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# Experimental Study on the Main Steam Line Break (MSLB) Accident Accompanied by the Loss of Shutdown Cooling System (SCS) based on the Risk/Performance Information Analysis

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## 1. Introduction

The probability of multiple failure accidents and the possibility of accident expansion due to over-design earthquakes, aging of nuclear power plants, and human errors, various researches have aimed at strengthening the nuclear power plant's defense-in-depth capability by enhancing the effectiveness of safety systems and accident management strategies. In particular, in accordance with the Nuclear Safety Act [1] amended in 2016 in Korea, nuclear safety goals that significantly strengthen severe accident-related requirements such as multiple failures, extreme disaster analysis, and probabilistic safety evaluation were introduced.

Verification and evaluation of the severe accident prevention capabilities, extreme disaster mitigation facilities and guideline are mostly dependent on deterministic safety analysis results. And there are only limited cases considering an operator’s actions for evaluation and verification of multiple failure accident management.

Thus, in this study, the integral effect test (IET) referring to the risk/performance information analysis on the multiple failure accident of operating nuclear power plant was conducted. Based on the test results of the IET, the accident management strategy and safety margin were evaluated to develop the accident management strategy optimization technology and to improve the safety margin

Referring to the risk/performance information analysis result on multiple failure accidents that may occur in operating nuclear power plants, a main steam line break (MSLB) accident accompanied by a loss of shutdown cooling system (SCS) was selected as the target scenario in this study.

The integral effect test was performed by utilizing the ATLAS (Advanced Thermal-Hydraulic Test Loop for Accident Simulation) test facility [2]. Based on the test results, the guidelines to amend overly conservative or unclear safety indicators were presented. In addition, risk/performance information was re-evaluated to the optimized accident management strategy.

## 2. Risk/Performance Information Analysis

In order to decide the multiple failure accident scenario for the ATLAS test, a multiple failure accident probabilistic safety assessment (PSA) model [3] for a MSLB was developed. And a probabilistic safety evaluation was performed to evaluate the possibility and impact of MSLB accident accompanied by the loss of SCS.

Based on the result, multiple failure accident scenario with high probability of occurrence and high conditional core damage probability was selected.

## 3. Integral Effect Test Utilizing ATLAS

### *3.1 Test Facility*

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 [4, 5]. Referring to the scaling analysis result, flow restrictors with inner diameter of 36.87 mm were inserted in both SGs outlet of steam i.e., secondary system.

To simulate the MSLB, the break was simulated at the upward pipe line of MSIV. The discharged inventory from the SG secondary system through the break was collected in the condensation tank and the integrated mass was measured. And the inventory from the intact SG secondary system through the ADV was also discharged into the condensation tank and its integrated mass was measured by the load cell.

### *3.2 Test Scenario*

Referring the risk/performance information evaluation result and the emergency operation procedure which is actually applied in the operating nuclear power plant, the detailed test scenario was determined as listed in the Table I.

Table I: Sequence of Major Events

| # | Event | Normalized time | Remark |
|---|---|---|---|
| 1 | Initiation of MSLB | 0.0150 | Break valves open |
| 2 | Reactor trip | 0.0156 | Reactor trip by LSGP signal Decay heat simulation |
| 3 | SG-1 Isolation | 0.0158 /0.0159 /0.0156 | Close of MSIV1, MFIV1/2, MSCV |
| 4 | Initiation of SIAS | 0.0218 /0.0237 | Refer to the primary system pressure, Injection through 4 cold-legs |
| 5 | Auxiliary feedwater injection | 0.0384 | SG-2, Rated flow rate |
| 6 | ADV open | 0.0158 | SG-2, Manual open by the operator with reactor trip |
| 7 | Stop of SI by the operator | 0.0943 | Refer to the pressurizer level, sub-cooling of the primary system |
| 8 | Shutdown cooling system operation condition | 0.6291 | Refer to the primary system pressure and Avg. temperature of hot-legs |
| 9 | Fail of the SCS operation | 0.6291 | Stop of the cooling operation by SG-2 |
| 10 | End of the test | 1.2014 | Secondary system pressure of SG-2 > MSSV opening pressure |

* LSGP: Low Steam Generator Pressure
SG: Steam Generator
MSIV: Main Steam Isolation Valve
MFIV: Main Feedwater Isolation Valve
MSCV: Main Steam Control Valve
SIAS: Safety Injection Actuation Signal
ADV: Atmospheric Dump Valve
SI: Safety Injection
MSSV: Main Steam Safety Valve

### *3.3 Test Result*

Considering the confidential problem of test data, all of the test results in this paper were normalized by an arbitrary value including the time frame.

