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Major Outcomes of the Second Phase of OECD-ATLAS International Joint Project

Kyoung-Ho Kang [a*], Jongrok Kim [a], Byoung Uhn Bae [a], Jae Bong Lee [a], Yusun Park [a],
Seok Cho [a], Nam Hyun Choi [a]

[a] *Korea Atomic Energy Research Institute, 111, Daedeok-Daero 989 Beon-Gil, Yuseong-Gu, Daejeon, 34057, Korea*

[*] *Corresponding author: khkang@kaeri.re.kr*

1. Introduction

Within the context of the OECD/NEA ATLAS project from April 2014 to March 2017 [1], a series of tests were performed to resolve key thermal-hydraulic safety issues related to multiple high risk failures highlighted from the Fukushima accident, by utilizing a thermal-hydraulic integral effect test (IET) facility of ATLAS (Advanced Thermal-Hydraulic Test Loop for Accident Simulation) [2] as shown in Fig. 1. Notwithstanding the distinguished achievement of the OECD/NEA ATLAS project, a general consensus between the project partners was reached to continue the second phase of project with an aim of enhancing the nuclear safety analysis technology and improving the best guidelines for accident management. In particular, the OECD/NEA ATLAS phase 2 project (hereafter, OECD-ATLAS2 project) focused on the validation of simulation models and methods for complex phenomena of high safety relevance to thermal-hydraulic transients in design basis accident (DBA) and beyond-DBA (BDBA).

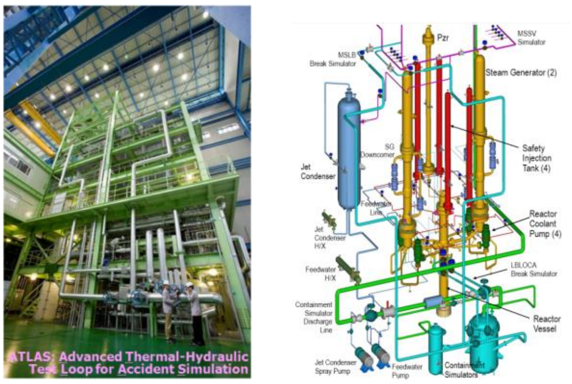

Fig.1. Photograph and schematic diagram of ATLAS

Under the framework of the OECD-ATLAS2 project, based on the project agreement, a total of 8 integral effect tests in 5 different topics were performed with the ATLAS facility. The OECD-ATLAS2 project started in October 2017 with a three-year period. Due to the worldwide crisis resulting from the COVID-19, however, the project was determined to be extended to the end of 2020. 18 organizations from 11 countries participated in the project as follows: Belgium (Bel V, ENGIE), China (SPICRI, CNPRI, NPIC), Czech (UJV), France (CEA, EDF), Germany (GRS), Japan (JAEA), Spain (CSN), Switzerland (PSI), UAE (FANR), USA (USNRC), and Korea (KAERI, KINS, KHNP-CRI, KEPCO E&C). Japan joined the OECD-ATLAS2 project as an in-kind contributor who provided the experimental data for the counterpart test against the large scale test facility (LSTF). The operating agency (OA) established a national consortium together with Korean nuclear players and contributed to this project by performing pre- and post-test calculations.

2. Summary of Test Results

The key outline of the tests performed in the OECD-ATLAS2 project is as follows:

Table I: Test matrix of OECD-ATLAS2 project

Topics	Number of tests	Remarks
B1-SBLOCA		
- SBLOCA w/o SIP under PAFS operation	1	Resolving the safety issues
B2-Passive Core Makeup		
- SBO with Hybrid SIT	1	
- SBLOCA with PECCS	1	Condensation model, w and w/o nitrogen
B3-IBLOCA		
- PZR Surgeline Break	1	Effect of break position and ECC injection
- DVI Line Break	1	Cliff Edge Effect
B4-Design Extension Conditions		
- SLB with SGTR	1	Long-term core PCT behavior during multiple failure accident
- Shutdown Coolability w/o RHRS	1	Effect of reflux condensation, accident sequence modeling
B5-Counterpart Test		
- Counterpart Test of LSTF SB-PV-07 (1% RPV top break SBLOCA)	1	Addressing the scaling issue
Total	8	

2.1 Test B1 series

An SBLOCA at a cold leg is one of the most important DBA. In order to mitigate the consequences of an SBLOCA transient, pertinent safety systems should be utilized. After the Fukushima accident, various passive safety systems have been proposed to improve the safety and reliability of an ultimate heat removal system without any operator action during DBA and BDBA transients. A PAFS is one of the advanced safety features which is intended to replace a conventional active auxiliary feedwater system. The driving force for PAFS is a natural convection mechanism; i.e., condensing steam in nearly-horizontal U-tubes submerged inside a large water pool. The experimental data with a single train of PAFS can be used to validate a prediction capability of safety analysis codes for the natural circulation phenomenon and asymmetric cooling effect. One test was performed to simulate an SBLOCA at a cold leg by utilizing a single train of PAFS in the framework of OECD-ATLAS2 project.

