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TRACE5 Assessment of 100% Direct Vessel Injection Line Break in ATLAS Facility

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ABSTRACT

The calculations using TRACE V5.0 patch2 code were conducted for 100% DVI line break test of ATLAS which is the first domestic standard problem (DSP-01). In the steady state conditions, the errors between the calculated and measured values are acceptable for most primary/secondary system parameters. From the transient calculations, the predicted sequence of events occurred some seconds later than that of experiment due to slow depressurization of the primary side after break. Before the LSC, the code could not predict well the behavior of downcomer and core collapsed water levels due to the under-prediction of the discharged mass and high flow rate returned to core from hot leg. After the LSC, the downcomer and core collapsed water levels drop rapidly and the code predicts relatively well the trend of downcomer and core water level. All RCPs suction legs or loop seals are cleared completely and suddenly. The predicted cladding temperature generally agrees well with the experiment except the peaking behavior. In the sensitivity studies, if the discharging mass increases arbitrarily, the downcomer and core water level before SIP injection shows better prediction result than that of base case. However, except for this, the overall behavior for main parameters shows the bad prediction results. For the effect of flow restriction from hot leg using the CCFL model, the predicted core water level was not significantly decreased compared to the base case. In conclusions, TRACE code has good capabilities to simulate the 100% DVI line break test of ATLAS. However, TRACE code including the choked flow and CCFL models needs to be improved and more detailed modeling is needed to predict more accurate results.

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EXECUTIVE SUMMARY

An integral effect test on the SBLOCA (Small-Break Loss of Coolant Accident) aiming at 100% DVI (Direct Vessel Injection) line break was conducted with the ATLAS in 2009 by KAERI (Korea Atomic Energy Research Institute). In this study, the calculations using TRACE V5.0 patch2 code were conducted for 100% DVI line break test of ATLAS which is the first domestic standard problem (DSP-01) to assess TRACE code capability to simulate the transient thermal-hydraulic behavior for SBLOCA.

For the modeling of ATLAS facility, the reactor vessel was modeled in three dimension using VESSEL component and flows within a coolant loop of the primary and secondary sides were modeled in one dimension using main hydraulic components such as PIPE and TEE in TRACE codes. Also, the emergency core cooling system, the heater rods and main control logics were properly modeled in order to consider the various conditions of this experiment. The steady state was determined by conducting a null transient calculation and the errors between the calculated and measured values are acceptable for most primary/secondary system parameters.

In an experiment, after initial steady-state conditions were reached, the DVI line break test was initiated by opening a break simulation valve at 199 seconds, and transient calculation was also conducted by setting up 199 seconds as the initial time. The predicted sequence of events occurred some seconds later than that of the experiment due to slow depressurization of the primary side after break. Before the SIP starts, the predicted pressurizer pressure decreases slowly compared to the experimental results because of the under-estimation of break flow according to the characteristic of choked flow model. Before the LSC, the code could not predict well the behavior of the downcomer and core collapsed water levels due to the under-prediction of the discharged mass and high flow rate returned to core from hot leg. After the LSC, the downcomer and core collapsed water levels drop rapidly and the code predicts relatively well the trend of the downcomer and core water level. In this study, all RCPs suction legs or loop seals are cleared completely and suddenly and the code predicted the LSC ~ 25 seconds later than the measured value due to the slow depressurization rate. The predicted cladding temperature generally agrees well with the experiment except the peaking behavior. This peak of cladding temperature is directly related to core collapsed water level dip before the LSC.

The sensitivity studies were performed to identify 1) the effect of discharging mass for the depressurization and 2) the effect of flow restriction from hot leg for the core collapsed water level. If the discharging mass increases arbitrarily, the downcomer and core water level before SIP injection shows better prediction results than that of the base case. However, except for this, the overall behavior of the main parameters shows the bad prediction results. For the effect of flow restriction from hot leg using the CCFL model, the predicted core water level was not significantly decreased compared to the base case. The results are different in comparison with that of RELAP5 code and we need to perform further studies to better understand this difference.

In conclusion, the TRACE code has good capabilities to simulate the 100% DVI line break test of ATLAS. However, there are some discrepancies in quantitatively predicting the primary pressure, break flow, the downcomer and core collapsed liquid level, and so on. Therefore, the TRACE code including the choked flow and CCFL models needs to be improved and more detailed modeling is needed to predict more accurate results.

FOREWORD

This report represents one of the in-kind contributions submitted to fulfill the bilateral agreement for cooperation in thermal-hydraulic activities between Korean Institute of Nuclear Safety (KINS) and the U.S. Nuclear Regulatory Commission (NRC) in the form of a Korean contribution to the NRC's Code Assessment and Maintenance Program (CAMP), the main purpose of which is to validate the TRAC/RELAP Advanced Computational Engine (TRACE) Code.

Since 2006, the integral effect test facility, ATLAS (Advanced Thermal-Hydraulic Test Loop for Accident Simulation) has been operated by Korea Atomic Energy Research Institute (KAERI), which was constructed to simulate the reactor coolant system (RCS) behavior during transients in its reference designs, OPR-1000 and the APR-1400. After a series of the direct vessel injection (DVI) line break tests in the ATLAS was complete for four break sizes, 5%, 25%, 50%, and 100%, KAERI proposed the first domestic standard problem (DSP-1) in 2009, aiming at assessing the capability of the existing codes for 100% DVI line break among the code users in Korea. At that time the TRACE code has not been used by any users. Just after termination of the DSP-1 program, KINS has reassessed the problem, using TRACE V5.0 patch2, with much information on the test specification provided by KAERI.

This report decribes the TRACE code assessment for 100% DVI line break in the ATLAS facility. According to a coordinated frame, NUSTEP for cooperation among Korean Utilities, KINS, research institutes, this report has been reviewed as the contribution by KINS and KAERI.

1. INTRODUCTION

A small break loss of coolant accident (SBLOCA) is characterized by relatively slow reactor coolant system (RCS) depressurization rates and by relatively slow mass loss from the RCS, compared to the design basis accident of large break loss of coolant accident (LBLOCA) [1]. Unlike the LBLOCA, the sequence of events following a small break in a LWR can evolve in a variety of ways. Operator actions, reactor design, ECCS set points, break size and location will have a bearing on how the SBLOCA scenario unfolds. Another principal difference is the domination of gravity effects in small breaks versus inertial effects in the large breaks [2]. In order to predict the thermal-hydraulic characteristics during SBLOCA adequately, code must have sufficient modeling capabilities to take these factors into account.

The DVI-adopted plants treat a DVI line break as another spectrum among the SBLOCAs in their safety analysis because the DVI nozzle directly attached to the reactor vessel is vulnerable to a postulated break from a safety viewpoint. The thermal hydraulic phenomena in the RPV down-comer are expected to be different from the cold leg injection (CLI) mode during the postulated design basis accidents. In the event of a DVI line break, the vapor generated in the core is introduced to the reactor pressure vessel (RPV) downcomer through the hot legs, the steam generators and the cold legs. Then the vapor should pass through the upper part of the RPV downcomer to be discharged through the broken DVI nozzle. Therefore, the behavior of the two-phase flow in the upper annulus downcomer is expected to be complicated and relevant models need to be implemented into the safety analysis codes in order to predict these thermal hydraulic phenomena correctly. So far there is not enough integral effect test data for the DVI line breaks which can demonstrate the progression of the DVI line break accident realistically and can be used for an assessment and improvement of safety analysis codes.

KAERI has been operating an integral effect test facility, ATLAS (Advanced Thermal-Hydraulic Test Loop for Accident Simulation) for accident simulations for the OPR-1000 and the APR-1400 [3]. After the ATLAS was modified to have a configuration for simulating the DVI line break accidents of the APR1400 at the beginning of 2008, sensitivity tests for different DVI line break sizes were performed by KAERI [3]. The Integral effect database for four break sizes were established; 5%, 25%, 50%, and 100%. The ATLAS has been used to provide the unique test data for the 2 (hot legs) x 4 (cold legs) reactor coolant system with direct vessel injection (DVI) of emergency coolant. The 100% DVI line break test was selected for the first domestic standard problem (DSP-01) in 2009 to enhance the understanding on the behavior of nuclear reactor systems with the DVI and to assess existing thermal-hydraulic analysis codes such as MARS, RELAP and so on. The TRACE code has not been used to simulate a DVI line break in the DSP-01 program.

The TRACE code is the thermal-hydraulic system code and has been developed by USNRC for a realistic analysis of thermal-hydraulics transients in pressurized water reactors [4]. TRACE has been designed to perform best-estimate analyses of loss-of-coolant accidents (LOCAs), operational transients, and other accident scenarios in pressurized light-water reactors (PWRs) and boiling light-water reactors (BWRs). It can also model phenomena occurring in experimental facilities designed to simulate transients in reactor systems. Models used include multidimensional two-phase flow, nonequilibrium thermo-dynamics, generalized heat transfer, reflood, level tracking, and reactor kinetics.

In this study, the calculations using TRACE V5.0 patch2 code released in the beginning of 2011 and comparison with experimental data were conducted for 100% DVI line break test of the ATLAS which is the first domestic standard problem (DSP-01) to assess TRACE code capability to simulate the transient thermal-hydraulic behavior for the DVI line break.

2. ATLAS FACILITY AND TEST DESCRIPTION

2.1 Overview of the ATLAS

The ATLAS is a large-scale thermal-hydraulic integral effect test (IET) facility for advanced pressurized water reactors (PWRs), APR-1400 and OPR-1000. It can simulate a wide variety of accident and transient conditions including large and small break LOCAs.

