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Steam Line Break Analysis Using RELAP5/MOD3.3 for Steam Generator Blowdown Load Assessment

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ABSTRACT

The RELAP5/MOD3.3 is generally used for best-estimate transient simulation of light water reactor coolant systems during postulated accidents in the Light Water Reactor (LWR). The RELAP5/MOD3.3 code is based on a non-homogeneous, and non-equilibrium model for a two phase system that is solved by a fast, partially implicit numerical scheme to permit economical calculation of system transients. This code is suitable for the analysis of transients and postulated accidents in LWR systems, including both large- and small-break loss of coolant accidents, as well as for the full range of operational transients.

For the evaluation of structural integrity for the steam generator in the Pressurized Water Reactor (PWR), the postulated accidents, such as the Steam Line Break (SLB) in the Advanced Power Reactor (APR1400) at the Korean domestic plants, are considered Design Basis Events (DBE). In order to evaluate the structural integrity of a steam generator during the SLB, the data for the thermo-hydraulic velocity, density and pressure are needed.

This study was performed to calculate thermal hydraulic parameters, such as thermo-hydraulic velocity, density and pressure, using the RELAP5/MOD3.3 code for the structural evaluation of the steam generator internals during the postulated SLB accidents.

The calculation results were verified by comparing with experimental data generated from the experimental facility ATLAS (Advanced Thermal-Hydraulic Test Loop for Accident Simulation)

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EXECUTIVE SUMMARY

The RELAP5/MOD3.3 is a computer code for best-estimate transient simulation of light water reactor systems during the postulated accidents. In this study, the possibility of the application of the postulated SLB accident analysis for a steam generator was investigated using the RELAP5/MOD3.3. For the verification and validation of the code analysis results, experimental tests were performed in the thermal-hydraulic integral effect test facility, ATLAS (Advanced Thermal-hydraulic Test Loop for Accident Simulation) at KEARI, located in Daejeon, Korea. The code analysis results for the thermal hydraulic response during the SLB accidents were compared to those of experimental tests. Some sensitivity studies were also performed.

From this study, major results of the analysis and experiments are summarized below.

- The RELAP5/MOD3.3 predicts well the thermal hydraulic behaviors of the ATLAS SLB experimental test.
- The dynamic pressure estimated with the RELAP5/MOD3.3 is slightly conservative compared to that of the ATLAS SLB experimental test data.
- The following conclusions are drawn from the results of the sensitivity studies.
- The Henry-Fauske critical flow model is recommended.
  - The break valve opening time of 1.0 milli-sec is recommended.
  - A maximum time-step-size less than or equal to or 10^{-4} sec is recommended.
  - The nodalization effect in the components around the steam line throat is
  - Negligible

ABBREVIATIONS AND ACRONYMS

ATLAS	Advanced Thermal-Hydraulic Test Loop for Accident Simulation
APR1400	Advanced Power Reactor 1400 MWe
SG	Steam Generator
DAS	Data Acquisition System
SLB	Steam Line Break
KAERI	Korea Atomic Energy Research Institute
MARS-KS	Multi-dimensional Analysis of Reactor Safety
RCS	Reactor Coolant System
RELAP	Reactor Excursion and Leak Analysis Program
US NRC	United States Nuclear Regulatory Commission

1 INTRODUCTION

The light water reactor (LWR) transient analysis code, RELAP5, was developed at the Idaho National Engineering Laboratory (INEL) for the U.S. Nuclear Regulatory Commission (NRC). Uses of code include analyses required to support rulemaking, licensing audit calculations, evaluation of accident mitigation strategies, evaluation of operator guidelines, and experiment planning analysis. RELAP5 has also been used as the basis for a nuclear plant analyzer.

The mission of the RELAP5/MOD3.3 development program was to develop a code version suitable for analysis of all transients and postulated accidents in LWR systems, including both large- and small-break loss-of-coolant accidents (LOCAs) as well as the full range of operational transients.

This study focuses on the applicability of the RELAP5/MOD3.3 for a guillotine break transient showing very sudden change of thermal hydraulic conditions in the system in a short period of time. That is, the code is analyzed to check whether or not it predicts well the rapid thermal hydraulic response in an SLB accident.

From a structural integrity point of view, the pressure difference between the internals of the steam generator is a critical parameter because the pressure difference is the only structural load during the guillotine break accidents. To verify the pressure difference inside the steam generator, the dynamic pressure near the break line was investigated and analyzed in relation to experimental test data.

2 DESCRIPTION OF THE ATLAS SG SLB EXPERIMENT

Experimental tests were performed to verify the RELAP5/MOD3.3 results. The tests were performed at a special facility for testing the thermal hydraulic integral effect. Transducers were installed to check accurately the dynamic pressure data. The experimental tests proceeded to reach a steady state condition and then the break was simulated with data logging. During the test, the major thermal-hydraulic parameters, such as dynamic and static pressures, local temperatures, and flow rates, were obtained in the course of an abrupt break of the steam generator steam line using the double rupture disk assembly. Also, the reproducibility of the test was checked by doing additional test cases observing the characteristics of the dynamic pressure during the tests. Details are shown in the following subsections.

2.1 Experimental Facilities

A thermal-hydraulic integral effect test facility, ATLAS (Advanced Thermal-hydraulic Test Loop for Accident Simulation), was used to perform tests for a steam line break in the steam generator. ATLAS has the same two-loop features as the APR1400 (Advanced Power Reactor 1400 MWe); but is a half-height and 1/288-volume scaled test facility with respect to APR1400. The fluid system of the ATLAS consists of a primary system, a secondary system, a safety injection system, a break simulating system, a containment simulating system, and auxiliary systems. Figure 2-1 shows a 3-dimensional view of the ATLAS.

The ATLAS has two steam generators and each steam generator consists of a lower plenum, a U-tube assembly, middle and upper SG vessels, two downcomer pipes, and other internals as shown in Figure 2-2.

A steam line break was simulated by installing a break spool piece in one of the steam line in the SG-1. Figure 2-3 shows the configuration of the break simulation system for the steam line break of the ATLAS steam generator. The break opening time is the most crucial factor influencing a blow-down load during a steam line break, so it should be simulated appropriately in the test. In order to make the break opening time as short as possible, a double rupture disc assembly was used in the test. Figure 2-4 shows the configuration of the double rupture disc assembly which consists of two rupture discs having different cutoff pressures for actuation.

The double rupture disc assembly works as followings: rupture disc-2 will be opened at first when the pressure in the “intermediate region” is increased up to a specified actuation pressure by operators. Subsequently, rupture disc-1 will be opened within very short period of time (about 1 milli-sec) by the driving force resulting from opening rupture disc-2.