The overall system pressure behavior is shown in Fig. 1. The primary system pressure decreased rapidly right after MSLB initiation due to the over cooling by the secondary system of the broken SG. But the primary system pressure was recovered by the safety injection actuation and the safety injection was stopped at 0.0943 normalized time by an operator when the sub-cooling of the primary system and the collapsed water level in pressurizer conditions are satisfied.

After that, the primary system pressure decreased continuously by supplying auxiliary feedwater and cooling operation using ADV on the intact SG secondary system.

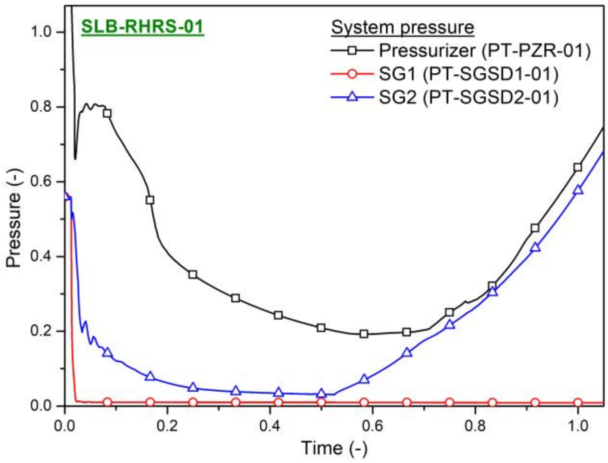

Fig. 1. System pressure behavior

The auxiliary feedwater flowrate that was supplied on the intact SG was determined from the scaling analysis result of ATLAS against OPR1000. The auxiliary feedwater was injected during 0.5891 normalized time with the total mass of 2,273.21 kg as shown in Fig. 2.

The total amount of coolant inventory of condensate storage tank (CST) is 600,000 gallon in OPR1000. The coolant inventory of CST is very important factor that can have an effect on the availability of auxiliary feedwater.

The total mass of auxiliary feedwater that was supplied in this test scenario can be converted 2.3006 $m^3$ with 1 atm, 50℃ condition. And it corresponds to 142,520 gallon as the OPR1000 nuclear power plant scale, referring to the ATLAS scaling analysis result against the OPR1000. Thus, based on this test result, we can deduce that the available time of auxiliary feedwater supply from the CST is 14.9 hours.

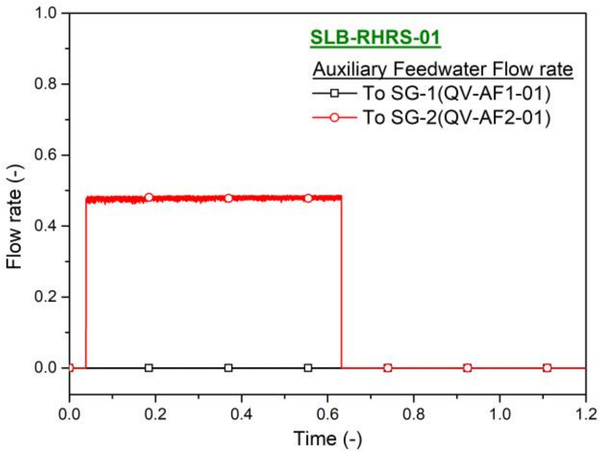

Fig. 2 Auxiliary feedwater supply

When the system reached at the SCS operation condition, the operator stopped the auxiliary feedwater injection and close the ADV and there was no cooling operation on the whole system after that.

