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Transient Analysis of Loss of Coolant Accident at Sub-atmospheric Pressure Locations 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 at sub-atmospheric pressure location, 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.

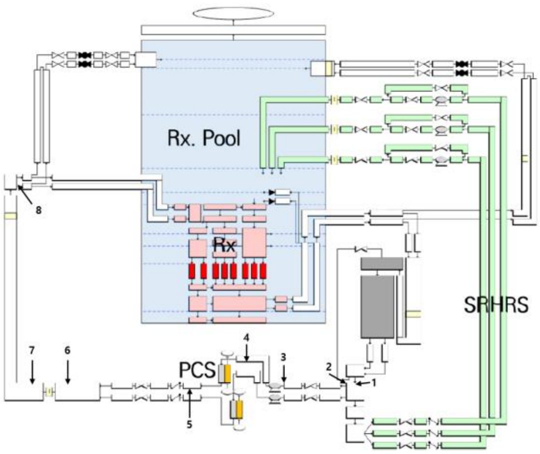


Fig. 1. Simplified diagram of the KJRR calculation Model

The LOCA accident in pipe rupture research varies depending on the location of the rupture. The elevated piping connecting the core outlet to the decay tank inlet maintains sub-atmospheric pressure during steady state operation due to the core pressure drop and elevation head differences. In the event of a break in this section, air ingress occurs, driven by the pressure differential between the confinement air and the internal piping. 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

A sub-atmospheric pressure pipe rupture accident in the primary cooling system refers to a scenario where a pipe break occurs at the highest horizontal section of the piping between the core outlet and the decay tank, as shown in Fig. 2. Since this section is maintained at a pressure below atmospheric pressure during normal operation, ambient air is ingested into the system upon pipe rupture. The entrained air travels through the primary cooling system piping to the decay tank, resulting in a gradual decrease in the tank's water level and a concomitant increase in differential pressure. Furthermore, the primary coolant displaced by the incoming air flows into the pool, causing a slight increase in the pool water level, as illustrated in Fig. 3.

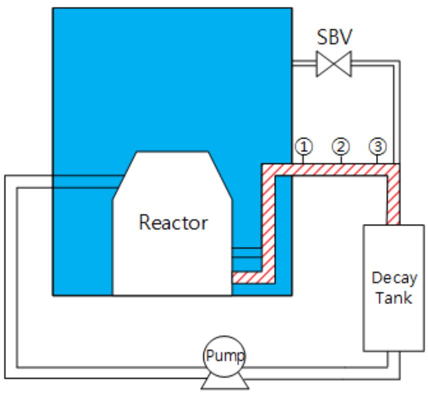


Fig. 2. Location of Sub-atmospheric pressure LOCA

When the differential pressure in the decay tank exceeds a specific threshold, the Reactor Protection System (RPS) simultaneously triggers reactor scram and pump trip signals, leading to the shutdown of both the reactor and the primary cooling system pumps. As the sub-atmospheric pressure at the uppermost piping of the primary cooling system core outlet is relieved upon pump shutdown, air ingestion ceases as shown in Fig. 4. Subsequently, the accident exhibits the same behavior as a conventional Loss of Coolant Accident (LOCA), characterized by coolant discharge.

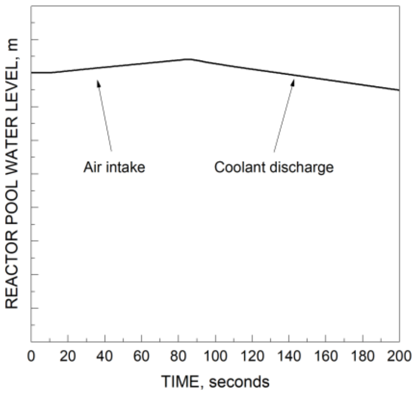
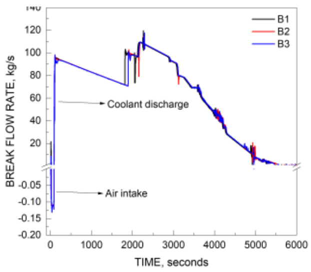


Fig.3 Pool level variation in early stage of LOCA

To identify the most limiting accident scenario, analyses were performed considering various break locations, break sizes, and single failure criteria.