In the B1.1 test, a 2 inch cold leg SBLOCA was simulated with total failure of safety injection pump under an operation of PAFS. A single train of PAFS was connected to a steam generator number 2 (SG-2) of ATLAS. When the collapsed water level of the secondary side in SG-2 reached 25 % in a wide range scale, PAFS started to operate. In the B1.1 test, an accident management action was taken by the secondary side depressurization of steam generator number 1 (SG-1). When the maximum heater rod surface temperature in the core reached 450 ^{o}C, the accident management action was initiated by fully opening an atmospheric dumping valve (ADV) of SG-1. After PAFS actuation, the secondary side water levels of steam generator maintained stable values. After accident management action, however, the secondary side water level and the secondary system pressure of SG-1 sharply decreased because the steam was vented out. Flow rates of loop-2 were larger than those of loop-1 which could be attributed to the asymmetric cooling by a single train of PAFS operation. The B1.1 test result showed that the reactor core was quenched after an operation of PAFS and accident management action.

2.2 Test B2 series

After the Fukushima accident, demand for safety enhancement in nuclear power plants (NPPs) has increased. The accident showed that for the prevention of the core meltdown, the core makeup at high pressure of reactor coolant system (RCS) is crucial, and that even in an SBO situation, the core makeup water must be supplied efficiently. The concept of a hybrid safety injection tank (H-SIT) is a passive safety injection system that allows high-pressure core makeup over the operating pressure of a light water reactor (LWR). The H-SIT can be pressurized equivalently to the RCS through a pipe connection between the H-SIT and the pressurizer, along with nitrogen charging, in which case the coolant can be injected by gravitational head between the RCS and the H-SIT. As a similar concept to the H-SIT, the passive emergency core cooling system (PECCS) can be pressurized equivalently to the RCS through a pipe connection between the safety injection tank (SIT) and a cold leg. It is worth investigating the thermal-hydraulic phenomena anticipated in these passive core makeup systems to produce clear knowledge of the actual phenomena and to provide the best guideline for accident management. Two tests were performed on the topic of the passive core makeup.

The target scenario for the B2.1 test was a prolonged SBO with operation of the H-SIT as a passive core makeup system. Pressure balance lines connecting the pressurizer to the H-SITs were in a keep-open status throughout the test period. In the B2.1 test, typical events of an SBO scenario were well reproduced. The secondary side of steam generators became empty, resulting from the inventory discharge through the cyclic opening and closing of the main steam safety valves (MSSVs) during the initial period of the transient. After the secondary side of the steam generators became dried out, the primary system pressure started to increase due to a degradation of the heat removal capacity of the steam generators. Periodic discharge of the primary inventory through a pilot operated safety relief valve (POSRV) of the pressurizer was observed. The hybrid safety injection tank number 1 (H-SIT-1) and hybrid safety injection tank number 2 (H-SIT-2) were activated to inject the coolant through the DVI nozzles with the first opening of a POSRV. The hybrid safety injection tank number 3 (H-SIT-3) and hybrid safety injection tank number 4 (H-SIT-4) were set to open when the maximum heater rod surface temperature in the core increased higher than 450 ^{o}C. The core was effectively cooled by the safety injection flow from the H-SITs. The B2.1 test result shows that the H-SITs had an effective core cooling performance as a passive safety feature.