The ATLAS has the following characteristics: (a) 1/2-height, 1/288-volume, full-pressure simulation of the APR1400; (b) geometrical similarity with the APR1400, including 2 (hot legs) x 4 (cold legs) reactor coolant loops, DVI of emergency core cooling water, integrated annular downcomer, etc.; (c) incorporation of specific design characteristics of the 1000-MW (electric) class OPR1000 such as a cold-leg injection and the low-pressure injection pumps, (d) a maximum 10% of the scaled nominal core power, and (e) simulation capability of broad scenarios, including the reflood phase of the large-break LOCA, small-break LOCA scenarios including the DVI line breaks, steam generator tube rupture, MSLB, mid-loop operation, etc [5]. Scientific design of the ATLAS was accomplished from the viewpoints of both a global and local scaling based on Ishii et al.[6]'s three-level scaling methodology.

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 as shown in Figure 2.1 and 2.2. The primary system includes a reactor pressure vessel, 2 hot legs, 4 cold legs, a pressurizer, 4 reactor coolant pumps (RCPs), and 2 steam generators (SGs). The arrangement of the primary loop of ATLAS including DVI lines is shown in Figure 2.3. When a starting signal is generated to simulate a DVI line break of the ATLAS, a quick-opening valve in Figure 2.4 is fully opened to discharge the RCS inventories from the RPV into the containment simulator through a broken DVI nozzle. The secondary system of the ATLAS is simplified to be of a circulating loop-type. The steam generated at steam generators is condensed in a direct condenser tank and the condensed feedwater is again injected to the steam generators. Most of the safety injection features of the APR1400 and the OPR1000 are incorporated into the safety injection system of the ATLAS. The safety injection system of the ATLAS consists of four safety injection tanks (SITs), two high pressure safety injection pumps (SIPs) which can simulate safety injection and long-term cooling, a charging pump for charging auxiliary spray, and a shutdown cooling pump and a shutdown heat exchanger for low pressure safety injection, shutdown cooling operation and recirculation operation. The break simulation system consists of several break simulating lines such as large break LOCA (LBLOCA), direct vessel injection (DVI) line break LOCA, SBLOCA, steam generator tube rupture (SGTR), main steam line break (MSLB) and feedwater line break (FLB), etc. Each break simulating line consists of a quick opening valve, a break nozzle and instruments as shown in Figure 2.4. It is precisely manufactured to have a scaled break flow through it in the case of LOCA tests. The containment simulating system of the ATLAS has a function of collecting the break flow and maintaining a specified back-pressure in order to simulate containment as shown in Figure 2.5. Besides, the ATLAS has some auxiliary systems such as a makeup system, a component cooling system, a nitrogen/air/steam supply system, a vacuum system, and a heat tracing system. Secondary and auxiliary systems are designed as simply as possible since the main focus of the IET using the ATLAS will be on the simulation of primary-system transient and accidents, except for a MSLB and a FLB. Figure 2.6 shows a photograph of the front view of the ATLAS facility. The information on the ATLAS and some test results can be found in the literatures [7,8,9,10,11].

