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Preliminary Analysis for a DVI Line Break of APR1400 using SPACE

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1. Introduction

Based on the operating experience for the nuclear power plant during a long period, the Intermediate Break Loss of Coolant Accident (IBLOCA) has considered as the Design Basis Accident (DBA) instead of the Large-Break Loss of Coolant Accident (LBLOCA). The nuclear industry and the regulatory organization in France have already excluded the LBLOCA in the DBA [1]. In Korea, the research project for the LOCA re-classification is on-going to exclude the LBLOCA in the DBA and develop the safety analysis methodology for the IBLOCA. As the early phase of the project, the phenomena identification and ranking table (PIRT) is now developing to enhance the phenomenological understanding for the IBLOCA as a DBA.

In case of APR1400, the direct vessel injection (DVI) type for the safety injection is adopted and the break of a DVI line can be considered as one of IBLOCA cases in the PIRT. Therefore, the characteristics of the DVI line break have to be investigated and the analysis results should be provided to develop the IBLOCA PIRT. The purpose of this study is to conduct the analysis for the DVI line break using the safety and performance analysis code for nuclear power plants (SPACE) and investigate the important parameters for the peak cladding temperature (PCT).

2. Analysis Methods

2.1 Steady State Condition

In this study, the SPACE input model for APR1400 LBLOCA analysis [2] was used with some modifications. The steady state condition is shown in Table 1.

2.2 DVI Line Break Model

The DVI break was simulated as shown in Fig. 1. The connection between SIT D and SIP (C562) was removed and the TFBC node having the atmospheric condition was connected with the upper downcomer region (C562-02) which is the same location of the SIT D injection line.

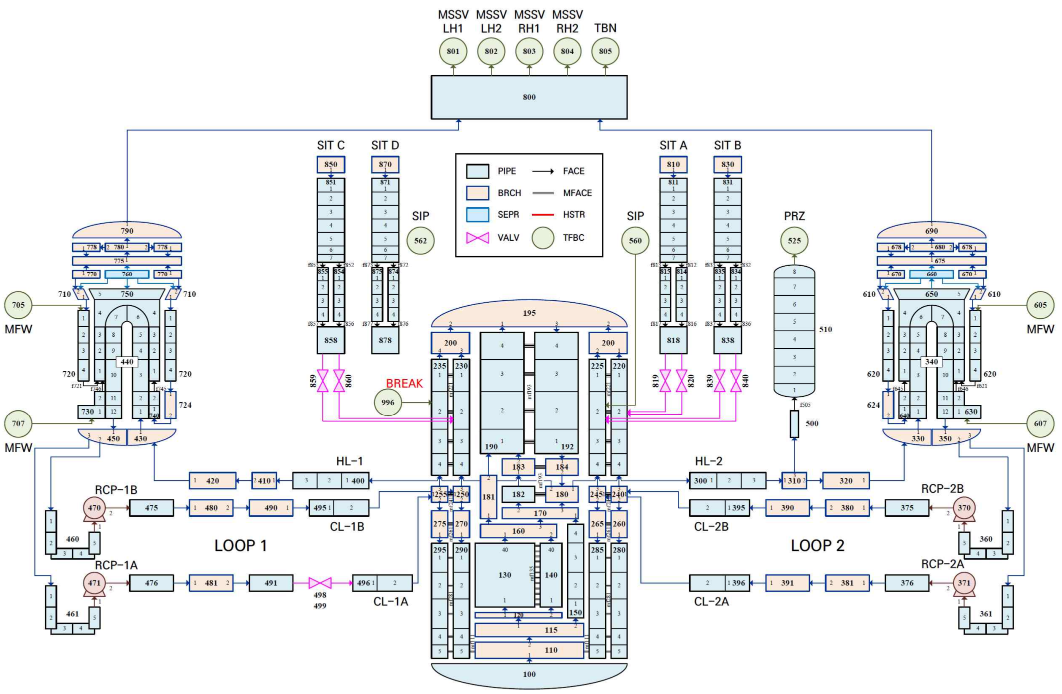

Fig. 1. Nodalization of the SPACE input model for APR1400

Table I: Steady state condition

System parameter	Design	SPACE
Reactor power (MWth)	3983	3983
Core flow rate (kg/s)	20361	20682.2
Hot FA flow rate (kg/s)	-	83.6
Average core flow rate (kg/s)	-	20598.6
Hot leg flow rate (kg/s)	10495.5	10660.4
Cold leg flow rate (kg/s)	5247.75	5330.2
Pressurizer pressure (MPa)	15.51	15.51
Core inlet temperature (K)	563.6	564.54
Core outlet temperature (K)	598.3	598.12

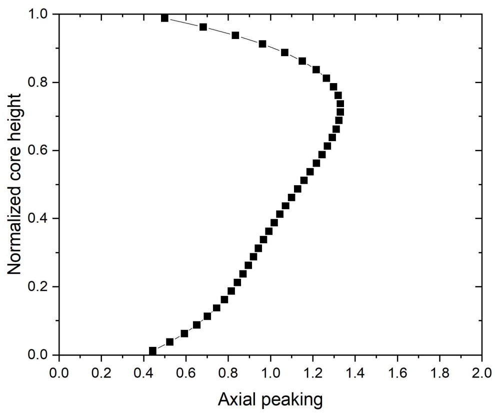

Fig. 2. Axial power distribution (F_z=1.33)

The guillotine break of a DVI line was simulated and the break size was 0.4 ft^2 (0.0371m^2) which corresponds to 8.1% of the cross-sectional area of a cold leg. For the critical flow at the break location, Henry-Fauske model was used with the discharge coefficient of 1.0 for all conditions and the non-equilibrium constant of 0.14.

