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Simulation of Multiple Steam Generator Tube Rupture (SGTR) Event Scenario

Kwang Won Seul, Young Seok Bang, and In Goo Kim
Korea Institute of Nuclear Safety
19 Guseong-dong, Yuseong-gu, Daejeon 305-338, Korea
k240skw@kins.re.kr

Taisuke Yonomoto and Yoshinari Anoda
Japan Atomic Energy Research Institute
2-4 Shirakata-shirane, Tokai, Ibaraki 319-11, Japan

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Abstract

The multiple steam generator tube rupture (SGTR) event scenario with available safety systems was experimentally and analytically evaluated. The experiment was conducted on the large scaled test facility to simulate the multiple SGTR event and investigate the effectiveness of operator actions. As a result, it indicated that the opening of pressurizer power operated relief valve was significantly effective in quickly terminating the primary-to-secondary break flow even for the 6.5 tubes rupture. In the analysis, the recent version of RELAP5 code was assessed with the test data. It indicated that the calculations agreed well with the measured data and that the plant responses such as the water level and relief valve cycling in the damaged steam generator were reasonably predicted. Finally, sensitivity study on the number of ruptured tubes up to 10 tubes was performed to investigate the coolant release into atmosphere. It indicated that the integrated steam mass released was not significantly varied with the number of ruptured tubes although the damaged steam generator was overfilled for more than 3 tubes rupture. These findings are expected to provide useful information in understanding and evaluating the plant ability to mitigate the consequence of multiple SGTR event.

Key Words : multiple steam generator tube rupture (SGTR) event, LSTF/ROSA-IV, RELAP5/MOD3.3, primary-to-secondary break flow

1. Introduction

The steam generator tube rupture (SGTR) can cause a direct flow path from the primary to the secondary system and result in the release of fission products into atmosphere. The consequence of the SGTR incident generally varies with the number of ruptured tubes, the availability of the safety systems, and timely operator actions following the event. Particularly, depending on the availability of the safety systems such as safety injection and auxiliary feedwater systems, the incident could be divided into two groups; one is the SGTR incident with emergency core cooling system (ECCS) and the other is the SGTR incident without ECCS. The former is a design basis accident of the nuclear power plant (NPP), and several experimental and analytical studies have been performed for the single and multiple SGTR incidents. In recent, the effectiveness of the emergency operating procedure (EOP) has been studied with different break sizes and event scenarios using the integral test facilities such as SEMISCALE [1], BETHSY [2], LSTF [3,4], and IIST [5]. They indicated that the intentional depressurization of the primary system combined with the secondary system cooling was effective in quickly terminating the incident. Meanwhile, the latter is a severe accident that can cause a core damage. These incidents also have been tested and analyzed with several test facilities [6,7,8], indicating that some alternative measures such as a large pressurizer relief valve or an operable accumulator are needed to mitigate the consequence of the incident without core uncovery.

The present study is focusing on the multiple SGTR event with an available ECCS. The event sequence is based on the real SGTR incident of the Mihama Unit 2 in Japan, occurred in 1991. The Mihama Unit 2 is a two-loop Westinghouse-type pressurized water reactor (PWR) with 500 MW electric power output. The incident was initiated by a double-ended break of a single steam generator U-tube. During the transient, the operator tried to open the power operated relief valve (PORV) on the pressurizer to stop the break flow by reducing the primary system pressure. However, the PORVs failed to open, and then the pressurizer auxiliary spray system was used instead. Thereafter, the primary system was depressurized and equalized to the secondary system pressure. This event scenario was simulated using the Large Scale Test Facility/Rig of Safety Assessment-IV (LSTF/ROSA-IV) in the Japan Atomic Energy Research Institute (JAERI) to investigate the detailed plant behavior [3].

The purpose of present study is to simulate the multiple SGTR event scenario using the LSTF/ROSA-IV facility and RELAP5 transient analysis code. Particularly, the effectiveness of operator action such as an opening of pressurizer PORVs during the multiple SGTR is experimentally investigated, and the predictability of the RELAP5 code and the coolant release into atmosphere during transient are also analytically studied. The event scenario in the LSTF experiment is similar to the single SGTR event occurred in Mihama Unit 2. However, the break size is modeled as an area of 6.5 ruptured U-tubes to simulate the multiple SGTR event, and the pressurizer PORV is used to depressurize the primary system pressure instead of the pressurizer auxiliary spray. In the analysis, a recent version of RELAP5/MOD3.3β code is assessed using the experiment data, and a sensitivity study is performed to investigate the effect of the number of ruptured tubes on the amount of the coolant release. This study is expected to provide useful information in understanding the plant responses and evaluating the current EOP against the single and multiple SGTR events.