In the B2.2 test, a cold leg SBLOCA with an operation of a PECCS was simulated. The PECCS has two major functions to mitigate a DBA situation. The first one is automatic depressurization of the primary system pressure through automatic depressurization valve (ADV) on pressurizer. The second function is passive safety injection using 2 high-pressure safety injection tanks (HP-SITs) and 2 safety injection tanks (SITs). With the start of a break, the primary system pressure decreased to the low pressurizer pressure (LPP) set-point, 10.7 MPa. The low pressurizer pressure signal isolated the secondary system. After the isolation, the secondary side water inventory continuously decreased to depletion with the open-close hysteresis of the MSSVs. High-pressure safety injection tank number 1(HPSIT-1) and high pressure safety injection tank number 3 (HPSIT-3) were activated when the pressurizer pressure decreased below 10.0 MPa. Continuous release of the primary system inventory induced an increase of cladding temperature in the core. ADV number 1 and 2 were actuated when the maximum heater rod surface temperature in the core increased higher than 380 ^{o}C and 410 ^{o}C, respectively. With safety injection from the HP-SITs and depressurization through automatic depressurization valves (ADVs) along with inventory depletion of the secondary side of the steam generators, the primary system pressure abruptly decreased below the activation set point of the SIT, 4.2MPa. The safety injections from HP-SITs were not effectively injected during the early phase of the transient. Only after the opening of the ADV number 1 and number 2, the safety injection water was injected to the RCS which resulted in an effective decrease of the maximum heater rod surface temperature in the core.

2.3 Test B3 series

An IBLOCA has been recognized as one of the important topics in terms of risk-informed regulation. There is a widespread opinion that the frequency of double-ended guillotine break of primary coolant circuit piping such as hot and cold legs of pressurized water reactor (PWR) is quite low. Therefore a rupture of an intermediate-size pipe is becoming relatively more important than ever in risk-informed regulation. Although there are available experimental data for an IBLOCA, it is relatively quite limited. Thus, two tests for IBLOCA transients with a pressurizer surgeline break and a DVI line break were performed.

The target scenario for the B3.1 test is an IBLOCA with a pressurizer surgeline break, which is corresponding to 10-inch break in APR1400. The test was composed of Run 1 and Run 2, according to the available number of SITs, with an aim of investigating asymmetric and multi-dimensional thermal-hydraulic phenomena. The pressurizer surgeline break induced a rapid depressurization of the primary system and a blowdown of the coolant in the reactor pressure vessel (RPV). The safety injection from the SIPs and the SITs supplied sufficient safety injection water to the RCS and any excursion behavior of the cladding temperature was not observed in the core during the whole transient of Run 1 and Run 2 tests. A reverse heat transfer at the U-tube in the steam generators made a superheated condition of the cold leg flow. The coolant in lower down-comer was sufficiently mixed and there was no significant asymmetric behavior of the coolant temperature at the lower plenum in both tests. In the Run 2 test excluding one SIT, a different behavior of the injection flow rate from SIT and the asymmetric temperature distribution were observed in the upper down-comer compared to the Run 1 test.

The B3.2 test was performed to simulate a DVI line break IBLOCA, which is corresponding to 8.5-inch break in APR1400. The DVI line break induced a steam pressure build-up in the core and an excursion behavior of the cladding temperature. The minimum water level in the core was observed at the moment of the loop seal clearance. Since the break nozzle was located at the DVI line, the clearance of an upper down-comer could make an effective flow path of the steam toward the break. The B3.2 test result showed that the reactor core was quenched after the flow path of the steam toward the break provided by the clearance of a loop seal and an upper down-comer. Compared to the B3.1 test result for a pressurizer surgeline break, the B3.2 test indicated that the break location at the DVI line could significantly affect the behavior of the core heat-up. While an excursion of the cladding temperature did not occurred in the B3.1 test even with a larger break area than the DVI line, the simulation of the DVI line break scenario showed a core heat-up until the clearance of a loop seal and an upper down-comer.

2.4 Test B4 series

Design extension conditions (DECs) such as an SBO, a multiple accident of an SLB accompanied by an SGTR, and a shutdown coolability without a residual heat removal system (RHRS), which have not been seriously considered from a viewpoint of DBA, were incorporated in the test matrix of the OECD-ATLAS2 project. Especially, various kinds of multiple failure accidents have attracted world-wide attention as a post-Fukushima action. Two tests were performed in the subject of DECs.