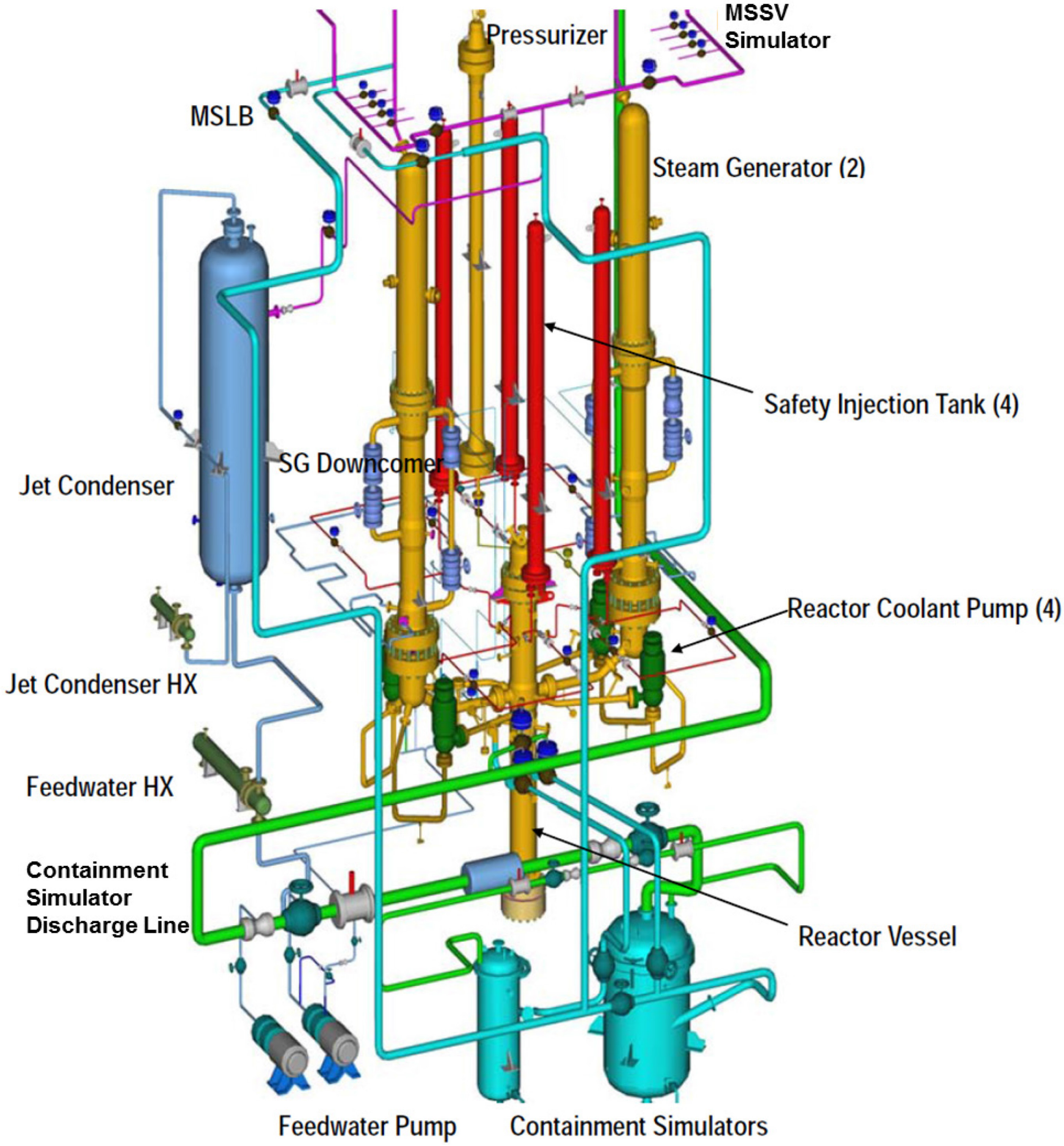

Figure 2-1 Major Component of the ATLAS

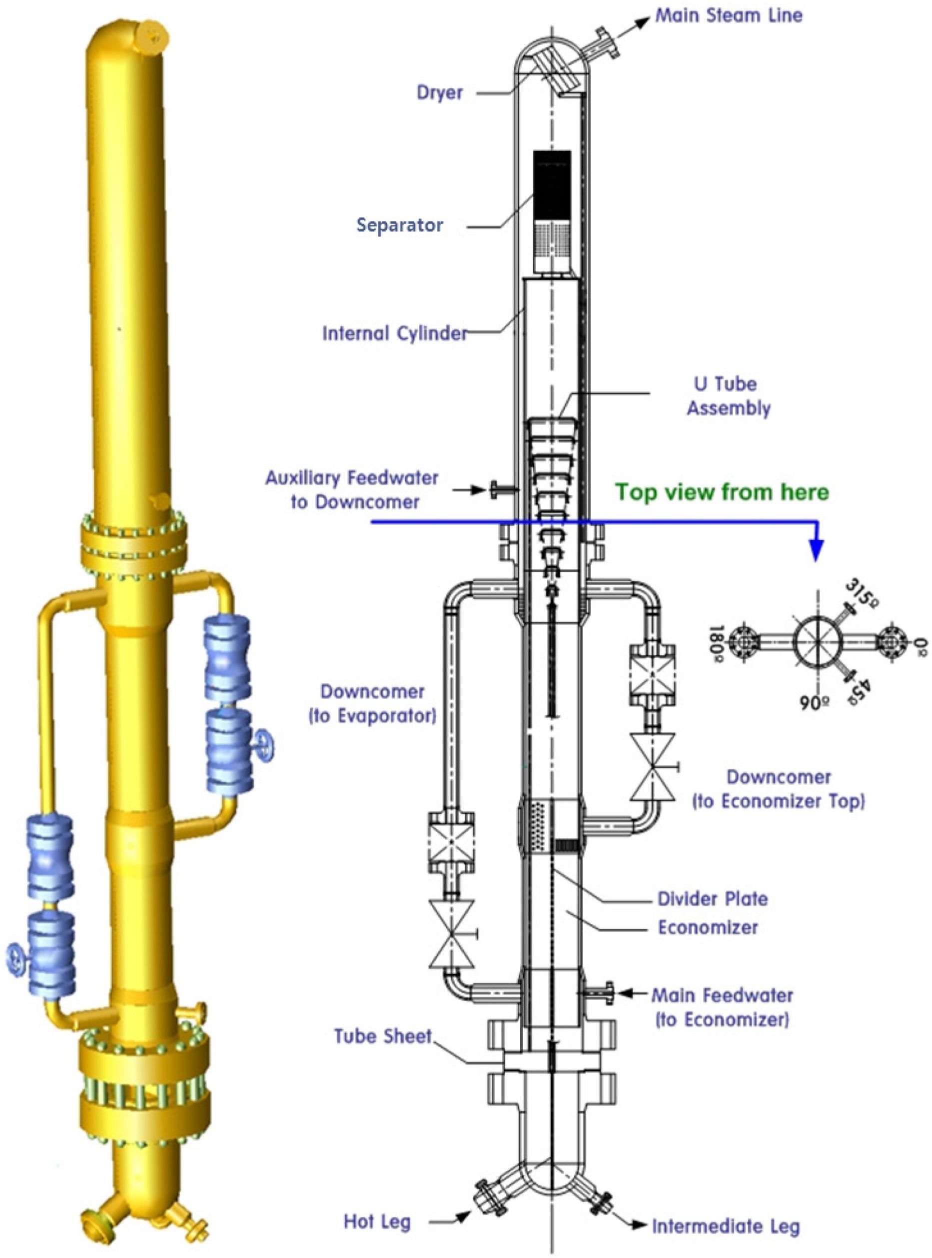

Figure 2-2 Steam Generator of the ATLAS

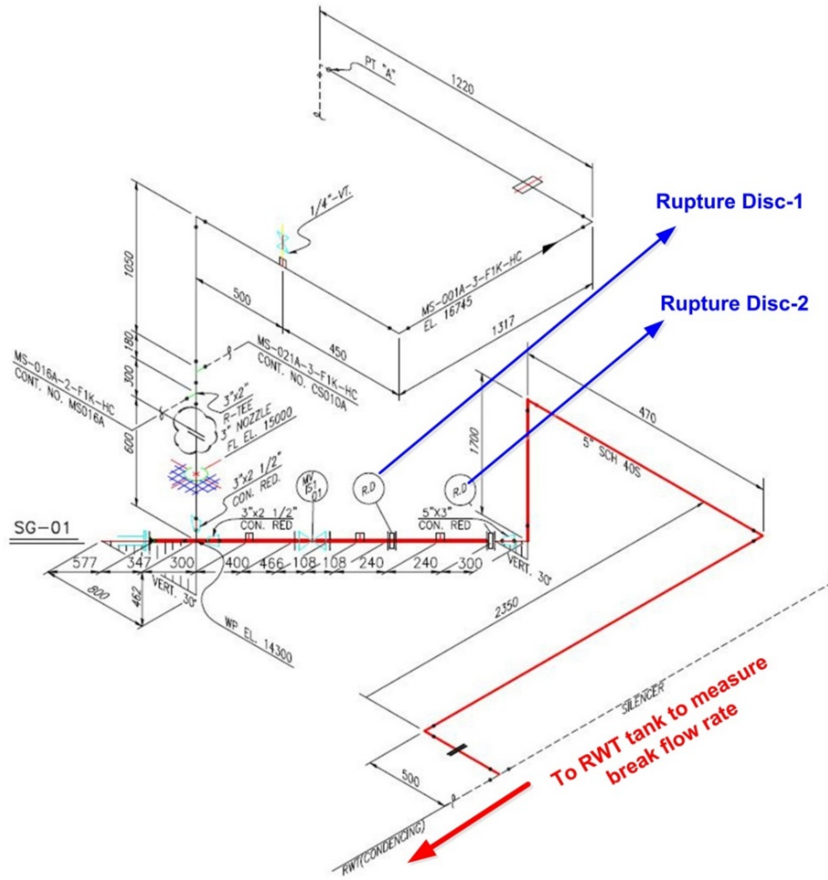

Figure 2-3 Configuration of the Break Simulation System

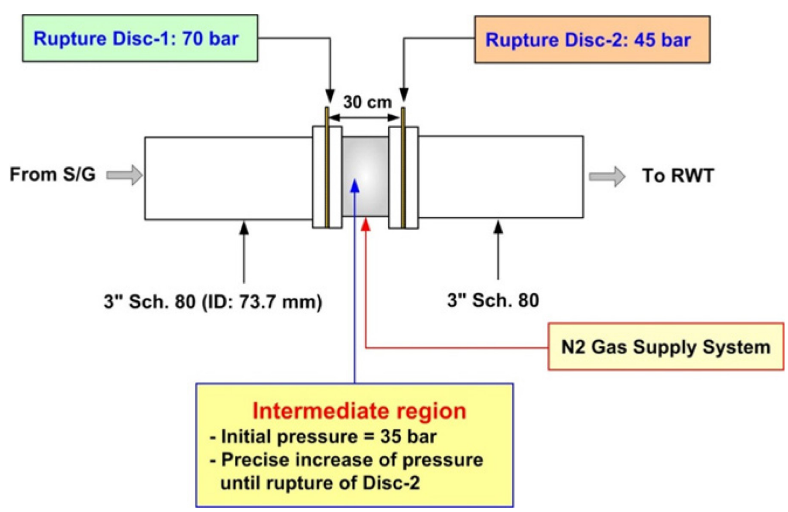

Figure 2-4 Configuration of the Double Rupture Disc Assembly

2.2 Instrumentation

In the ATLAS test facility, 1,300 instrument were installed for measurement of several thermal-hydraulic parameters in the components. The measuring locations that would be affected most after the break were selected. Table 2-1 shows a list of the measuring locations, parameters, and tag name of sensors. Figure 2-5 shows the locations of measurement ID in the ATLAS steam generator.