2. Experiment on the Multiple SGTR Event

2.1. Experimental Facility

The LSTF/ROSA-IV test facility is a 1/48 volumetrically scaled model of a Westinghouse type four-loop PWR with 3,423 MWt [9]. The facility with the same elevation as the reference PWR includes a reactor pressure vessel, two symmetric primary loops, and secondary systems. Four primary loops of the reference PWR were represented by two loops, intact and broken loops, with an equal volume. Each primary loop includes an active steam generator with 141 full height U-tubes and a reactor coolant pump (RCP). The pressure vessel contains a core with full-length fuel of 1,104 rods simulating rod bundle, a cylindrical downcomer surrounding the core, upper and lower plena, and upper head. The core can simulate decay heat up to 14 % of the 1/48-scaled nominal PWR core power. The primary horizontal legs were sized to conserve the length-to-square root of diameter ratio ($L/\sqrt{D}$) as well as the scaled volumes to properly simulate the two-phase phenomena. The pressurizer with heater and spray system was connected to hot leg in the intact loop and scaled to conserve the length-to-diameter ratio as well as the scaled volumes.

In the secondary side, main feedwater system, boiler section, steam separator, steam lines and several valves such as relief and safety valves, main steam isolation valve (MSIV), turbine throttle and bypass valves were installed to simulate the similar steam flow to the reference plant. Also, the engineered safety systems such as a high pressure safety injection (HPSI) system connected to hot and cold legs and the auxiliary feedwater (AF) systems connected to each steam generator were installed to simulate the transient behavior.

The SG U-tube break in the reference plant was simulated using a break unit, which was placed outside the steam generator to measure break flow. As shown in Fig. 1, the break line was connected between the broken SG inlet plenum and the bottom of the secondary boiler section. This configuration represents that the U-tube rupture occurs near the steam generator tube sheet. A break orifice, quick-opening valve, and venturi flowmeter were installed on the break line. The break orifice diameter is 10.1 mm corresponding to the double-ended rupture of 6.5 U-tubes, which is equivalent to 1.0 % hot leg break.

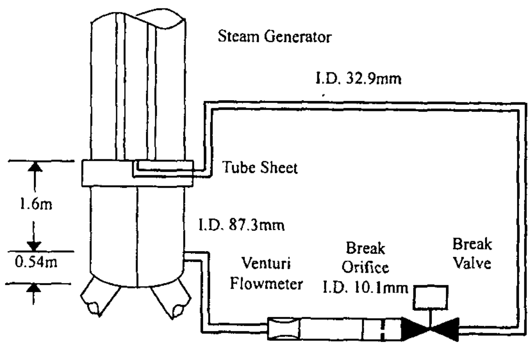

Fig. 1. Break Unit of LSTF

The LSTF also has more than 2,300 of instruments for measuring of major parameters at various locations. All signals are sent to data acquisition systems and recorded on magnetic disks with a digital form. The uncertainties of major parameters are approximately 0.53 % of full-scale range for pressure, 0.735 % for fluid temperature, 1.67 % for mass flow rate, and 0.32 % for water level.

2.2. Experimental Results

The initial conditions were set up with nominal operating conditions of a typical four-loop PWR, 15.52 MPa of a primary pressure and 589.3 and 559.9 K of hot and cold leg fluid temperatures in the intact loop. The core inlet flow rate was 14 % of the scaled nominal flow rate because of the limited core power capacity up to 14 % of the volumetrically scaled rated power, 10 MW. The SG secondary pressure and water temperature were 6.891 MPa and 558.2 K, respectively. The test sequence and logic were simulated based on

