Experimental Study on the Intermediate Break Loss of Coolant Accident (IBLOCA) with Total failure of Safety Injection Pumps

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1. Introduction

In Korea, the nuclear safety regulation was revised to define the safety regulation clearly, in 2015. In the revised nuclear safety regulation, the "Multiple failure" is defined as "Over the single failure, the failure is occurred in more than two component and loss of their safety function". This multiple failure accident is considered as an accident having a high core damage frequency (CDF) in the not only deterministic safety analysis (DSA) but also probabilistic safety analysis (PSA) method [1].

Especially, an intermediate-size break loss of coolant accident (IBLOCA) with the loss of safety injection was selected as the additionally considered multiple failure accident for all operating nuclear power plant. Even though this accident has relatively low probability to occurrence, ($9.8e^{-9}$/year for APR1400) the core damage frequency is high. And there is no measure to prevent the core damage according to the deterministic safety analysis evaluation result.

Thus, an integral effect test to simulate an IBLOCA with total failure of safety injection pumps (SIPs), which was named IB-SIP-02 test, was performed to investigate the thermal hydraulic phenomena during this multiple failure accident using the ATLAS (Advanced Thermal-Hydraulic Test Loop for Accident Simulation) facility [2].

The objective of this test is to investigate the general thermal hydraulic phenomena during a multiple failure accident of an IBLOCA with total failure of SIPs. In addition, an accident mitigation strategy for preventing the core damage, such as its measure and the grace period for initiation of the accident management action, will be evaluated.

2. Description of the Test Facility

2.1 ATLAS Test Facility

ATLAS was designed to model a reduced-height primary system of APR1400 (Advanced Power Reactor 1400 MWe) which has the 1/2-height, 1/144-area and 1/288-volume scales for APR 1400 [2]. ATLAS can be used to provide the unique test data for a loop arrangement of 2 hot legs and 4 cold legs for the reactor coolant system with a direct vessel injection (DVI) of emergency core cooling (ECC) system. ATLAS can simulate full pressure and temperature conditions of APR1400. The total inventory of the primary system is 1.6381 m^3.

The fluid system of ATLAS consists of a primary system, a secondary system, a safety injection system, a break simulation system, a containment simulation system, and auxiliary systems.

The primary system includes a reactor pressure vessel (RPV), two hot legs, four cold legs, a pressurizer, four reactor coolant pumps, and two steam generators. The secondary system of ATLAS is simplified to be a circulating loop-type. The steam generated at two steam generators is condensed in a direct condenser tank, and the condensed feedwater is re-circulated to the steam generators. A scaling method of the ATLAS design [3] and the detailed design and description of ATLAS facility can be found in the literature [2].

2.3 Break Simulation System

Generally, a break accident with the break area which is larger than 10 % of the primary loop pipe flow area is considered as an intermediate size break. In the IB-SIP-02 test, the break was simulated on the cold-leg (1A) with 10 inch break size for prototype power plant with downward direction. The inner diameter and the length of the break nozzle were designed to be 18.0 mm and 226.0 mm, respectively as shown in Figure 1. The break nozzle area is 2.544×10^{-4} m^2, which corresponds to 11.4 % of the cold leg flow area of APR1400.

The break simulation unit starts from the cold leg and it is connected to a refueling water tank (RWT) to measure the accumulated mass by a load cell.

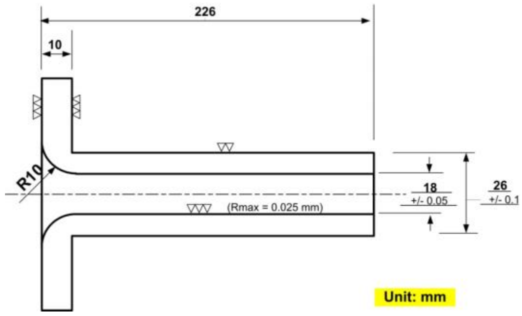

Fig. 1. Schematic diagram of the break nozzle

3. Test Procedure

When the whole system reached a specified initial test condition, the steady-state conditions of the primary and the secondary systems were maintained for more than 30 minutes. After that, the test was initiated by opening of a break valve.

The primary system pressure decreased rapidly due to the break. The reactor scram signal was actuated when the primary system pressure reached the specified pressure of low pressurizer pressure (LPP) trip condition. With the generation of the reactor scram signal, the turbine stop signal was actuated so the main steam control valves (MSCVs) and the main steam isolation valves (MSIVs) were closed. The main feedwater isolation signal (MFIS) was generated to close the main feedwater isolation valves (MFIVs) and the main feedwater pumps stopped so steam generators were isolated.

