Test result on a small break loss-of-coolant accident simulation for the control rod driving mechanism nozzle rupture with a failure of safety injection pump using the ATLAS facility

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1. Introduction

The safety issues related to the structural integrity of the upper head penetration nozzle of the nuclear reactor were raised in 2002 and 2012 as problems with the RPV wall thinning around the control rod drive mechanism (CRDM) penetration nozzle in the Davis Besse nuclear power plant in the United States and the micro-cracking on the control element drive mechanism of Younggwang Unit 3 [1, 2]. Recently, the safety issue on an upper head penetration nozzle of a reactor pressure vessel (RPV) was raised again as 35 of the 84 welds in which stress corrosion cracks were found during the preventive maintenance of Hanbit Unit 5 were disapproved [3]. Circumferential cracking of the penetration nozzle can lead to a small break loss-of-coolant accident (SBLOCA) at the RPV upper head.

In general, when an SBLOCA occurs at the upper head of a RPV, the characteristic of the thermal-hydraulic phenomenon is that the break flow is mainly discharged in a vapor phase compared to the other located SBLOCAs such as a break at hot legs, cold legs, and direct vessel injection lines. Therefore, it was reported that the coolant inventory in the primary system was relatively well preserved under a safety injection pump (SIP) operation, resulting in a later core heat-up [4]. However, as a multiple failure accident scenario, an SBLOCA at the RPV upper head with a failure of SIP can damage a reactor core if a proper accident management (AM) action is not implemented.

In this study, a multiple failure accident of an SBLOCA due to a break of two CRDM nozzles with a failure of SIP was simulated by using a thermal-hydraulic integral effect test facility, ATLAS (Advanced Thermal-hydraulic Test Loop for Accident Simulation) [5]. Regarding an operator's AM action, 50 % opening of an atmospheric dump valve (ADV) of steam generator was taken according to the pre-specified maximum heater rod surface temperature in the core. In addition, the auxiliary feedwater (AFW) was supplied when the secondary side collapsed water level reached a wide-range of 25 % in both steam generators.

The main objective of this study is to investigate the thermal-hydraulic phenomena during an SBLOCA at RPV upper head with a failure of SIP focused especially on the break flow behavior related to the water level in the RPV upper head and the loop seal clearing (LSC) behavior associated with the pressure build-up through the RPV bypass line.

Considering the confidential problem, all the data in this paper are presented with the normalized value using random numbers.

2. Descriptions for the test

ATLAS is a large-scale thermal-hydraulic integral effect test facility for APR1400 (Advanced Power Reactor 1400 MWe), OPR1000 (Optimized Power Reactor 1000 MWe) and APR+ (Advanced Power Reactor Plus), which can simulate the overall thermal-hydraulic behavior of major systems and components during transient and accident conditions at prototypical pressure and temperature conditions. The detailed design and description for ATLAS facility can be found in the literature [6].

2.1 ATLAS configuration

This study presents an experimental result of the test named CRDM-SIP-03. Figure 1 shows a schematic diagram of a loop connection of ATLAS for the CRDM-SIP-03 test. The fluid system of ATLAS consist of a primary system, a secondary system, a safety injection system, a break simulating system, auxiliary systems.

In the test, the break simulation system consists of a quick opening valve (OV-BS-01), a break nozzle, and related instruments as shown in Figure 2. The inner diameter of the break nozzle was determined to be 7.12 mm which corresponds to 1/203.6 of the break area for two CRDM nozzles of APR1400. The break flow from the broken CRDM nozzles was sent to the refueling
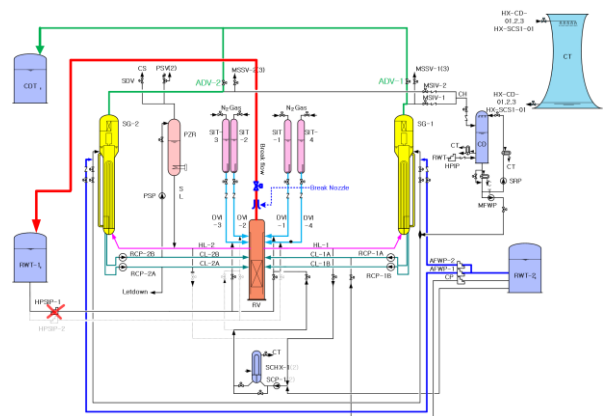

