Experimental Study on the Counterpart Test of LSTF 1% SBLOCA at Reactor Pressure Vessel Top with Accident Management Action

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1. Introduction

After the Fukushima accident, interest on the safety assessment of the existing nuclear power plant increased. And such interest is expanded to further demand for experiments which can give information related to the transient state of nuclear power plant. Thus, importance of the integral effect test (IET) result performed at the ATLAS facility, which is the third largest IET facility in the world, is increasing, not only within Korea but also in the nuclear R&D programs such as the OECD/NEA international joint project. In order to resolve key thermal-hydraulic safety issues related to multiple high risk failures highlighted from the Fukushima accident, various accident scenarios were considered as test items of the second phase of OECD/NEA ATLAS international joint project (hereafter, OECD-ATLAS2 project).

As one of the test items performed under the OECD-ATLAS2 project, the B5.1 test was defined as a counterpart test with respect to the LSTF SB-PV-07 test. The target scenario for the SB-PV-07 test was a 1% small-break loss-of-coolant accident (SBLOCA) at reactor pressure vessel (RPV) top in a pressurized water reactor (PWR) under assumption of total failure of high pressure injection (HPI) system. In this test, an accumulator (ACC) tank injection was assumed to be available and the two kinds of accident management (AM) actions, manual injection of HPI as the first AM action and the 2nd system depressurization as the second AM action, were utilized [1].

The objective of this test is to investigate the general thermal hydraulic phenomena during an SBLOCA with accident management actions. As a counterpart test of the LSTF SB-PV-07 test, the test result was analyzed to identify scaling characteristics of ATLAS test facility compared with other IETs.

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 was developed by the Korean nuclear industry.

ATLAS 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 the 2(hot legs) x 4(cold legs) reactor coolant system with a direct vessel injection (DVI) of emergency core cooling (ECC). ATLAS can simulate full pressure and temperature conditions of APR1400. The total inventory of the primary system is 1.6381 m^3.he secondary system of ATLAS is simplified to be a circulating loop-type. The steam generated at two steam generators (SGs) is condensed in a direct condenser tank, and the condensed feedwater is re-circulated to the SGs.

A detailed design and description of the ATLAS facility can be found in the literature [3].

2.2 Scaling analysis between ATLAS and LSTF

To determine the test conditions of B5.1, a scaling analysis between LSTF and ATLAS was performed.

First, the various design parameters were compared between LSTF and ATLAS from the perspectives of volume, area, and length. Among them, two parameters, i.e., the effective heating length of core and the primary system inventory, were selected as major scaling parameters. The scaling ratios of length (l_{oR}) and volume ($l_{oR}d_{oR}^2$) were determined as 0.52 and 0.20, respectively. The scaling ratio of diameter (d_{oR}), 0.62, was obtained based on the length (l_{oR}) and volume ($l_{oR}d_{oR}^2$) scaling ratios. By utilizing these scaling parameter ratios, the ratios of other global scaling parameters such as time, flow rate, and power were determined. The detailed procedure of scaling strategy can be found in the reference [4].

The initial and boundary conditions of the B5.1 test were obtained by analyzing the LSTF SB-PV-07 test result and applying the scaling result on it.

2.3 Break Simulation System

In the LSTF SB-PV-07 test, the break was simulated by using a sharp-edge orifice mounted at the downstream of a horizontal pipe that was connected to an upper head nozzle. The inner diameter of orifice was 10.1 mm. To simulate the break in the ATLAS facility as similar as possible with the SB-PV-07 test, the break simulation unit was installed. The nozzle design of ATLAS at the top of RPV is shown in Fig. 1. The break simulation unit starts from the RPV upper head nozzle and is connected to a refueling water tank (RWT) to measure the integrated mass of the break flow. The break valve was installed on the horizontal part, which is higher level than the top of the RPV, of the break simulation unit. An orifice was installed at the end of the RPV upper head nozzle. The orifice has also a sharp-edged shape with an inner diameter of 5.41 mm. The inner diameter of the orifice was determined from the scaling analysis result.