First, three arbitrary break locations were selected along the uppermost piping, as shown in Fig. 2. Since this section is a horizontal pipe with no elevation change, the internal pressure remains nearly uniform. Accordingly, as shown in Fig. 4 and 5, it was confirmed that the air ingestion rate, coolant discharge rate, and water level variations remain nearly identical regardless of the break location. Furthermore, due to the negligible differences in overall thermal-hydraulic behavior, the Critical Heat Flux Ratio (CHFR) and fuel temperature also exhibited nearly consistent values.



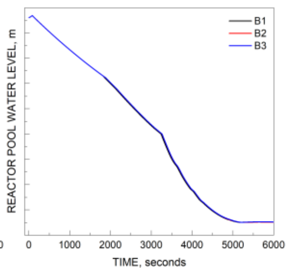
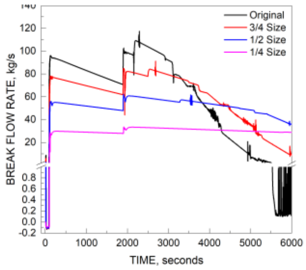


Fig. 4. Break flow rate Fig.5. Pool level

Fig. 6 and 7 illustrate these variations according to the break size. A break size spectrum analysis—conducted at 3/4, 1/2, and 1/4 of the maximum 6-inch break—confirmed that transients progress more rapidly with larger break sizes. However, as shown in Figures 8 and 9, the minimum CHFR and maximum fuel temperature occur during the early stage of the accident, which is attributed to the immediate reactor trip upon the occurrence of the break. Therefore, from the perspective of the minimum CHFR and maximum fuel temperature, no significant differences were observed across various break sizes.



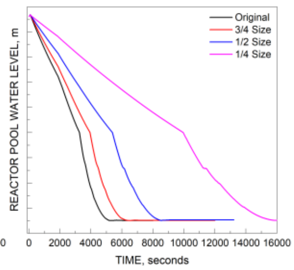
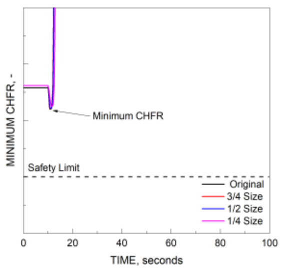


Fig. 6. Break flow rate Fig. 7. Pool level



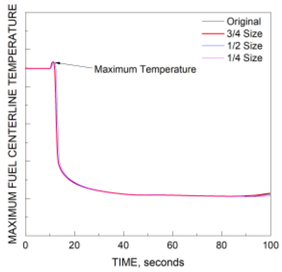


Fig. 8. CHFR Fig. 9. Centerline Fuel Temp.

Finally, transient analyses were performed considering single failure criteria, including the unavailability of one SRHRS train, failure of one primary cooling system check valve, and failure of one siphon breaker valve at either the inlet or outlet piping.

The results indicated that the maximum fuel temperature and minimum CHFR remained consistent regardless of the single failure, aligning with previous sensitivity analysis results. However, as shown in Fig. 10, the failure of one SRHRS train resulted in a higher coolant discharge rate until approximately 2,000 seconds, when the remaining SRHRS pumps were activated, leading to the most rapid decrease in the pool water level (Fig. 11).

Collectively, these results confirm that there are no significant differences in accident progression based on break location, size, or single failure during a sub-atmospheric pressure pipe rupture. Furthermore, it was verified that reactor safety is fully ensured in terms of the minimum CHFR and maximum fuel temperature.

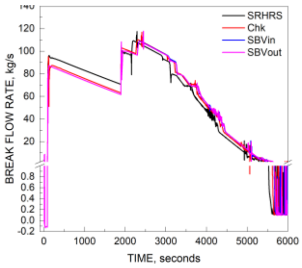


Fig. 10. Break flow rate

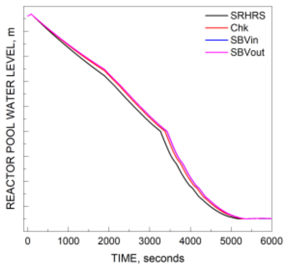


Fig. 11. Pool level

3. Conclusion

A transient analysis using the RELAP5/MOD3.3 program was performed to analyze a LOCA accident at sub-atmospheric pressure locations in the KJRR. Based on the accident analysis results, the overall safety of the reactor was verified, and fuel integrity was specifically confirmed through the evaluation of the minimum CHFR and maximum fuel temperature.

Acknowledgement

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REFERENCES

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