Table 1. Timing of Major Events After Tube Rupture

Major Events [seconds]	Test	Calculation
• Tube break	0	0
• Reactor trip signal	50.6	49.3
• Turbine trip	55.0	54.3
• Safety injection signal	66.2	63.5
• RCP speed control	130.6	129.3
• Main feedwater restart	110.2	107.5
• HPSI into cold legs	76.2	73.5
• HPSI into hot legs	366.2	363.5
• Auxiliary feedwater into SG-I	330.2	327.5
• Charging into cold legs	1371.4	1290.6
• SG-B MSIV close	650.6	649.3
• SG-B relief valve open/close	three times	three times
• SG-I relief valve open	650.6	649.3
• SG-I relief valve close	1250.0	1289.2
• Pressurizer PORV open	1146.0	1147.3
• Pressurizer PORV close	1277.0	1287.5
• Intact loop RCP restart	1847.6	1849.3
• Termination	3000.0	3000.0

the real incident of Mihama Unit 2 [10]. The core power and the rotational speed of RCP were controlled by a computer-controlled curve based on the plant data.

Table 1 shows the event sequences and timing of the major events in the experiment and calculation. The experiment was initiated by opening a valve downstream break orifice. The reactor trip signal was automatically taken place at 50.6 s after tube rupture due to low pressure in the pressurizer, despite the back-up heater on with 54 kW to pressurize the primary system. The SI signal was also automatically generated below 12.87 MPa and the SI flow started to inject into cold legs and hot legs at 76.2 s and 366.2 s, respectively. The RCP speed was controlled about 130.6 s to simulate loop mass flow of the primary system, and the main feedwater was restarted about 110.2 s for about 220 s to simulate the coolant inventory in the secondary system of the real plant incident. The affected SG was manually isolated at 600 s after reactor trip, and simultaneously the relief valve in the intact SG was opened for the primary system cooling. The pressurizer PORV was manually open at 1,146 s when water temperature in the hot leg was below 547.2 K. Thereafter, the pressure in the primary and secondary systems was quickly equalized and the primary-to-secondary flow via ruptured tubes was terminated. Finally, the SI flow was turned off when the pressurizer water level recovered above 1.0 m and the RCP in the intact loop was restarted about 1,797 s after reactor trip to recover the plant.

The test results indicated that overall responses to the multiple SGTR incident were similar to the single SGTR event [3] although it had faster evolutions of major sequences such as reactor trip, automatic SI actuation, and relief valve opening than in the single tube rupture event. It also indicated that the opening of the pressurizer PORV, as an operator action to mitigate the consequence of the event, was significantly effective in depressurizing the primary coolant system and providing a quick termination of the primary-to-secondary break flow. It means that the current EOP, although it was developed against the single SGTR event, is effective to mitigate the release of coolant to the environment even for the 6.5 tubes rupture incident. The details of thermal-hydraulic behavior during transient are discussed in section 3.2 with calculation results.

3. Analysis on the Multiple SGTR Event

3.1. Modeling and Calculation Conditions

The transient analysis code, RELAP5/MOD3.3β, which was recently released by U.S. Nuclear Regulatory Commission [11], was used to simulate the multiple SGTR event scenario. The RELAP5 is

