Transactions of the Korean Nuclear Society Autumn Meeting
Changwon, Korea, October 30-31, 2025

Evaluation of the Accident Management Strategies on a Loss of Safety Injection under a Small Break Loss of Coolant Accident (SBLOCA) Condition Based on the ATLAS Test Results

Yusun Park*, Byoung Uhn Bae, Jae Bong Lee, Jun Yeong Jung, Jongrok Kim, Seok Cho, Won Pil Baek, Kyoung Ho Kang

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

**Corresponding author: yusunpark@kaeri.re.kr*

***Keywords :** SBLOCA, IET, Accident management strategy.

1. Introduction

According to the safety and regulatory standards that were strengthened domestically and internationally after the Fukushima accident, interest in multiple failure accidents has increased with the introduction of the concept of design extension conditions. Operator actions are very important factor in accident management strategies for multiple failure accidents. However, there is still a lack of evaluation and validation about the effectiveness of the accident management strategies against the multiple failure accidents in which various operator actions are considered.

Therefore, a risk/performance information analysis was performed on multiple failure accidents by comprehensively considering the effects of enhanced safety standards, regulatory changes, and revisions of emergency operating procedures. As a result, from the view point of the core damage probability, it was found that a loss of safety injection accompanied by a small break loss of coolant accident had the highest risk. In particular, the conditional core damage frequency was evaluated to be 100%. Thus, this multiple failure accident was selected as the target scenario in this study.

For the selected multiple failure accident, two demonstration tests were conducted using a thermal-hydraulic integral effect test facility, ATLAS (Advanced Thermal-Hydraulic Test Loop for Accident Simulation), to evaluate the effectiveness of safety systems and operator actions based on the accident management strategy currently in place. Two tests have the same initial and boundary conditions and have the same scenario except for the initiation time of the first accident management action. Based on the test results, guidelines were proposed for the optimization of the accident management strategy.

2. Integral Effect Test Using ATLAS

ATLAS is a thermal-hydraulic integral effect test facility that can simulate wide range of accident scenarios at prototypic pressure and temperature conditions for APR1400 and other pressurized water reactors. The detailed information of ATALS can be found in the reference [2]. To simulate the OPR1000 condition, scaling analysis between OPR1000 and ATLAS was performed first [3, 4]. Referring to the scaling analysis result, the volume scaling ratio of ATLAS against OPR1000 was determined as 1/206.5.

To simulate an SBLOCA, a break unit was connected on the cold leg vertically. A break valve and a break simulation nozzle were installed on the break unit. The inner diameter of the break nozzle is 4.20 mm that corresponds to 0.45% of cold leg area of OPR1000 nuclear power plant. The discharged inventory from the reactor coolant system through the break was collected in the condensation tank (CDT).

3. Test Scenarios

Referring the risk/performance information evaluation result and the emergency operation procedure of the operating nuclear power plant, the detailed test scenario was determined. From the initiation of the SBLOCA, the reactor trip is induced by the low pressurizer pressure signal. Simultaneously with the reactor trip signal, the core power decreases to follow the decay heat curve and the secondary system is isolated.

Due to the assumption of the loss of safety injection, the high-pressure safety injection is not activated in spite of the safety injection actuation signal. To induce the rapid depressurization of the primary system, the operator opens and operates atmospheric dump valves on two steam generators keeping the target cooling rate, as the first accident management action. From the view point of the Risk/performance evaluation, the operator action to mitigate the accident is granted after 30 minutes from an accident initiation time. Thus, in this study, two cases were considered. In the first scenario (SBLOCA-OPR-01), the first accident management action is initiated after 30 minute and, in the second test (SBLOCA-OPR-02), it was realized within 30 minutes from the high-pressure safety injection failure event.

When the collapsed water level in the steam generator secondary system decreased to the set point value, the auxiliary feedwater is supplied. The first accident management action is initiated at 0.5683 normalized time in the SBLOCA-OPR-01 test and 0.2843 normalized time in the SBLOCA-OPR-02 test, respectively, after safety injection actuation signal is initiated.

To achieve more aggressive system depressurization, opening the safety valve on the pressurizer is planned as the second accident management action. In the SBLOCA-OPR-01 test, the second accident management action is implemented at 0.1000 normalized time after the first accident management action initiated. In the SBLOCA-OPR-02 test, the second accident management action is planned to initiate when the ADVs were opened 95% to keep the target cooling rate.

When the primary system pressure decreases successfully, the four safety injection tanks supply the coolant to the reactor pressure vessel through 4 cold-legs. And the shutdown cooling system operates when the primary system is cooled down to the set point condition for shutdown cooling system operation.

4. Test Results

The actual event sequence that was realized in two tests is listed in the Table I. Considering the confidentiality of test data, all of the test results in this paper were normalized by an arbitrary value including the time frame.

