Preliminary Experiment of Natural Convection Heat Transfer Characteristics of Horizontal Narrow Gap Rectangular Channel

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1. Introduction

In the process of fuel replacing in research reactors, nuclear fuel can fall into the reactor pool due to the carelessness of the operator or mechanical failure of the tools. The nuclear fuel of a research reactor, due to its length-wise shape, lays horizontally when it falls to the bottom.

Vertically-standing nuclear fuel can be effectively cooled by forced convection and natural convection, which ensures the integrity of the fuel. However, when the fuel is placed horizontally, the side plates or fuel plates block the flow of coolant, which reduces the cooling performance.

This study performed a preliminary experiment to investigate the natural convection cooling characteristics under the limited conditions as mentioned above.

2. Methods and Results

In this section, the critical heat flux at horizontal conditions was predicted by two methods in order to design the test section. Subsequently, a test section and experimental apparatus, simulating horizontally positioned nuclear fuel, were fabricated based on the predicted critical heat flux.

2.1 Estimation of Heat Flux

To manufacture the test section, the target research reactor must be selected and the target heat flux and cooling channel shape must be determined. In this study, the Kijang Research Reactor(KJRR) was selected as the target research reactor. The design parameters of the KJRR are presented in Table 1.

Table I: Design Parameter of Target Research Reactor

Parameter	Value
Power	15 MW
Number of Fuel Assembly	22 EA
Number of Fuel Plate	21 EA
Heated Length	600 mm
Channel width	66.6 mm
Channel Gap	2.35 mm
Target Power of One Plate	1.3 kW

The target heat flux of the test section must cover the maximum heat flux during fuel withdrawal and be able to reach the level at which CHF occurs. The maximum heat flux during fuel withdrawal was estimated under the condition of withdrawing the fuel immediately after the reactor is shut down. The result is about 1.3 kW per fuel plate, which is about 4% of the normal power.

The critical heat flux (CHF) at horizontal conditions was predicted by two methods. The first method was based on the study by Kim et al.[1], and the second method was based on the study by Park et al[2].

Kim et al. conducted CHF experiments on one-side heating in narrow gap rectangular channels. The gap of the channel was changed to 1, 2, 5, and 10 mm, and the angle of the channel was changed from vertical (90 degrees) to horizontal (180 degrees). According to the experimental results, the CHF value in the horizontal channel is about 10-20% of the vertical value. With this result and the CHF value in the vertical channel, the CHF value in the horizontal condition can be predicted.

The critical heat flux (CHF) in vertical square channels was calculated using the SPACE-RR code. The square channel CHF correlation in the SPACE-RR code is the Sudo-Kaminaga correlation, which is suitable for Kijang research reactor conditions. The SPACE-RR code modeling is shown in Fig. 1. The calculation results using the method of Kim et al. showed that CHF in horizontal narrow gap rectangular channels occurs when the heat flux is about 20-40 kW/m^2.

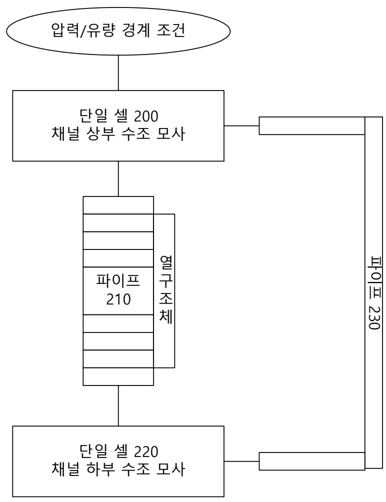

Fig. 1. SPACE-RR modeling

The second method for predicting CHF is an estimate by Park et al. Park applied the energy balance, force balance, and Taylor instability theory to propose a CHF model for narrow gap rectangular channels under pool saturated water boiling conditions. The proposed model is presented in equations (1)-(3). The CHF value calculated by the Park model is 25-50 kW/m2.

$$q_{CHF} = h_{fg}\rho_g\left(\sigma g \Delta\rho\right)^{0.25}\cdot\frac{B}{A}\cdot\frac{1}{N} \qquad (1)$$

$$A = 1 + 6.7\times10^{-4}\left(\frac{\rho_l}{\rho_g}\right)^{0.6}\left(\frac{L}{\delta}\right) \qquad (2)$$

$$B = \frac{\cos\alpha}{8}\left(\frac{4}{\sqrt{3}\pi} + \frac{\sqrt{3}\pi}{\theta}\right)^{0.5} \qquad (3)$$

The CHF value in horizontal channels predicted by the two methods is approximately 20-50 kW/m^2. Therefore, the target heat flux was set to this value and the test section was manufactured.

2.2 Test section and Experimental Apparatus

The test section was manufactured as shown in Figure 2, using the target heat flux calculated in Section 2.1 and the specifications of the KJRR. The heating section was made of STS, and the outer surface of the heating element was insulated with Bakelite.