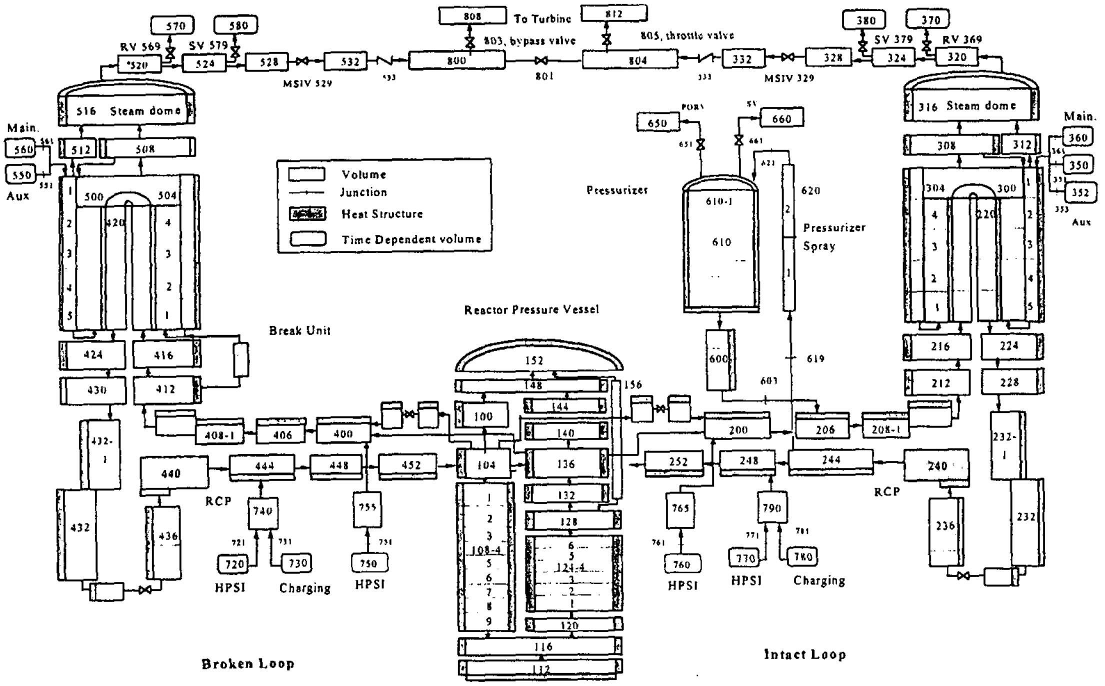

Fig. 2. LSTF Nodalization for RELAP5 Assessment

an internationally well-recognized best-estimate system analysis code, based on a non-homogeneous and non-equilibrium model for one-dimensional two-phase system. Basically, this code solves six field equations including constitutive models and correlations, and it has been improved in its models and correlations such as critical flow model, wall condensation, interfacial drag correlation, steam table, and numerical schemes. Main computer system used in calculation is a personnel computer with 700 MHz of CPU speed.

Figure 2 shows the nodalization to simulate the LSTF facility with RELAP5 code. The modeling is based on 177 hydraulic volumes connected by 186 junctions and 173 heat structures. The reactor pressure vessel elements (volumes 100 to 156) include the volumes corresponding to downcomer, lower and upper plena, core, upper head, and guide thimble channel. The core was simply modeled as a single channel with 6 hydraulic volumes.

Two primary loops are represented by an intact loop (volumes 200 to 252) and a broken loop (volumes 400 to 452) in a nearly symmetrical way. Each loop consists of a hot leg, SG inlet and outlet plena, U-tube, loop seal, RCP, and cold leg. The pressurizer is connected to hot leg in the intact loop through the surge line elements. Secondary sides of two SGs (volumes 300 to 399 and 500 to 599) are simulated for a downcomer, boiling section, steam separator and steam dome. Main steam line and header including the relief valves and turbine throttle and bypass valves are also modeled to simulate an accurate steam flow during transient.

Safety injection systems are modeled by time-dependent volumes and junctions, and connected to hot and cold legs to simulate the emergency

Table 2. Steady-State Conditions for Test and Calculation

Major Parameters (Intact/Broken)	Test	Calculation
• Core power [MWt]	10	10
• Primary pressure [MPa]	15.52	15.52
• Hot leg temperature [K]	589.3/588.4	589.9/589.9
• Cold leg temperature [K]	559.9/558.8	560.8/560.8
• Pressurizer water level [m]	2.713	2.705
• Loop mass flow [kg/s]	31.47/31.39	30.82/30.81
• Primary total mass [kg]	-	5578.1
• SI water temperature [K]	300.2	300.2
• Secondary pressure [MPa]	6.891/6.908	6.891/6.889
• SG water temperature [K]	558.2/559.2	557.8/557.8
• SG water level [m]	9.19/9.28	9.16/9.24
• SG total mass [kg]	-	2510.1/2530.1
• Main steam flow [kg/s]	2.65/2.63	2.767/2.748
• Feedwater temperature [K]	495.0/300.2	495.0/300.2

core cooling system. Secondary water feeding systems such as main and auxiliary feedwater systems are also modeled as time-dependent volumes and junctions. The break unit representing the multiple ruptured U-tubes is modeled with same configuration as the LSTF test, in which includes break valve and orifice modeled as junctions. All intact U-tubes are simply modeled as an averaged single channel.

Table 2 represents the comparison of initial conditions between the test and calculation. The calculated values were obtained from the steady-state run. Major parameters in the primary and secondary systems well agreed with the measured values.

Boundary conditions during transient were set up based on the specified operational set points and control logic performed in the experiment. Particularly, the power decay and RCP coastdown were modeled using the same curves as in the experiment, and the SI and auxiliary feedwater flow rates were modeled as a function of the system pressure based on the experimental data.