Actual decay of the core power started with a specified delay time after the actuation of scram signal and it followed the scaled decay power curve. The decay heat in the reactor core was simulated to be 1.2 times the ANS-73 decay curve for a conservative condition.

When the primary system pressure decreased to the set point of the SIP actuation, the coolant injection was not realized at this point because the total failure of SIPs was assumed in the test scenario.

With the continuous and rapid decrease of the primary system pressure, safety injection tank (SIT) actuation signal was initiated. Four SITs were actuated and coolant was supplied through four DVI lines. After the rated inventory from SITs was fully delivered to the reactor coolant system (RCS), SITs were isolated to prevent the nitrogen gas inflow.

Due to the relatively large break size, the pressure and collapsed water level in the reactor pressure vessel decreased quickly. Thus, repairing of one train SIP by the operator was set up as an accident management (AM) action. The initiation condition of this AM action was the later condition either 30 minutes (prototype power plant time, [4]) after break initiation or the maximum heater rod surface temperature exceeding 500 ^{o}C.

By the initiation of the AM action, coolant was supplied to the RCS and the system was cooled down stably. The test was terminated by the operator's decision after the system condition reached to the shutdown cooling system operation.

The sequence of test scenario is listed in Table I.

Table I: Sequence of event

No	Description	Remark (set point)	Non-dimensional Time
1	IBLOCA Start	Break valve open	0.1007
2	LPP Trip	Primary system pressure	0.1067
3	Reactor Scram	Coincidence with LPP	0.1067
4	MSCV Close	With delay time after reactor scram	0.1070
5	MSIV Close	With delay time after reactor scram	0.1078
6	MFIV Close	With delay time after reactor scram	0.1090
7	SIT injection	Primary system pressure	0.1530
8	Excursion of heater rod surface Temp.	Maximum heater rod surface temperature	0.8130
9	AM Action	30 minutes after break initiation or rated value of the maximum heater rod surface temp.	0.8130
10	Quenching	-	0.8423
11	Test end	Shout down cooling condition	1.0000

4. Test Result

Considering the confidential problem of test data, all of the test results in this paper were normalized by an arbitrary value including the time frame.

Figure 2 shows the variation of the system pressures. With start of the transient, the primary system pressure started to decrease rapidly with opening of the break valve. When the primary system pressure reached a specified value, the scram signal was generated.

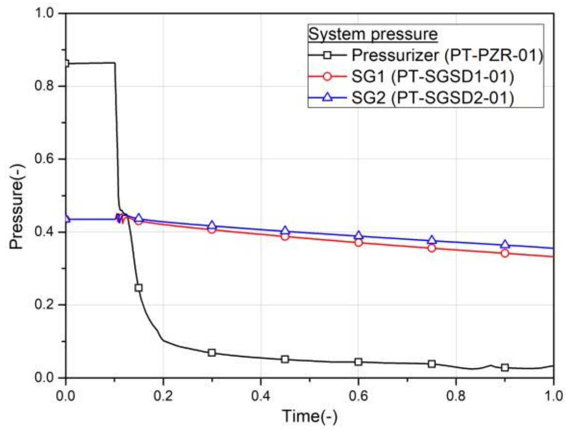

Fig.2. Variation of the system pressure

With the scram signal, steam generators were isolated from the system and it led to an increase in the secondary system pressure to higher than the set point of the opening of a main steam safety valves (MSSVs). A periodic discharge of the secondary system inventory through the MSSVs induced the pressure fluctuation of the secondary system during the early short period of the transient. But it had no significant effect on the primary system behavior except the short period of the pressure plateau. The secondary system pressure decreased gradually through the whole test period due to the depressurization and the stable management of the accident of the whole system.

In the IB-SIP-02 test, the collapsed water levels in the core and the down-comer decreased very drastically from the early transient period as shown in Figure 3. The collapsed water level in the core decreased under the top of the active core level in short time from the transient starts. Thus the excursion of heater rod surface temperature started at 0.1180 of non-dimensional time as shown in Figure 4.

