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BREAK SIZE EFFECT ON THE TRANSIENT THERMAL-HYDRAULIC BEHAVIOR DURING THE STEAM GENERATOR TUBE RUPTURE ACCIDENT

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Abstract

In order to simulate the SGTR accident of the APR1400, integral effect tests were performed by simulating a double-ended rupture of a single and five U-tubes. Following the reactor trip, the primary system pressure decreased and the secondary system pressure increased until the MSSVs was opened to reduce the secondary system pressure. Break area affected the timings of the major events observed in the tests. Less heat transfer to the secondary side caused by earlier actuation of the safety injection pumps had more influence on the secondary pressure of the affected steam generator than the break flow.

Introduction

The steam generator tube rupture (SGTR) accident is one of the design basis accidents, which has a unique feature of the penetration of the barrier between the reactor coolant system (RCS) and the secondary system resulting from failure of steam generator U-tubes. The SGTR has an importance in safety due to a concern of a containment bypass of radioactive inventory. In the course of the SGTR, the radioactive coolant breaking through broken steam generator U-tubes mixes with the shell-side water in an affected steam generator. The break flow from ruptured U-tubes can increase a water level and a pressure of the affected steam generator. Following a reactor and a turbine trips, the main steam safety valves (MSSVs) can be open to mitigate an increase in the secondary system pressure. Meanwhile, the SGTR can provide a direct flow path from the primary to the secondary systems resulting in the release of fission products into the atmosphere. It is generally known that the break flow rate from the primary to the secondary sides is the most important factor affecting the overall thermal-hydraulic behaviors such as the depressurization rate of the RCS, the water level increase and the pressurization rate of the secondary system, and the consequent MSSV opening time, etc. The break flow rate through the broken U-tubes depends on the primary-to-secondary system differential pressure, the primary coolant subcooling, the break area, and the break location along the U-tubes [1].

A postulated SGTR event of the APR1400 (Advanced Power Reactor 1400 MWe) was experimentally investigated with the thermal-hydraulic integral effect test facility, ATLAS (Advanced Thermal-hydraulic Test Loop for Accident Simulation) [2]. In order to simulate the SGTR accident of the APR1400, the SGTR-HL-04 and the SGTR-HL-05 tests were performed by simulating a double-ended rupture of a single and five U-tubes at the hot side of the ATLAS steam generator. The main objectives of these tests were not only to provide physical insight into the system response of the APR1400 during a transient situation of the

SGTR but also to establish integral effect test database for the validation of the SPACE (Safety and Performance Analysis Computer Code) [3], which is now under development by the Korean nuclear industry. Even though a complete rupture of a U-tube has very low possibility for the vertical Inconel U-tubes of the pressurized water reactors (PWRs), it was agreed that a double-ended rupture of a single U-tube is worth being investigated as the most limiting SGTR from a viewpoint of safety. In order to investigate the effects of the U-tube break area on the transient thermal-hydraulic behavior, two cases for a single tube rupture and five tubes rupture have finally been selected as the representative cases to validate a safety analysis code such as the SPACE code.

1. Description of the ATLAS

The reference plant of the ATLAS is the APR1400, which has a rated thermal power of 4000 MW and a loop arrangement of 2 hot legs and 4 cold legs for the RCS. The ATLAS also incorporates some specific design features of the Korean standard nuclear power plant, the OPR1000 (Optimized Power Reactor 1000 MWe), such as a cold-leg injection mode for a high pressure and a low pressure safety injection modes. The ATLAS can be used to investigate the multiple responses between the systems for a whole plant or between the subcomponents in a specific system during anticipated transients and postulated accidents. Besides, the ATLAS can be used to provide the unique test data for the 2(hot legs) x 4(cold legs) reactor coolant system with a direct vessel injection (DVI) of emergency core cooling (ECC); this will significantly expand the currently available data bases for code validation.

Parameters	Scaling ratio	ATLAS design
Length (height)	l_{oR}	1/2
Diameter	d_{oR}	1/12
Area	d_{oR}^{2}	1/144
Volume	$l_{oR}\ d_{oR}^{2}$	1/288
Velocity	$l_{oR}^{1/2}$	1/1.414
Time	$l_{oR}^{1/2}$	1/1.414
Core power	$l_{oR}^{1/2}\ d_{oR}^{2}$	1/203.6
U-Tube diameter (steam generator)	$l_{oR}^{1/2}$	1/1.414
Number of U-tubes (steam generator)	-	1/72
Flow rate	$l_{oR}^{1/2}\ d_{oR}^{2}$	1/203.6
Pressure drop	l_{oR}	1/2

Table 1 Major scaling parameters of the ATLAS.