The dynamic pressure was measured using Kistler dynamic pressure transducers with a measurement frequency of 10000 Hz. In order to precisely estimate the rupture time of the discs, additional dynamic pressure transducers (Dynamic-P-04 and Dynamic-P-05) were installed in the pipe line of the break simulation system as shown in Figure 2-6.

shows the analysis uncertainty levels of each group of instruments.

Table 2-1 List of Measuring Locations and Parameters

ID	Location	Dynamic Pressure	Static Pressure	Fluid Temperature	Mass Flow rate
1	Outside of main steam line				QV-MS1-01
2	Top of steam dome	Dynamic-P-02	PT-SGSD1-01		
3	Outlet region of steam separator	Dynamic-P-01		TF-SGSD1-03	
4	Downcomer feedwater line			TF-MF1-02	QV-MF1-02
5	Downcomer cold side			TF-SGDC1-04	
6	Downcomer hot side			TF-SGDC1-02	
7	Economizer feedwater line			TF-MF1-01	QV-MF1-01
8	Inside of lower plenum from hot leg			TF-SGP1-01	
9	Inside of lower plenum to cold leg			TF-SGP1-02	
10	Hot leg side		PT-HL1-01		QV-HL1-01B
11	Cold leg side				QV-CL1A-01B QV-CL1B-01B

Table 2-2 Uncertainty Level of Instruments

Items	Unit	Uncertainty
Static Pressure	MPa	0.039 %
Dynamic Pressure	bar	1.02 %
Differential Pressure	kPa	0.23 %
Collapsed Water Level	m	0.17 %
Temperature	^{o}C	maximum 2.4 ^{o}C
Flow rate	kg/s	0.053 %

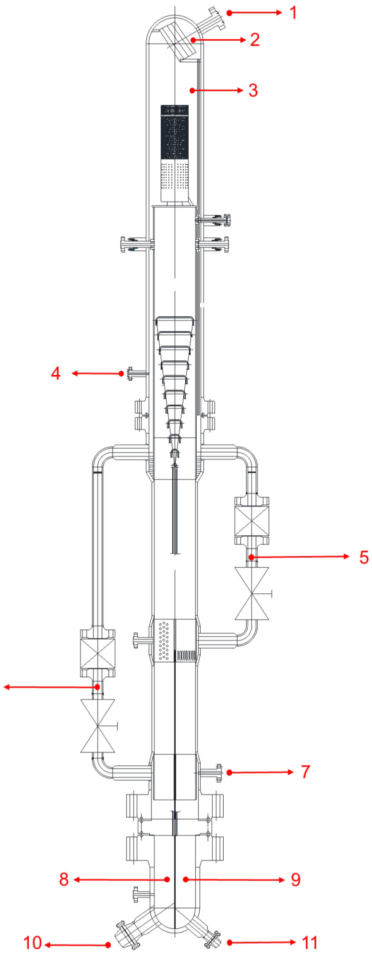

Figure 2-5 Locations of Measurement ID in the Steam Generator of the ATLAS

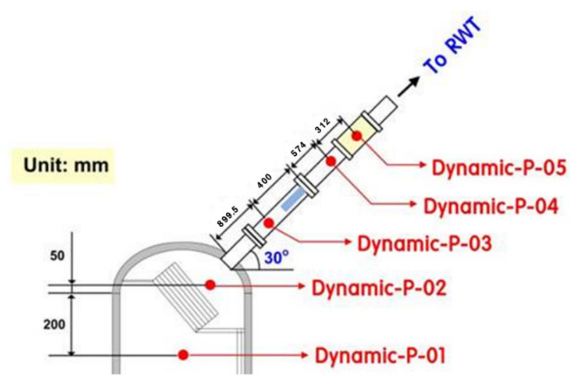

Figure 2-6 Detailed Locations of Dynamic Pressure Sensors Installed in the SG

2.3 Experimental Procedure

After a steady state condition was achieved in both primary and secondary systems, a break in the steam generator steam line was simulated according to the following procedures.

- ATLAS main data is logged.
- After about 10 minutes from start of the ATLAS data logging, sub-DAS is started to acquire the dynamic pressure data in the steam generator.
- After a few sec from the start of the sub-DAS data logging, OV-MSIV1-01 valve is closed to block the main steam line from the break simulation line.
- After a few sec from closing of the OV-MSIV1-01 valve, nitrogen gas is supplied into the “Intermediate region” as shown in Figure 2-4 to rupture the disc-2. Subsequently, the disc-1 is opened by the driving force resulting from rupture of the disc-2.
- After acquiring experimental data for about 30 sec, Sub-DAS and ATLAS data logging are stopped.

The major events of the tests is listed in Table 2-3.

Table 2-3 Sequence of Major Events

Events	Time (sec)	Description
Heat Up	~ 18,000	Heating up process
Test Standby	~ 600	Steady state condition
Test Start	0.0	ATLAS main data logging start
	600.0	Sub-DAS data logging start (Dynamic Pressure)
	607.5	Blocking of main steam line
Break	613.0	SG-1 steam line break 100% open
Test End	630.0	Data recording stop

2.4 Experimental Conditions

The tests aimed to obtain major thermal-hydraulic parameters, such as dynamic and static pressures near the break location, local temperatures, and flow rates during the steam line break of the steam generator. Two tests, named SLB-DS-01 and SLB-DS-02, were performed using the double rupture disc assembly in order to simulate a sudden break of the steam generator steam line. The reason for performing SLB-DS-02 was to confirm the reproducibility of the SLB tests, and the reproducibility of the tests was confirmed by comparing the SLB test results. Therefore, only the SLB-DS-01 data was used to compare with the RELAP5/MOD3.3 calculation results.

Table 2-4 presents the actual initial conditions measured in these two tests. In

Table 2-4, STDEV stands for standard deviations of each designated values. Even though there are some discrepancies between the target values and the measured values, the measured initial conditions of the primary and secondary systems are acceptable considering the standard deviations of each value, and the characteristics of the integral effect test.

Table 2-4 Actual Initial Parameters of the ATLAS Steady-State Condition

Design Parameter	Target Value	Measured Value SLB-DS-01 (Value(1)/STDEV(2))	Measured Value SLB-DS-02 (Value(1)/STDEV(2))	Remark (Sensor ID)
REACTOR PRESSURE VESSEL				
Normal Power (MW)	1.56	1.630 / 0.0008	1.624 / 0.0006	Heat loss: about 80 kW
Pressurizer Pressure (MPa)	15.5	15.5 / 0.006	15.5 / 0.008	PT-PZR-01
Core Inlet Temperature (^oC)	290.7	289.8 / 0.17	290.0 / 0.15	TF-LP-02G18
Core Outlet Temperature (^oC)	324.2	325.6 / 0.14	325.7 / 0.13	TF-CO-07G14,18,21,25
STEAM GENERATOR				
Steam Flow Rate (kg/s)	0.444	0.382 / 0.007 0.418 / 0.001	0.377 / 0.006 0.409 / 0.004	SG-1 (QV-MS1-01) SG-2 (QV-MS2-01)
Feed Water Flow Rate (kg/s)	0.444	0.434 / 0.003 0.426 / 0.002	0.397 / 0.007 0.403 / 0.007	SG-1 (QV-MF1-01, 2) SG-2 (QV-MF2-01, 2)
Feed Water Temperature (^oC)	232.2	234.2 / 0.16 233.6 / 0.17	233.1 / 0.13 232.6 / 0.13	SG-1 (TF-MF1-01) SG-2 (TF-MF2-01)
Steam Pressure (MPa)	7.83	7.82 / 0.009 7.82 / 0.009	7.84 / 0.005 7.84 / 0.005	SG-1 (PT-SGSD1-01) SG-2 (PT-SGSD2-01)
Steam Temperature (oC)	293.5	295.2 / 0.11 295.4 / 0.09	295.4 / 0.08 295.5 / 0.08	SG-1 (TF-SGSD1-03) SG-2 (TF-SGSD2-03)
PRIMARY PIPING				
Cold Leg Flow (kg/s)	2.0	1.94 / 0.016	1.90 / 0.017	Cold leg average (QV-CL1A,1B,2A,2B-01B)

Notes)

(1) Average Value at -10 to 300 sec

(2) STDEV: Standard Deviation at -10 to 300 sec

3 RELAP5 INPUT MODEL

3.1 ATLAS Facility Steam Generator RELAP5 Input Model

The RELAP5/MOD3.3 input model was prepared from the MARS-KS code input model originally created at KAERI. The MARS-KS code input model consists of a reactor pressure vessel, primary piping, steam generators, a pressurizer, steam lines, a safety injection system, feedwater and the turbine system, and reactor coolant pumps. Figure 3-1 shows the MARS-KS code input model nodalization.