3.2. Comparisons of Analysis Result with Experiment

Figure 3 shows the pressure behavior measured and calculated in the primary and secondary systems during transient. A rapid pressure drop after tube rupture in the primary system and pressure cycling due to relief valve open and close in the secondary system are shown in both cases. Figure 4 shows the break flow rate through the ruptured tube, depending on the pressure difference between both systems. In overall, the calculations reasonably agree with the measured data. Particularly, the transient behavior in the broken SG such as three times of RV open and close was well predicted, and also the pressure behavior associated with operator actions such as the pressurizer PORV opening, was reasonably

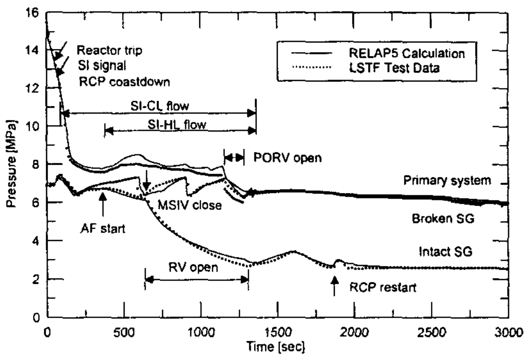

Fig. 3. Primary and Secondary System Pressures

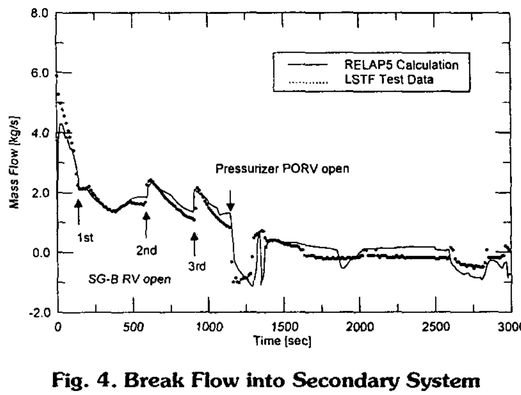

Fig. 4. Break Flow into Secondary System

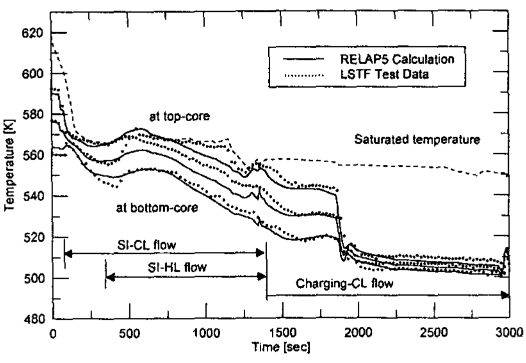

Fig. 5. Water Temperature in the Core Region

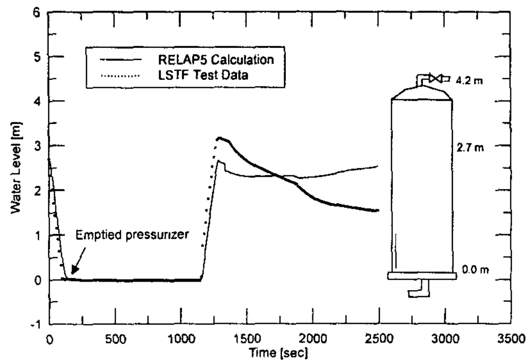

Fig. 7. Collapsed Water Level in the Pressurizer

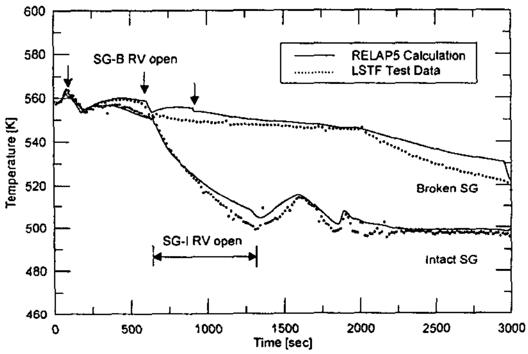

Fig. 6. Water Temperature in the Steam Generator Boiling Region

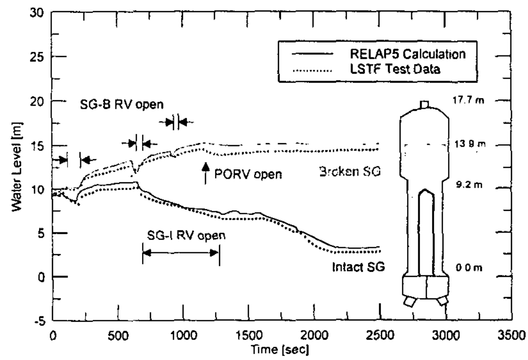

Fig. 8. Collapsed Water Level in the Steam Generator

calculated.