The ATLAS is designed according to the well-known scaling method suggested by Ishii and Kataoka [4] to simulate various test scenarios as realistically as possible. It is a half-height and 1/288-volume scaled test facility with respect to the APR1400. The main motive for adopting the reduced-height design is to allow for an integrated annular down-comer where multi-dimensional phenomena can be important in some accident conditions with a DVI operation. According to the scaling law, the reduced height scaling has time-reducing results in the model. For the half-height scaled facility, the time for the scaled model is $\sqrt{2}$ times faster than the prototypical time. The friction factors in the scaled model are maintained the same as those of the prototype. The hydraulic diameter of the scaled model is maintained the same as that of the prototype to preserve the prototypical conditions for the heat transfer coefficient. Major scaling parameters of the ATLAS are summarized in Table 1.

The fluid system of the ATLAS consists of a primary system, a secondary system, a safety injection system, a break simulation system, a containment simulation system, and auxiliary systems. The primary system includes a reactor pressure vessel (RPV), two hot legs, four cold legs, a pressurizer, four reactor coolant pumps (RCPs), and two steam generators (SGs). The total inventory is 1.6247 m^3, which was validated by the actual inventory measurement. The total height of the ATLAS is about 30 m, i.e. 10 m underground and 20 m above ground. The secondary system of the ATLAS is simplified to be a circulating loop-type. The steam generated at two steam generators is condensed in a direct condenser tank, and the condensed feedwater is re-circulated to the steam generators. The ATLAS uses water as the working fluid and is scaled for prototypic pressure and temperature conditions. Figure 1 shows a schematic diagram of loop connection, which presents the elevations of the ATLAS major components.

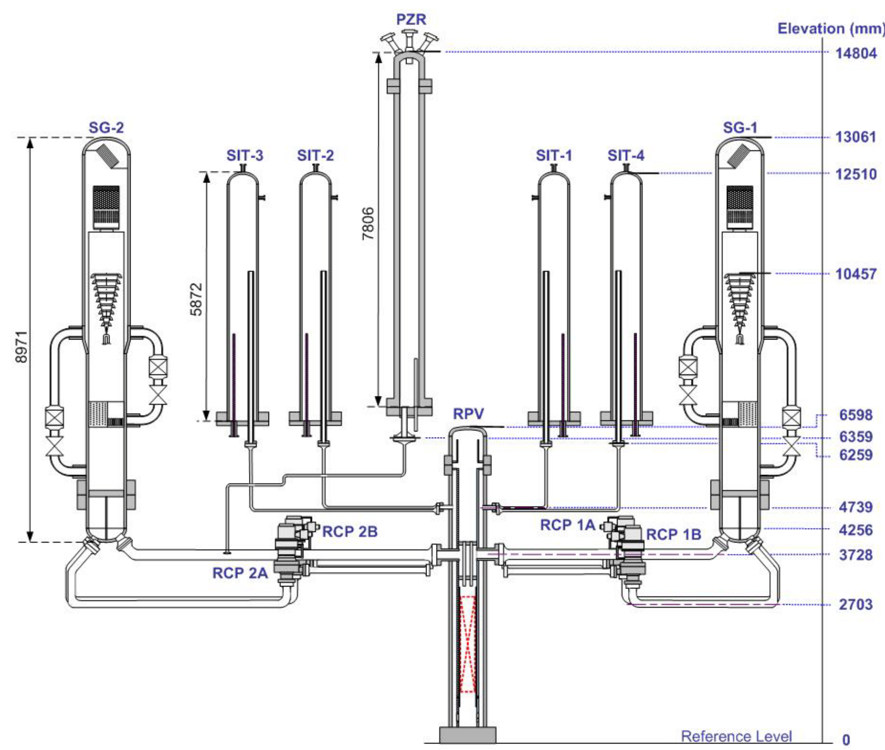


Figure 1 Schematic diagram of loop connection.

2. Experimental conditions and procedures

Two tests, named as SGTR-HL-04 and SGTR-HL-05, were performed to simulate a double-ended rupture of a single and five U-tubes at the hot side of an affected steam generator (SG-1 in Figure 1). In the present tests, considering the safety analysis results for the SGTR accident of the APR1400 [5], a reactor trip was assumed to occur by an increase in the steam generator level, i.e., a high steam generator level (HSGL) trip signal. In addition, a single-failure of a loss of a diesel generator, resulting in the minimum safety injection flow to the RPV, was assumed to occur in concurrence with the reactor trip. Therefore, the safety injection water from the safety injection pump (SIP) was only available through the DVI-1 and -3 nozzles, and the safety injection water from the safety injection tank (SIT) was available through all of the DVI nozzles. Since the primary system pressure was maintained above the set-point of the SIT, 4.03 MPa during the present tests period, the SIT water was not supplied.

