Integral Effect Test on a Loss of Safety Injection accompanied by a Small Break Loss of Coolant Accident based on the Risk/Performance Information

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1. Introduction

After the Fukushima accident, Korea Hydro & Nuclear Power (KHNP) included the “Emergency Operation Procedure Verification Procedure” in the accident management plan of operating nuclear power plants for design-based accidents (DBAs) and multiple failure accidents. In order to achieve the enhanced nuclear power plant safety, it is essential to strengthen the ability to respond to multiple failure accidents by optimizing accident management strategies from a comprehensive perspective of both deterministic and probabilistic safety analysis.

By referring to the risk/performance information analysis result [1] for multiple failure accidents in operating nuclear power plants, a loss of safety injection under a small break loss of coolant accident (SBLOCA) was selected as the target scenario since it has a significant impact on the core damage frequency. For the selected multiple failure accidents, an integral effect test was conducted using ATLAS (Advanced Thermal-Hydraulic Test Loop for Accident Simulation) test facility. In the test, the results of the risk/performance information analysis was considered for the currently applied accident management strategy.

Based on the present test results, the effectiveness of accident management actions and safety systems was evaluated against the multiple failure accidents and optimization of accident management strategy was suggested.

2. Integral Effect Test Utilizing ATLAS

ATLAS was designed to model a reduced-height primary system of APR1400 (Advanced Power Reactor 1400 MWe) and it can simulate full pressure and temperature conditions of the prototype nuclear power plant. The detailed information of ATALS can be found in the reference [2]. To simulate the OPR1000 (Optimized Power Reactor 1000 MWe) 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 point. Flow restrictors with an inner diameter of 36.87 mm were inserted in both steam generators (SGs) outlet.

The discharged inventory from the reactor coolant system through the break was collected in the condensation tank (CDT) and the integrated mass was measured by the load cell to estimate the break flow rate. And the inventory from the secondary system through the atmospheric dump valves (ADVs) was discharged into the Refueling water storage tank (RWT) and its integrated mass was measured by the load cell.

3. Test Result

Referring the risk/performance information evaluation result and the emergency operation procedure of the operating nuclear power plant, the detailed test scenario was determined. The major event sequence 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
1	Initiation of SBLOCA	0.1000	Break valve open
2	Reactor trip	0.1413	Reactor trip by LPP signal Decay heat simulation
3	SGs Isolation	0.1417 /0.1430 /0.1440	Close of MSIVS, MFIVs, MSCV

4	MSSV operation	0.1437	Cyclic operation referring to the secondary system pressure
5	Initiation of SIAS	0.1413	Referring to the primary system pressure
6	Fail of the SI operation	0.1413	No coolant injection to the system
7	1st AM - ADVs open	0.5683	Both SGs, keep the target cooling rate
8	2nd AM : pressurizer safety valve open	0.7150	Keep the target cooling rate
9	SIT operation	-	Referring to the primary system pressure, Injection through 4 cold-legs
10	Excursion of heater rod surface temperature	0.7680	Increase of the heater rod surface temperature
11	End of the test	0.7680	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 during the transient. Right after the break initiation, the primary system pressure rapidly decreased. As the MSSVs were operating, the primary system pressure kept maintained in equilibrium with the secondary system pressure. After ADVs were opened (at 0.5683 normalized time) as the first accident management (AM) action, the operation of MSSVs stopped and the system pressures decreased gradually. The opening of the pressurizer safety valve (at 0.7150 normalized time) as the second AM action was initiated at 0.1467 normalized time after the 1st AM action but it did not affect the efficient depressurization of the primary system.

The first accident management action was initiated at 0.5683 normalized time with opening ADVs on both SGs. The first AM action was conducted by the manual operation of ADVs by operators keeping the target cooling rate which was determined by averaged temperature of two hot legs. As shown in Fig. 2, the hot leg temperatures rapidly decreased at the beginning of the operator's AM action, but decreased gradually as the operator manually operated the ADVs with maintaining the required cooling rate of the system, continuously.

