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Transient Analysis of Loss of Coolant Accident in the KJRR

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1. Introduction

The Kijang Research Reactor(KJRR) is currently under construction and is intended as a multipurpose research reactor. This study analyzed the loss-of-coolant accident, one of the representative accidents that could occur in the KJRR.

2. Methods and Results

2.1 Calculation model and method

The transient analysis for KJRR was performed using the RELAP5/MOD3.3 code[1]. Fig. 1 shows a simplified model of the system used for transient analysis. The transient analysis covered the reactor pool, reactor, primary coolant system (PCS), and the Safety Residual Heat Removal System (SRHRS). Other systems were excluded from the analysis because they had little impact on the accident analysis or could not be guaranteed to function during an accident scenario.

In addition, the setpoints and activation times of the Reactor Protection System (RPS) and Engineered Safety Features (ESF) were determined with conservative assumptions.

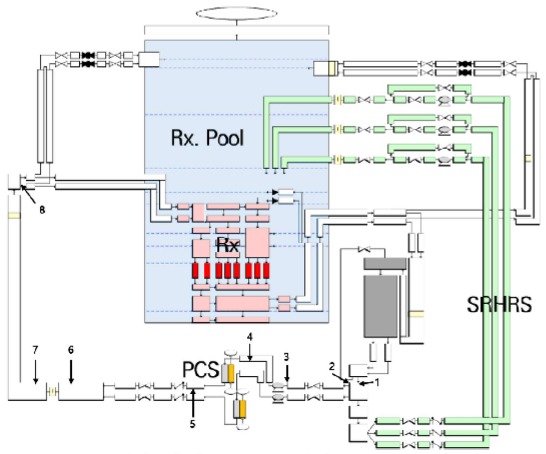

Fig. 1. Simplified diagram of the KJRR calculation Model

The LOCA accident in pipe rupture research varies depending on the location of the rupture. From the reactor outlet to the decay tank, the pressure inside the pipe is lower than atmospheric pressure, so if a rupture occurs, air enters from the confinement. In contrast, if a rupture occurs in other sections, coolant is discharged, similar to a typical LOCA accident. In this study, analysis was conducted only on accidents where coolant is discharged, similar to a typical LOCA accident.

2.2 Calculation results

To select the most limiting break accident, a break spectrum analysis was performed. Fig. 2 shows the results of break flow rate analysis according to the break location. As seen in the figure, the largest flow rate is observed when breaks occur at sections 1, 2, and 3. This is estimated to be due to these sections being located at the lowest point of the PCS piping, resulting in the highest head pressure. Therefore, one of sections 1, 2, or 3 was selected for accident analysis.

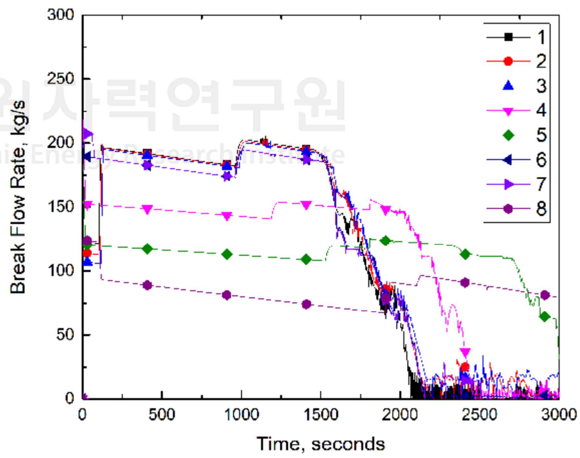

Fig. 2. Break flow rate with various points

When a break occurs, the sequence of events is as follows: if a sudden break occurs in the PCS piping, coolant is released through the break point. As the coolant release causes the reactor pool level to drop(Fig.3), reaching the reactor shutdown setpoint of the Reactor Protection System (RPS), the reactor shuts down. Simultaneously, the primary coolant system pump also stops due to the RPS signal, and the SRHRS activates to provide additional cooling to the core.

If the accident progresses and the pool water level drops below a certain level, the RPS signal causes the siphon break valve to open, preventing additional water discharge from the pool, and the pool level stabilizes.