The target scenario of the present study is the double-ended guillotine break of a single and five U-tubes at 4.03 m above the tube sheet bottom in the APR1400. In the test, however, the double-ended guillotine break of a U-tube cannot be directly simulated. The reverse break flow from the cold side of the steam generator was not taken into account in the present test for simplicity. Instead, the break spool was installed in an external break simulating pipe and the primary inventory was discharged from the hot side of the lower plenum to the upper location of the steam generator secondary hot side. Because the ATLAS is a half-height test facility with respect to the APR1400, the discharging location was 2.015 m above the inlet of the U-tube. Figure 2 shows the piping arrangement of the break simulation system which consists of a break simulation valve (OV-SGTR1-01), an orifice flow meter, and a break nozzle. With the current piping arrangement, the corresponding break location of the APR1400 preserving the scaling law was 4.03 m above the inlet of the U-tube.

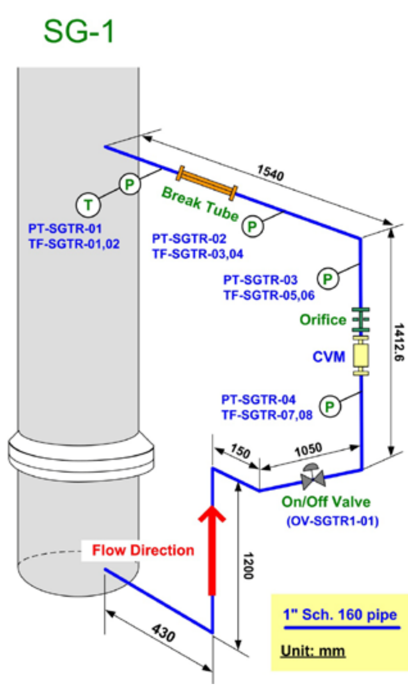


Figure 2 Piping arrangement of break simulation system.

2.1 Determination of test conditions

The present test conditions were determined by a pre-test calculation with a best-estimate thermal-hydraulic safety analysis code, MARS-KS [6]. First of all, a transient calculation was performed for a single and five U-tubes rupture of the steam generator of the APR1400 to obtain the reference initial and boundary conditions. A best-estimate safety analysis methodology, which is now commonly accepted in nuclear industry, was applied to the transient calculation of the APR1400. A single failure assumption for a safety injection system was assumed in the MARS calculation; four SITs and two of four SIPs were available. The initial and boundary conditions for the present test were obtained by applying the scaling ratios shown in Table 1 to the MARS calculation results for the APR1400.

Table 2 summarizes actual initial conditions of the SGTR-HL-04 and the SGTR-HL-05 tests. The decay heat was simulated to be 1.2 times that of the ANS-73 decay curve for the conservative condition. The initial heater power was controlled to be maintained at about 1.64 MW, which was equal to the sum of the scaled-down core power (1.56 MW) and the heat loss rate of the primary system (about 80 kW). The heater power was then controlled to follow the specified decay curve after 48.1 and 32.6 seconds from the opening of the break simulation valve (OV-SGTR1-01) in the SGTR-HL-04 and the SGTR-HL-05 tests, respectively.

Design Parameter	Target Value	Measured Value	
		SGTR-HL-04	SGTR-HL-05
Normal Power (MW)	1.56	1.631	1.623
Pressurizer Pressure (MPa)	15.5	15.4	15.6
Core Inlet Temperature (^oC)	290.7	289.9	290.5
Core Outlet Temperature (^oC)	324.2	324.7	325.8
Net Thermal Power (MWt)	0.78	0.752, 0.724	0.774, 0.766
Steam Flow Rate (kg/s)	0.444	0.382, 0.412	0.401, 0.433
Feed Water Flow Rate (kg/s)	0.444	0.427, 0.411	0.441, 0.437
Feed Water Temperature (^oC)	232.2	234.8	235.6
Steam Pressure (MPa)	7.83	7.83	7.85
Secondary Side Level (m)	5.0	5.0, 4.97	5.02, 4.98
Re-circulation ratio (-)	11.0	11.5, 8.8	10.9, 8.6
Cold Leg Flow (kg/s)	2.0	2.2	2.2

Table 2 Initial conditions of the present tests.

2.2 Test procedures

When the whole system reached a specified initial condition for the test, as shown in Table 2, the steady-state conditions of the primary and the secondary systems were maintained for more than 30 minutes. After this steady-state period, the main test was started by opening the break simulation valve, OV-SGTR1-01. Initial water level of the steam generator was set to be 5.0 m. For the reactor trip to be induced by a HSGL trip signal, the HSGL trip signal was set to be actuated at 5.05 m of the steam generator water level. When the HSGL signal occurred, the RCP and the pressurizer heater were stopped, and the main feedwater isolation valves (MFIVs) and the main steam isolation valves (MSIVs) were closed with pre-specified delay times. The closing of the MFIVs and the MSIVs is equivalent to the containment isolation of the APR1400. As the SGTR accident progressed, the primary system pressure decreased below 10.7244 MPa and the SIP was actuated with a pre-specified delay time of 28.28 seconds.