Fig. 1 Schematic flow diagram of the ATLAS highlighting the break flow collection and AFW supply systems
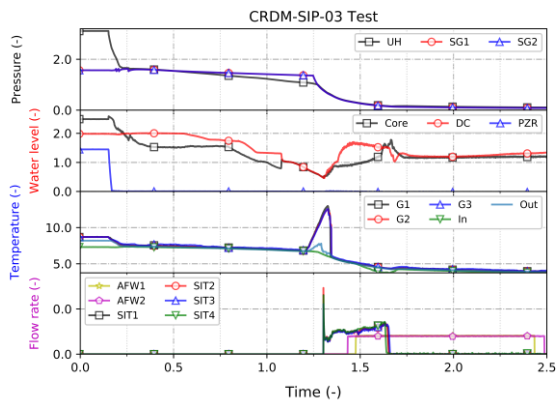
Fig. 2 A schematic diagram of the break simulation system
(Unit: mm)

water tank number 1 (RWT-1) for collection and measurement. The discharged flow from the secondary system through an ADV opening was collected and measured in the condensation tank (CDT). The flow area of the ADV was set to 4.607e-4 m² according to the 50 % ADV opening.

The safety injection water from safety injection tanks (SITs) was supplied through direct vessel injection (DVI) nozzles on an upper down-comer of the RPV. The water temperature, water level, and pressure of SITs were set to approximately 50 °C, 3.7 m, and 4.3 MPa, respectively. The coolant used for the AFW was supplied from the RWT-2, and the AFW temperature and flow rate were set to approximately 50 °C, 0.2 kg/s, respectively.

The decay heat was simulated to be 1.2 times that of the ANS-73 decay curve from a conservative point of view. The initial heater power was controlled to be maintained at about 1.664 MW, which was equal to the sum of the scaled-down core power (1.566 MW) and the heat loss rate of the primary system (about 98 kW) for the test.

2.2 Test procedures

After the steady-state period, the test was started by opening the break valve, OV-BS-01.With the break, the primary system pressure decreased rapidly and a low pressurizer pressure (LPP) signal was generated. When the LPP signal occurred, the reactor scram and turbine trip signals were generated. The main feedwater

Table 1: Sequence of events and set-points on the CRDM-SIP-03 test

No	Event	Set-points	Non-dimensional Time (-)
1	Test start	Break valve open	0.1515
2	LPP Trip	Primary system pressure	0.1855
3	Reactor Scram	Coincidence with LPP	0.1855
4	MSIS	With delay time after reactor scram	0.1875
5	MFIS	With delay time after reactor scram	0.1895
6	Core power decrease	With delay time after reactor scram for 8% power simulation	0.1915
7	Loop seal clearing	Collapsed water level on the intermediate leg (SG side) is zero	1.0775
8	Start of Excursion of heater rod surface Temperature	Maximum heater rod surface temperature	1.2010
9	AM Action	With the set-point of the maximum heater rod surface temperature	1.2490
10	SIT initiation	Down-comer pressure	1.3020
11	SIT fluid device simulation	With set-point on the SIT's collapsed water level	1.4870
12	SIT injection termination	With set-point on the SIT's collapsed water level	1.4885
13	AFW injection start	25 % water level of SG secondary inventory	1.4755
14	AFW injection stop	40 % water level of SG secondary inventory	2.4340
15	Test end	Shout down cooling condition	2.5000

LPP: Low Pressurizer Pressure
MSIS: Main Steam Isolation Signal
MFIS: Main Feedwater Isolation Signal
MSSV: Main Steam Safety Valve
AFW: Auxiliary Feedwater

isolation valves and the main steam isolation valves were closed with pre-specified delay times. Despite the occurrence of LPP signal, emergency core cooling water from SIP could not be supplied due to an assumption of a total failure of SIP. Until the primary system pressure decreased to the set-point for an actuation of SIT, 4.03 MPa, the core heat-up occurred due to the inventory loss through the break. The AM action of 50 % opening of an ADV of steam generator was taken to mitigate an accident consequence according to the pre-specified maximum heater rod surface temperature in the core. The AFW was injected at 25 % of the SG secondary side water inventory and stopped at 40 % of the water inventory. Table 1 summarizes the sequence of the events observed in the present test.