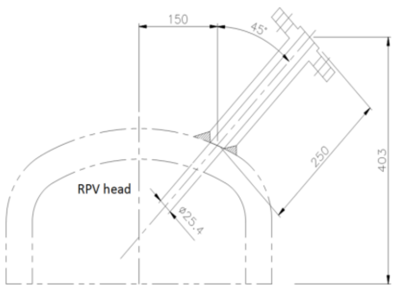

Fig. 1. Schematic diagram of the break nozzle

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 and the reactor scram signal was actuated when the primary system pressure reached the specified pressure. With the generation of the reactor scram signal, the turbine stop signal was actuated. The main feedwater pumps stopped and a main feedwater isolation signal (MFIS) was generated to close the main feedwater isolation valves (MFIVs). Main steam isolation valves (MSIVs) were also closed.

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 of the LSTF SB-PV-07 test.

By the break at the top of RPV, the primary system inventory decreased and the surface temperature of core heater rod started to increase. As the first AM action, when the maximum core exit temperature reached a setting value, the manual injection of the safety water from HPI system into cold-legs in both loops was started. Due to the operation of HPI system, the primary system pressure decreased continuously. When the primary system pressure decreased to a specified pressure, ACCs were injected to all four cold-legs.

The second AM action was initiated when the primary system pressure reached a specified pressure after ACC injection. The second AM action was taken by the secondary side depressurization by opening atmospheric dump valves (ADVs) of SG-1 and SG-2, simultaneously. With initiation of the second AM action, the auxiliary feedwater also started to supply to both SGs at the scaled flow rate of the LSTF SB-PV-07 test.

By initiation of the second AM action, the system was cooled down stably and the test was terminated by the operator's decision. The sequence of events of B5.1-S1 test is listed in Table Ⅰ.

Table I: Sequence of event

No	Description	Remark (set point)	SB-PV-07	B5.1
1	Break valve open	Manual open	0.0000	0.0000
2	Reactor scram	Primary system pressure	0.0058	0.0068
3	Initiation of core power decay	Generation of scram signal	0.0085	0.0108
4	Turbine trip	Generation of scram signal	0.0060	0.0068
5	MFIV Close	Generation of scram signal	0.0063	0.0068
6	MSIV Close	Generation of scram signal	0.0082	0.0068
7	Manual injection of HPI	Maximum core exit temperature	0.2317	0.6285
8	Initiation of ACC	Primary system pressure	0.3817	0.6703
9	SG-1/2 2nd side depressurization	Primary system pressure	0.5068	0.6733
10	Aux. injection into SG1/2	Primary system pressure	0.5112	0.6733
11	Test ends	Operator's decision	0.8693	1.0328

4. Test Result

Considering the confidential problem of test data provided under the OECD/NEA international joint project, all of the test results in this paper were normalized by an arbitrary value including the time frame.

Fig. 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. With the scram signal, SGs 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 valve (MSSV). A periodic discharge of the secondary inventory through the MSSVs induced the pressure fluctuation of the secondary system, however, it had no significant effect on the primary system behavior.

When the first accident management action, manual injection of HPI, was initiated, the decreasing rate of the primary system pressure increased slightly but the first AM action did not have a significant effect on the depressurization of the primary system, in the B5.1 test. However, in the LSTF SB-PV-07 test, the first management action was actuated earlier, at 0.2317 of non-dimensional time, than the B5.1 test so the tendency of the primary system pressure decrease occurred earlier in the LSFT SB-PV-07 test than that of the B5.1 test. With the ACCs injection, the primary system pressure started to decrease rapidly. It made the depressurization of the primary system under the set point of the second AM action actuation, SG secondary side depressurization by opening ADVs. With starting of the second AM action, the secondary system pressures decreased rapidly.

With the second AM action initiation, the primary system pressure decreased significantly and the system was stabilized finally. When the coolant injection from ACCs was terminated as the collapsed water level in the ACC reached the top of the stand pipe, the depressurization rate of the primary system decreased.