Contrary to the affected steam generator (SG-1), the water level of the intact steam generator (SG-2) decreased continuously and reached the set-point of the auxiliary feedwater actuation signal (AFAS) in the SGTR-HL-04 test. Injection of the auxiliary feedwater recovered the water level of the SG-2 and it became similar to that of the SG-1. However, the level of the SG-2 did not reach the set-point of the AFAS in the SGTR-HL-05 test. Table 3 shows the sequence of the major events observed in the SGTR-HL-04 and the SGTR-HL-05 tests. The break area affected the timings of the major events observed in the present tests. The larger break area, the faster the major events progressed.

Event	Set-Point	Time (sec)	
		SGTR-HL-04	SGTR-HL-05
Break open	Operator's action	206	202
HSGL (Reactor	SG 2nd level = 5.05 m	283	207
MFIS, MSIS	HSGL trip + 7.07 s delay	291	215
MSSV first opening	SG 2nd pressure @ 8.1 MPa	303	222
Decay power	Break open + time delay	254	235
SIP	LPP (RCS pressure < 10.7214 MPa) + 28.28 sec delay	1416	451
SIT	PT-PZR-01 < 4.03 MPa	-	-
AFAS on/off	SG 2nd level = 2.67 m / 3.9	3153/3911	-

Table 3 Summary of major sequence of events.

2.3 Design of the break nozzle

In order to simulate the SGTR accident of the APR1400 as realistically as possible, a boundary flow scaling approach was taken from a break flow rate point of view. During the SGTR, the break flow can be choked or not depending on the differential pressure between the primary and the secondary systems. In either case, the break flow rate in the ATLAS should be scaled down appropriately with respect to the APR1400. Based on the boundary flow scaling criteria, the break flow rate should be preserved. Taking into account the velocity scaling ratio of the ATLAS, i.e., $u_R = l_{oR}^{1/2}$, it can be expressed as Eq. (1).

$$\left[\frac{a_{break}}{a_o}\frac{u_{break}}{u_o}\right]_R = 1, \qquad \left[\frac{a_{break}}{a_o}\right]_R u_{break,R} = u_{oR} = l_{oR}^{1/2}. \tag{1}$$

As for the choking flow, since the break flow is determined by the critical flow, the velocity ratio of break flow ($u_{break,R}$) becomes one. And as for the non-choking flow, the break flow is determined by the differential pressure between the primary and the secondary systems. In order to preserve the break flow rate, from Eq. (1), the break area in the ATLAS test should be scaled down with respect to the APR1400 as the following Eqs. (2) and (3) for choking and non-choking flow cases, respectively.

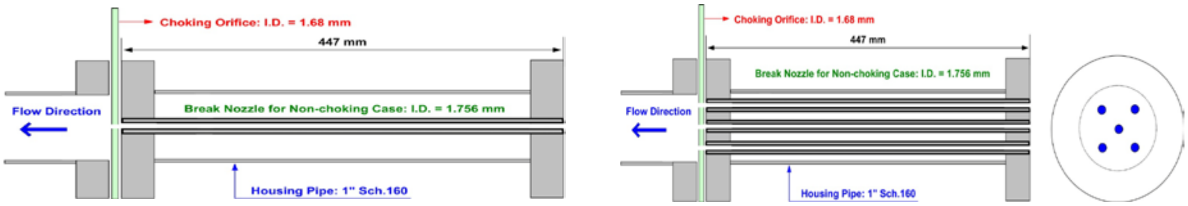
Choking flow condition: $a_{break,R} = a_{oR} l_{oR}^{1/2}$, (2)

Non-choking flow condition: $a_{break,R} = a_{oR} l_{oR}^{1/2} \cdot \left[f\frac{l}{d} + K\right]_R^{1/2} \cdot \Delta P_R^{-1/2}$. (3)

According to the scaling factor of the ATLAS, a break area ratio becomes 1/203.6 if a choking flow condition is assumed. On the other hand, since the velocity decreases by a factor of $\sqrt{2}$ due to the half-height scale of the ATLAS, the pressure loss coefficient $[f\ l/d + K]$ should be double the prototypic value in the test if a non-choking flow condition is assumed. In this study, the break was simulated using the external break simulating pipe and therefore the diameter of the external break simulating pipe in the test was increased by a factor of $\sqrt{2}$ compared to that of the double-ended guillotine break. Table 4 shows the break areas for a single and five tubes rupture cases of this study.

Test ID	SGTR-HL-04		SGTR-HL-05	
Break condition	Single tube rupture (mm)		Five tubes rupture (mm)	
	Single-ended	Double-ended	Single-ended	Double-ended
Choking	1.19	1.68	2.66	3.76
Non-choking	1.46	2.06	3.26	4.61

Table 4 Break nozzle diameters of a single and five tubes ruptures.



(a) SGTR-HL-04 test (b) SGTR-HL-05 test

Figure 3 Detailed geometry of the break nozzle.