The target scenario for the B4.1 test was a multiple failure accident of an SLB accompanied by a SGTR. The B4.1 test was started by opening two break valves at the main steam line. The secondary system pressure of the SG-1 decreased rapidly and the reactor scram signal was actuated when the secondary system pressure of the SG-1 reached 6.11 MPa. When the wide-range level of SG-1 decreased to 0.1 m, a SGTR was initiated. Due to the inventory loss by the SGTR, the primary system pressure decreased. When the primary system pressure reached 10.72 MPa, the SIP actuation signal was activated. After that, the RCS cooled down with recovery of the collapsed water level of secondary side in SG-1. Finally, the test was terminated with the operator’s decision when the recovered collapsed water level of SG-1 was over 7.0 m. In the B4.1 test, an SLB made the primary system pressure decrease as the excessive heat removal through the break at the steam line. With the SGTR, the primary system inventory moved to the steam generator secondary side. After the injection of the auxiliary feedwater and the safety injection water from SIPs, however, the primary system pressure became stabilized and the collapsed water level in the secondary side of affected steam generator was recovered. From the present test result it can be concluded that the whole system can be successfully cooled-down with the proper operation of safety systems against this kind of multiple failure accident.

The B4.2 test was performed to simulate a loss of RHRS during a mid-loop operation. The main purpose of this test was not only to investigate thermal hydraulic transient in the RCS during a loss of RHRS but also to evaluate the effectiveness of reflux condensation and safety injection from a SIT on shutdown coolability. In the B4.2 test, the pressurizer manway was opened and the initial water level in the primary system was the centerline of the hot leg to simulate a mid-loop operation. The secondary system inventories were empty and 5.0 m for SG-1 and SG-2, respectively. The SITs with variable initial pressure conditions were utilized in the present test. In the B4.2 test, the top part of the core was uncovered and the excursion of the heater rod surface temperature in the core occurred. The safety injection water from the safety injection tank number 1 (SIT-1) was supplied until the internal pressure of the SIT-1 was kept higher than the pressure of RPV down-comer. At the end, the SIPs were actuated when the heater rod surface temperature in the core exceeded 500 ºC since the safety injection water from SITs was not supplied due to a very small pressure difference between the primary system and the SIT-1. The B4.2 test result showed that the existence of secondary system inventory and the location of the pressurizer cause the asymmetric thermal-hydraulic behavior in the RCS, and the safety injection from SIT and SIP can make up the uncovered core with the coolant and cool down the RCS during a mid-loop operation with a loss of RHRS.

2.5 Test B5 series

Even though a lot of integral effect tests have been performed for the past decades by utilizing various large scale facilities, the scaling issue is one of the remaining major safety issues under debate between regulatory authorities and utilities. The scaling inherent in a certain facility needs to be justified before its data are used for a safety analysis. It was agreed at the project review group (PRG) meeting that ATLAS could be utilized to reproduce one of the scenarios of LSTF in order to address the scaling issues.

The B5.1 test was defined as a counterpart test with respect to the LSTF SB-PV-07 test which simulated a 1% SBLOCA at RPV upper head under assumptions of total failure of high pressure injection system and non-condensable gas inflow to the primary system from accumulator tanks. The initial steady-state conditions were achieved at a scaled power based on the core power that was supplied in the SB-PV-07 test. An SBLOCA at RPV upper head was successfully simulated using the ATLAS facility as a counterpart test. When the maximum core exit temperature (CET) reached 623 K, the coolant was manually injected from the high pressure injection system into cold legs in both loops as the first accident management action. The whole core was quenched after the first accident management action, and then the accumulator system was actuated in both loops when the primary system pressure reduced to 4.51 MPa. After the scaled inventory was injected into the RCS from accumulator tanks, the accumulator tanks were not isolated from the RCS so the nitrogen gas could flow into the RCS. When the primary system pressure decreased to 4 MPa, the secondary system depressurization was initiated by opening the atmospheric dump valves (ADVs) in both steam generators as the second accident management action. At the same time with the second accident management action, the auxiliary feedwater injection was actuated. 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, 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 intermediate leg, inner structure of upper head, and the location of the active core between two facilities which resulted from the different design of prototype NPP for each facility. They can affect the pressure difference between the upper-head and the downcomer region of RPV.

3. Conclusions

The second phase of OECD/NEA joint project utilizing an integral effect test facility of ATLAS has been successfully progressed from October 2017 to December 2020. A total of 8 integral effect tests in 5 different topics were carried out and 18 organizations from 11 countries participated in this project. Utilizing the established IET database, simulation models and methods for complex phenomena of high safety relevance to thermal-hydraulic transients in DBA and BDBA were validated. The present OECD-ATLAS2 project aimed at the safety enhancement of operating NPPs by simulating the various accident transients in connection with the safety analysis technology. The thermal-hydraulic behaviors related to a passive core makeup, an IBLOCA, and a multiple failure accident such as an SLB combined with an SGTR, were investigated in a systematic manner.

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