Fig. 3 Overall thermal-hydraulic behavior for the CRDM-SIP-03 test

3. Experimental results

Overall thermal-hydraulic behaviors are presented in Figure 3. When the break of the two CRDM penetration nozzles was initiated by opening the break valve, the primary system was rapidly depressurized to the secondary system pressure which bounded at the set-point of main steam safety valves. The primary system pressure decreased slowly due to the reduced decay heat and also the change of the break flow quality. Figure 4 (upper) shows the integrated mass of the break flow from the primary system and the discharged flow from the secondary system measured by load cells. And Figure 4 (lower) indicates the mass flow rate of break flow from the primary system. During the initial blow-down period (approximately 0.1007 ~ 0.1925 normalized time) the break flow was effectively discharged as a liquid dominant state. After the decrease of upper head water level in the RPV, the break flow was discharged as a steam generated from the core until the ADV opening. Since the break area was relatively small and located at the top of the RPV, the water inventory in the core was well preserved. However, the
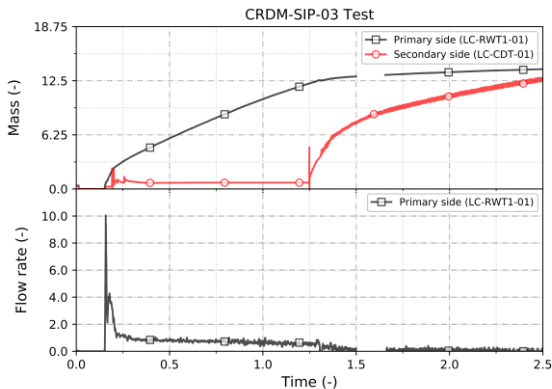
Fig. 4 Integrated mass and flow rate of the break flow from the primary system and the discharged flow from the secondary system
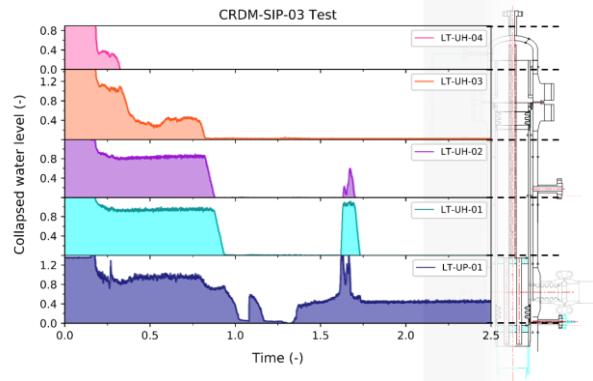
Fig. 5 Collapsed water levels in the RPV upper head

water inventory in the core decreased significantly when there was no more coolant in the U-tube and the lower plenum of the steam generator. After the decrease of the water inventory in the core was stopped with the water supply by LSC phenomenon, it was maintained until the initiation of SITs injection. The secondary system cooling effect by the ADV opening as the AM action resulted in the decrease of the break flow rate by reducing the pressure and temperature of the primary system. Figure 6 shows the collapsed water levels of vertical piping in intermediate legs (ILs). When the water level of the steam generator side piping in IL2A decreased and reached the horizontal piping, the coolant in the loop seal was abruptly supplied to the core through the down-comer. As a result, the core water level suddenly increased at 1.0775 normalized time as shown in Figures 3 and 5 (LT-UP-01). In the present test, LSC phenomenon was triggered by differential pressure (DP) between hot leg (or core) and the cold leg (or down-comer), which is dependent on the manometric force between the bottom of the loop seal and the core or down-comer water levels [7]. In Figure 7 (upper), DP between the cold leg and hot leg indicates that the steam generated from the core pushed the hot leg to cold leg through the ILs before the LSC occurrence. This pressure build-up was due to the steam transfer from the down-comer to the upper head though
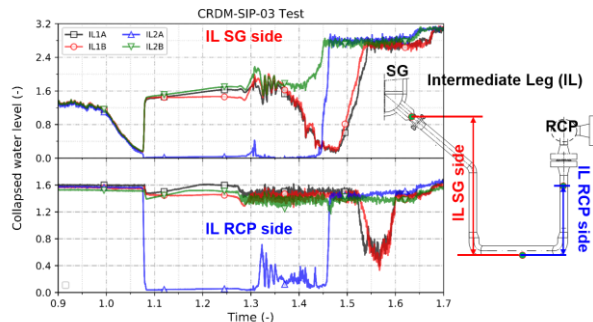