Since the break area for the choking flow condition is smaller than that for the non-choking flow condition, the break area for the choking flow condition should be preferentially taken into account in a break nozzle design. In this study, the break nozzle was designed as a combination of a choking orifice and a break tube to satisfy the scaling law of the break flow rate for both choking and non-choking flow conditions. In the SGTR-HL-04 test, the orifice having a diameter of 1.68 mm was installed at the end of the break nozzle to simulate the choking flow condition. Furthermore, in order to preserve the break flow rate for the non-choking flow condition as well, the diameter and the length of the break tube were determined to be 1.756 mm and 447 mm taking into account the available diameters of commercial tubes. If the same approach is taken into account for the selection of the scalable dimension of a break tube in the SGTR-HL-05 test, the diameter and the length of the break tube become 4.57 mm and 2329 mm for the non-choking flow condition, respectively. However, this break tube could not be easily manufactured and also it might distort the flow behavior inside the break tube. Therefore, five-parallel assembly of single U-tube rupture simulated break tube was used in the SGTR-HL-05 test. A detailed geometry of the break nozzle used in the present test is shown in Figure 3.

3. Discussions on the experimental results

3.1 Overall thermal-hydraulic behaviours

When the SGTR event was initiated by opening the break simulation valve, OV-SGTR1-01, the RCS depressurized until the SIPs were actuated as shown in Figure 4. Depressurization rate of the RCS in the SGTR-HL-05 test was larger than that in the SGTR-HL-04 test due to the relatively large break size. Supply of safety injection (SI) water mitigated the RCS depressurization rate due to compensation of the RCS inventory. Depressurization rate of the RCS was estimated to be 3.92 kPa/sec and 23.07 kPa/sec in the SGTR-HL-04 and the SGTR-HL-05 tests, respectively. Figure 5 shows the variation of the secondary system pressure. Following the reactor trip, the secondary pressure increased until the MSSVs were opened to reduce the secondary system pressure. Subsequent to the peak in the secondary system pressure of the steam generators, the secondary system pressure decreased, resulting in closure of the MSSVs. Then, the secondary system pressure started to increase again until it reached the MSSV set-point because the steam generators were isolated due to the previous MSIS and the MFIS actuations. The MSSVs in two steam generators showed almost simultaneous opening and closing behaviors. However, the time difference of the actuation of the MSSVs in two steam generators became larger as time went by.

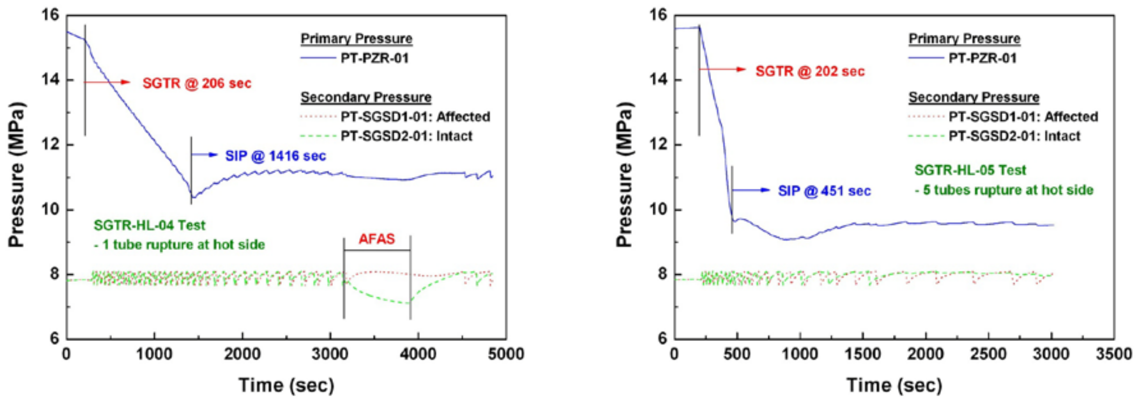


Figure 4 Variation of system pressures.

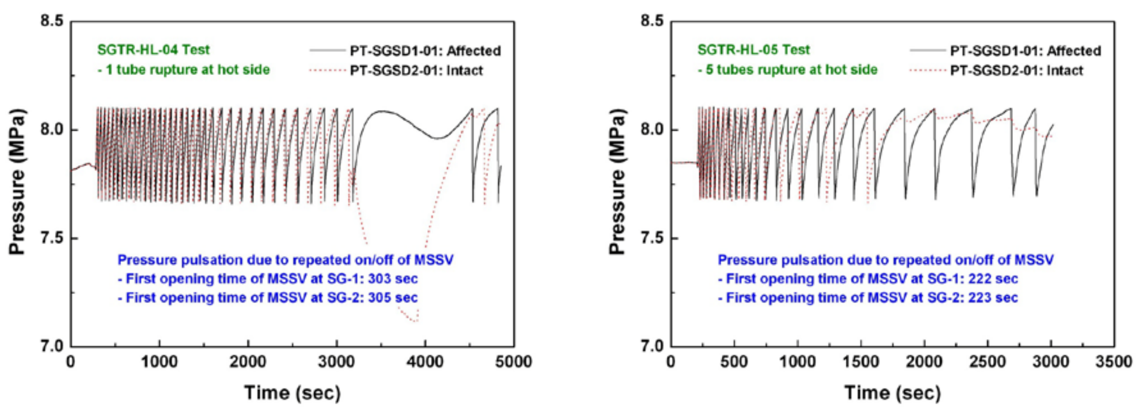


Figure 5 Variation of secondary system pressures.