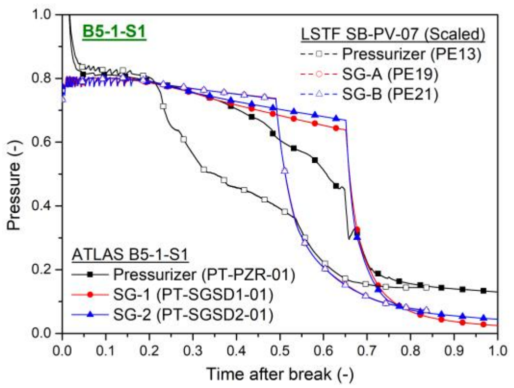

Fig.2. Variation of the system pressure

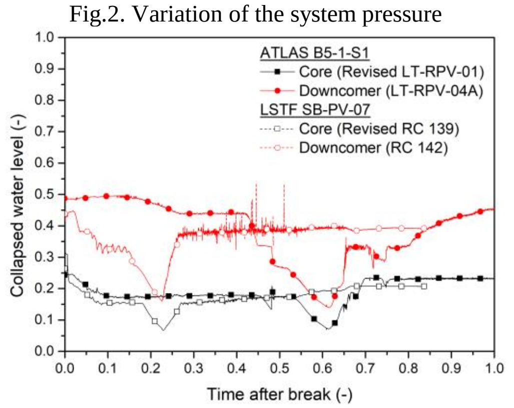

Fig. 3. Collapsed water levels in the RPV

Comparison of the collapsed water levels from the bottom of the active core of the ALTAS B5.1 test and the LSTF SB-PV-07 test were shown in Fig. 3. Due to the break flow, the collapsed water level of the reactor core decreased rapidly during the early stage of the transient. The water level decreased continuously near the top of the active core. However, loop seal clearing occurred at 0.5483 of non-dimensional time in the loop 1A and 2A in the ATLAS B5.1 test, and the collapsed water level of the core increased near the top of the active core again.

In the SB-PV-07 test, the collapsed water levels in the core and the down-comer decreased from the early transient period. The decreased inventory might be transferred to the upper head region through the control rod guide tube and contributed to forming the collapsed water level in the upper head region. It brought about the earlier behavior of the excursion of the fuel rod surface temperature and the first AM action. So the decrease and recovery of the collapsed water level in the RPV was processed faster in the LSTF SB-PV-07 test than that of the B5.1 test/

The collapsed water level of SGs after initiation of the second AM action is shown in Fig. 4. With opening of ADVs as the second AM action, the collapsed water levels decreased rapidly and they kept nearly constant after supply of auxiliary feedwater. After some time had passed, however, the collapsed water levels of both SGs were recovered. The difference between two steam generators could be resulted from the different auxiliary feedwater flowrate and different heat transfer characteristics through SG u-tubes.

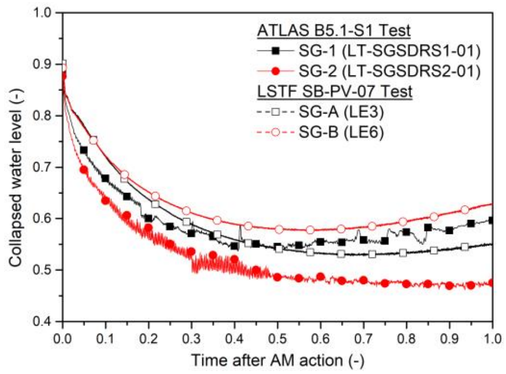

Fig. 4. Collapsed water level of SGs

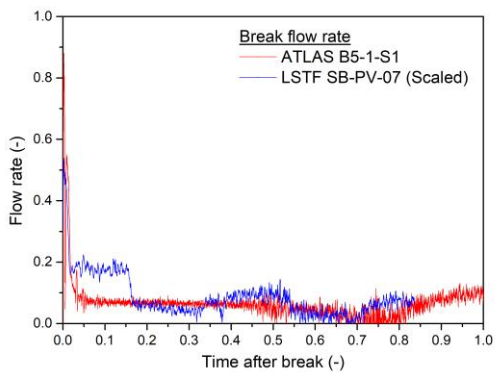

Fig. 5. Break flow rate

Fig. 5 shows the break flow rate of both tests. In the B5.1 test, the integrated mass of break flow was measured using a load cell installed beneath RWT (0.17% uncertainty) and the break flow rate was derived as the differential of the time-integrated break flow that was evaluated from the mass increase of the RWT. In the SB-PV-07 test, the break flow rate was also derived as the differential of the time-integrated break flow evaluated from the liquid level increase in the storage tank (3.35% uncertainty). After the high peak flow rate at the initiation of the transient, the break flow rate varied according to the tendency of the primary system pressure. After 0.8 non-dimensional time after break in the B5.1 test, the break flow rate increased without any significant increase of the primary system pressure. This can be attributed to the recovery of the collapse water level in the RPV upper head region.