The pressure in the steam generators first increased in the early phase of transient due to the closing of the turbine throttle valves after turbine trip. Next, the pressure in the affected SG increased three times due to the break inflow from the primary system and heat transfer across the SG U-tubes. In particular, the third and fourth pressure increases were caused dominantly by the ascending water level because there was little heat transfer across the U-tubes due to the reduced loop flow after the RCP stopped.

The break flow, as shown in Fig. 4, remained high before the PORV opened, but it rapidly dropped and temporarily reversed after the PORV opened. Thereafter, the primary-to-secondary flow nearly stopped due to the equalized pressure between the both systems.

Figures 5 and 6 show the water temperature in the core region and boiling section of the steam generator, respectively. The overall trend of the calculated water temperature in the core region agreed well with the measured data despite the simple core model of one-dimensional single channel. This means that the safety injection flow and charging flow into the primary system were appropriately simulated. The decay heat in the core is removed by natural circulation flow to the secondary sides after the RCP stops, i.e., about 400 s. That core heat is transferred mostly to the intact SG across the U-tubes wall, and partially transferred to the broken SG through the break flow. Particularly, during the intact SG RV open interval, the heat flux across the U-tubes wall significantly increased, and then the coolant temperature in the core gradually decreased as shown in Fig. 5. It means that the primary coolant could be effectively cooled down by the RV open in the intact SG. Finally, the coolant temperature dropped by the forced circulation flow when the RCP restarted about 1,847.6 s.

Figures 7 and 8 show the collapsed water level in the pressurizer and both steam generators. The water level in the pressurizer rapidly dropped from nominal water level and emptied about 98.6 s after tube rupture. Despite the safety injection into the primary system, it was not recovered because of the large break flow and the coolant shrinking. After the pressurizer PORV opened, the water level started to rise quickly due to the rapid pressure drop. The calculation agreed well with the test data until the time that the water level in the pressurizer was recovered. Also, the trend of the water level calculated in the steam generators agreed well with the measured data.

In the test facility as a real plant, there is a steam separator at 13.9 m above the tube sheet of steam generator. In the present study, the time for the water to reach this level is considered as the steam generator is overfilled. This SG overfilling time was estimated about 777.6 s in the calculation, while 907.4 s in the experiment. A little faster overfilling in the calculation was due to the over-increase of the water level in the early phase of transient after the first RV close. Meanwhile, the temporary decrease during the second and third RV opens was reasonably predicted.

3.3. Discussion on the Stratified Flow in the Hot Leg

The experiment showed that some vapor in the pressurizer surged into the hot leg of the intact loop after the pressurizer was emptied. Thus, the hot leg was horizontally stratified about 100 s after tube rupture. This horizontal stratified flow regime with less than 0.1 of void fraction was also predicted in the calculation.

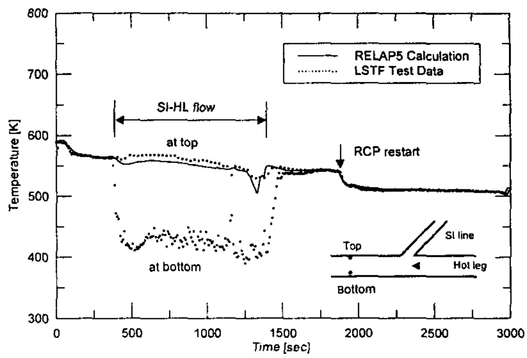

Fig. 9. Water Temperature in the Hot Leg of Intact Loop

However, the experiment also showed that the fluid temperature in the hot leg was thermally stratified due to the cold water injected from the SI line connected to upper part of the hot leg. As shown in Fig. 9, the measured water temperature in the upper part of the hot leg was significantly higher than that in the bottom part. Meanwhile, the calculation shows slightly lower averaged water temperature than the top fluid temperature. This thermally stratified flow occurred in the hot leg during safety injection period was not predicted in the calculation because of the limitation of the one-dimensional analysis model.