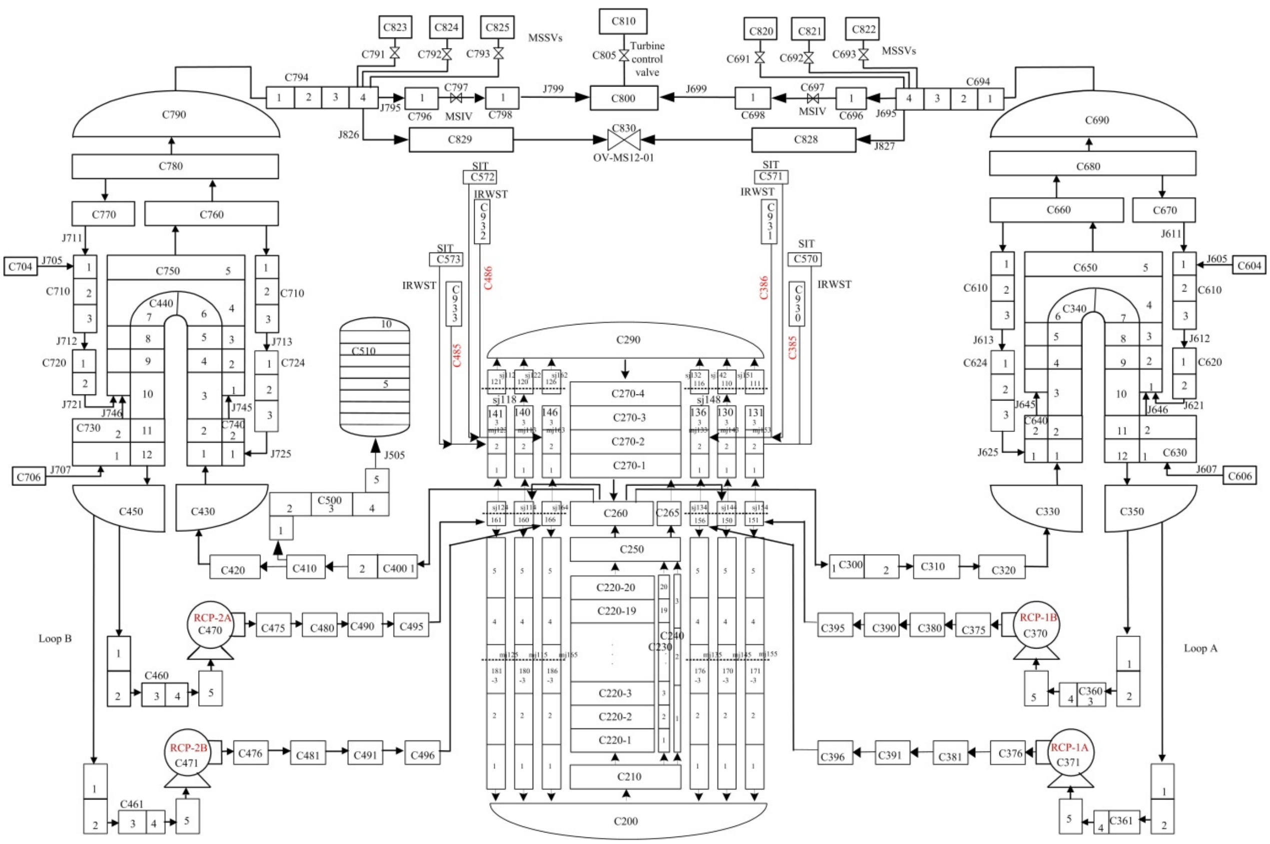

Figure 3-1 MARS-KS Code ATLAS Nodalization

To simulate the steam line break test, the RELAP5/MOD3.3 input model took one steam generator from the MARS-KS code input model and modified it accordingly for analysis. Nodalization of the RELAP5/MOD3.3 input model for the ATLAS steam line break is shown in Figure 3-2. The model is based on 78 volumes connected by 38 junctions and 15 heat structures.

The primary side of the steam generator consists of primary inlet/outlet plenum and U-tubes.

- The primary inlet/outlet plenums (C330/C350) are modeled as BRANCH components and are connected to RCS loops (C300/C390/C391) and the primary side of the U-tubes (C340). RCS loop are modeled as TIME DEPENDENT VOLUME components and used as the boundary condition in the primary side of the steam generator.
- The U-tubes consist of 12 volumes in a PIPE component.

The secondary side of the steam generator consists of economizer, evaporator, riser, downcomer, hot and cold side downcomer pipe, separator, bypass, steam dome, economizer feedwater line, donwcomer feedwater, main steam line and steam break line.

- The economizer (C630) is modeled as a PIPE component and has 2 volumes.
- The evaporator is divided into the hot side evaporator (C640/C650) and the cold side evaporator (C651). The hot side evaporator is modeled as a PIPE component and has 6 volumes and the cold side evaporator is also modeled as a PIPE component and has 4 volumes. The hot/cold side evaporators (C650/C651) are also connected by cross flow which is modeled as a MULTIPLE JUNCTION component.
- The riser (C659) is modeled as a BRANCH component connected to the evaporator and the separator.
- The downcomer (C610) is modeled as an ANNULUS component and has 3 volumes.
- The hot/cold side downcomer pipes (C624/C620) are modeled as PIPE components and have 3 and 2 volumes, respectively.
- The separator (C660) is modeled as a SEPARATOR component.
- The bypass (C670) is modeled as a SINGLE VOLUME component.
- The steam dome (C680/C690) is modeled as BRANCH components.
- The economizer feedwater line consists of an economizer box (C615) and economizer feedwater pipes (C617/C618). The economizer box and the economizer feedwater pipes are modeled as PIPE components and consists of 14 volumes. The economizer feedwater pipes are connected to TIME DEPENDENT VOLUME components (C700/C701) used as boundary condition to the economizer feedwater line.
- The downcomer feedwater (C604) is modeled as a TIME DEPENDENT VOLUME component used as boundary condition to the downcomer feedwater and connected to the downcomer by a TIME DEPENDENT JUNCTION component (J605).
- The main steam line (C800/C810/C694/C696/C698) is modeled as SINGLE VOLUME components and a TIME DEPENDENT VOLUME component used as boundary condition to the main steam line connected by a VALVE component. The main steam line consists of 5 volumes.
- The steam break line (C910/C920/C930/C940) is modeled as SINGLE VOLUME components, a PIPE component, and a TIME DEPENDENT VOLUME component and has 15 volumes. A TIME DEPENDENT VOLUME component (C940) provides the boundary condition as the atmosphere and is connected by a VALVE component (J931).