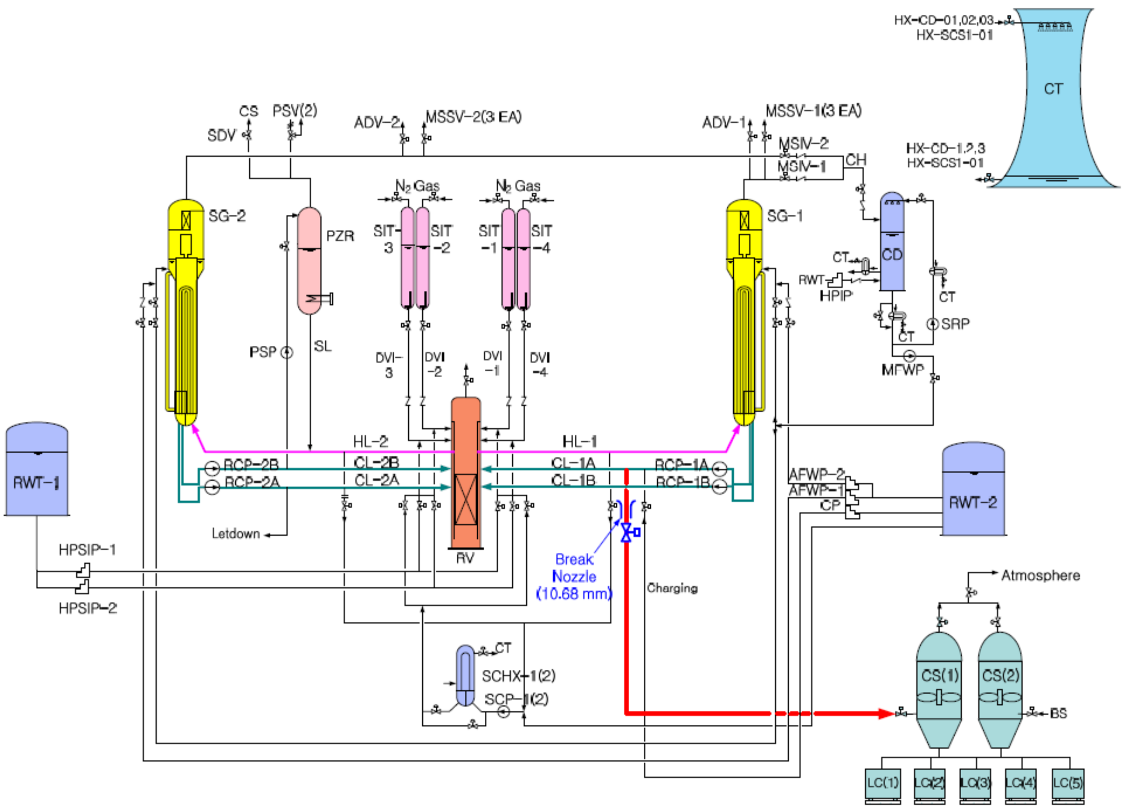


Figure 2.1 Schematics of the ATLAS Facility

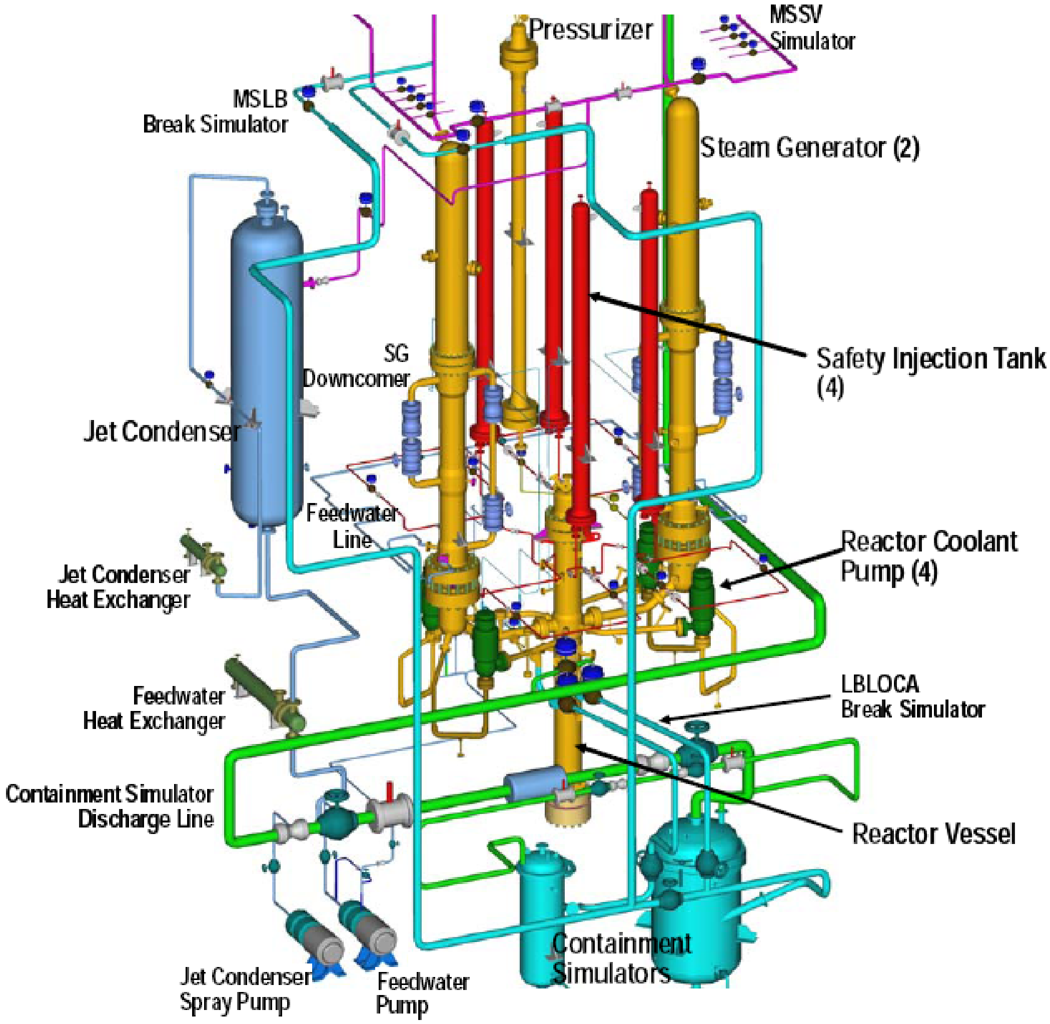


Figure 2.2 Isometric Configuration of the ATLAS Facility

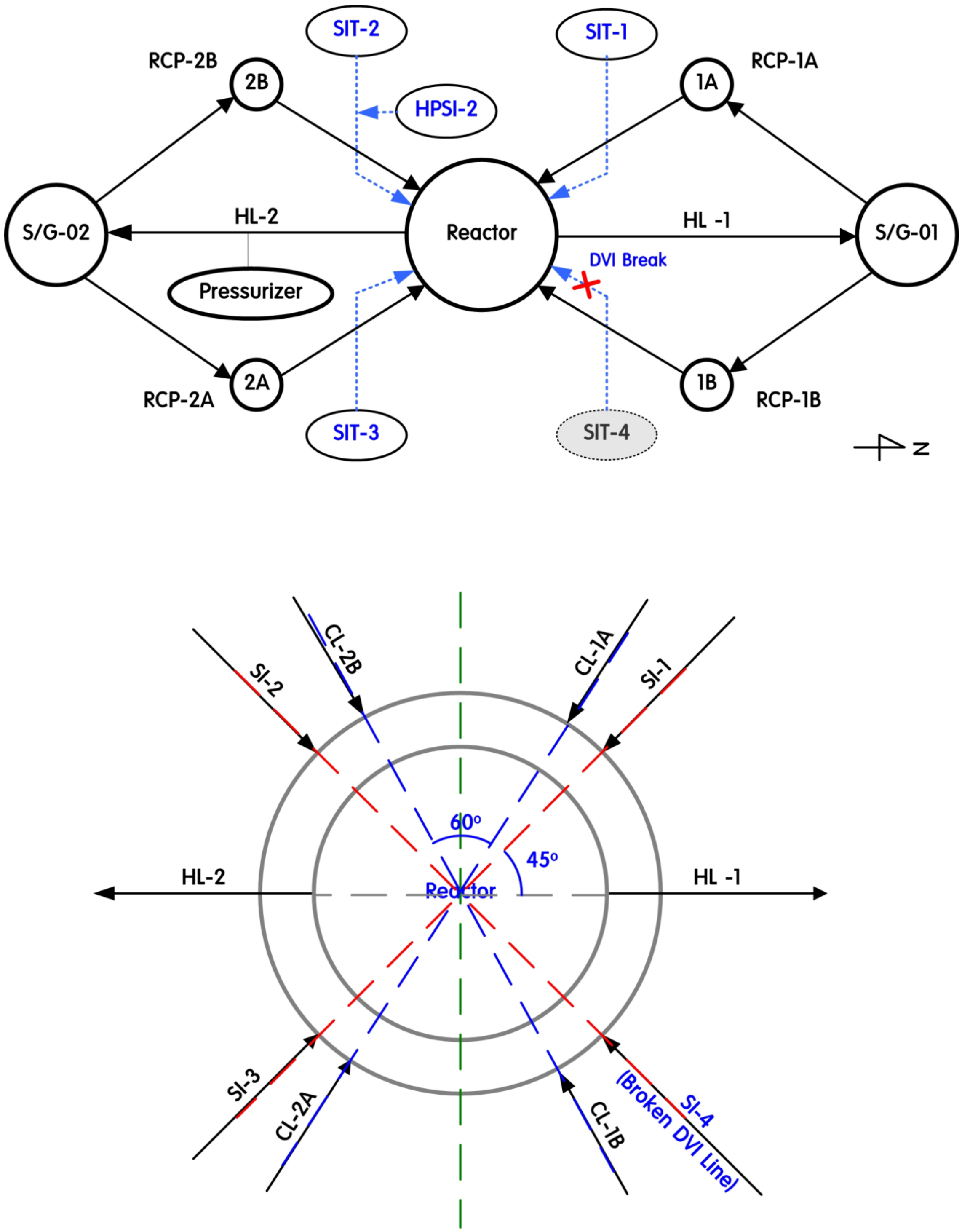
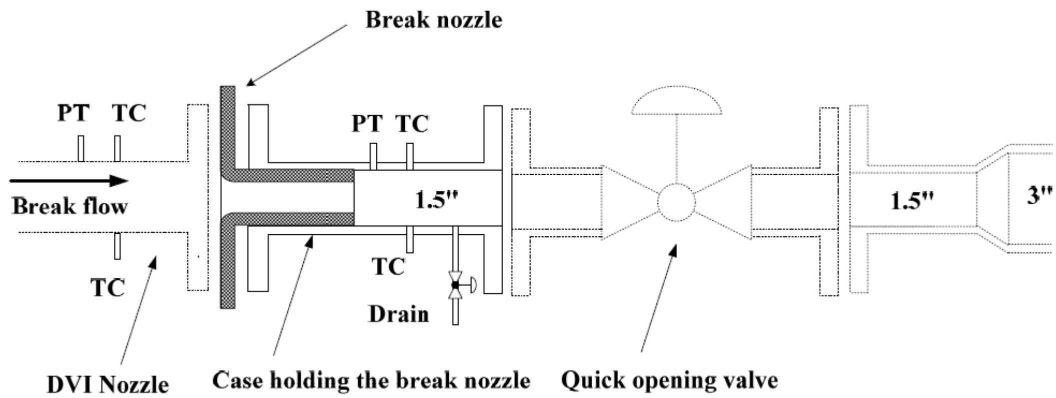


Figure 2.3 Arrangement of the Primary Loop of the ATLAS



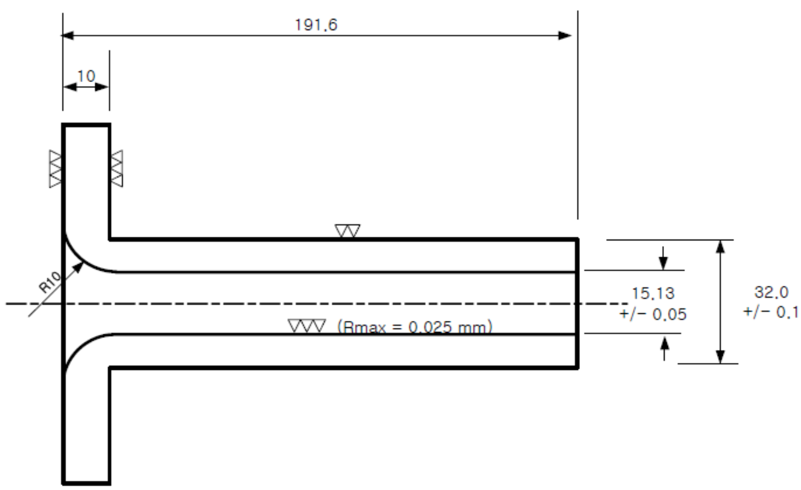


Figure 2.4 Configuration of the Break Simulation System for the DVI Line Break Tests

