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Experimental Study on the Safety Injection Pump (SIP) Failure Accompanied by the Steam Line Break (SLB)

Yusun Park[a*], Byoung Uhn Bae[a], Jong Rok Kim[a], Jae Bong Lee, Hae Min Park, Yong Chul Shin[a], Kyung Ho Min[a], Nam Hyun Choi[a] , Kyoung Ho Kang[a]
[a]*Thermal Hydraulics Safety Research Division, Korea Atomic Energy Research Institute*
989-111 Daedeok-daero, Yuseong-gu, Daejeon, 305-333, Republic of Korea
[*]*Corresponding author: yusunpark@kaeri.re.kr*

1. Introduction

A SIP failure accompanied by a SLB accident is one of the multiple failure accident. In the revised nuclear safety regulation in Korea, March 2016, the "Multiple failure" is defined as "Over the single failure, the failure is occurred in more than two component and loss of their safety function". The multiple failure accidents show the relatively high occurrence probability and the CDF.

Thus, an SIP failure accompanied by a SLB accident simulation test was performed as an example of a multiple failure, to investigate the thermal hydraulic phenomena during an accident.

The test ID of this test was named as SLB-SIP-02 test and it was performed on 24th October, 2017. The overall thermal hydraulic phenomena that can be anticipated in the scenario were observed. An analysis of the experimental results on the overall thermal-hydraulic behaviors in the reactor coolant system (RCS) during a transient period is explained in this study. In addition, we can obtain the insights about the accident management procedures in the case of this kind of multiple failure accident.

2. Description of the Test Facility

In this study, a thermal-hydraulic integral effect facility, ATLAS (Advanced Thermal-hydraulic test Loop for Accident Simulation) was utilized to simulate thermal hydraulic phenomenon. The ATLAS facility is designed according to the well-known scaling method suggested by Ishii and Kataoka [1, 2]. A detailed design and description of ATLAS can be found in the literature [3].

The target scenario of this SLB test is a double-ended guillotine break of a main steam line. Fig. 1 shows a piping arrangement of the break simulation system and Fig. 2 shows a detailed design of the flow restrictor which was installed at the outlet of a steam generator. The main steam line from the steam generator 1 was modeled as the guillotine break at the upstream of a main steam isolation valve (MSIV).

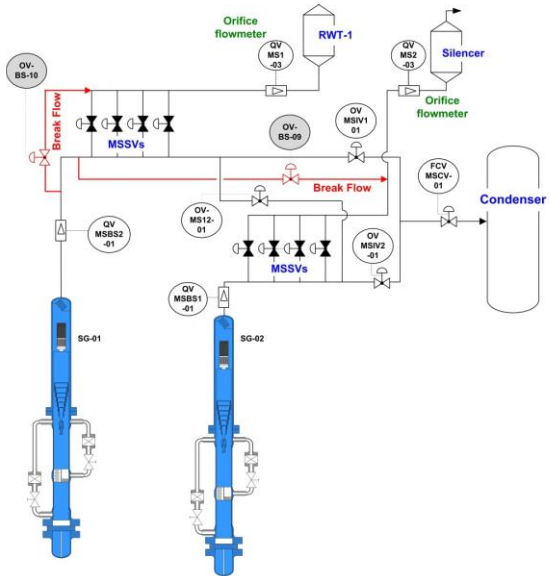
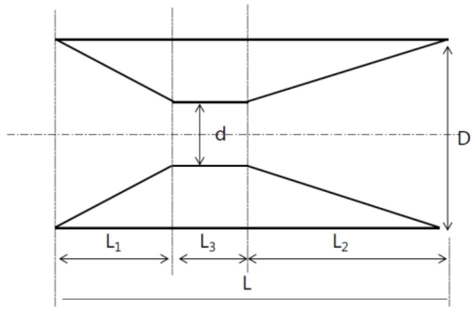


Fig.1. Piping arrangement of the break simulation system



Type	D	d	L1	L2	L3	L
Chorking	73.0	38.6	88.48	140.07	75.63	304.18
Non-Chorking	73.0	45.9	69.70	110.34	59.58	239.62

* Unit : mm

Fig.2. Design of a flow restrictor

3. Test Procedure

In Table 1, the initial steady state and boundary conditions of the SLB-SIP-02 test were summarized and compared with those of the APR1400. These initial and boundary conditions were obtained by applying the scaling analysis to the MARS-KS calculation results for APR1400 analysis. When the whole system reached specified initial conditions, the steady-state conditions of the primary and secondary systems were maintained for more than 30 minutes. After that, data saving was started during a steady state period from -10 to 300 seconds. Measured values during this initial steady period were averaged and they are listed in Table 2.