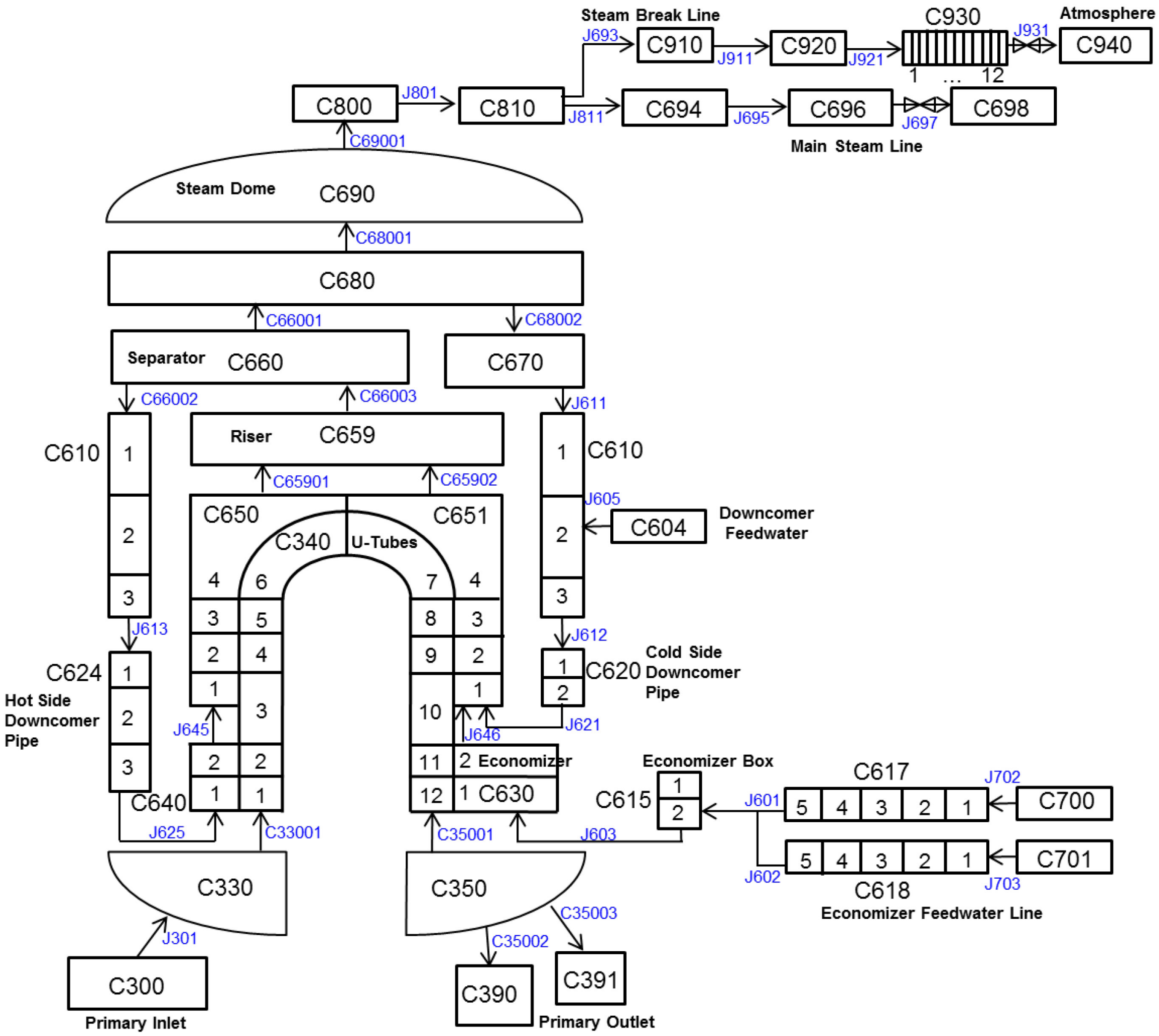

Figure 3-2 The ATLAS Steam Generation RELAP5/MOD3.3 Nodalization

4 RELAP5 ANALYSIS RESULTS

4.1 Steady State Calculation

In order to achieve a stable initial condition, the steady state calculation was performed for 3000 sec. The following controllers were used for the first 3000 sec:

- The downcomer feedwater flow rate proportional controller
- The economizer feedwater flow rate proportional controller
- The steam dome pressure proportional controller

The other controlled parameters (feedwater temperature, primary coolant temperature, primary coolant pressure, and primary coolant flow rate) were entered as boundary conditions.

Table 4-1 shows the calculated parameters compared to the experiment. The calculated steam dome pressure was a little different from that of experimental value. Because the measured steam dome pressure was lower than the saturation pressure at the given steam dome temperature, the saturation pressure at that given steam dome temperature was used as the steam dome pressure in the steady state calculation. The calculated steam flow rate was a little different from that of experimental value. Although the calculated steam and feedwater flow rates were equal, their measured values were not equal. It is noted that the instrument used for the water flow rate is more accurate than for the steam flow rate. Therefore, the calculated steam and feedwater flow rates were adjusted to the experimentally measured feedwater flow rate.

Once a stable condition was obtained, the downcomer and economizer feedwater controllers were de-activated and replaced by a relevant boundary condition for the transient analysis.

Table 4-1 Steady State Results

Parameters	Measured(1)	Calculated
Primary system		
Hot leg pressure (MPa)	15.571±0.006	15.571
Inlet plenum temperature (K)	599.3±2.4	598.7
Outlet plenum temperature (K)	566.4±2.4	564.3
Cold leg flow rate (kg/s)(2)	1.8975±0.001	1.8975
Secondary system		
Thermal power (MW)	0.749	0.749
Steam dome pressure (MPa)	7.812±0.003	8.013
Steam dome temperature (K)	568.3±2.4	568.3
Steam flow rate (kg/s)	0.382±0.0002	0.42822
Economizer feedwater temperature (K)	507.3±2.4	507.3
Downcomer feedwater temperature (K)	503.5±2.4	503.5
Economizer feedwater flow rate (kg/s)	0.392±0.0002	0.3922
Downcomer feedwater flow rate (kg/s)	0.036±0.00002	0.036

Notes)

(1) Average value during 0 to 600 sec
(2) Average of cold legs A and B

4.2 Transient Calculation

In the SLB experiment (SLB-DS-01), after an initial steady-state condition was reached, this condition was maintained for about 600 sec. The steam line break test was initiated by closing the main steam line valve at 607.5 sec and opening the break line valve at 613.1 sec.

The RELAP5/MOD3.3 calculation was also conducted for null transient analysis during 600 sec. The break was initiated by closing the 697 valve at 607.5 sec and opening the 931 valve at 613.1 sec as in the SLB experiment (SLB-DS-01). Table 4-2 shows the sequence of events in both the calculation and the ATLAS experiment.

Table 4-2 Sequence of Events

Events	ATLAS SLB Test [SLB-DS-01] (sec)	RELAP5/MOD3.3 Calculation (sec)	Remark
Steady state condition	0.0 to 600.0	0.0 to 600.0	null transient
Blocking of main steam line	607.5	607.5	697 valve close
SG-1 steam line break 100% open	613.1	613.1	931 valve open
Test end	630.0	630.0	

In the transient calculation, the Henry-Fauske critical flow model was used and the maximum time step size was 0.000001 sec during the period from 613.0 to 613.2 sec. The RELAP5/MOD3.3 calculation data were compared with experimental data in Figure 4-1 through Figure 4-6. The experimental data are labeled “Exp.”, and the RELAP5/MOD3.3 calculation data as “Cal.”.

4.2.1 Accumulated Mass of Break Flow

The RELAP5/MOD3.3 calculation was in good agreement with the experiment data for accumulated mass of break flow, as presented in Figure 4-1. The Henry-Fauske critical flow model was used for the critical flow model. To obtain better agreement between calculated and experimental results, 0.41 was used for the discharge coefficient and 0.14 (default) was used for the thermal non-equilibrium at the throat (junction 921).

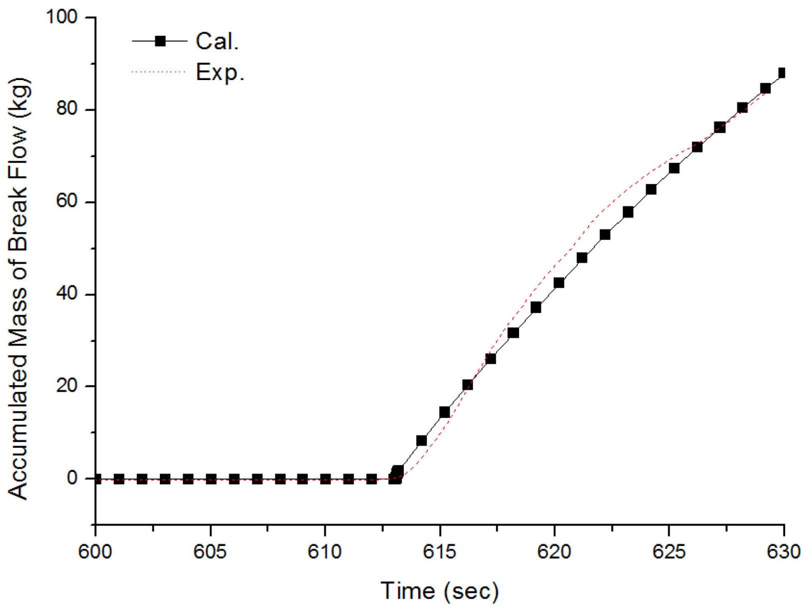

Figure 4-1 Accumulated Mass of Break Flow

4.2.2 Flow Rate

The main steam line flow behaved differently in the RELAP5/MOD3.3 calculation and in the experiment. The main steam line valve (valve 697) closed quickly in the RELAP5/MOD3.3 calculation, but the main steam line valve was closed slowly (manually) in the experiment. After the break valve (rupture disk) opening, leakage appeared in the main steam line, because the main steam line isolation valve did not provide perfect isolation. Because, the meaningful thermal hydraulic parameters were acquired from the experiment during a short period of time (approximately 0.1 to 0.2 sec), this difference is immaterial.