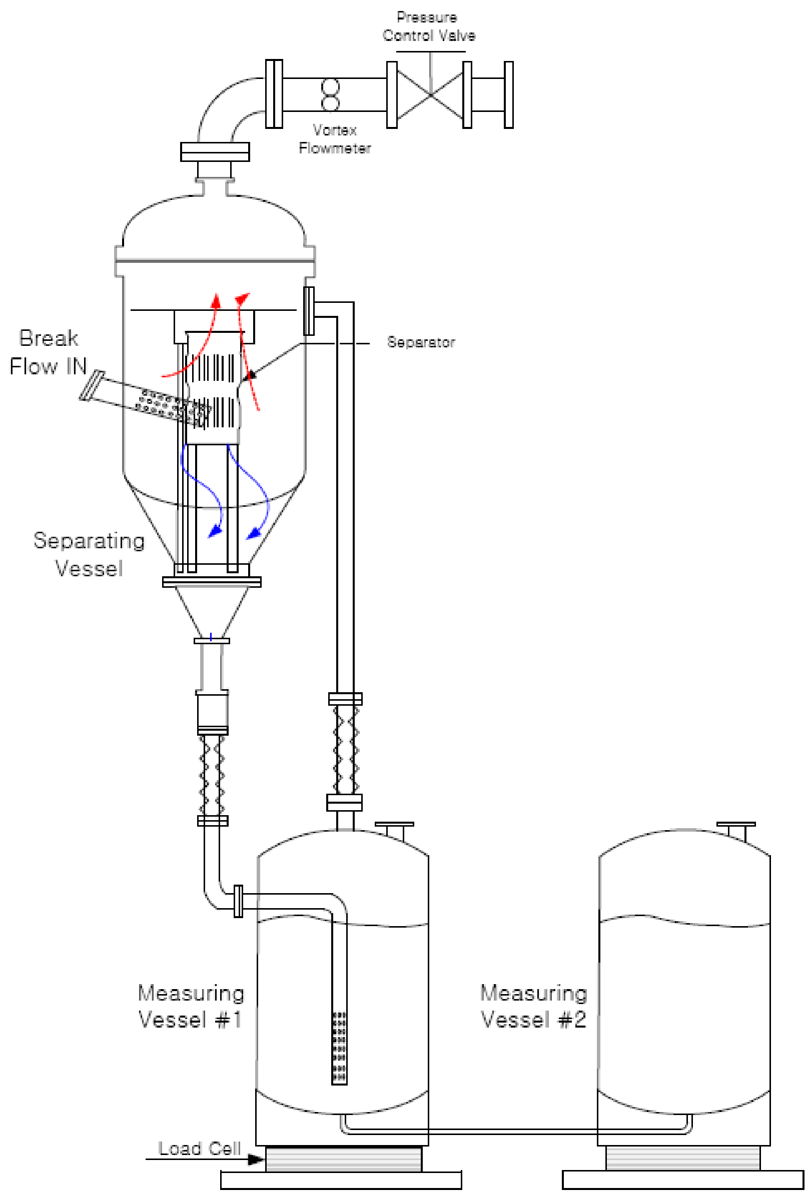


Figure 2.5 Schematics of the Containment Simulation System

Figure 2.6 Photograph of ATLAS

2.2 Test Procedure

The 100% DVI line break of the ATLAS was conducted according to the following experimental procedure [11]. Basically, the experimental conditions for the present tests were determined by a pre-calculation with the best-estimate thermal hydraulic code, MARS3.1. First of all, a transient calculation was performed for the DVI line break of the prototypic plant, APR1400 to obtain the reference initial and boundary conditions. The safety injection system of the APR1400 has 4 mechanically separated hydraulic trains. They are also electrically separated by 2 divisions, implying that each emergency diesel powers 2 hydraulic trains. The pre-calculation was conducted with the assumption of loss of off-site power simultaneously with the break and the worst single failure as a loss of a diesel generator, resulting in the minimum safety injection flow to the core. Furthermore, the safety injection flow to the broken DVI-4 nozzle was not credited. Therefore, the safety injection flow by the safety injection pump (SIP) was injected only through the DVI-2 nozzle opposite to the broken DVI-4 nozzle.

Table 2.1 Comparison of the Sequence of a 100% DVI Line Break

Events	**APR1400 (time, sec)**	**ATLAS (time, sec)**	
Break open	0	0	
Low pressurizer pressure trip (LPP)	20.9		If pressurizer pressure < 10.72 MPa
Pressurizer heater trip	LPP+0.0 sec	LPP+0.0 sec	Delay time is reduced by a square root of 2
Reactor scram & RCP trip	LPP+0.5 sec	LPP+0.35 sec	
Turbine isolation	LPP+0.1 sec	LPP+0.07 sec	
Main feedwater isolation	LPP+10.0 sec	LPP+7.07 sec	
Safety injection pump start	LPP+40.0 sec	LPP+28.28 sec	
Low upper downcomer pressure trip (LUDP)	LUDP	LUDP	If downcomer pressure < 4.03 MPa
Safety injection tank (SIT) start	LUDP+0.0 sec	LUDP+0.0 sec	
Low flow turndown of the SIT			If water level of the SIT is less than the specified set point

For the safety injection flow by the 4safety injection tanks (SIT), 3 SITs except for the SIT connected to the broken DVI-4 nozzle were available to provide the safety injection flow into the core. As for the core power, a conservative 1973 ANS decay heat curve with a 1.2 multiplication factor was used in the transient calculation. In the DVI line break, the containment back-pressure does not affect the progression of this transient, because a choking condition is maintained throughout this transient. Therefore, the containment back-pressure was not an important control parameter in the present test.

Based on the calculated sequence of events of the DVI line break for the APR1400, the initial and boundary conditions for the present integral effect test were determined. The delay time required for an initiation of the safety injection systems such as the safety injection pump and the safety injection tank were reduced by a square root of 2 according the scaling law of the ATLAS. Also, the delay times for an isolation of the secondary feedwater or steam supply systems were also scaled down by the scaling ratio of the ATLAS [5]. The detailed sequence of events applied to the present test is summarized in Table 2.1.

The 100% DVI line break of the ATLAS was performed at the same pressure as the prototypic plant, APR1400. The temperature distribution along the primary loop was also preserved. The primary inventory was heated with core heaters to its specified steady state condition and was pressurized by a pressurizer until the primary system reached a steady state condition. During the primary heat-up process, the secondary system was also heated up to a specified target hot condition by controlling the heat removal rate from the primary system. At a steady state condition, the core power generated by electrical heaters was balanced by the energy removed by the secondary system. The obtained steady state condition was maintained constant to stabilize the system behavior of the ATLAS for more than 10 minutes.

During the heat-up process, several crucial components influencing the boundary conditions of the test were controlled by operators. The 4 core bypass flow control valves were controlled to have the predetermined stem positions to have a scaled core bypass flow rate. The initial water levels and pressures of the safety injection tanks were controlled to have these specified values. The refueling water storage tank (RWT) was filled with water to its initial level of 50% and the water inventory was electrically heated to its pre-determined temperature of 50^{o}C, and then the water was circulated through the injection line so as to preheat the line up to the same temperature as the water.

Subsequent to the heat-up process, several initialization procedures were taken to obtain the required initial and boundary conditions for the DVI line break. The primary coolant flow rate was reduced to 8% of the scaled value to have the same temperature distribution along the primary loop. Operation experience showed that the required 8% of the primary coolant flow rate was achieved in a natural circulation condition. The containment pressure control valve was opened fully to simulate an atmospheric pressure during the present test. All connecting pipe lines and three-way valves of the containment simulating system were aligned to measure the break flow rate separately.

After a steady state condition in the whole ATLAS system was maintained for more than 10 minutes, the transient test was commenced. First of all, data logging was initiated to log all measurement points in a steady state condition. After the initial data logging was completed for about 200 seconds, the DVI line break test was initiated by opening a quick-opening break valve, OV-BS-03 at the break spool piece. A DVI line break was simulated by installing a break spool piece at one of the DVI nozzles. The configuration of the break spool piece is shown in Figure 2.4. It consists of a quick opening valve, a break nozzle, a case holding the break nozzle, and a few instruments. A pressure transducer and 2 thermocouples were installed both upstream and downstream of the break nozzle. Detailed geometry of the break nozzle for the present DVI line break tests is shown in Figure 3.4. The break nozzle was installed vertically downward at the discharge line of the DVI nozzle. The quick opening valve was opened within 0.5 seconds by operators when the test was initiated. The break flow was discharged to the containment simulating system.

When the pressurizer pressure reached a specified pressure of 10.72MPa, the low pressurizer pressure (LPP) signal was automatically generated by embedded control logics. The heaters of the pressurizer and all tracing heaters in the primary system were tripped at the same time of the LPP signal. The RCP was automatically tripped with a time delay of 0.35 seconds after the LPP signal. The main steam and the main feed water line were isolated with a time delay of 0.07 seconds and 7.07 seconds after the LPP signal, respectively. The isolation of the secondary system requires a simultaneous actuation of several valves in the pipe line. It was done by the programmed control logics without operator intervention. Operation of the SI pump was triggered by the LPP signal with a time delay of 28.3 seconds. The initiation of the SI pump requires an alignment of the valves located in the supply line. This alignment was also completed automatically by the control logic without any time delay.

When the downcomer pressure of the reactor vessel became lower than the specified pressure of 4.03MPa, the SIT started to deliver the high SI flow to the reactor vessel through the three DVI nozzles by fully opening the flow control valve. When the water level of the SIT reached a specified set point, the stem of the flow control valve was lowered to a specified position to supply a required low injection flow rate. When the water level of the SIT was lowered to a specified empty set point, the flow control valve was fully closed for the nitrogen gas, not injected into the reactor vessel. The transient was terminated with the end of the data logging when it was judged by operators that all the major phenomena have already taken place.

3. MODELING INFORMATION

TRACE code has been developed as the unified code for the reactor thermal hydraulic analyses in USNRC [4]. The Symbolic Nuclear Analysis Package (SNAP) is a suite of integrated applications designed to simplify the process of performing engineering analysis and it provides a flexible framework for creating and editing input for engineering analysis codes as well as extensive functionality for submitting, monitoring, and interacting with the codes [12]. The SNAP currently can support most analysis codes developed in USNRC, for example the CONTAIN, COBRA, FRAPCON-3, MELCOR, PARCS, RELAP5 and TRACE codes. In this study, the Model Editor which is one of SNAP client applications used to simulate the DVI line break test of ATLAS with TRACE codes.

As mentioned previously, one of the objectives of the ATLAS development was to understand the complicated multi-dimensional phenomena like the downcomer boiling and the ECC water bypass which were important safety issues of the APR1400. The safety analysis codes such as RELAP5 have some limitation in order to predict these thermal hydraulic phenomena correctly since it is a kind of one-dimensional based codes. However, the TRACE code has the capability to simulate a 3D Cartesian- and/or cylindrical-geometry flow by using 3D components named VESSEL [4]. This is important in the prediction of the ECC bypass and the steam-water interaction in the downcomer during a LOCA. In this study, the reactor vessel including the core and the downcomer was modeled in 3 dimensions using the VESSEL component and flows within a coolant loop of the primary and secondary sides were modeled in one dimension using main hydraulic components such as PIPE and TEE in TRACE codes. The hot leg, cold leg, steam generator, steam line, and reactor coolant pump (RCP) were included in the primary and secondary systems. Also, the emergency core cooling system (ECCS) like the safety injection tank (SIT) and the safety injection pump (SIP) and the containment which is the boundary condition of LOCA were modeled. In order to simulate the heater rod in ATLAS, 2 heat structures, one was the average heaters and the other was the hot heater, were implemented in the each volume of core. Main control logics were modeled by the control block in TRACE.