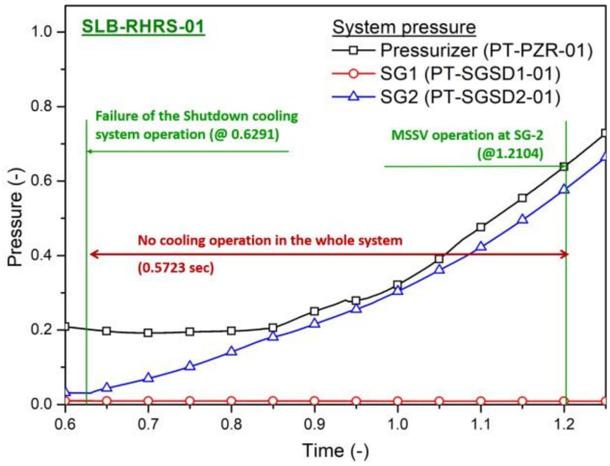


Fig. 3 System behavior after SCS operation failure

As shown in Fig. 3, due to the decay heat transfer from the reactor core to the SG secondary system, the secondary system pressure of SG-2 increased and it reached at the MSSV opening pressure, finally.

So, the time interval from the failure of SCS operation at 0.6291 normalized time to the first open of MSSV as 1.2014 normalized time can be regarded as the coping time of an operator without any cooling operation on the system and it corresponds 4 hours and 30 minutes as the OPR1000 nuclear power plant time scale.

## 4. Re-evaluation of Risk/Performance Information

Based on the ATLAS test results, major events were selected to re-evaluated the risk/performance information in consideration of two aspects: ① occurrence frequency, which is the possibility of multiple failure accidents ② core damage frequency, which is the effect of multiple failure accidents.

As a result, the following four cases were selected.

- · When the heat removal by the secondary system is conducted, the operator finds an alternative water supply source before the CST is depleted and maintains the secondary system heat removal (AFOPHALTWT)
- · When the SCS operation condition is reached, the operator performs the SCS operation (SCOPHSDCOP)
- · When the operator fails the SCS operation after the system reaches the SCS operation condition, the operator restarts the auxiliary water supply pump or starting feed water pump to maintain secondary system heat removal (MXOPHMSHR)
- · When the pressure of the reactor coolant system is high, the operator opens the Pilot Operated Safety Relief Valve (POSRV) to decompress the primary system and inject the coolant (SDOPHLATE)

Among them, the SCOPHSDCOP case is a major event that is selected for re-evaluation in the view point of both the possibility and impact of multiple failure accidents.

Table II: Risk/Performance Information Re-evaluation Results

| Event | Decreased ratio (%) | |
|---|---|---|
| | Major event failure probability | Core damage Frequency |
| AFOPHALTWT | 75.4 | 44.7 |
| SCOPHSDCOP | 1.3 | 0.6 |
| MXOPHSDCOP | 48.8 | 13.3 |
| SDOPHLATE | 27.7 | 23.7 |

Selected four cases were re-evaluated using the Level-1 PSA model for risk assessment and the K-HRA method [6] used in calculating the probability of human error in the PSA model for risk assessment of the multiple failure accident.

The re-evaluation results are summarized in Table II. According to the re-evaluation result, there were several major events that have a little effect on the risk improvement. But it was confirmed that the risk of the most major events were improved then the ATLAS test results were applied.

## 5. Conclusions

The Level-1 PSA assessment model was developed to determine the multiple failure accident scenarios for ATLAS integral effect test and to evaluate the safety margin of the accident management strategy.

Using the developed PSA model, the accident scenario was evaluated in the viewpoint of the possibility and impact of multiple failure accidents on operating nuclear power plant of OPR1000. As a result, MSLB accident accompanied by the loss of SCS was selected for the target scenario in this study.

The integral effect test using the ATLAS facility was performed considering the operation of the safety systems and operator's accident management strategies that are currently applied on the OPR1000 in case of the selected multiple failure accident.

Based on the test results, the main conclusions were obtained as follows:

- · As a result of calculating the inventory of the CST, which is the water source of auxiliary water supply, the time that the operator can use auxiliary water supply from the CST was found to be 14.9 hours in case of OPR1000.
- · After failure of the initiation SCS, the coping time (till the first opening of the MSSV) that an operator can have without any cooling operation

on the system was evaluated 4 hours and 30 minutes as the OPR1000 time scale**.**

Referring this ATLAS test results, risk/performance information re-evaluation was performed for four accident management cases that were selected in the viewpoint of the possibility and impact of the multiple failure accident. According to the re-evaluation result, it was confirmed that the risks were improved in a number of major events when the test results were applied.

ACKNOWLEDGEMENT

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