On the contrary to the SGTR-HL-04 test, the MSSV of the intact steam generator (SG-2) was not opened any more after 1500 seconds in the SGTR-HL-05 test. Since the five U-tubes rupture was simulated in this test, the break flow rate was relatively higher than that of the SGTR-HL-04 test. Large discharge of the primary inventory resulted in rapid depressurization of the primary system and consequently early injection of the SIP in the SGTR-HL-05 test. Also the mass of the injected SI flow in the SGTR-HL-05 test was larger than that in the SGTR-HL-04 test. Supply of cold ECC water by the SIPs reduced the fluid temperature in the hot leg. Accordingly, fluid temperature difference between the hot leg and the cold leg was relatively small in the SGTR-HL-05 test as shown in Figure 6. It indicates that less energy was transferred to the secondary side compared with the single U-tube rupture case of the SGTR-HL-04 test. This caused the retardation of the MSSV opening of the SG-2. Meanwhile, the secondary pressure of the affected steam generator (SG-1) is more likely to increase due to higher break flow than the single U-tube rupture case. However, less heat transfer to the secondary side caused by earlier actuation of the SIPs had more influence on the secondary pressure of the affected steam generator than the break flow.

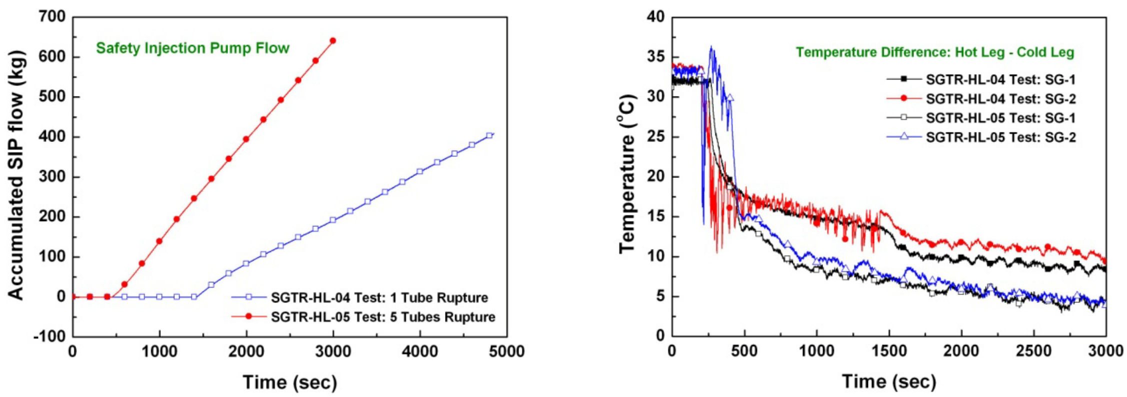


Figure 6 Variation of accumulated SIP flow and fluid temperature differences.

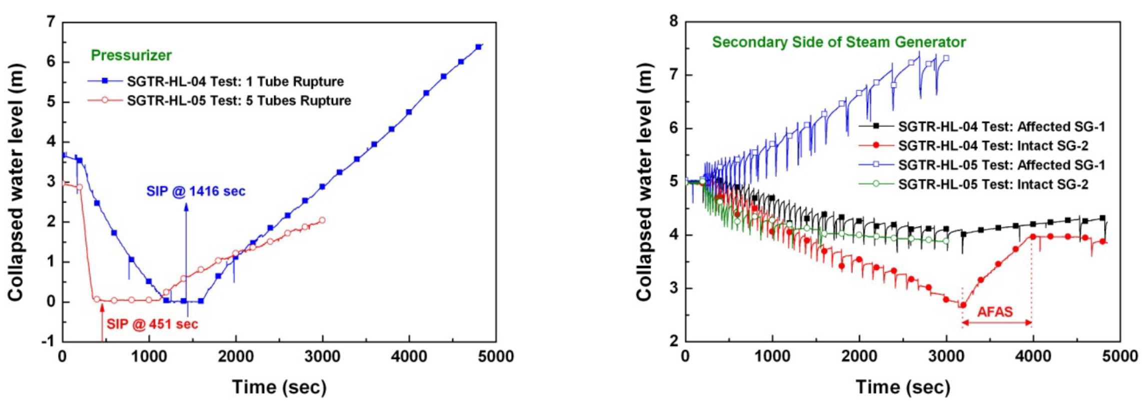


Figure 7 Variation of collapsed water level in the pressurizer and the steam generator.