3.1 <u>Main Loop Modeling</u>

The APR1400 is the prototype of ATLAS basically and it has 2 (hot legs) x 4 (cold legs) reactor coolant loops. Therefore, the composition of main components and the connection between the components in ATLAS is almost equal to those in APR1400. The pressurizer is connected to 2-side loop of primary system as shown in Figure 3.1. Each loop includes a hot leg, a steam generator, 2 reactor coolant pumps (RCPs), and 2 cold legs. Also, all passive heat structures including U-tubes of steam generator in each loop are modeled in this study.

- ○ The hot leg is connected from the VESSEL component modeled as the core to the steam generator. The hot leg consists of 4 volumes which are modeled with 3 PIPE components. The surge line of pressurizer is connected to the hot leg as the cross flow junction and the hot leg is horizontally connected to the VESSEL component.

- ○ The inlet/outlet plenum of SG is modeled as the PIPE component and is connected from the hot/cold leg to the U-tube. The U-tubes (V-340/V-440) in SG consist of 12 volumes in the PIPE component as shown in Figure 3.1. The riser (V-650/V-750) in the secondary side of SG is modeled as the PIPE component with 5 volumes and 4 volumes of riser exchange the heat with those in the primary side. The separator (V-660/V-770) in SG is modeled as the SEPARATORS component.

- ○ The pump suction leg is modeled as the PIPE component by 5 volumes and is connected from the SG to the RCP. The angle of flow path in those properly considers the experimental data. Also, in order to predict the loop seal clearing (LSC) better, the height of intermediate legs is modeled in consideration of the location of instrument in experiments.

- ○ The reactor coolant pump (RCP) is modeled as the PUMP component. The PUMP component in TRACE describes the interaction of the fluid with a centrifugal pump [4]. Since the PUMP component has the generic pump model including heat, torque, speed model, pump-curve data, etc., the pump behaviors like the pressure differential across the pump impeller and the pump impeller's angular velocity for steady-state and transient conditions can be calculated by properly specifying the initial conditions.

- ○ The cold leg is connected from the reactor vessel and the RCP and is modeled as 4 PIPE components. Also, it is horizontally connected to the VESSEL component.

- ○ The pressurizer is modeled by the PIPE component with 10 volumes. The empty time of pressurizer can influence the fuel temperature behavior due to the core rewetting which the core is quenched by the water in the pressurizer as it flows into the upper plenum through the hot leg. In the steady-state condition, the FILL component connected to the upper region of pressurizer is used to set the boundary condition of pressure.

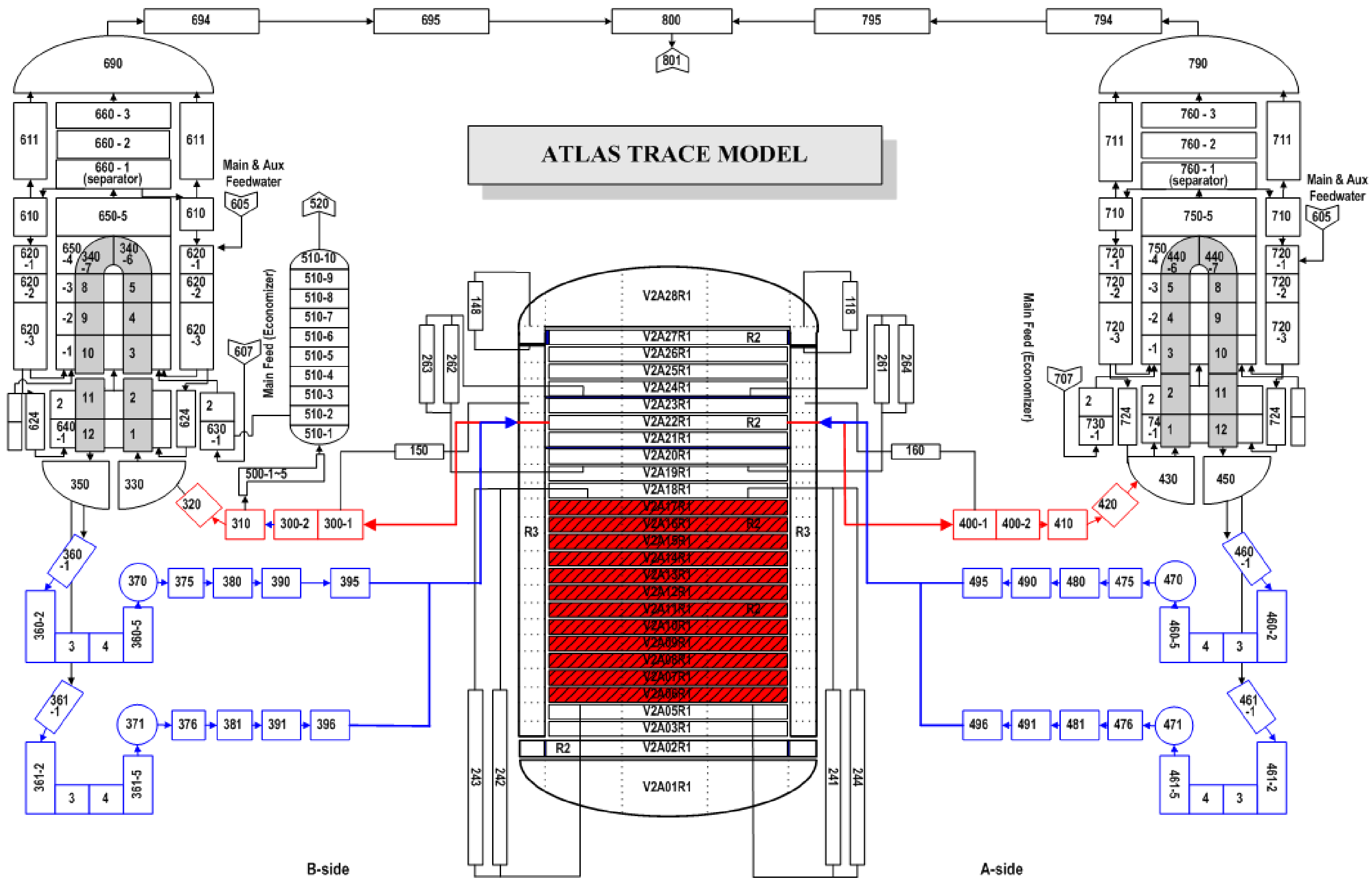


Figure 3.1 ATLAS Nodalization for TRACE

3.2 Reactor Vessel Modeling

The modeling of reactor vessel is performed by the VESSEL component and it includes the downcomer, upper and lower plenum, upper head, core, etc. The reactor vessel of ATLAS consists of the lower plenum, active core, upper plenum and upper head along the axial direction as shown in Figure 3.2. It is also made up of the core region and the downcomer along the radial direction. All heat structures for the reactor vessel including the heater rods to simulate the fuel assembly are modeled in this study. The detailed information on the TRACE modeling for ATLAS can be found in the literature [13].

- ○ The downcomer is the outermost region of the VESSEL component and consists of 6 volumes along the azimuthal direction and 26 volumes along the axial direction. In the calculations by using the one-dimensional codes like RELAP5, the downcomer is generally modeled as 2 ~ 6 separated PIPE components and the single junctions are used to connect azimuthally between the neighboring PIPE components to simulate the complicated multi-dimensional behaviors in the downcomer. However, this may have the limitation to correctly predict these phenomena. In this study, the reactor vessel is modeled as the cylindrical multi-dimensional component and it can produce the better results for the multi-dimensional behaviors. The DVI line is modeled by considering the injection position of design data in ATLAS.

- ○ The lower plenum is composed of 2 radial, 6 azimuthal and 5 axial volumes and the boundary of axial volumes agrees with the geometric boundary of ATLAS. The lower second volume contains the flow skirt in which the flow exchanges occur between the downcomer and the core.

- ○ The upper plenum is located between the fuel alignment plate (FAP) and the upper guide structure support plate (UGSSP) and it is connected from the core exit to the hot leg and the upper head region in the vessel. The upper plenum consists of 2 radial, 6 azimuthal and 3 axial volumes and the middle of axial volumes is connected to the hot leg of each loop. The upper head is made up of 2 radial, 6 azimuthal and 5 axial volumes as shown in Figure 3.2 and the bypass flow between the downcomer and the upper head is also modeled.

- ○ The active core consists of 2 radial, 6 azimuthal and 12 axial volumes and 12 flow channels is modeled at the specific axial level. The TRACE code has a tool to model the heat structure like the general thermal hydraulic codes such as RELAP5. The HTSTR component in TRACE evaluates the dynamics of conduction, convection and gap-gas heat transfer. Also, all fluid components, for example PIPE, TEE, etc. which include input for a pipe wall, internally have HTSTR components that provide for the simulation of the heat transfer from the pipe wall to the fluid [4]. Therefore, all fluid components including VESSEL and the heater rod in ATLAS is modeled by using the HTSTR component. There are 396 heater rods in the core of ATLAS. The two HTSTR components are modeled in each volume of 12 flow channels. One is to simulate the average heat structure to consider the number of heater rods and guide tubes in each volume as shown in Figure 3.3. The other is to model the hot heat structure which has 1.4 times heat power as the average value of one heater rod. Table 3.1 shows the applied decay power level with the variation of time.

○ Several bypass flow paths in the reactor vessel are also modeled in this study. It is important to predict the bypass flow correctly in the steady-state calculation since the modeling of bypass flow may influence the transient results. From the reference calculation by the MARS code and the design of ATLAS, 4 bypass flow paths are considered as follows.