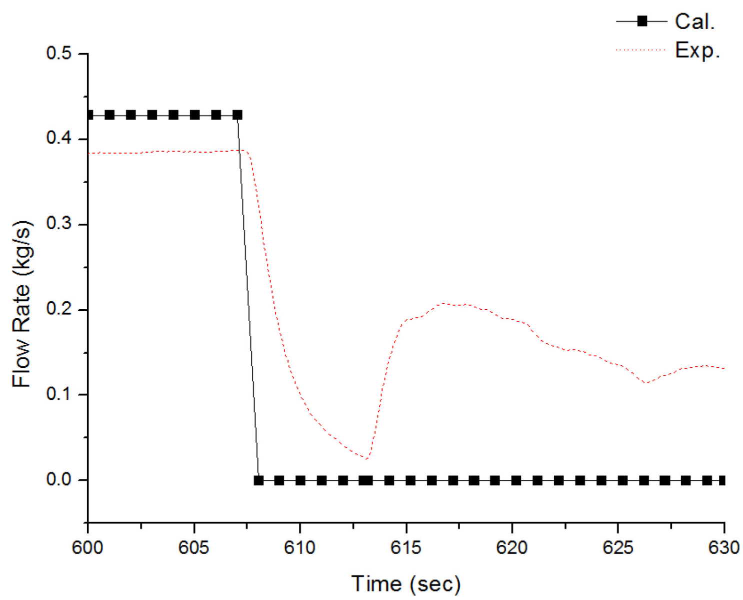

Figure 4-2 Flow Rate at the Main Steam Line

4.2.3 Temperature

The temperature calculated by the RELAP5/MOD3.3 was in good agreement with experimental data, as presented in Figure 4-3 and Figure 4-4, which show the temperature at the hot side downcomer and cold side downcomer, respectively. The temperatures at hot/cold side downcomers were slightly over-predicted after 623 sec.

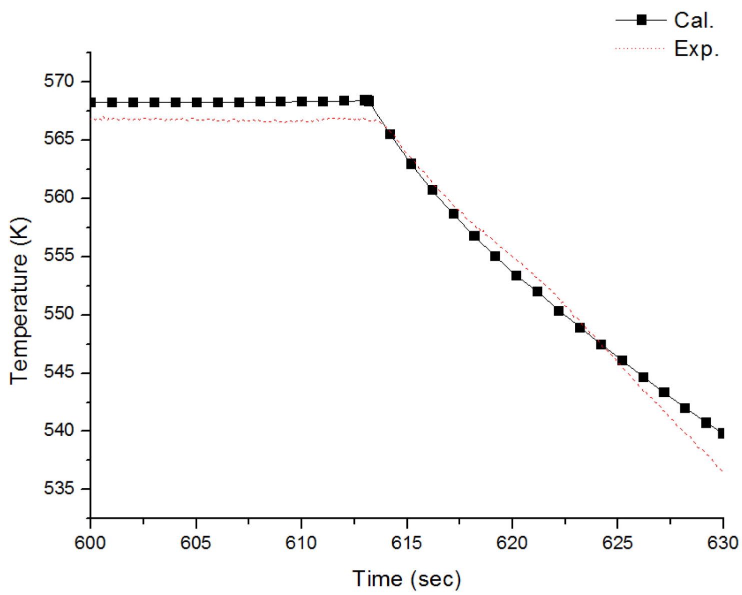

Figure 4-3 Temperature at the Hot Side Downcomer

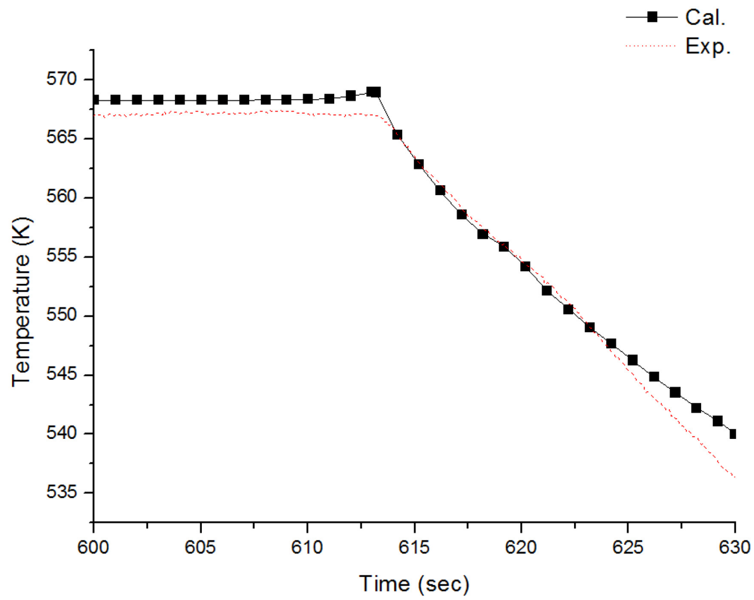

Figure 4-4 Temperature at the Cold Side Downcomer

4.2.4 Pressure

The RELAP5/MOD3.3 calculation was in good agreement with experimental data for pressure at the steam dome region, as presented in Figure 4-5. After 622 sec, the steam dome pressure began to show deviations between the RELA5/MOD3.3 calculation and the experimental results. The experimental steam dome pressure decreased more rapidly because of the steam leak through the main steam line, which was not perfectly isolated, as presented in Figure 4-2.

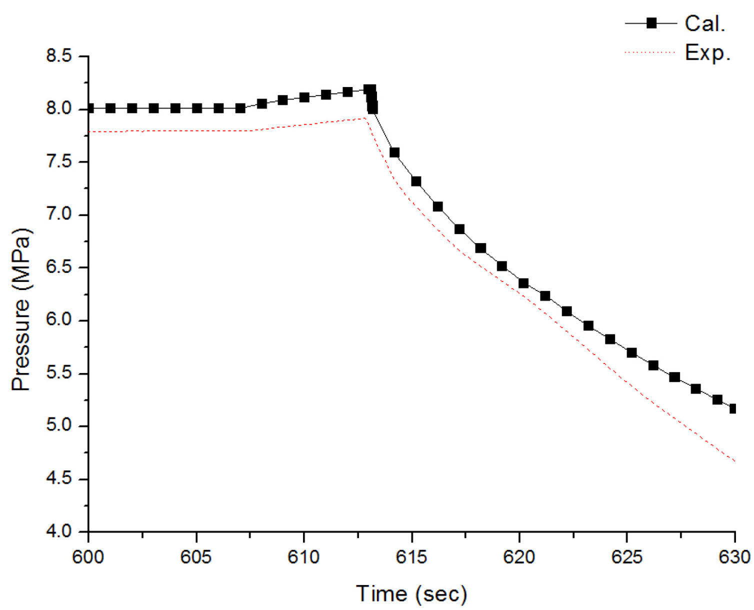

Figure 4-5 Pressure at the Steam Dome

4.2.5 Dynamic Pressure

The RELAP5/MOD3.3 calculation conservatively predicted the dynamic pressure for the steam line break of the steam generator, as presented in Figure 4-6. Locations of the dynamic pressure measurements are shown in Figure 2-6. The maximum dynamic pressure was observed at dynamic pressure - 03 which is 0.134 MPa in experiment and 0.145 MPa in the RELPA5/MOD3.3 calculation. The dynamic pressures - 01 and 02 are not shown, because they were too small to compare.

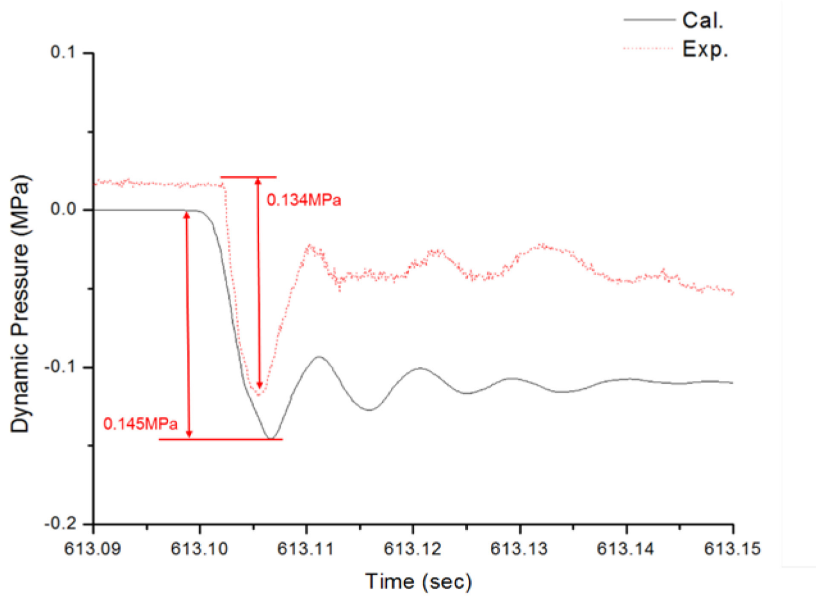

Figure 4-6 Dynamic Pressure - 03

4.3 Sensitivity Study

Sensitivity studies were performed to investigate the important factors affecting the dynamic pressure. The important factors are the critical flow model, the break valve opening time, the time step size, and nodalization. Detailed results are presented in the following sub-sections.