Figure 7 shows the variation of the collapsed water level in the pressurizer and the secondary side of the steam generators. In the SGTR-HL-04 test, due to the break flow, the collapsed water level of the affected steam generator showed milder decrease than that of the intact steam generator even though there were level fluctuations resulting from the discharged flow through the MSSVs. Especially, in the SGTR-HL-05 test, the collapsed water level of the affected steam generator continuously increased and the affected steam generator became filled with water which could be attributed to the relatively large break flow. The supply of the auxiliary feedwater recovered the decrease in the water level of the intact steam generator after about 3160 seconds in the SGTR-04 test. However, the level of the SG-2 did not reach the set-point of the AFAS in the SGTR-HL-05 test. Contrary to the secondary side of the steam generators, the collapsed water level of the pressurizer continuously increased after the actuation of the SI water injection in both tests.

3.2 Variation of break flow rate

In the present tests, the break flow rate was directly measured with an orifice flow meter. The break flow discharged through the break nozzle can be two-phase mixture flow. On the other hand, since inner diameter of the external pipe is relatively large by 20.7 mm, the flow regime at the upstream of the break nozzle is estimated to be single-phase water. The orifice plate of the orifice flow meter causes a static pressure difference between the upstream and the downstream sides of the plate. The mass flow rate (m) can be obtained using Eq. (4) [7].

$$m = \frac{C}{\sqrt{1-\beta^4}} \varepsilon \frac{\pi}{4} d^2 \sqrt{2\Delta P \rho_1} \quad \Rightarrow \quad K \equiv \frac{2\Delta P}{\rho V^2} = \frac{1}{(C\varepsilon)^2} \cdot \left(\frac{1-\beta^4}{\beta^4} \right). \tag{4}$$

In this study, the orifice flowmeter was installed at the upstream of the break nozzle for the break flow to be single-phase water flow at the measurement location. Separate effect tests were performed to check a flow regime at the measurement location of the break flow. In the separate effect tests, the primary inventory under the high pressure and the high temperature conditions of 7.7 MPa and 292 ^{o}C was discharged to the atmosphere. The differential pressure corresponded to the differential pressure between the primary and the secondary systems in the present SGTR-HL-04 and SGTR-HL-05 tests. The discharged inventory was collected and its mass was measured by the load cell. The accumulated mass measured by the orifice flowmeter shows nearly the same value measured by the load cell. This result indicates that single-phase water was discharged at the measurement location of the orifice flowmeter in both cases of a single tube and five tubes ruptures.

Figures 8 and 9 show the variation of the accumulated break flow and the break flow rate, respectively. In this study, as a complementary method to the direct measurement of the break flow, a RCS inventory-based break flow estimation method was applied. This method is based on a mass balance of the reactor coolant inventory between a change in the RCS inventory and the ECC injection flow rate during the progression of the break. In Figure 8, the accumulated break flow estimated by the RCS inventory change is also presented for comparison. The accumulated mass of the break flow was similar in both cases as shown in Figure 8. Break flow rates show similar trends of the variation of the differential pressure between the primary and the secondary systems. As for the test condition of the present tests, the calculated break flow rates using the Henry-Fauske critical flow model [8] are 0.118 kg/s and 0.589 kg/s which are larger than the maximum break flow rates measured in the SGTR-HL-04 and the SGTR-HL-05 tests, respectively. It could be confirmed that the break flow was discharged as single-phase water at the location of the break flow measurement and the measurement accuracy was acceptable in the present tests.

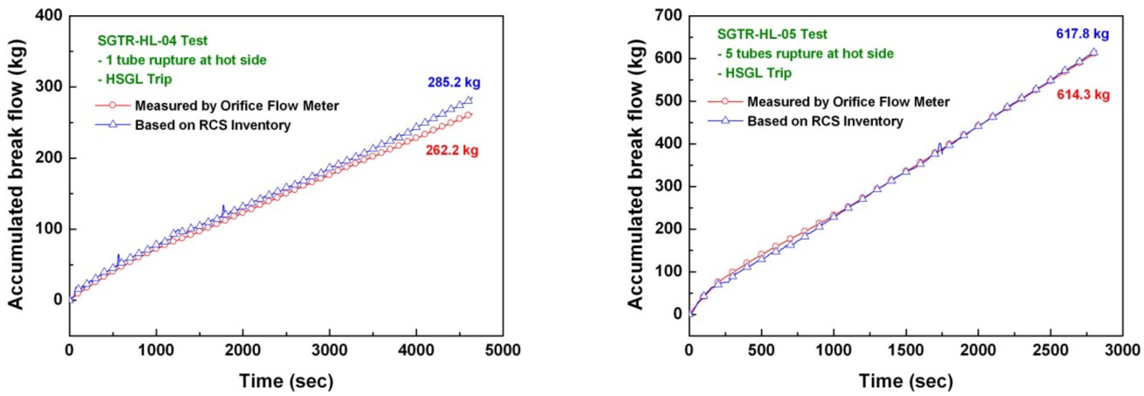


Figure 8 Variation of accumulated break flow.