- The bypass flow between the downcomer and the upper head
- The bypass flow from hot leg nozzle gap to the downcomer
- The bypass flow through the control element assembly (CEA) guide tubes
- The bypass flow through the core guide tubes

Table 3.1 Applied Decay Power Level with the Variation of Time

Time after break (sec)	Total Avg. Rod Power (kW)	Total Hot Rod Power (kW)	Factor
0.0	1499.022	67.416	1.000
24.1	1499.022	67.416	1.000
25.5	1442.808	64.887	0.9625
26.9	1405.333	63.202	0.9375
28.3	1349.119	60.674	0.9000
35.4	1199.217	53.932	0.8000
42.5	1105.528	49.719	0.7375
49.5	1011.839	45.505	0.675
56.6	955.626	42.977	0.6375
63.7	918.150	41.292	0.6125
70.7	899.413	40.449	0.6000
84.9	843.199	37.921	0.5625
99.0	805.724	36.236	0.5375
113.2	768.248	34.550	0.5125
127.3	749.511	33.708	0.5000
141.4	730.773	32.865	0.4875
212.2	637.084	28.651	0.4250
282.9	599.608	26.966	0.4000
353.6	562.133	25.281	0.3750
424.3	543.395	24.438	0.3625
495.0	524.657	23.595	0.3500
565.7	513.415	23.089	0.3425
636.4	505.919	22.752	0.3375
707.1	490.929	22.078	0.3275
1000.0	468.444	21.067	0.3125
3510.0	312.546	14.056	0.2085
7046.0	245.090	11.022	0.1635

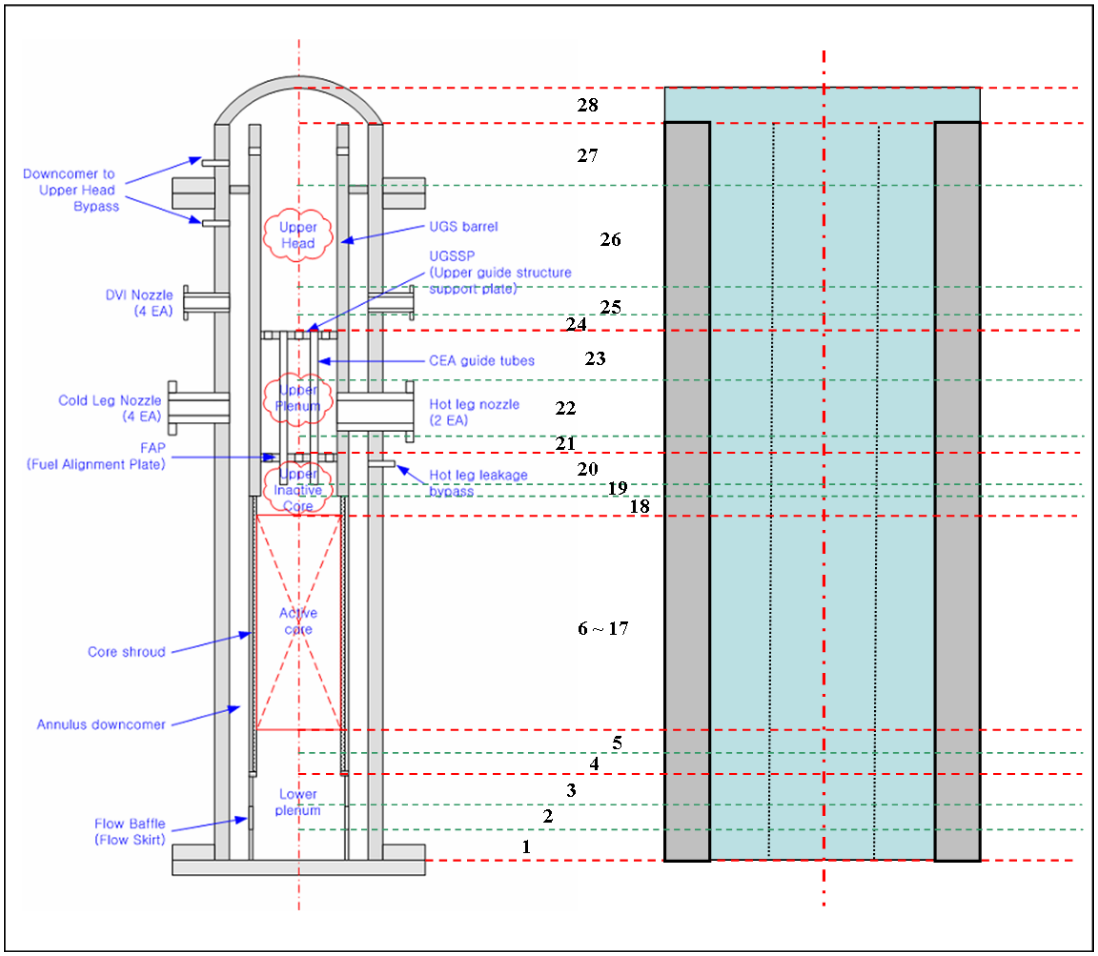


Figure 3.2 Reactor Vessel Modeling of ATLAS

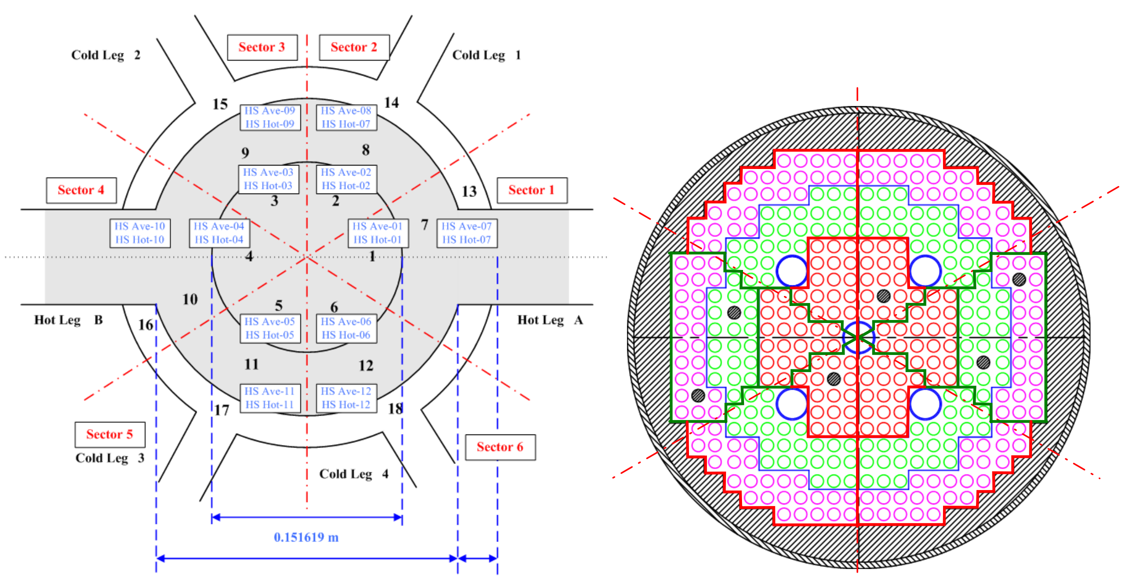


Figure 3.3 Heat Structure Modeling for ATLAS Core

3.3 <u>Emergency Core Cooling System (ECCS) & Break Modeling</u>

The reference plant of the ATLAS is the APR1400 as mentioned before. The APR1400 adopts new safety injection features such as four mechanically independent safety injection systems, a DVI system, fluidic devices in each SIT, and an elimination of a low pressure safety injection system. Each of the four SITs of APR1400 has a fluidic device which passively controls the discharge flow rate into the reactor coolant system. In the APR1400, a high flow condition is changed to a low flow condition due to a fluidic device during operation of the SIT. Also, the APR1400 adopts the in-containment refueling water storage tank (IRWST) and it results in the improvement of ECCS reliability by skipping the action to switchover to the recirculation mode. The ATLAS incorporates all safety injection design features of OPR1000 and APR1400. The emergency core cooling system in the APR1400 is shown in Figure 3.4.

The 100% DVI line break experiment of the ATLAS was performed with the assumption of loss of off-site power simultaneously with the break and the worst single failure as a loss of a diesel generator, resulting in the minimum safety injection flow to the core. Therefore, the safety injection flow through the safety injection tank (SIT) and the safety injection pump (SIP) to the broken DVI-4 nozzle was not credited. Thus, the safety injection flow by the safety injection pump (SIP) was injected only through the DVI-2 nozzle opposite to the broken DVI-4 nozzle. The safety water in the safety injection tank was injected through the 3 DVI nozzles except for the broken DVI-4 nozzle.

In ATLAS modeling of TRACE, the safety injection tank is modeled as the PIPE component and the pipe type is selected as the accumulator. The major parameters (e.g. water level, volumetric flow, discharge volume) are automatically calculated in the accumulator type. Two valves are modeled at the downstream of SIT and then the start of SIT injection, switchover to low flow region due to a fluidic device and termination of SIT injection can be simulated by using proper control logics in the PIPE and VALVE components as shown in Figure 3.5. The switchover to low flow region can be modeled by changing the flow area of valve at a specific water level of SIT. The safety injection pump is modeled as the FILL component. The injected flow rate is adjusted according to the pressure of the downcomer volume injected and the injecting time is determined by the trip signal for safety injection due to the low pressurization of RCS. Table 3.2 shows the mass flow rate of safety injection pump with respect to the downcomer pressure. The initial conditions (e.g., water temperature, water level, pressure, etc.) are determined as the experimental data. The double ended guillotine break of DVI-4 line is considered in transient calculations.