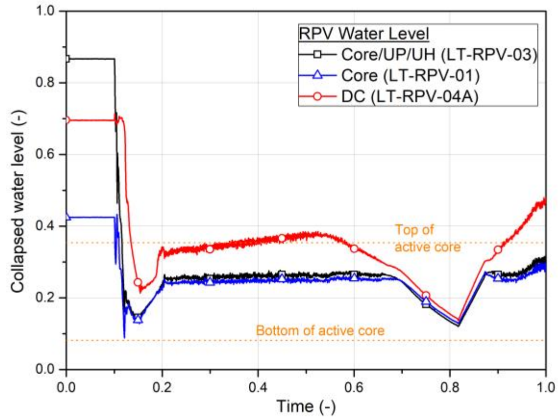

Fig. 3. Collapsed water levels in the RPV

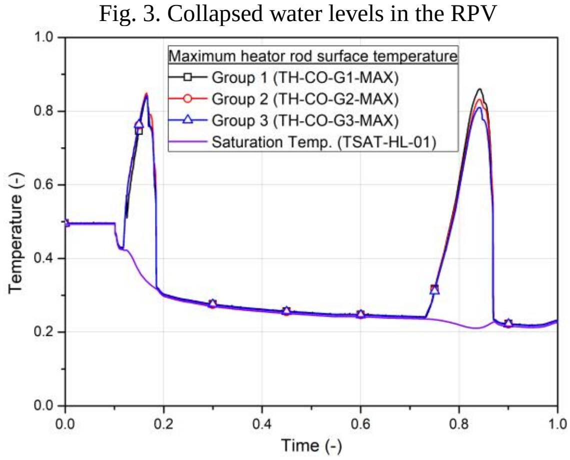

Fig. 4. Heater rod surface temperature behavior

During the test period, a loop seal clearing (LSC) phenomenon occurred. At the early stage of the transient, around 0.1233 of the non-dimensional time, collapsed water levels of loop seals started to decrease and the most of the loop seals, except the primary loop of 1B, were empty and they were not recovered till the end of the test, as shown in Figure 5 and Figure 6.

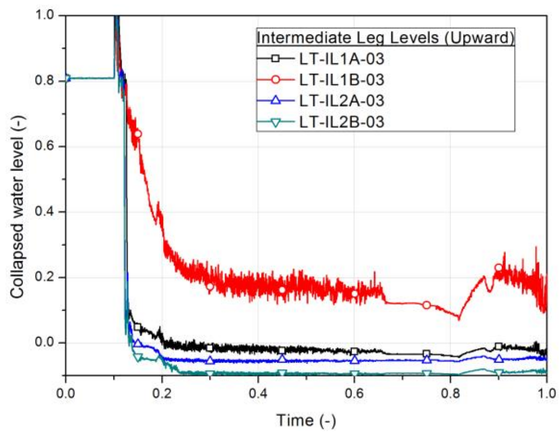

Fig. 5. Collapsed water level of intermediate legs (Upward)

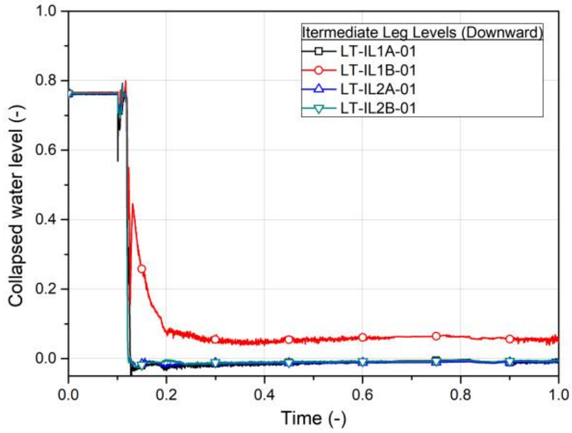

Fig. 6. Collapsed water level of intermediate legs (dowmward)

The LSC, of course, contributed to the recovery of the collapsed water level in the core. However, the LSC phenomenon occurred at the very early stage of the transient when the loss of the primary system inventory was very drastic. Thus the effect of LSC on the recovery of the collapsed water level in the core was negligible and the excursion of the heater rod surface temperature was followed with the large amount of loss of the primary system inventory through the break.

SITs were actuated in soon due to the primary system depressurization as shown in Figure 7 and Figure 8. During the safety water injection from SITs, the flow rate of the safety injection water was switched to the low flow region by the fluidic device simulation and they were injected to the RCS till the rated inventory was fully delivered and terminated around 0.5000 of the non-dimensional time.

The injected coolant from the SITs through DVI lines helped the recovery of the collapsed water level in the RPV. So the increase of heater rod surface temperature stopped as shown in Figure 4. However, right after the termination of SITs injection, the collapsed water level in the down-comer started to decrease and the collapsed water level in the core also decreased in soon due to the continuous loss of the coolant through the break.

It took time to decrease the collapsed water level in the RPV low enough to increase heater rods surface temperature. 30 minutes (prototype power plant time) after break initiation, which is one of the initiation condition of the AM action, had passed without heater rods surface temperature excursion. Thus an operator could have the time to take the AM action for more than 30 minutes in this test scenario.

However, heater rods surface temperatures increased again with continuous loss of the primary coolant without any make-up. When the maximum temperature of the heater rod surface exceeded 500 ºC, the AM action was initiated by the operator at 0.8130 of the non-dimensional time. Repairing of one train SIP by the operator was assumed to be the AM action. So SIP-1 and SIP-3 were actuated to inject the coolant to the RPV through DVI lines with the rated flow rate according to the primary system pressure. The core heaters were quenched by the AM action with recovery of the collapsed water level in the RPV. And the system was cooled down stably with continuous supply of the coolant to the system even though the break flow rate was kept at a constant rate till the late period of the transient, as shown in Figure 9.