After this 300 seconds steady-state period, the transient was started with opening break valves, OV-BS-09 and OV-BS-10. By the SLB, the pressure of SG-1 decreased rapidly and the reactor was tripped by the low steam generator pressure (LSGP) trip. After reactor trip, the core power started to decrease following the decay curve. When the collapsed water level of an affected steam generator secondary side decreased to 2.78 m, the auxiliary feedwater injection signal was induced. Because of the continuous discharge of break flow from the SG-1 secondary system, the collapsed water level of the SG-1 secondary side was not recovered and the auxiliary feedwater was supplied during the whole test period.

The primary pressure decreased rapidly right after opening break valves by the blowdown of an affected steam generator, SG-1. Even though the primary pressure decreased lower than 10.72 MPa which is the general set point of the safety injection pump actuation pressure, the safety injection pumps were fail to operate and no ECCS water was injected into the primary system. However, the primary system was cooled down continuously because of the heat removal from the SG-1. Thus, the test ended when the system was cooled down enough by the operator's decision. The sequence of event of the SLB-SIP-02 test is summarized in Table 3.

Table 1 Initial and boundary conditions

Design parameter	APR1400 (Steady state)	ATLAS target value
Primary system condition		
Normal power(MWt)	3983	1.63
Pressurizer pressure(MPa)	15.5	15.5
Core inlet temperature(°C)	290.7	290.1
Core outlet temperature(°C)	324.2	293.9
Secondary system condition		
Steam pressure(MPa)	6.9	7.31
Steam temperature(°C)	293.5	288.8
Steam flow rate(kg/s)	1152.4	0.444
Feedwater flow rate(kg/s)	1152.4	0.444
Feedwater temperature(°C)	232.2	232
Primary piping		
Cold leg flow rate(kg/s)/Loop	5540.1	16.4
Hot leg/Cold leg temperature(°C)	323.3 /290.1	393.8 /290.1

Table 2 Initial steady state of the SLB-SIP-02 test

Design Parameter	Target Value	Measured Value (Value/STDEV)
Reactor Pressure Vessel		
Normal Power (MW)	1.56	1.625/ 0.0025
Pressurizer Pressure (MPa)	15.5	15.48 / 0.0053
Pressurizer Level (m)	3.8	3.19 / 0.0175
Core Inlet Temperature (°C)	290.7	290.5 / 0.0881
Core Outlet Temperature (°C)	324.2	294.9/ 0.0703
Steam Generator		
Net Thermal Power (MWt)	0.78	0.7555 / 0.0056
Steam Flow Rate (kg/s)	0.444	0.4004/0.0219
Feed Water Flow Rate (kg/s)	0.444	0.4278/0.0019 0.4183/0.0057
Feed Water Temperature (°C)	232.2	231.8/0.1150, 231.2/0.1228
Steam Pressure (MPa)	7.83	7.31 / 0.0024, 7.31 / 0.0023
Steam Temperature (°C)	293.5	291.1/0.0914, 291.1/0.0611
Secondary Side Level (m)	5.0	5.05 / 0.0100 5.06 / 0.0095
Primary Piping		
Cold Leg Flow (kg/s)	2.0	17.2084/0.0401