The break system of the ATLAS is greatly simplified by using a single junction and BREAK component which is opened instantly at 199 seconds and the default critical flow model of TRACE, Ransom-Trapp model [14] is applied to the single junction [15]. The containment conditions are defined in a break table in the BREAK component. Table 3.3 shows the containment conditions with variation of time that is used in transient calculations. Figure 3.5 shows the modeling of the downcomer and the ECCS and Figure 3.6 shows the plane figure for the connection between the reactor vessel and the ECCS.

Table 3.2 Mass Flow Rate of SIP with respect to the Downcomer Pressure

Downcomer Pressure (MPa)	SIP Flow Rate (kg/sec)
0.10	0.323158
0.45	0.318567
0.79	0.313965
1.14	0.309055
1.48	0.304144
2.17	0.294039
2.86	0.283579
4.24	0.260868
5.62	0.235408
7.00	0.20624
8.38	0.170933
9.75	0.123685
10.10	0.107735
10.44	0.088879
11.13	0.0

Table 3.3 Containment Conditions with Variation of Time

Time after break (sec)	Pressure (Pa)	Quality
0.0	100930.0	1.0
3.0	155260.0	1.0
4.0	166990.0	1.0
6.0	161720.0	1.0
11.0	183760.0	1.0
13.0	195840.0	1.0
14.0	202100.0	1.0
17.0	215630.0	1.0
19.0	230260.0	1.0
23.0	239740.0	1.0
29.0	207910.0	1.0
35.0	194080.0	1.0
38.0	191060.0	1.0
42.0	188690.0	1.0
46.0	204010.0	1.0
56.0	181200.0	1.0
67.0	170200.0	1.0
79.0	168250.0	1.0
91.0	171990.0	1.0
167.0	140090.0	1.0
238.0	120720.0	1.0
261.0	116060.0	1.0
497.0	104370.0	1.0
1001.0	102870.0	1.0
2000.0	102490.0	1.0
1000000.0	100000.0	1.0

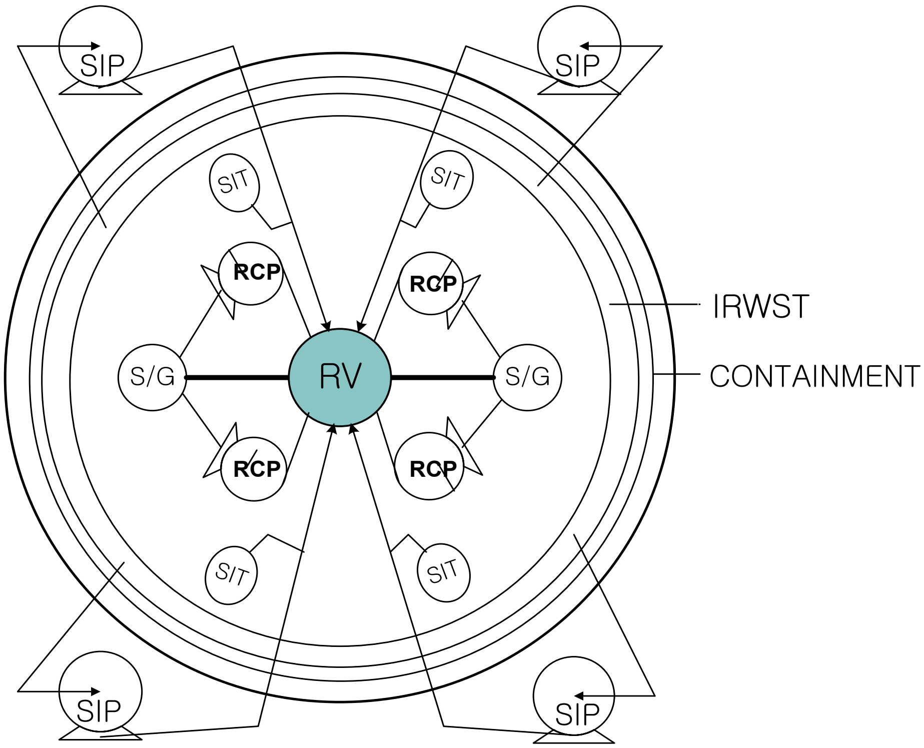


Figure 3.4 Emergency Core Cooling System (ECCS) in APR1400

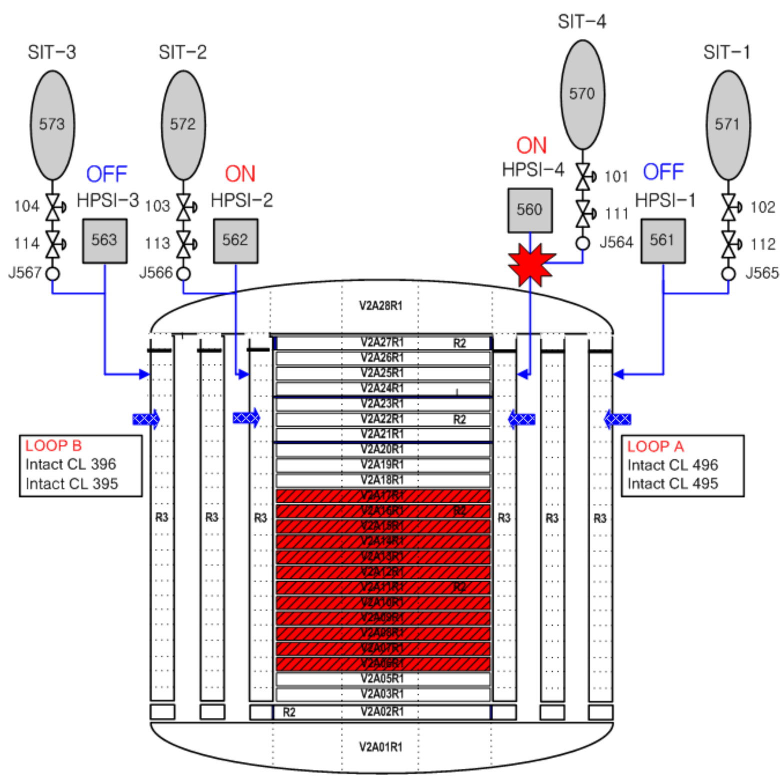


Figure 3.5 TRACE Model for the Reactor Vessel and ECCS

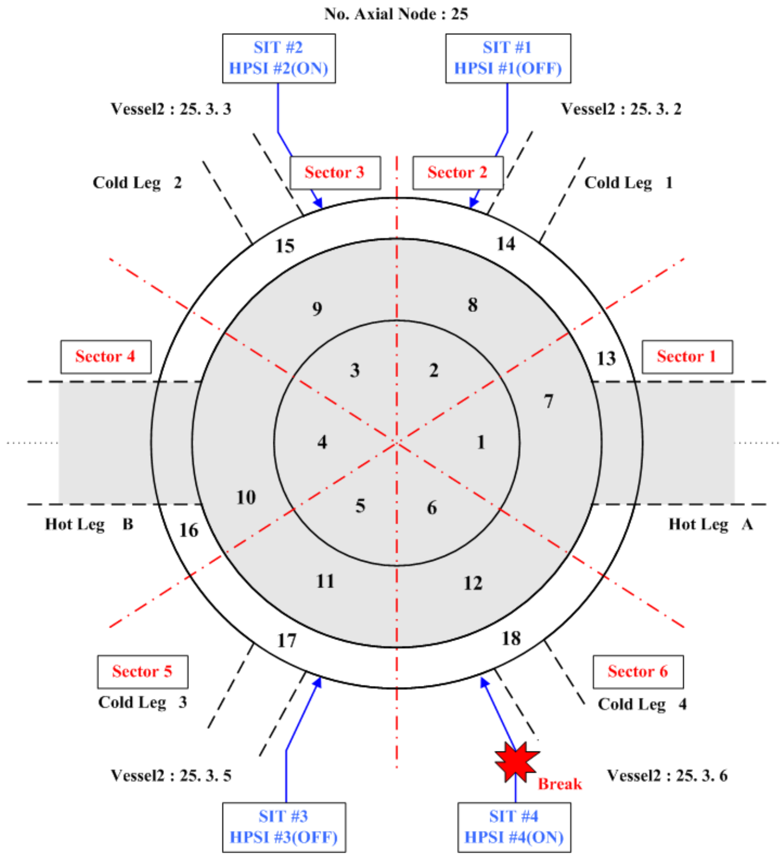


Figure 3.6 Plane Figure for the Connection between the Reactor Vessel and the ECCS

4. STEADY STATE ANALYSIS

The steady state was determined by conducting a null transient calculation and the steady state calculation results which are the initial conditions of 100% DVI line break test are shown in Table 4.1.