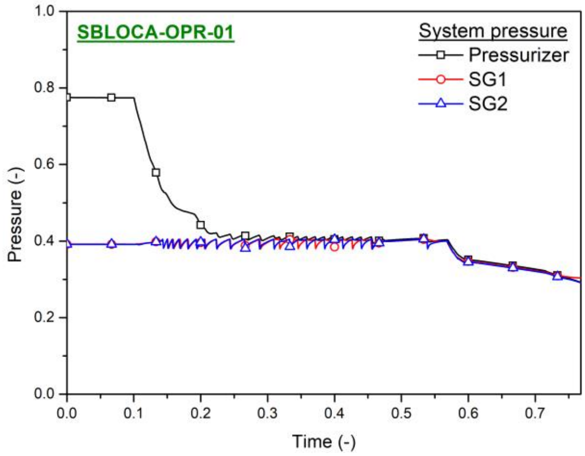

Fig. 1. System pressure behavior

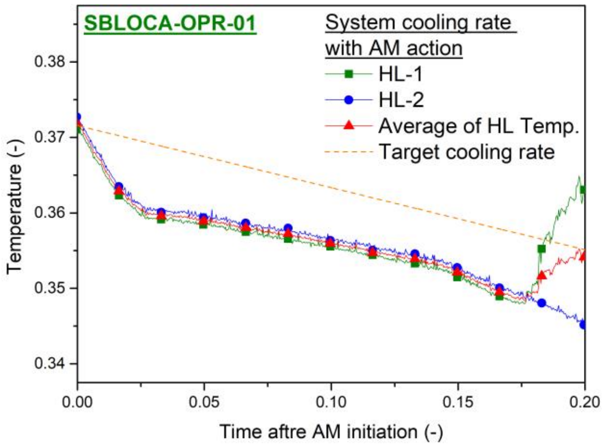

Fig. 2. System cooling with AM action

Fig. 3 shows the collapsed water level behavior in the reactor pressure vessel (RPV). As the break started, the collapsed water level in the RPV began to decrease. The loop seal clearing occurred at 0.4727 normalized time (SG side) and at 0.4760 normalized time (reactor coolant pump (RCP) side) so the water level in the core increased instantly at that time. However, due to the continuous release of coolant through the break, the water level in the core decreased again. After the second AM action of opening the pressurizer safety valve was initiated, the collapsed water level in the RPV decreased 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, resulting in accelerated heating of the core.

As AM actions were initiated, the collapsed water level in the RPV decreased rapidly. The core heaters were completely exposed to steam, and the heater rod surface temperature began to rise rapidly from 0.6647 normalized time, as shown in Fig. 4. The AM actions planned in this accident scenario were intended to induce operation of the safety injection tanks by intentionally depressurizing the primary system pressure, but they were not sufficient to mitigate the core heaters overheating. As the result, the excursion of the core heater rod surface temperature progressed more rapidly than the depressurization of the system by the AM actions. The core heater rod surface temperature reached to the core heater protection set value so the core power was stopped automatically by the control logic and the test was terminated.

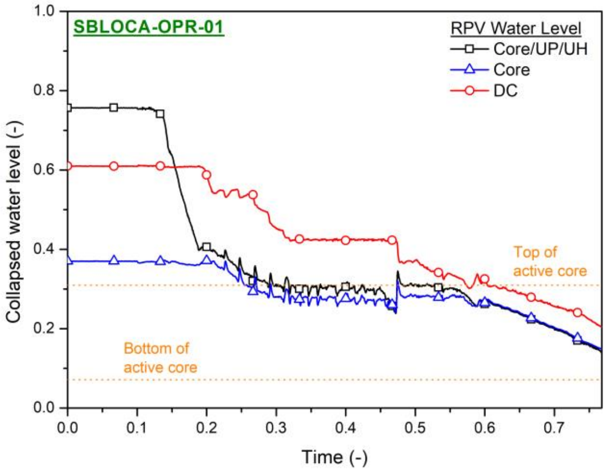

Fig. 3. Collapsed water level in the RPV

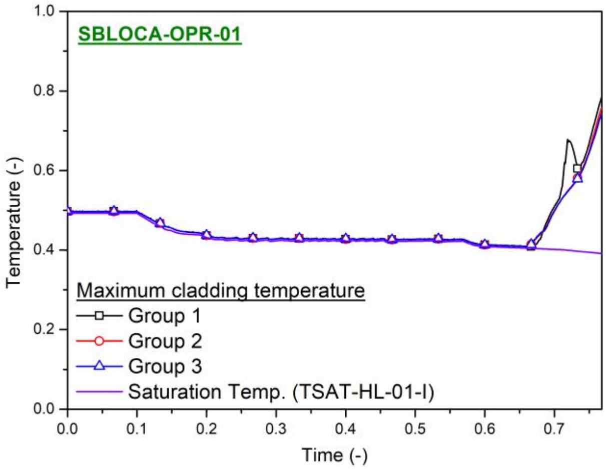

Fig. 4. Heater rod surface temperature

From this test result, it took 0.5647 normalized time from the break initiation to the excursion of the surface temperature of the core heaters (at 0.6647 normalized time).

This corresponds to 0.7987 normalized time based on an OPR1000 nuclear power plant, meaning that operators can secure a response time of about 0.8000 normalized time, less than one hour, in this kind of multiple failure accident.

In order to cool down the system through secondary system operation when safety injection fails under the loss of coolant accident condition, it is important to secure coolant inventory in the primary system while the system cools down to the pressure at which the safety injection tank operates. To achieve this, it was confirmed that AM actions must be initiated as soon as possible. In addition, for more aggressive primary system depressurization, it can be recommended that depressurizing the system using the pressurizer spray system rather than opening the pressurizer safety valve is helpful considering the coolant inventory of the primary system.

4. Conclusions

An integral effect test on multiple failure accidents was performed based on the risk/performance information. As a result of the risk/performance information analysis of an operating nuclear plant, OPR1000, a loss of safety injection accompanied by a small break loss of coolant accident was selected as the target multiple failure accident scenario in this study.

Using the ATLAS test facility, an integral effect test was successfully conducted. As a result of evaluating the effectiveness of the safety system and operator's accident management AM actions, it was found that if safety injection is not performed when an SBLOCA occurs, the AM actions to depressurize the reactor coolant system as quickly as possible is required to cool down the system stably. In addition, it was also recommended that AM measures using the pressurizer water spray system can be more effective than operator actions that accelerated the loss of coolant from the system, such as opening the pressurizer safety valve.

The present test results can be used to develop the AM strategy optimization methodology with various test and safety evaluation results of multiple failure accident. In addition, this test data and analysis result can be used for the development and validation of the best-estimate integrated analysis platform.

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