type (Position)	Temperature (℃)			Linear power (W/cm)
	Fuel center	Inner cladding	Tube surface	
U-Zr (Top)	1099.4	763.4	99.2	673
U-Zr (Bottom)	1119.5	770.1	111.3	456
U-Zr-5Ce (Top)	1104.5	762.9	99.1	716
U-Zr-5Ce (Bottom)	1123.5	769.7	111.3	486

Convection heat transfer coefficient
: 4.46 (3.28) W/cm^2.℃

4. Conclusions

To conduct a verification of the SFR nuclear fuel, irradiation testing is required. For this test, a stable capsule design is essential. A thermal design of the capsule was carried out for the second irradiation test of SFR nuclear fuel. Owing to the high temperature of the bottom position, caution is needed when designing the capsule. In consideration of the linear power and temperature of the cladding, it was decided to design the gap to be 0.06 mm. After determining the linear power of the nuclear fuel with reference to the results, the thickness of the Hf shielding material put in the capsule will be determined. The Hf shielding material serves as a controller of the neutron flux coming into the capsule.

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