Integral Effect Test on 17% Cold Leg IBLOCA for Investigation of Scaling Effect

Byoung-Uhn Bae*, Kyoung-Ho Kang, Yu-Sun Park, Jong-Rok Kim, Hyung-Gil Jun, Nam-Hyun Choi, Ki-Yong Choi
Korea Atomic Energy Research Institute, 989 - 111, Daedeok-daero, Yuseong-gu, Daejeon
*Corresponding author : *bubae@kaeri.re.kr*

1. Introduction

Consideration of the IBLOCA (Intermediate-Break Loss-of-Coolant Accident) in the design of a nuclear power plant can reduce unnecessary burden of the safety-related issues, since acceptance criteria of the emergency core cooling system (ECCS) is currently based on large break LOCA (LBLOCA) which induces the highest peak cladding temperature (PCT).[1] Also, as nuclear power plants are being aged and up-rated, the IBLOCA can become a concern to improve the operating efficiency. Currently, the Unites States Nuclear Regulatory Commision (USNRC) has chosen an IBLOCA as a design-basis accident (DBA) and the safety regulation of France replaced the LBLOCA by the realistic IBLOCA as the DBA.

To validate the safety analysis code and address safety issues for the IBLOCA, the database for the IBLOCA test in an integral effect test (IET) facility is essential. In particular, an experimental data from the counterpart test can resolve the scaling issues in the scaled-down test facilities. The objective of this study is to investigate thermal hydraulic phenomena during the IBLOCA by the counterpart test for the 17% cold leg break in ATLAS (Advanced Thermal-hydraulic Test Loop for Accident Simulation) facility, which was the A4.1 test in the OECD-ATLAS project. This integral effect test would contribute to evaluate the scaling effect of the ATLAS design and test condition.

2. Test Condition

2.1 ATLAS Facility

After completing an extensive series of commissioning tests in 2006, KAERI (Korea Atomic Energy Research Institute) started the operation of ATLAS and completed a series of integral effect tests of the reflood phase of a LBLOCA and small break LOCA (SBLOCA) scenarios that included a direct vessel injection (DVI) line break and a cold leg break [1]. The reference plant of ATLAS is the APR1400, which has a rated thermal power of 4000 MW and a loop arrangement of 2 hot legs and 4 cold legs for the reactor coolant system (RCS). ATLAS is a half-height and 1/288-volume scaled test facility with respect to the APR1400. ATLAS was designed according to the three-level scaling method suggested by Ishii and Kataoka [3,4] to simulate various scenarios as realistically as possible. Figure 1 shows a schematic diagram of a loop connection of ATLAS.

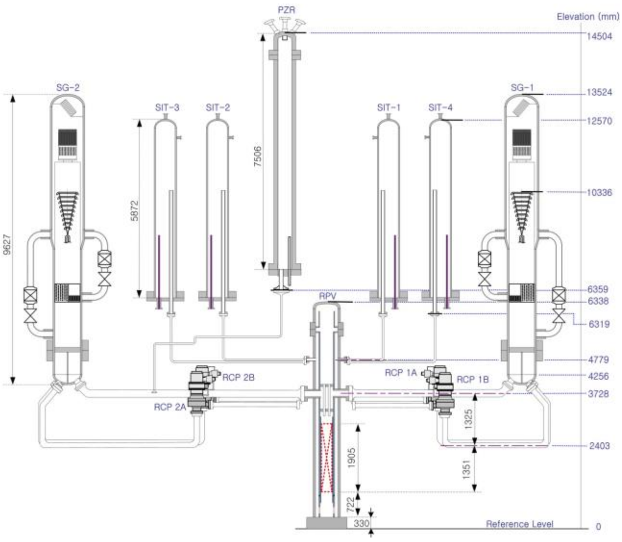

Fig. 1 Schematic diagram of loop connection of ATLAS

2.2 Counterpart test condition

The A4.1 test is a counterpart test for the IBLOCA transient with 17% break of a cold leg, which refers the test performed in the LSTF (Large Scale Test Facility). The LSTF simulates a Westinghouse-type four-loop PWR by a full-height two-loop system in 1/48 volume scale [5]. The reference test conditions of the LSTF (Test IB-CL-03) are summarized as follows [6].

1) Break size (flow area) is 17% cold leg break to simulate a double-ended Guillotine break (DEGB) of ECCS nozzle.
2) An upward long break nozzle is located on the top of cold leg in loop without pressurizer (PZR) for better simulation of break flow through broken ECCS piping
3) Loss of off-site power concurrent with the scram of reactor
4) High pressure injection (HPI), accumulator (ACC) and low pressure injection (LPI) systems in the loop with PZR (loop-A) only, as well as single-failure of diesel generators related to flow rates of both HPI and LPI systems
5) Non-condensable gas in ACC tank may flow into cold leg.
6) Total failure of auxiliary feedwater
7) Following thresholds due to the maximum fuel rod surface temperature for the LSTF core protection and power controlling system :

958K=70%, 961K=35%, 966K=13%, 977K=5%, 1003K=0%, of pre-determined value.