Table 3 Sequence of Events

Description	Time (sec)	Remark(Set-point)
SLB Start	303	OV-BS-09, OV-BS-10 Open
MFIS	303	Coincidence with break
MSCV Close	303	Coincidence with break
LSGP Signal	309	SG Pressure < 6.11 MPa
Reactor Trip	309	Coincidence with LSPG
RCP Trip	310	LSGP + 1.0 sec delay
MSIS	314	LSGP + 3.54 sec delay
Decay power start	323	Reactor trip + delay (12.07 sec in table)
AFAS	319	SG-1 level < 2.78(m)
Aux Injection into SG-1	363	AFAS+43.45 sec delay LT-SGSDRS1-01 level = 2.78m/3.9m(on/off)

3. Experimental Result

The total core power decreased after reactor trip following the decay curve. Fig. 3 shows that the core power was simulated as anticipated during the whole test period. As the collapsed water level of an affected steam generator, SG-1, decreased to 2.78 m, the auxiliary feedwater supplied to the SG-1 with rated constant flow rate as shown in Fig. 4.

Fig. 5 and Fig. 6 show the variation of system pressures and collapsed water levels, respectively. The pressure and the collapsed water level of an affected steam generator, SG-1, decreased rapidly right after the initiation of SLB. The collapsed water level of the pressurizer decreased also because of the excessive heat removal by the steam line break. The primary pressure decreased in the early stage of the transient and it turned to increase. This is the time that the inventory of an affected steam generator was dried out. During a certain period of time, the heat removal by the steam generator secondary side was not enough to overcome the amount of heat supplied to the core decay power. However, by the continuous supply of auxiliary feedwater with continuous decrease the decay power, the heat removal from the primary system exceeded the supplied core power. Thus, the primary system pressure decreased again and the system cooled down successfully.

The collapsed water level of an intact steam generator, SG-2, showed almost constant level after isolation. The pressure of SG-2 decrease during the whole test period by the inverse heat transfer to the primary system.

As shown in Fig. 7, flow rates of the loop-1 showed higher flow rate than that of the loop-2. This means that the natural circulation in the loop-1 was occurred actively.

The SLB break flow was collected in the condensation tank and the integrated mass was measured as shown in Fig. 8. In the early stage of transient, the break flow was discharged rapidly and it was kept with the almost constant flow rate during the whole test period.