Table I: Sequence of Major Events

#	Event	Normalized time		Remark
		SBLOCA -OPR-01	SBLOCA -OPR-02	
1	Initiation of SBLOCA	0.1000	0.1000	Break valve open
2	Reactor trip	0.1413	0.1397	Reactor trip by LPP signal Decay heat simulation
3	SGs Isolation	0.1440	0.1423	Close of MSIVS, MFIVs, MSCV
4	MSSV operation	0.1437	0.1413	Cyclic operation referring to the secondary system pressure
5	Initiation of SIAS	0.1413	0.1413	Referring to the primary system pressure
6	Fail of the SI operation	-	-	No coolant injection to the system
7	1st AM - ADVs open	0.5683	0.2843	Both SGs, keep the target cooling rate
8	Auxiliary feedwater supply	-	0.6587	
9	2nd AM : pressurizer safety valve open	0.7150	-	Keep the target cooling rate
10	SIT operation	-	-	Referring to the primary system pressure, Injection through 4 cold-legs
11	Excursion of heater rod surface temperature	0.6650	0.7887	Increase of the heater rod surface temperature
12	End of the test	0.7680	0.9207	Core power turned off by core heaters protection control logic

*ADV: Atmospheric Dump Valve
AM: Accident Management
LPP: Low Pressurizer Pressure
MFIV: Main Feedwater Isolation Valve
MSCV: Main Steam Control Valve
MSIV: Main Steam Isolation Valve
MSSV: Main Steam Safety Valve
SG: Steam Generator
SI: Safety Injection
SIAS: Safety Injection Actuation Signal
SIT: Safety Injection Tank

Fig. 1 shows system pressure behavior of two tests during the transient. Right after the break, the system pressure decreased sharply. The MSSVs were actuated, maintaining equilibrium between the primary system pressure and the secondary system pressure of the steam generator, after which the pressure began to drop again. The operator opened the ADVs as the first accident management action, causing an initial rapid decrease in secondary-side pressure. With sustained ADV operation to maintain the prescribed depressurization rate, the system pressure subsequently decreased more gradually. The second operator action was conducted in the SBLOCA-OPR-01 test as opening the pressurizer safety valve at 0.7150 normalized time. But it had little effect on system depressurization.

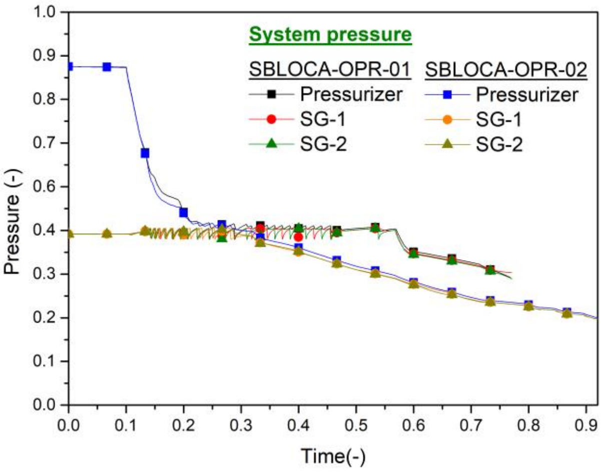

Fig. 1. System pressure behavior

Fig. 2 shows the collapsed water level in the reactor pressure vessel. With the onset of the break, the collapsed water level in the reactor pressure vessel decreased sharply. Loop Seal Clearing (LSC) phenomenon were observed in both tests, which temporarily recover of the core water level. However, continuous coolant discharge through the break caused the core water level to decline further. And after the second accident management action was initiated, the water level in the reactor pressure vessel dropped even more rapidly. In other words, the 2nd AM action which was taken to quickly depressurize the primary system pressure led to rapid depletion of coolant in the RPV before the depressurization effect of the primary system occurred.

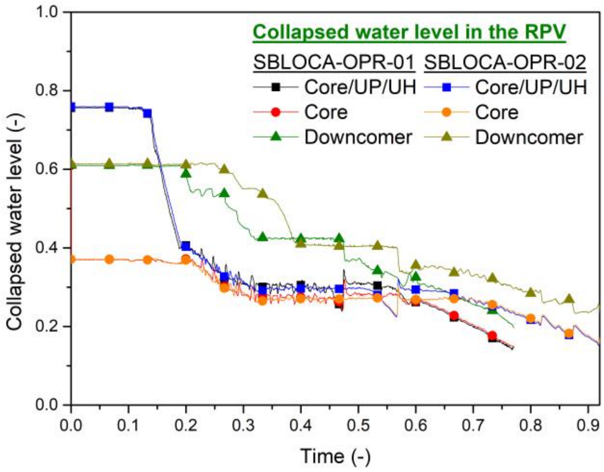

Fig. 2. Collapsed water level in the RPV

Fig. 3 shows the integrated mass of break flow. Break flow behavior showed similar trend between two tests. This can be explained in the same context as the similar behavior of collapsed water level changes in the reactor pressure vessel.