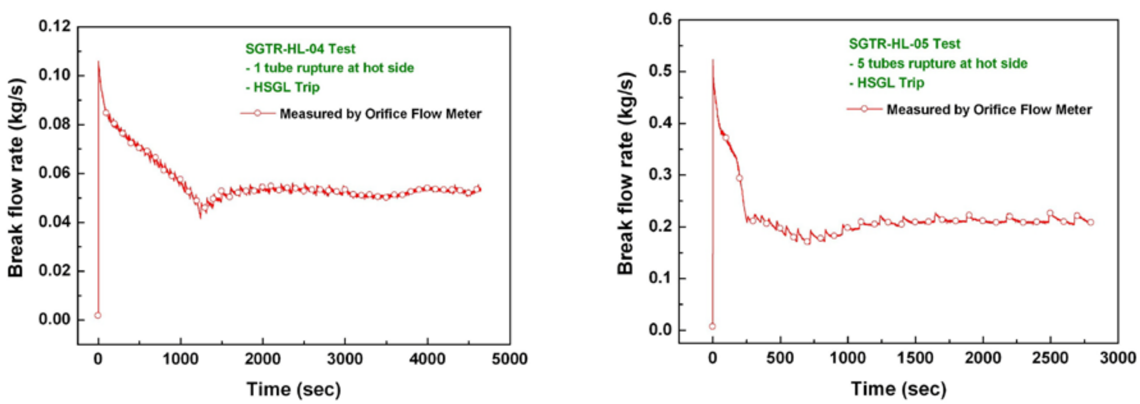


Figure 9 Variation of break flow rates.

4. Conclusion

In order to simulate the SGTR accident of the APR1400, integral effect tests were performed by simulating a double-ended rupture of a single and five U-tubes. Following the reactor trip, the primary system pressure decreased and the secondary system pressure increased until the MSSVs was opened to reduce the secondary system pressure. Break area affected the timings of the major events observed in the tests. On the contrary to the SGTR-HL-04 test, the MSSV of the intact steam generator (SG-2) was not opened any more after 1500 seconds in the SGTR-HL-05 test. Large discharge of the primary inventory resulted in rapid depressurization of the primary system and consequently early injection of the SIP in the SGTR-HL-05 test. Supply of cold ECC water by the SIPs reduced the energy transfer to the secondary side compared with the single U-tube rupture case. Meanwhile, the secondary pressure of the affected steam generator (SG-1) is more likely to increase due to higher break flow than the single U-tube rupture case. However, less heat transfer to the secondary side caused by earlier actuation of the SIPs had more influence on the secondary pressure of the affected steam generator than the break flow. This integral effect test data will be used to evaluate the prediction capability of existing safety analysis codes of the MARS and the RELAP5 as well as the SPACE code. Furthermore, this data can be utilized to identify any code deficiency for a SGTR simulation, especially for DVI-adapted plants.

5. References

[1] T. Kuroda et al., “Break Flow Modeling for a Steam Generator Tube Rupture (SGTR) Incident in a Pressurized Water Reactor (PWR),” JAERI-M 93-236, Japan Atomic Energy Research Institute, 1993.

[2] W.P. Baek, C.H. Song, B.J. Yun, et al. “KAERI Integral Effect Test Program and the ATLAS Design,” Nuclear Technology, Vol. 152, No. 183, 2005.

[3] K. Y. Choi et al., “Development of a Wall-To-Fluid Heat Transfer Package for the SPACE Code,” Nuclear Engineering and Technology, Vol. 41, No. 9, 2009, pp. 1143-1156.

[4] M. Ishii and I. Kataoka, “Similarity Analysis and Scaling Criteria for LWRs Under Single Phase and Two-Phase Natural Circulation,” NUREG/CR-3267, ANL-83-32, Argonne National Laboratory, 1983.

[5] Korea Electric Power Corporation, “KNGR Standard Safety Analysis Report,” Chapter 15, 2001.

[6] K. Y. Choi et al., “MARS Input Data for 8 % State Calculation of the ATLAS,” KAERI/TR-3046/2005, Korea Atomic Energy Research Institute, 2005.

[7] H. S. Bean, “Fluid Meters: Their Theory and Application,” The American Society of Mechanical Engineers, 1971.

[8] B. D. Chung et al., “MARS Code Manual Volume V: Models and Correlations,” KAERI/TR-3872/2009, Korea Atomic Energy Research Institute, 2009.