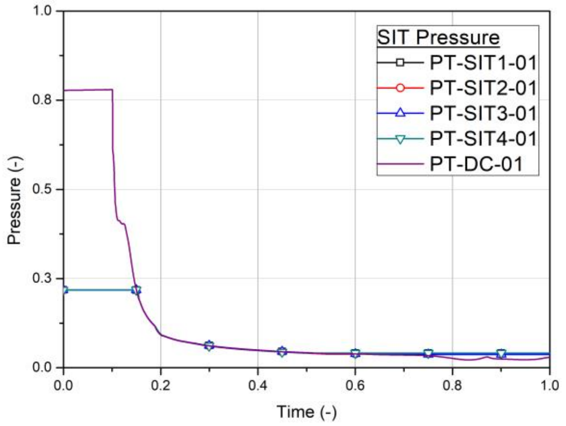

Fig. 7. Pressure behavior of SITs

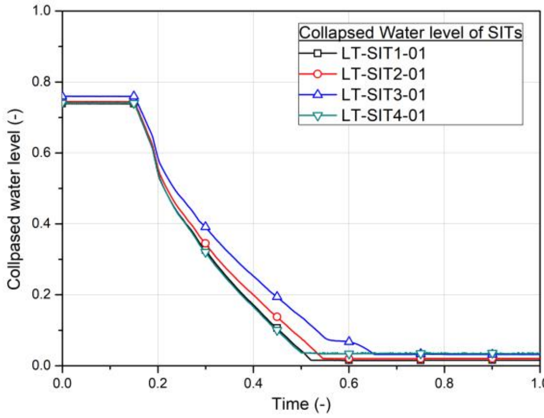

Fig. 8. Collapsed water levels in the SITs

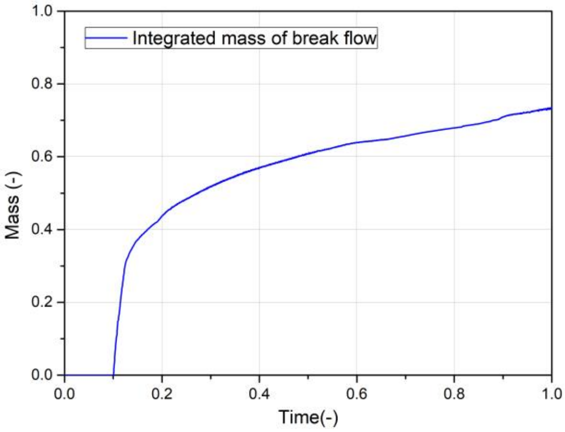

Fig. 9. Integrated mass of the break flow

5. Conclusions

An IBLOCA with the total failure of SIPs was successfully simulated using the ATLAS facility, and the major thermal hydraulic phenomena that can typically occur in this kind of scenario were observed in the IB-SIP-02 test.

During the transient of the IB-SIP-02 test, major thermal-hydraulic parameters such as the system pressures, the collapsed water levels, and the heater rod surface temperature behavior were measured and analyzed. The major findings of the present test are summarized as follows:

- The loop seal clearing phenomenon occurred at the early stage of the transient. However, the effect of the recovery of the collapsed water level in the core by the LSC was negligible due to the significant inventory loss through the break.
- Due to the drastic decrease of the primary system pressure by the large break size, SITs were actuated in short time from the first occurrence of the heater rod surface temperature excursion. The coolant injected from SITs contributed to preventing the core damage in the early period of the transient.
- In this test scenario, the AM action was initiated when the maximum heater rod surface temperature exceeded 500 ºC. That is, the operator can have the time to take the AM action for more than 30 minutes in this kind of multiple accident situation.
- Repairing one train of SIPs was assumed to be the AM action so the coolant from one train of SIPs was injected to the RCS. It quenched the core heaters and the reactor coolant system became stabilized. It might be fail to prevent the core damage without additional make-up of the coolant. Thus, we can conclude that the operation of only one train of SIP is enough to cool down the system if it works on the proper time.

From the present test results, we could identify the accident mitigation strategy for preventing the core damage, such as its measure and the response time to take the accident management action, in this kind of the multiple failure accident. These integral effect test data can also be used to evaluate the prediction capability of the safety analysis codes.

ACKNOWLEDGMENTS

The authors would like to acknowledge to the Ministry of Science and ICT of Korea for their financial support of this project (NRF-2017M2A8A4015028).

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