In this kind of accident scenario, the liquid level in the upper head is an important parameter since it can affect the amount of break flow. In the LSTF facility, there was an upper core support plate in the upper head region and it was kept the certain water level during the transient. However, there was no inner structure in the ATLAS upper head and the upper head region of ATLAS was empty from the early transient stage, as shown in Fig. 6. Thus the overall break flow rate in the B5.1 test was smaller than the scaled break flow rate of the SB-PV-07 test. It caused the late progress of accident scenario in the B5.1 test compared with the SB-PV-07 test.

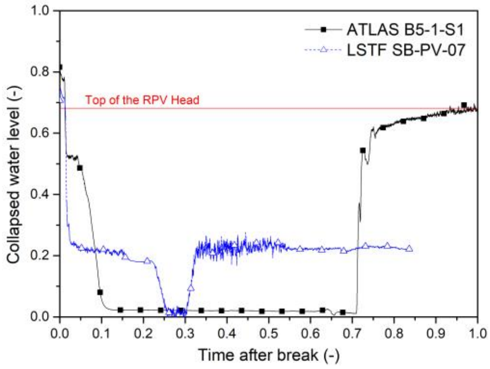

Fig. 6 Collapsed water level in the upper head region

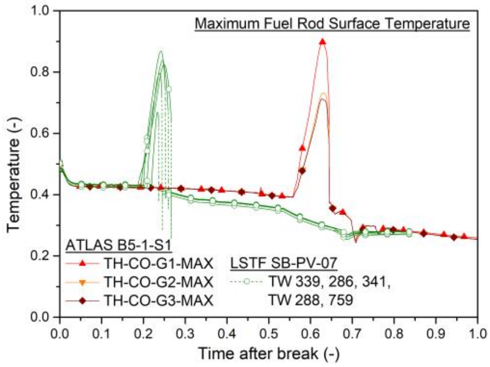

Fig. 8. Maximum fuel rod surface temperature

During the transient period of B5.1 test, the excursion of the heater rod surface temperature occurred as shown in Fig. 7. After loop seals were reformed, the collapsed water level in the core decreased rapidly under the top of the active core due to the continuous loss of the primary inventory through the break. Thus the heater rod surface temperatures started to increase. When the core exit temperature (CET) reached a set value of the first AM action, HPI system was actuated. The coolant injected from HPI system contributed to the quenching of the core heaters after some time delay to be required for delivering the coolant from an injected cold leg to the RPV core.

4. Conclusions

During the transient of B5.1 test, major thermal-hydraulic parameters such as the system pressures, the collapsed water levels, the flows in the primary loops, and the fluid temperatures were measured and analyzed. The major findings of the B5.1 test are summarized as follows:

- An SBLOCA at RPV top was successfully simulated using the ATLAS facility, and the major thermal hydraulic phenomena that can typically occur in this kind of scenario were observed.
- The overall sequence of transient scenario progressed later in the ATLAS B5.1 test than that of the LSTF SB-PV-07 test. This is mainly due to the different break flow rate between two tests. ATLAS and LSTF have different inner geometry of the RPV upper head and it can have a significant effect on the RCS inventory distribution, especially during the early transient period.
- Loop seal clearing phenomenon, which did not occur in the SB-PV-07 test, occurred in the B5.1 test clearly. This can be attributed to the different design of intermediated-leg, inner structure of upper head region, and the location of the active core between two facilities which resulted from the different design of prototype nuclear power plant for each facility. They can affect the pressure difference between the upper-head and down-comer region of the RPV.

From the B5.1 test results, we could identify the difference of scaling characteristics between two integral effect test facilities of ATLAS and LSTF. These integral effect test data can be used to evaluate the prediction capability of the safety analysis codes.

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