Fig. 2. Test section

An experimental apparatus consisting of two pools and a test section was manufactured as shown in Fig. 3 to simulate horizontal natural convection. The test section was connected to the pools on both ends with bellows pipes, as the bellows are flexible and easy to adjust to horizontal.

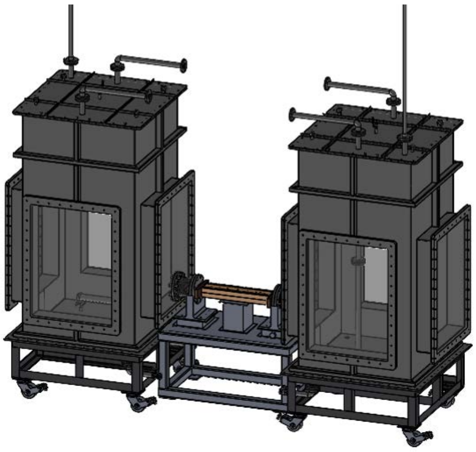

Fig. 3. Drawing of Experimental Apparatus

In this experiment, the temperature of the heated wall was measured at five points in the longitudinal direction, and the temperature of the inlet and outlet coolant was also measured. The temperature measurement positions and sensor numbers are presented in Fig. 4.

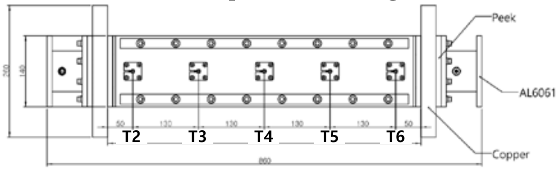

Fig. 4. Location of sensor position and number

3. Test Results and Discussion

In a research reactor, spent fuel is cooled in the core for one day after the reactor is shut down, and then moved to the spent fuel storage pool. At this time, the spent fuel is cooled down to 2% of the full power. Therefore, it can be assumed that decay heat of the spent fuel under normal conditions are less than 2% of the full power.

This preliminary experiment was performed under the condition of 2% of the full power, which is 600W, according to the above assumption. The power was slowly increased to 600W in about 6 minutes, as shown in Fig. 5, and then maintained at a steady state for about 5 minutes.

The experimental results are shown in Fig. 6. As can be seen in the figure, the temperature reached close to 100 degrees Celsius in all parts except for the 6th thermocouple (outlet side). At this temperature, it is estimated that nucleate boiling (ONB) has occurred. After ONB occurred, the wall temperature oscillated due to the bubbles that were generated and disappeared in the cooling channel However, there was no significant temperature rise, indicating the absence of critical heat flux (CHF). Thus, it was confirmed that there were no issues with nuclear fuel integrity under the given conditions.

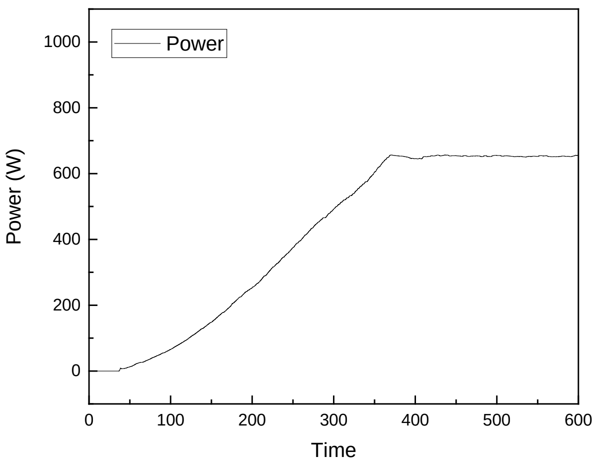

Fig. 5. Power Curve of Preliminary Experiment

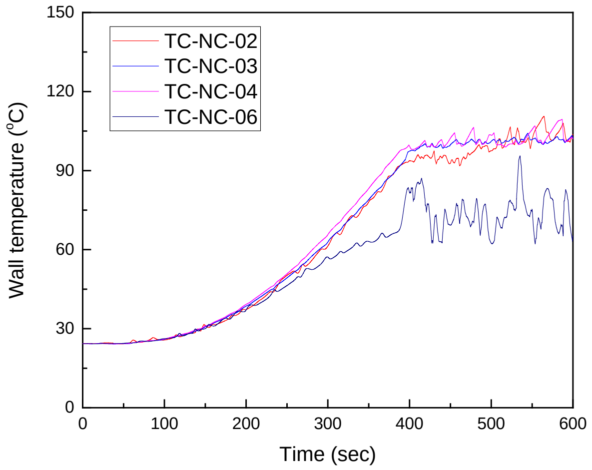

Fig. 6. Experimental results

4. Conclusions

A test section and an experimental apparatus were fabricated and tested to confirm the cooling characteristics of research reactor spent fuel in a horizontal position. The preliminary experiment results showed that the nuclear fuel integrity is maintained when a drop accident occurs under the normal conditions of spent fuel transportation.

ACKNOWELGEMENTS

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