3. Test Result

According to the agreement of the OECD-ATLAS project, the test data should be confidential until the year of 2020. So that, all of the test results in this paper including the time frame were divided by an arbitrary value and plotted on the non-dimensional axis.

Figures 2 and 3 show the thermal power in the core and pressure behavior of the primary and secondary systems in the A4.1 test. After the initiation of the break, a rapid depressurization of the primary system induced the core trip and the decay heat curve in the reactor core was properly simulated in the ATLAS test when compared to the LSTF test, until the core power in the LSTF test was abruptly decreased by the power protection logic. There was no activation of the core power protection logic according to the maximum cladding temperature in the ATLAS test. The primary system pressure presented a plateau, then loop seal clearance made it decreased again. As revealed in the results, overall transient of the pressure in the RCS was appropriately preserved in the ATLAS counterpart test, while the LSTF test did not show a plateau of the primary system pressure due to a slow occurrence of the loop seal clearance. Difference in the primary system pressure during the later period between two tests was relevant to the different transient of the core power.

Collapsed water level in the core region was compared in Fig. 4. The coolant in the reactor pressure vessel was depleted rapidly after the break. The core water level was recovered at the moment of the loop seal clearance and the ACC injection, after which it was maintained around a center of the active core. In the LSTF test, the difference in timing of the loop seal clearance made the core water level recovered earlier and higher than the ATLAS test result. After the loop seal clearance, the recovered core water level was decreased until the ACC actuation. The collapsed levels in the LSTF test were maintained higher than those of the ATLAS counterpart test during the later period of the transient due to the reduced core power and steam generation.

Figure 5 shows the maximum cladding temperature measured in the ATLAS test. The heaters of Group 1 had a larger heat flux to scale the maximum heat flux in the LSTF. It made the maximum temperature on the Group 1 heaters much higher than those of Group 2 or 3. Injection of the ECC water from the ACC contributed to quench the core effectively and decrease of the cladding temperature. Due to difference of the core water level and the uncovered position in the active core region between two tests as shown in Fig. 4, a higher value of the maximum cladding temperature was observed in the LSTF test.