2.3 Assumptions for Transient Analysis

The axial power distribution in the core and the reactivity coefficient table used in this study were the same with those applied in the LBLOCA safety analysis [2]. The axial power distribution had a slightly top-skewed shape having F_z of 1.33, as shown in Fig. 2.

The reactor coolant pump (RCP) trip and the secondary system isolation were assumed to occur in the same time with the break. Due to the isolation of the secondary system, the increase of the secondary system pressure and the opening of MSSV were predicted and the MSSVs and their setpoints were simulated.

Table II: Main event sequences for DVI line break

Event	Time (sec)
PCT	8 (699K)
Reactor trip	22.1
SIP injection	60.9
Crossover of 1st/2nd pressure	99.1
SIT injection	200.6

Also, it was assumed that one diesel generator was failed as the single failure. In this case, two SIPs were not operated owing to the single failure and additionally one SIP and one SIT was not operated due to a DVI line break. Consequently, one SIP and 3 SITs were normally operated during transient.

3. Transient Analysis Results

Based on the steady state condition, the break model and the assumptions described in section 2, the transient analysis for the DVI break was conducted using SPACE. The main results for the sequence and event time are listed in Table II.

In the SPACE analysis results, the RCP trip occurred and the secondary system was isolated simultaneously with a DVI line break. The coolant was discharged from the simulated DVI line break (Fig. 3) and the primary system pressure was rapidly decreased (Fig. 4). At 22 seconds after break, the reactor trip occurred with low pressure setpoint (10.72 MPa) of the primary system and a SIP was actuated with 40 seconds delay.