In the experiment, the pressurizer PORV manually opened to depressurize the primary system when the fluid temperature in the hot leg of the intact loop was below 547.2 K. It is one step of the emergency operating procedures. However, in the calculation, the PORV was modeled to open at a little lower fluid temperature, 543.4 K to compensate the one-dimensional model limitation. As a result, this modified set point delayed the PORV opening time about 120 s and gave closer calculation results to the experiment.

4. Sensitivity Analysis

4.1. Sensitivity Calculations

With the RELAP5 code assessed with the multiple SGTR test data, sensitivity study on the number of ruptured tubes was performed to investigate the plant responses in the viewpoint of coolant release into atmosphere. In general, the coolant release would increase as the number of ruptured tubes increases. Several valves such as relief valves, steam dump valves, or turbine bypass valves in the secondary side are known to be possible paths even though the release path depends on the plant design. Actually, the plant incident of the Mihama Unit 2 showed that the relief valves on the affected steam line was the major release path. Thus, in the present sensitivity calculation, the relief valve in the damaged SG was considered as a major coolant release path to environment.

Figure 10 shows the steam flow rate through the relief valves on the affected SG for the 6.5 tubes rupture (TR) in the experiment and calculation. It indicates that the RELAP5 code predicted well the RV open and close timings and its flow rate. It means that the pressure behavior of the damaged SG affecting the RV cycling was appropriately simulated as previously discussed in Fig. 3. In the sensitivity calculation on the number of ruptured tubes, 1 up to 10 tubes, the same event sequence as the test scenario and the same code options were applied. Therefore, the reactor and turbine trip signals and SI actuation signal are automatically generated depending on the number of ruptured tubes. The pressurizer heater and main feedwater are also automatically controlled until the reactor trip occurs. The auxiliary feedwater into the intact SG is also delivered automatically until the SG water level is overfilled. The sensitivity calculations are performed until the primary and secondary systems are balanced in pressure behavior.

4.2. Effect on the Number of Ruptured Tubes

As the number of ruptured tubes increases the primary-to-secondary break flow increases. The large break flow leads to rapid pressure drop in the primary system, which results in earlier reactor trip and safety injection actuation. Figure 11

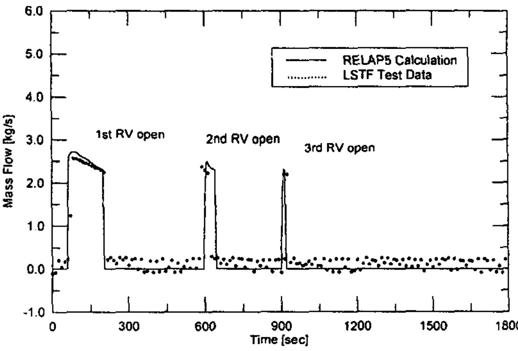

Fig. 10. Steam Flow Rate Through Relief Valve in the Broken SG

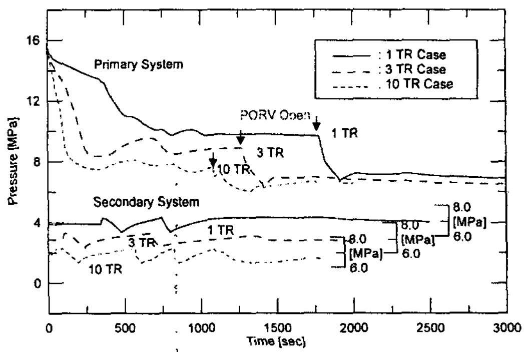

Fig. 11. System Pressure Behavior (Sensitivity Calculation)