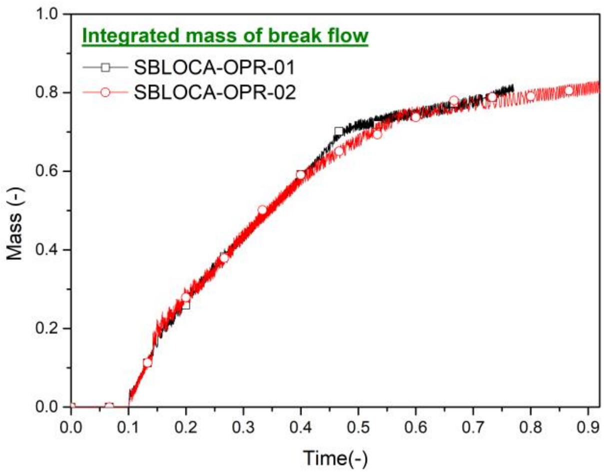

Fig. 3. Integrated mass of the break flow

Once the operator actions were initiated, the collapsed water level in the reactor pressure vessel decreased. As the most part of active core was uncovered, their surface temperature began to rise steeply as shown in Fig. 4. In both tests, the temperature increase progressed faster than the depressurization effect achieved by operator actions. And before SIT injection could occur, the heater surface temperature reached the core protection limit. Thus, the core heaters were turned off by the core protection logic and the tests were terminated.

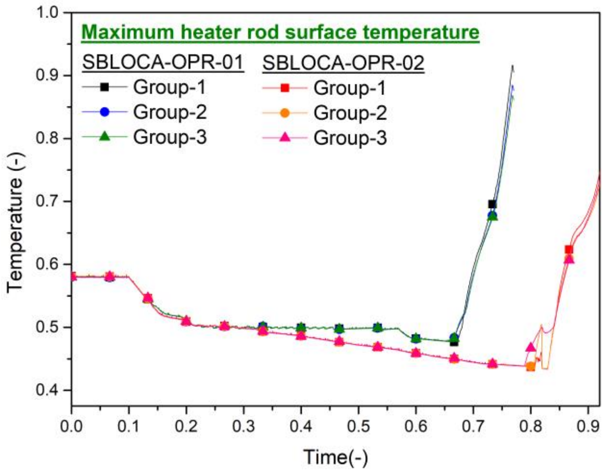

Fig. 4. Heater rod surface temperature

4. Conclusions

After SBLOCA was initiated, the operator attempted to cool down the system through the secondary side of the steam generator as the first accident management action. When the core heater surface temperature began to rise, the operator implemented a more aggressive depressurization measure by opening the pressurizer safety valve, in the SBLOCA-OPR-01 test. However, this action had no significant effect on depressurization of the primary system while it accelerated depletion of the primary coolant inventory, resulting in a rapid increase in core heater surface temperature.

The time from the onset of the break to the excursion of the heater rod surface temperature was approximately 0.5646 normalized time and 0.6900 normalized time, respectively. This time can be considered as the operator's response time to prevent the severe accident. Comparing two test results, if high pressure safety injection fails and cooling is attempted via the secondary system, operator actions must be initiated as early as possible to secure more response time for operators and to secure more coolant inventory in the reactor pressure vessel until the SIT actuation pressure is reached.

Based on these two test results, it was confirmed that accident management actions which were considered in these two tests were insufficient to reduce the system pressure to the SIT actuation setpoint, underscoring the need for earlier initiation of operator interventions. Nevertheless, considering the limited primary coolant inventory, it is recommended to consider using pressurizer spray for system depressurization rather than the opening a safety valve on the pressurizer.

The present test results can be used to develop the accident management strategies optimization methodology with various tests and safety evaluation results of multiple failure accident.

Acknowledgement

This work was supported by the National Research Foundation of Korea (NRF) grant funded by the Korea government (MSIT) (Grant No. RS-2022-00144111).

REFERENCES

[1] "AIMS-PSA Release 2 Reference Manual (Rev.1.2)," Korea Atomic Energy Research Institute (2015).
[2] J. B. Lee et al., "Description Report of ATLAS Facility and Instrumentation (Third Revision)," KAERI/TR-8106/2020, Korea Atomic Energy Research Institute (2020).
[3] K. Y. Choi et al., "Scaling Analysis Report of the ATLAS Facility," KAERI/TR-5465/2014, Korea Atomic Energy Research Institute (2014).
[4] Hyun-Sik Park et al., "Design Calculation Sheet for ATLAS Fluid System," KAERI/TR-3333/2007, Korea Atomic Energy Research Institute (2007).