4.3.1 Critical Flow Model

The critical flow model is an important factor in the steam line break analysis. Thus, a sensitivity analysis of the critical flow model was performed. The critical flow models used for this sensitivity analysis were the Henry-Fauske critical flow model, the Modified Henry-Moody critical flow model and the Original RELAP choked flow model.

Figure 4-7 shows the behavior of the dynamic pressure depending on the critical flow model. The Henry-Fauske model predicted the dynamic pressure more conservatively than the Modified Henry-Moody and the Original RELAP models. To obtain conservative results, the Henry-Fauske critical flow model is recommended for the steam line break of the steam generator.

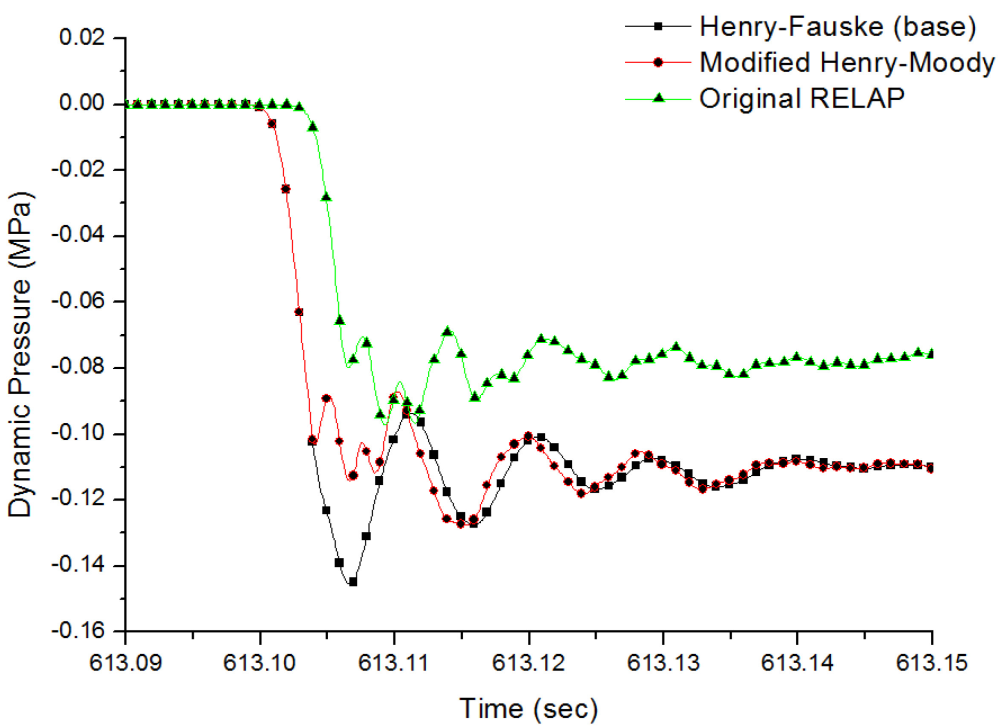

Figure 4-7 Dynamic Pressure - 03 for the Critical Flow Model

4.3.2 Break Valve Opening Time

The break valve opening time of 1.0 milli-sec is required for the steam line break analysis by the Korea Institute of Nuclear Safety. In the test, the maximum break valve opening time was 1.4 milli-sec. Thus, a sensitivity analysis was performed to find out the effect of changes in the break valve opening time. The analysis ranged from 1.0 to 2.5 milli-sec.

Figure 4-8 shows the behavior of the dynamic pressure depending on the break valve opening time. The maximum dynamic pressure increased as the break valve opening time decreased to 1.0 milli-sec. When the break valve opening time was 1.0 and 2.5 milli-sec, the maximum dynamic pressure was 0.145 and 0.144 MPa, respectively.

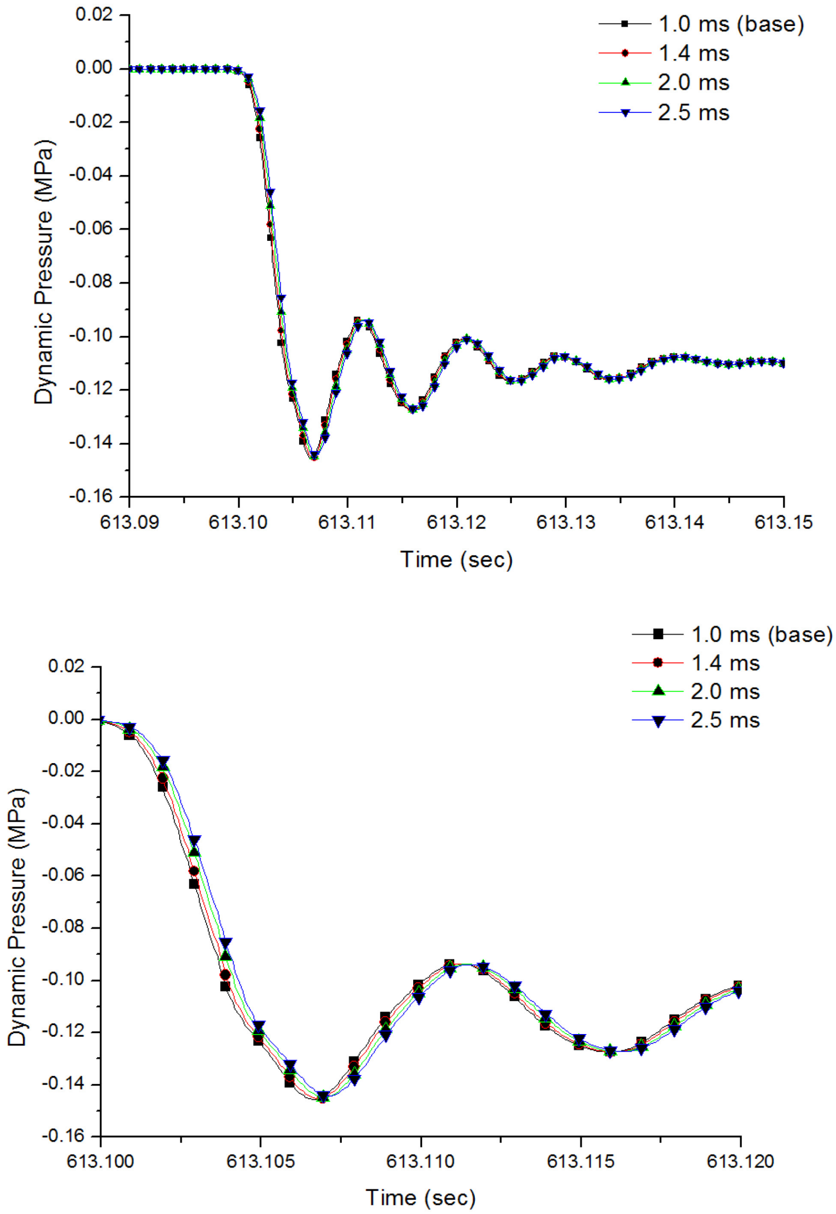

Figure 4-8 Dynamic Pressure - 03 for the Break Valve Opening Time

4.3.3 Time Step Size

After opening of the steam line break valve, the maximum dynamic pressure was reached within 0.007 sec. Thus, the time step size is an important factor in steam line break analysis and a sensitivity study for time step size was performed. The analysis for the maximum time step size ranged from 10^{-7} to 10^{-3} sec.

Figure 4-9 shows that the maximum dynamic pressure increased as the maximum time step size decreased to 10^{-6} sec, and that the dynamic pressure was almost the same below that value. A maximum time step size less than or equal to 10^{-4} sec is recommended to obtain a most conservative maximum dynamic pressure. When the maximum time step size was 10^{-4} sec, the calculated maximum dynamic pressure was 0.136 MPa (0.134 MPa in the experiment).