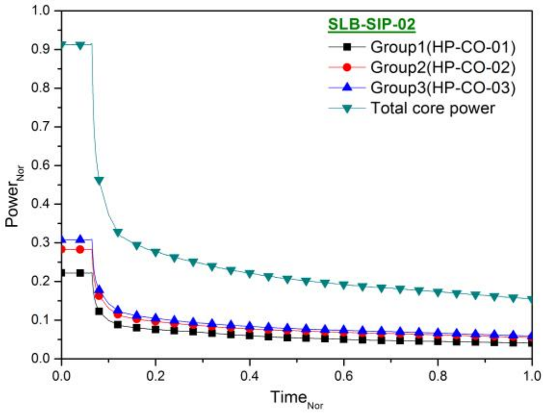


Fig. 3. Supplied core power

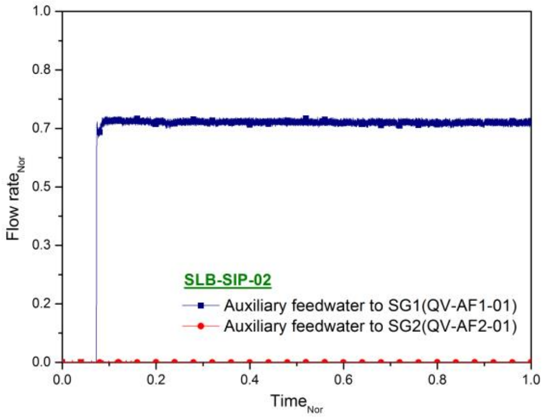


Fig. 4. Auxiliary feedwater flow rate

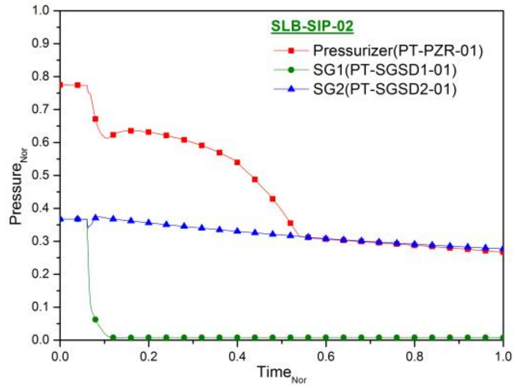


Fig. 5. Variation of system pressures

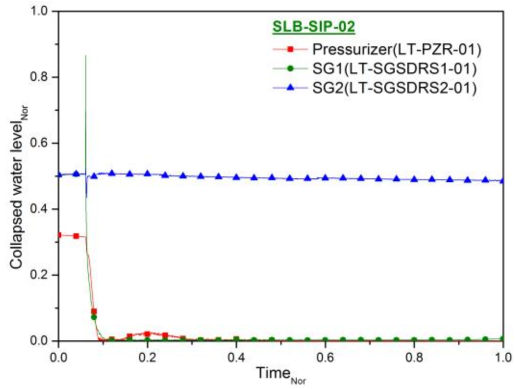


Fig. 6. Variation of collapsed water levels

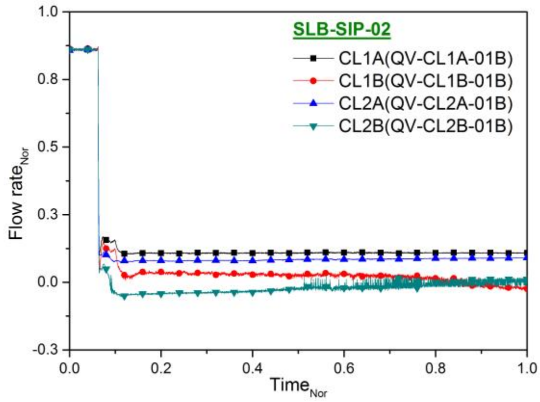


Fig. 7. Loop flow rates

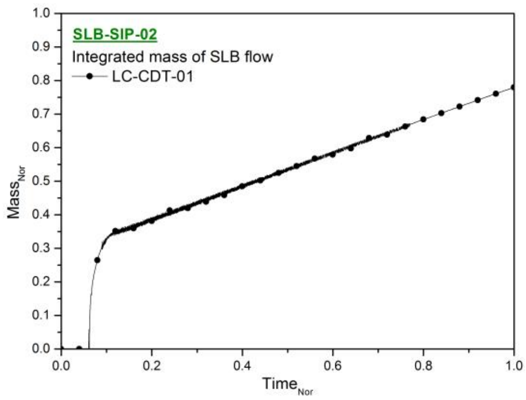


Fig. 8. Integrated mass of the break flow

4. Uncertainty Analysis

The uncertainty of the measured experimental data is analyzed in accordance with a 95% confidence level. According to the ASME performance test code 19.1[4], the uncertainty interval of the present results is given by the root-mean-square of a bias contribution and a precision contribution. The bias and precision errors were evaluated from the data acquisition hardware specifications and the calibration results performed once per year, respectively. Table 4 summarizes the analyzed uncertainty levels of each group of instruments.

Table 4 Uncertainty levels of instruments

Items	Unit	Uncertainty
Static Pressure	MPa	0.56%
Differential Pressure	kPa	0.46%
Collapsed Water Level	m	2.21%
Temperature	ºC	Max. 2.4ºC
Flow rate	kg/s	2.65%
Heater Power	W	± 0.595%
Heat Flux (Tube, Pool side)	W/m^2	± 15.3%
Heat Transfer Coefficient (Tube side)	$W/m^2 K$	± 3.1%
Heat Transfer Coefficient (Pool side)	$W/m^2 K$	± 2.6%
Load Cells (RWT/CDT)	kg	0.17%/0.06 %

5. Conclusions

The SLB-SIP-02 test was performed to simulate a steam line break with a total failure of safety injection pumps. The major thermal-hydraulic phenomena which can be occurred during a transient period were realized successfully.

By the SLB accident, the primary system was cooled down due to the excessive heat removal. Despite the total failure of safety injection pumps, the continuous heat removal through the steam generator secondary side was kept by the injection of the auxiliary feedwater.

Thus we can conclude that the reactor system can be cooled-down stably in the case of this multiple failure accident condition.

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