After sufficient residual heat removal from the core, the SRHRS pump stops operating, and a flap valve automatically opens, forming a natural circulation path within the core. Subsequently, the core is cooled over the long term by natural circulation using the residual coolant remaining in the pool.

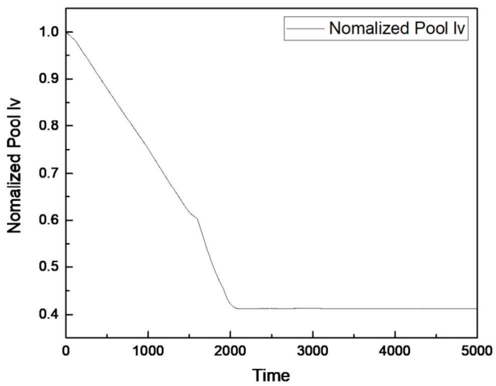

Fig. 3. Normalized Pool Level

Fig. 4 and 5 show the fuel temperature and Critical Heat Flux Ratio (CHFR) during the accident sequence. The fuel temperature rapidly decreases after reactor shutdown in the initial phase of the accident and remains around 50°C. When the SRHRS pump stops operating and the system switches to natural convection, the temperature increases somewhat. After natural convection stabilizes, the temperature is maintained at a certain level.

The CHFR increases rapidly after reactor shutdown and then drops sharply when the system switches from forced to natural convection around 2000 seconds. Subsequently, as latent heat decreases, CHF also increases gradually. Examining the fuel temperature and CHF, sufficient margin is confirmed during the LOCA accident.

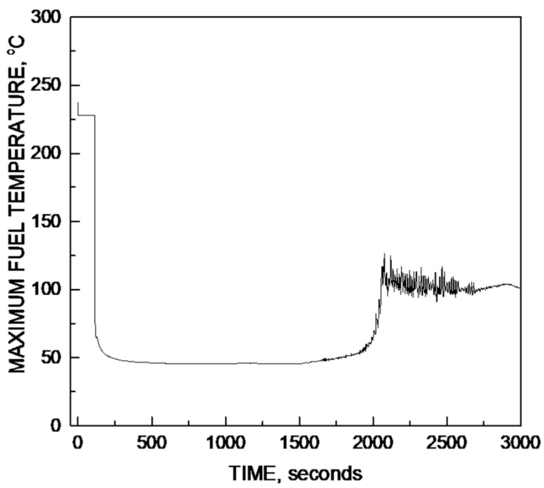

Fig. 4. Maximum Fuel Temperature

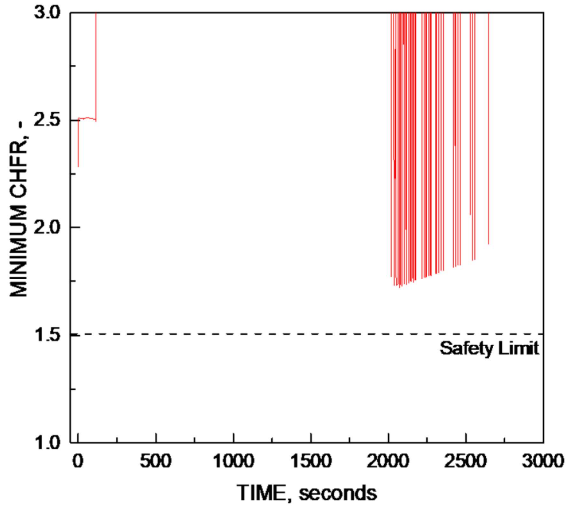

Fig. 5. Critical Heat Flux Ratio

3. Conclusion

A transient analysis using the RELAP5/MOD3.3 program was performed to analyze a LOCA accident in the KJRR. Based on the accident analysis results, the safety of the fuel temperature and the reactor pressure vessel was confirmed from the CHFR perspective.

Acknowledgement

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REFERENCES

[1] RELAP5/Mod3.3, Code Manual Volume V, User's Guideline, NUREG/CR-5535/Rev1, 2001