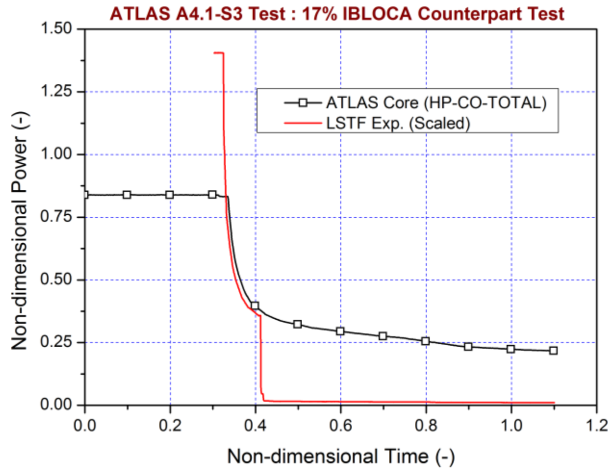

Fig. 2 Core power in the A4.1 test

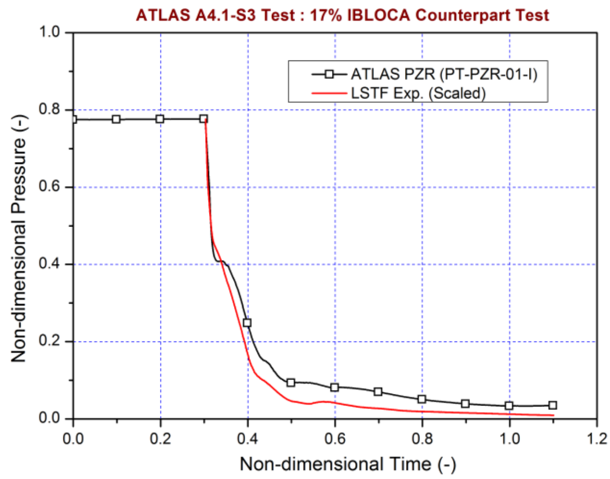

Fig. 3 Primary system pressure in the A4.1 test

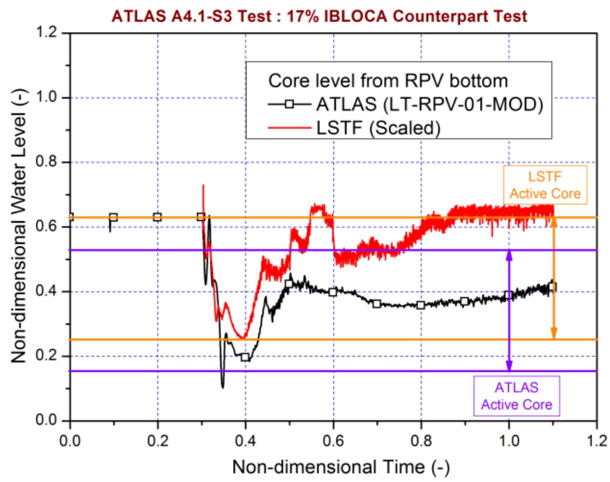

Fig. 4 Core water level in A4.1 test

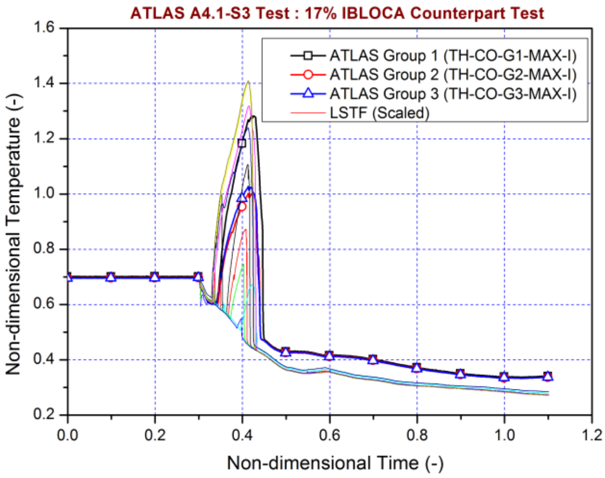

Fig. 5 Maximum heater surface temperature in A4.1 test

Figure 6 shows the liquid flow rates measured in the cold legs. The broken cold leg (1A) showed a negative flow after the initiation of the break, while the flow rates in intact cold legs increased due to a larger pressure difference induced by the break. After the loop seal clearance, the liquid flow rate in the loop was reduced to zero and then injection of the ACC and the LPI made an increase of the flow rate in intact cold legs. As interpreted from the loop seal behavior, the flow rate in the cold leg 1B did not show a significant value due to the partial loop seal blockage during the later period of the transient.

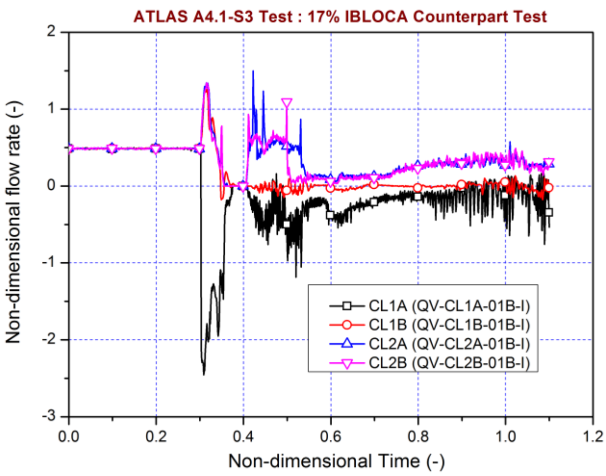

Fig. 6 Cold leg flow rate in A4.1 test

4. Conclusions

The A4.1 test of the OECD-ATLAS project was performed to simulate a cold leg IBLOCA as a counterpart test with respect to the LSTF IBLOCA test with 17 % cold leg break. The main purposes of this test were to investigate thermal hydraulic transient during a cold leg IBLOCA and to evaluate the scaling characteristics of the ATLAS design. In the A4.1 test, a single failure of the HPI/LPI and a total failure of the auxiliary feedwater to the secondary system were assumed.

The LSTF test data were scaled down according to the scaling methodology and directly compared to the ATLAS test result. It showed that overall sequence of major events and thermal hydraulic phenomena including transient behavior of the system pressure, temperature and the break flow were reasonably reproduced in the ATLAS test. A lower maximum cladding temperature was observed in the ATLAS test when compared to the LSTF test, due to difference of the loop seal clearing characteristics, the core water level and the uncovered position in the active core region.

This experimental data of the A4.1 test can be used to evaluate the prediction capability of existing safety analysis codes and identify any code deficiency in predicting the IBLOCA transient. Also, more detailed investigation and analysis with comparing to the LSTF test data will contribute to enhance the understanding on scaling issues.

REFERENCES

[1] S. M. Bajorek, “Realistic Small- and Intermediate-Break Loss-of-Coolant Accident Analysis Using WCOBRA/TRAC”, Nuclear Technology, 136, pp. 50-62, 2001.
[2] K.H. Kang et al., “Revised Description Report of ATLAS Facility and Instrumentation,” KAERI/TR-5974/2015, Korea Atomic Energy Research Institute, 2015.
[3] K. Y. Choi et al., “Scaling Analysis Report of the ATLAS Facility,” KAERI/TR-5465/2014, Korea Atomic Energy Research Institute, 2014.
[4] M. Ishii and I. Kataoka, “Similarity Analysis and Scaling Criteria for LWRs Under Single Phase and Two-Phase Natural Circulation,” NUREG/CR-3267, ANL-83-32, Argonne National Laboratory, 1983.
[5] Mitsuhiro SUZUKI et al., “ROSA-V Large Scale Test Facility (LSTF) System Description for the Third and Fourth Simulated Assemblies,” JAERI-Tech 2003-037, Japan Atomic Energy Research Institute, 2003.
[6] Japan Atomic Energy Agency, Final Data Report of ROSA-2/LSTF Test 2 (Cold Leg Intermediate Break LOCA IB-CL-03 in JAEA), 2013.