Fig. 6 Collapsed water levels in the intermediate legs indicating the SG side (upper) and the RCP side (lower)
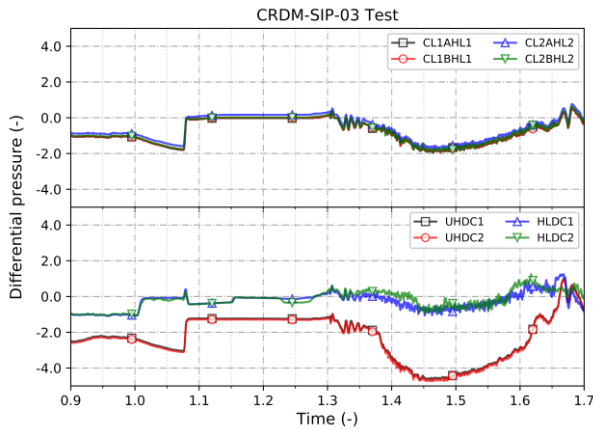
Fig. 7 Differential pressures between hot legs to cold legs and between down-comer to upper head of RPV and hot legs through the bypass line in the ATLAS

the RPV bypass line. In Figure 7 (lower), DP between the upper head and down-comer shows the steam flow from the down-comer to upper head before the LSC occurrence (at 1.0775 normalized time). Although the present test was not an SBLOCA with the break located at cold leg, the steam flow through the RPV bypass line caused the LSC phenomenon with the pressure difference between the hot leg and cold leg. Due to the coolant supply to the core by the LSC, the excursion of heater rod surface temperature was delayed in the present test.

4. Conclusions

A simultaneous double-ended guillotine break of two nozzles of the CRDM on the upper head of the RPV was simulated using the ATLAS facility. The test was performed to simulate an SBLOCA that occurs along with a failure of SIP as a multiple failure accident. The AM action of 50 % opening of an ADV of a steam generator was taken to mitigate an accident consequence according to the pre-specified maximum heater rod surface temperature in the core.

The major thermal-hydraulic phenomena such as the system pressure, the collapsed water level in the RPV, the break flow mass and the maximum heater rod surface temperature were presented and discussed. Especially, the break flow behavior and the LSC behavior associated with the pressure build-up through the RPV bypass line were analyzed.

The major findings of the present test are summarized as follows

- The discharged flow rate through the CRDM nozzle depended on the break flow quality. Except for initial blow-down phase, the break flow was discharged as a steam. As a result, the water inventory in the core was well preserved.

- During an SBLOCA at the RPV upper head, the LSC phenomenon was observed. When the steam flow from the down-comer to the upper head formed through the RPV bypass line, the LSC occurred, and the coolant in the loop seal was supplied to the core through the down-comer.

The present integral effect test data are expected to provide a technical insight for resolving the safety issue of the penetration nozzle impairment recently raised in Hanbit Unit 5 nuclear power plant. In addition, they can contribute to resolving the safety issues and enhancing the safety analysis technology for a multiple failure accident of an SBLOCA with a failure of SIP.

ACKNOWLEDGMENTS

REFERENCES

[1] K.J.Kim "Regulatory positions for in-service inspection," Nuclear Safety & Security Information Conference 2015, KINS, (2015).
[2] USNRC, "Davis-Besse reactor vessel head degradation lessons-learned task force report" LLTF Report, U.S. NRC (2002).
[3] Nuclear Safety and Security Commission, "Special inspection status and future plans for identifying suspicions of maintenance of the reactor head of Hanbit Unit 5," 129th Nuclear Safety and Security Commission, (2020).
[4] K.H. Kang et al., "Test report on the small break loss of coolant accident simulation for the CEDM nozzle rupture with the ATLAS," KAERI/TR-4856/2013, Korea Atomic Energy Research Institute (2013).
[5] K.Y.Choi et al., "Recent achievement and future prospects of the ATLAS program," Nuclear Engineering and Design 354, 110168, (2019).
[6] J. B. Lee et al., "Description Report of ATLAS Facility and Instrumentation (Third Revision)," KAERI/TR-8106/2020, Korea Atomic Energy Research Institute (2020).
[7] Y.S. Kim and K.H. Kang, "Comparison of loop seal clearing behavior between SBLOCA and SBO-related tests in the ATLAS facility," Annals of Nuclear Energy 132, 734-740, (2019).