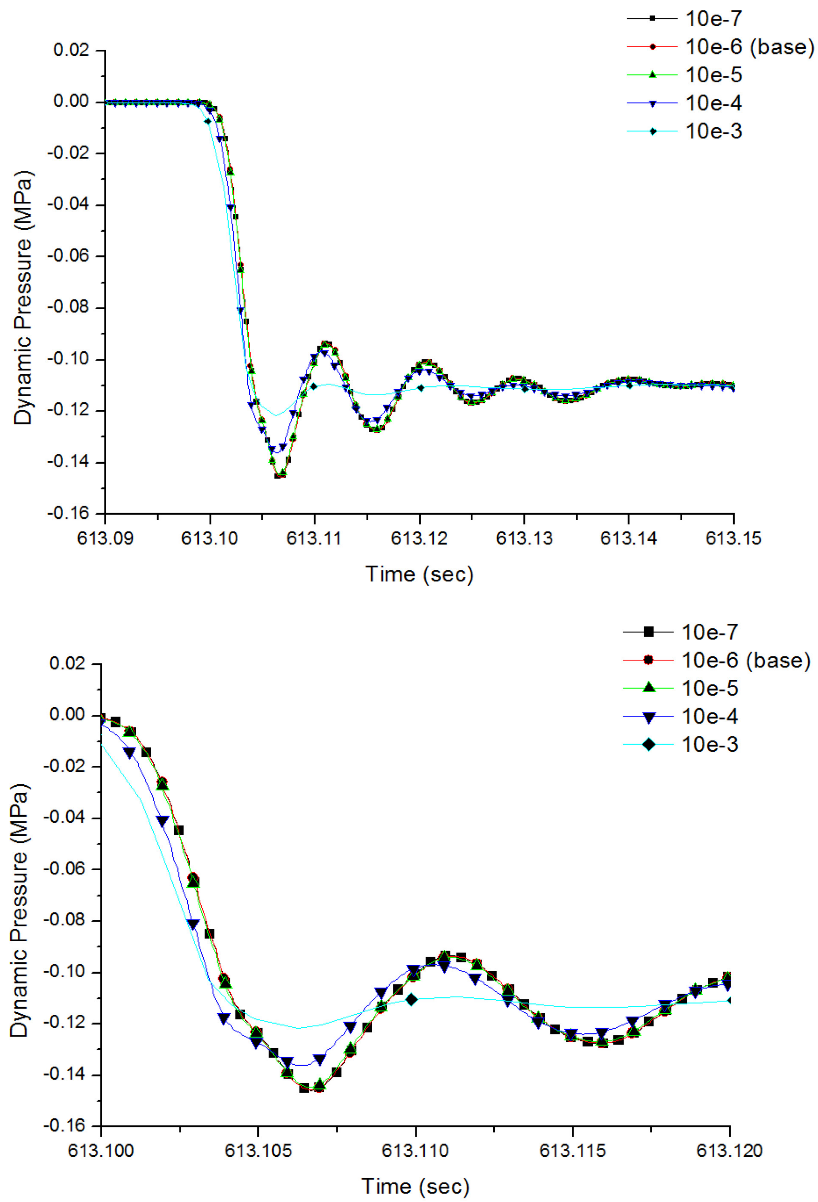

Figure 4-9 Dynamic Pressure - 03 for the Time Step Size

4.3.4 Nodalization

In the steam line break analysis, the nodalization in the components around the steam line throat could influence the analysis results. Thus, a sensitivity analysis on the nodalization was performed by varying the number of volumes in the PIPE component (C920). The numbers of C920 volumes chosen for the sensitivity analysis were 1, 3, and 6.

The results of the sensitivity analysis on the nodalization are shown in Figure 4-10. The maximum dynamic pressure decreased as the number of volumes increased. However, because the difference was not significant, the effect of the nodalization in the components around the steam line throat is negligible.

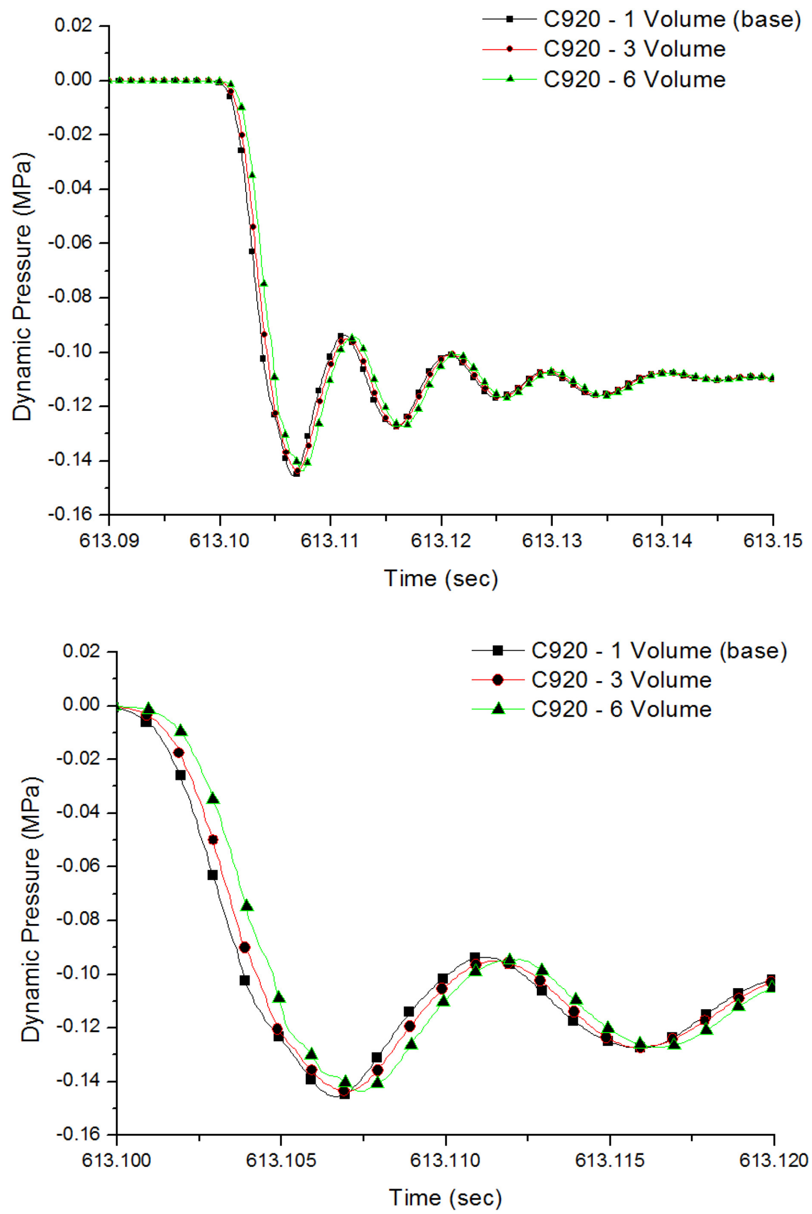

Figure 4-10 Dynamic Pressure - 03 for the Nodalization

5 CONCLUSIONS

The steam line break accident was simulated using RELAP5/MOD3.3. The calculation results were compared to the experimental data. As a result, the pressure, temperature, and accumulated mass behaviors were found to be well predicted, and the maximum dynamic pressure was predicted conservatively. Thus, it is concluded that use of the RELAP5/MOD3.3 should be acceptable for calculation of the steam generator blowdown load from steam line break accident.

6 REFERENCES

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10. SUPPLEMENTARY NOTES
K. Tien, NRC Project Manager

11. ABSTRACT (200 words or less)

The RELAP5/MOD3.3 is generally used for best-estimate transient simulation of light water reactor coolant systems during postulated accidents in the Light Water Reactor (LWR). The RELAP5/MOD3.3 code is based on a non-homogeneous, and non-equilibrium model for a two phase system that is solved by a fast, partially implicit numerical scheme to permit economical calculation of system transients. This code is suitable for the analysis of transients and postulated accidents in LWR systems, including both large- and small-break loss of coolant accidents, as well as for the full range of operational transients.
For the evaluation of structural integrity for the steam generator in the Pressurized Water Reactor (PWR), the postulated accidents, such as the Steam Line Break (SLB) in the Advanced Power Reactor (APR1400) at the Korean domestic plants, are considered Design Basis Events (DBE). In order to evaluate the structural integrity of a steam generator during the SLB, the data for the thermo-hydraulic velocity, density and pressure are needed.
This study was performed to calculate thermal hydraulic parameters, such as thermo-hydraulic velocity, density and pressure, using the RELAP5/MOD3.3 code for the structural evaluation of the steam generator internals during the postulated SLB accidents.
The calculation results were verified by comparing with experimental data generated from the experimental facility ATLAS (Advanced Thermal-Hydraulic Test Loop for Accident Simulation)

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