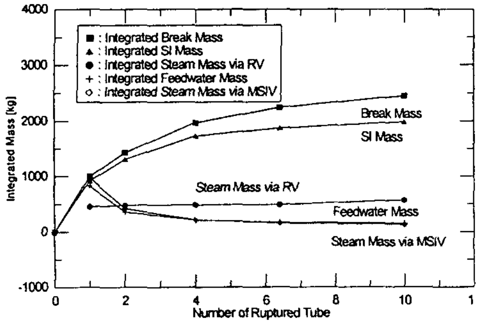

Fig. 12. Integrated Flow Mass (Sensitivity Case)

shows the pressure behavior with the number of ruptured tubes. The reactor trip for 1, 3, and 10 TR cases was estimated to occur at about 341.5, 92.0, and 38.9 seconds, respectively. After the reactor trip, the SI automatically starts and core is cooled. The SI flow rate also increases due to the low primary pressure. The larger the break size, the earlier the primary system cooldown and the pressurizer PORV open. The PORV opening time was estimated about 1,440, 1,184, and 1,094 seconds after reactor trip for 1, 3, and 10 TR cases, respectively. However, all cases of the pressure behavior in the secondary system were very similar except with different RV opening times and cycling in the damaged SG. Actually, three times of RV open were calculated in the 10 TR case while two times in the 1 TR or 3 TR cases.

Figure 12 shows the integrated masses on the number of ruptured tubes until the PORV open. There is no more relief valve open and coolant release from the secondary side after the PORV open. The figure shows that the steam outflow through the turbine throttle valve was balanced to the main feedwater flow before the turbine trip. And the integrated break and SI masses increased as the number of ruptured tubes increased. It also indicates that the integrated steam mass through the RV was not significantly varied with the number of ruptured tubes even though there was a difference in the number of RV cycling during transient. It is because most of steam was discharged out during the first and second RV open interval as shown in Fig. 10. However, the water level in the damaged SG was estimated about 11.9, 13.9, and 15.6 m for 1, 3, and 10 TR cases, respectively. It means that the affected SG was overfilled for the cases of more than 3 TR.

The LSTF facility represents a full-height and volumetrically large scaled-down facility, but the fluid volume-to-surface ratio is smaller than in the full-sized plant. Also, the four-loop PWR was simply modeled as two symmetrical loops and the break unit was installed outside the steam generator. These LSTF specific configurations could give a little different response to the SGTR incident from the real plant. Actually, the experimental data, obtained from the simulation of the real SGTR incident occurred in the Mihama Unit 2 plant [3], showed that the number of RV cycling in the damaged SG was not same as the real event, i.e., three times in the plant incident but once in the experiment. Then, it should be limited in directly applying the present study results to real NPPs.

5. Conclusions

The multiple SGTR event scenario with available safety systems was evaluated experimentally and analytically. The experiment was conducted on the LSTF/ROSA-IV facility to simulate the multiple SGTR event scenario and investigate the effectiveness of operator action to depressurize the primary system. As a result, it indicated that the opening of the power operated relief valves on the pressurizer was significantly effective in quickly terminating the primary-to-secondary break flow even for the 6.5 tubes rupture.

In the analysis, the recent version of RELAP5 code was assessed with the test data. It indicated that the transient calculation agreed well with the measured data and that the plant responses such as the water level and relief valve cycling in the affected steam generator were reasonably predicted. However, the code with one-dimensional model showed some limitations in predicting the multi-dimensional phenomena such as a thermal stratification in the hot leg.

Finally, the sensitivity study on the number of ruptured tubes, 1 up to 10 tubes rupture, showed that the coolant release into atmosphere was not strongly dependent on the break sizes. Particularly, the integrated steam mass released was not significantly varied with the number of ruptured tubes, although the damaged steam generator could be overfilled for more than 3 tubes rupture.

These findings may have a limitation in directly applying to the real power plant because of the scalability of the test facility, the model limitations of the calculation, and the disign differences. However, it is expected to provide useful information in understanding the plant responses to the multiple SGTR event and evaluating the effectiveness of operator actions to mitigate its event consequences.

Nomenclature

AF : Auxiliary Feedwater
ECCS : Emergency Core Cooling System
EOP : Emergency Operating Procedure
HPSI : High Pressure Safety Injection
IIST : INER Integral System Test
LSTF : Large Scale Test Facility
LOBI : LWR Off-normal Behavior Investigators
MSIV : Main Steam Isolation Valve
NPP : Nuclear Power Plant
PORV : Power Operated Relief Valve
PWR : Pressurized Water Reactor
RCP : Reactor Coolant Pump
RELAP : Reactor Excursion and Leak Analysis Program
RV : Relief Valve
ROSA-IV : Rig of Safety Assessment - IV
SG-I/B : Intact/Broken Steam Generator
SGTR : Steam Generator Tube Rupture
SI-HL/CL : Safety Injection into Hot/Cold Legs
TR : Tube Rupture

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