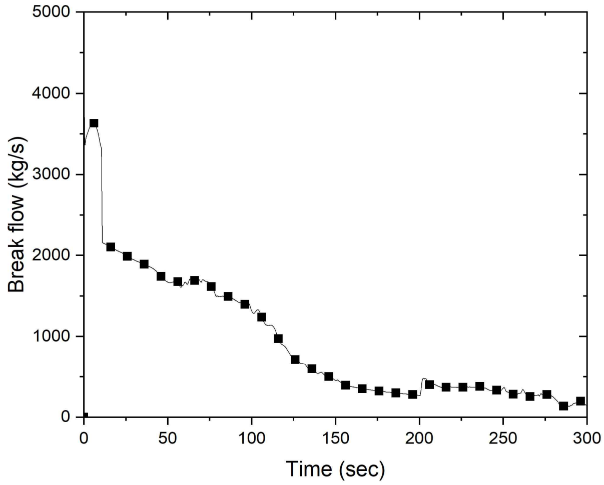

Fig. 3. Flow rate at DVI line break

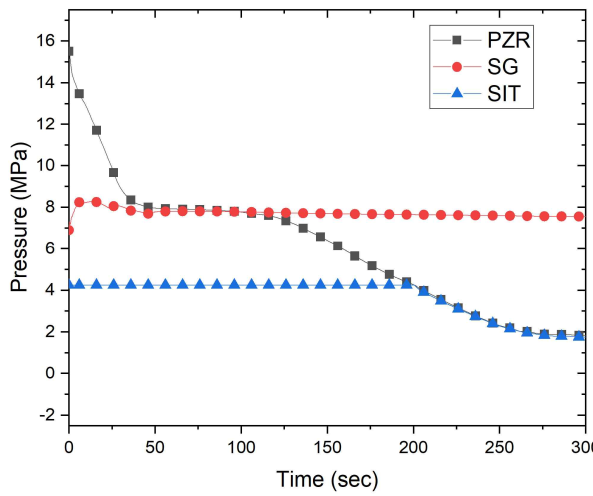

Fig. 4. Pressure of main systems

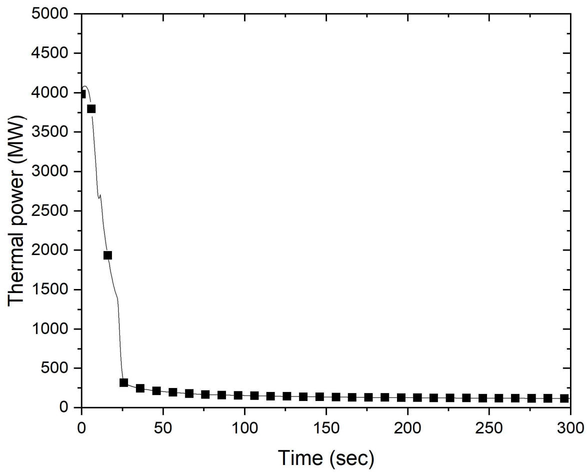

Fig. 5. Core thermal power

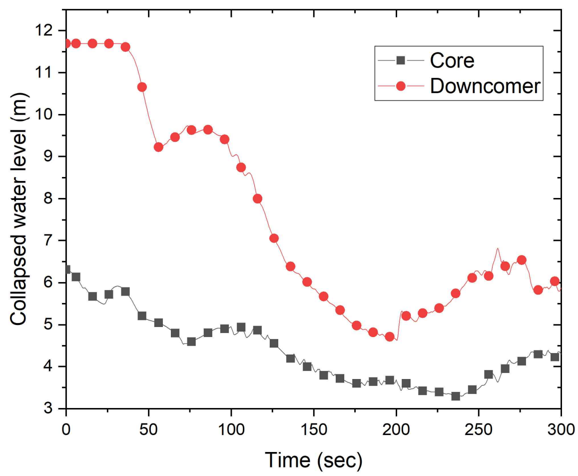

Fig. 7. Water level in core and downcomer

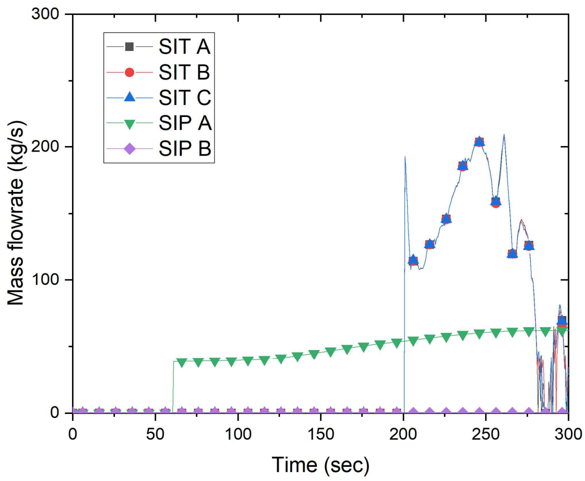

Fig. 6. Flowrate of safety injection

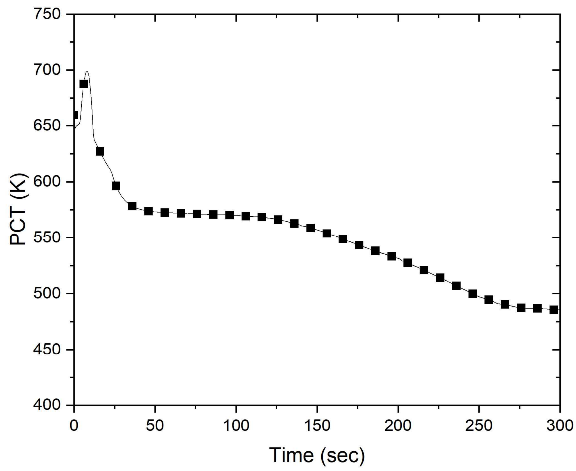

Fig. 8. Peak cladding temperature

The secondary system pressure was increased due to main steam and feedwater isolation and heat transfer from the primary system. Therefore, the MSSV was open during ~40 seconds at early phase. At 99 seconds after break, the secondary system pressure became higher than the primary system pressure. After that crossover of system pressure, the heat was inversely transferred from the secondary to the primary systems.

As shown in Fig. 5, the core power was rapidly decreased right after break owing to negative reactivity of the moderator density. After the reactor trip, the core power was more rapidly decreased and only the decay heat was generated.

The primary pressure was continuously decreased to ~2 MPa. When the primary pressure was 4.245 MPa, the cooling water in 3 SITs was injected. The flowrate of the safety injection was shown in Fig. 6.

The water levels in the core and the downcomer were shown in Fig. 7. After break, the water levels in the core and the downcomer were rapidly decreased. After beginning of the SIT injection, the water level in the downcomer was increased and the core water level was also recovered with ~30 seconds delay.

The PCT was shown in Fig. 8. The core heat-up occurred at early phase owing to the RCP trip before the reactor trip and the PCT was 699 K in this phase. When the core water level was close to the lowest level, the core was not heated up. It means the core heat-up due to the boil-off did not occur.

3. Conclusions

Based on the LBLOCA safety analysis methodology, the SPACE analysis was conducted for the DVI line break, additionally with the assumptions of the failure of a diesel generator and RCP trip and secondary system isolation at break.

The core heat-up occurred at the early phase due to the RCP trip before the reactor trip. The coolant was discharged and the primary system pressure was decreased to ~2 MPa. As a consequential event, the core water level was decreased. However, the core heat-up did not occur during the boil-off phase due to the high core water level.

As further studies, the SPACE analysis is needed using SBLOCA analysis methodology because the size of a DVI line break was relatively small.

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