The core power in the calculation was set to 1.566 MW which is the value subtracting the heat loss of 88 kW from the experimental core power, because adiabatic boundary conditions are applied in the system [16]. The hot leg and cold leg temperatures are little lower than in the experimental data. However, the calculated temperature difference between the hot leg and cold leg is ~ 33 K, and this value is almost same as that of experiment. Therefore, it is considered that energy conservation of experiment is well simulated. In the case of momentum conservation, the RCS flow rate in calculation is a little different from the experiment. If we do a simple energy balance calculation using the measured hot leg and cold leg temperatures, the RCS flow rate assuming 1.5666 MW core power is 2.05 kg/sec which is almost the same as the code predicted values. As mentioned previously, in experiment, the 8% of the primary coolant flow rate was achieved in a natural circulation condition without operating the reactor coolant pump (RCP). On contrary, forced circulation condition using the RCP should be maintained in calculation. It is considered that the difference of RCS flow rate between the experiment and calculation results from the type of flow in the system.

As shown in Table 4.1, the differences between the calculated and measured values are acceptable for most of the primary/secondary system parameters. Some initial conditions of ECCS are considered conservatively in comparison to the experiment. The initial pressure of containment is exactly the same as the experimental data.

Table 4.1 Steady State Calculation Results

Parameter	Experiment	Calculation	Remarks
Primary system			
- Core power (MW)	1.644 (heat loss 88 kW)	1.654 (w/ heat loss)	
- PRZ pressure (MPa)	15.49	15.5	
- Core inlet temp. (K)	564.0	564.4	
- Core outlet temp. (K)	598.1	596.9	
- Hot leg temp. (K)	599.2	596.2	Ave. of 2 HL
- Cold leg temp. (K)	566.2	562.7	Ave. of 4 CL
- RCS flow rate (kg/s)	2.2	2.05	Ave. of 4 CL
Secondary system			
- Steam-dome pressure (MPa)	7.86	7.81	Ave. of 2 SG
- Steam temp. (K)	569.0	566.5	Ave. of 2 SG
- Feedwater (FW) temp. (K)	509.0	505.4	
- FW flow rate to economizer (kg/s)	0.34	0.399	Ave. of 2 SG
- FW flow rate to downcomer (kg/s)	~0.0	0.044	Ave. of 2 SG
ECCS			
- SIT pressure (MPa)	4.23	4.03	Ave. of 4 SIT
- SIT temp. (K)	323.7	322.1	Ave. of 4 SIT
- SIT level (m)	5.28	5.26	Ave. of 4 SIT
- RWT temp. (K)	316.7	322.0	Ave. of 2 SIP
Containment			
- Pressure (MPa)	0.10	0.10	

5. TRANSIENT ANALYSIS

In the experiment, after initial steady-state conditions were reached, the DVI line break test was initiated by opening a break simulation valve at 199 seconds, and transient calculation was also conducted by the opening trip in the FILL component at 199 seconds as the initial time. The predicted sequence of events is compared with that of the ATLAS experiment as shown in Table 5.1.

Table 5.1 Sequence of Events

Event	ATLAS Experiment (sec)	TRACE Calculation (sec)	Remarks
Steady state condition	< 199.0	< 199.0	
Break open	199.0	199.0	
Low pressurizer pressure (LPP)	219.0	229.6	10.72 MPa
Turbine trip	219.1	229.1	LPP + 0.07 sec
Reactor trip by LPP		229.9	LPP + 0.35 sec
RCP trip		229.9	LPP + 0.35 sec
Decay power start	223.0	223.0	
Main feedwater isolation	226.0	236.7	LPP + 7.07 sec
SIP injection signal	246.0	257.9	LPP + 28.28 sec
Max. PCT	290.0	-	
Loop Seal Clearing (LSC)	~295/~297/~319/~320	~321/~322/~322/~322	
SIT injection start	431.0	426.0	

When the pressurizer pressure decreases below 10.72 MPa, the reactor/RCP trip and main steam isolation occur logically by the low pressurizer pressure (LPP) signal. However, in the experiment, core power decay was initiated with time delay of 4.0 seconds after the break, and this condition was applied in calculation as shown in Table 3.1. The main feedwater is isolated with time delay of 7.07 seconds after the LPP signal, logically. SIP was injected with time delay of 28.28 seconds after LPP, and SIT injection was initiated when the downcomer pressure was reduced below 4.03 MPa. After the calculation was started at 199 seconds, the pressurizer pressure decreased and reached to a low pressurizer set pressure at ~ 229.6 sec. The LPP time of TRACE is ~ 10 seconds later than that of the ATLAS experiment. The decay of heater power in TRACE is modeled to start at same time (~ 223.0 seconds) with the experiment. The safety injection pump (SIP) injection is predicted ~ 10.0 seconds later than that of experiment due to the delay LPP time. The safety injection tank (SIT) starts to inject ~ 426 seconds because of rapid reduction of pressure after the loop seal clearing (LSC).

In the general, for small or intermediate break, the core is uncovered and the cladding temperature increases. At this calculation, we can identify the following specific scenarios of SBLOCA for the plant with a U-tube steam generator.

○ Blowdown phase : After break occurs abruptly, the primary pressure is reduced rapidly. As soon as it reaches the set point of low pressurizer pressure, the control rod drops and the reactor scrams. Also, the reactor coolant pump is tripped simultaneously. After the safety injection signal happens, the emergency cooling water is injected into the RCS. During the blowdown phase, most of the RCS is filled with the liquid phase and the break flow is discharged as the subcooled or saturated liquid phase. Consequently, the reduction of primary pressure slows down at the extent of slightly higher value than the secondary pressure. At this time, the primary and secondary pressures are determined according to the core decay heat, the heat removal of SG, the set point of safety valves of SG, etc. Figure 5.1 shows the typical blowdown phase in this calculation.

○ Natural circulation : Lastly in the blowdown phase, the RCS reaches the quasi-equilibrium condition. This can continue for several hundred seconds according to the break size. During this period, the loop seal is full in the liquid phase. The coolant is discharged continuously and the vapor begins to be generated from the upper U-tube of steam generator. The vapor generation is propagated into the upper plenum through the upper head and the phase separation is accelerated. The decay heat in the core is removed from the discharged flow and the steam generators. The steam in U-tubes does not find the effective path since the loop seal is filled with the liquid and the steam generated in core is trapped in the RCS. Therefore, the liquid with low quality is discharged from the break. Figure 5.2 shows the typical natural circulation phase in this calculation.

○ Loop Seal Clearing (LSC) : As the water level of cold leg side in SG approaches the upper suction line (loop seal) of RCS, the trapped steam in the RCS can be released from the break position. This is called loop seal clearing (LSC). Also, the loop seal clearing can occur abruptly as soon as the pressure in upper region of vessel becomes larger than the hydrostatic pressure of liquid phase in the loop seals. The loop seal clearing can happen in the partial or entire loops. The discharged flow is changed from the low quality liquid to the steam. At just before loop seal clearing, the core can be uncovered due to the decrease of the collapsed core level according to the vessel pressurization. After the loop seal clearing, the pressure unbalance is resolved and the collapsed water level is covered gradually. Figure 5.3 shows the RCS conditions after the loop seal clearing.

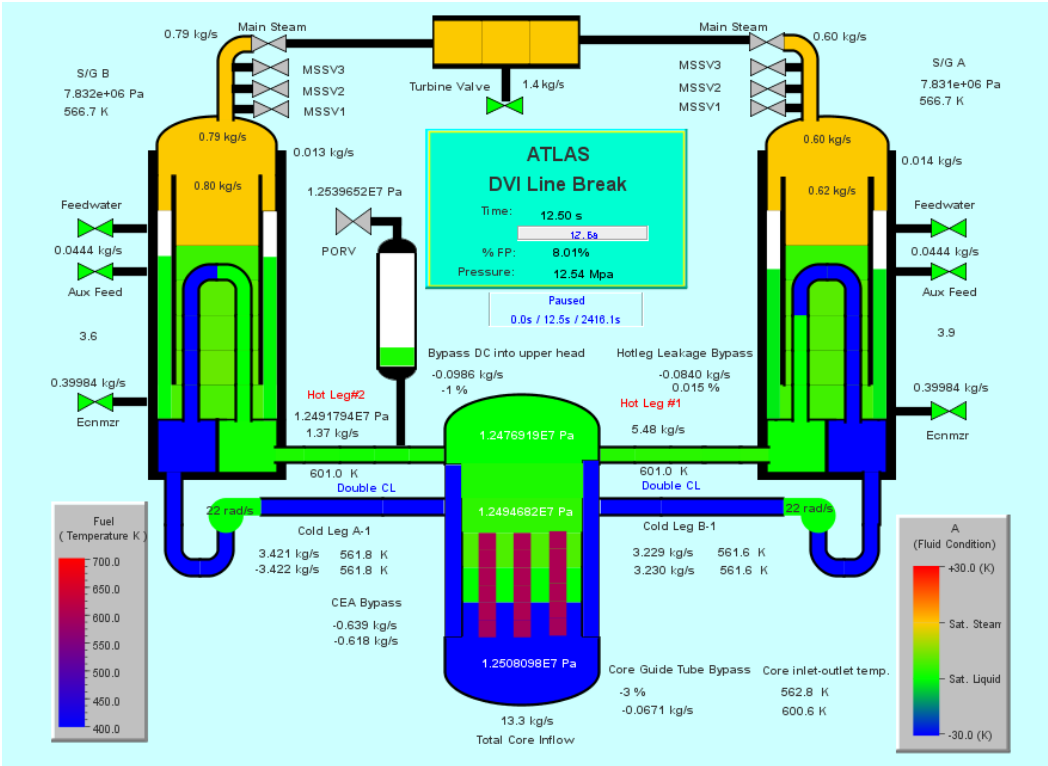


Figure 5.1 Blowdown Phase (at 12.5 seconds after break)

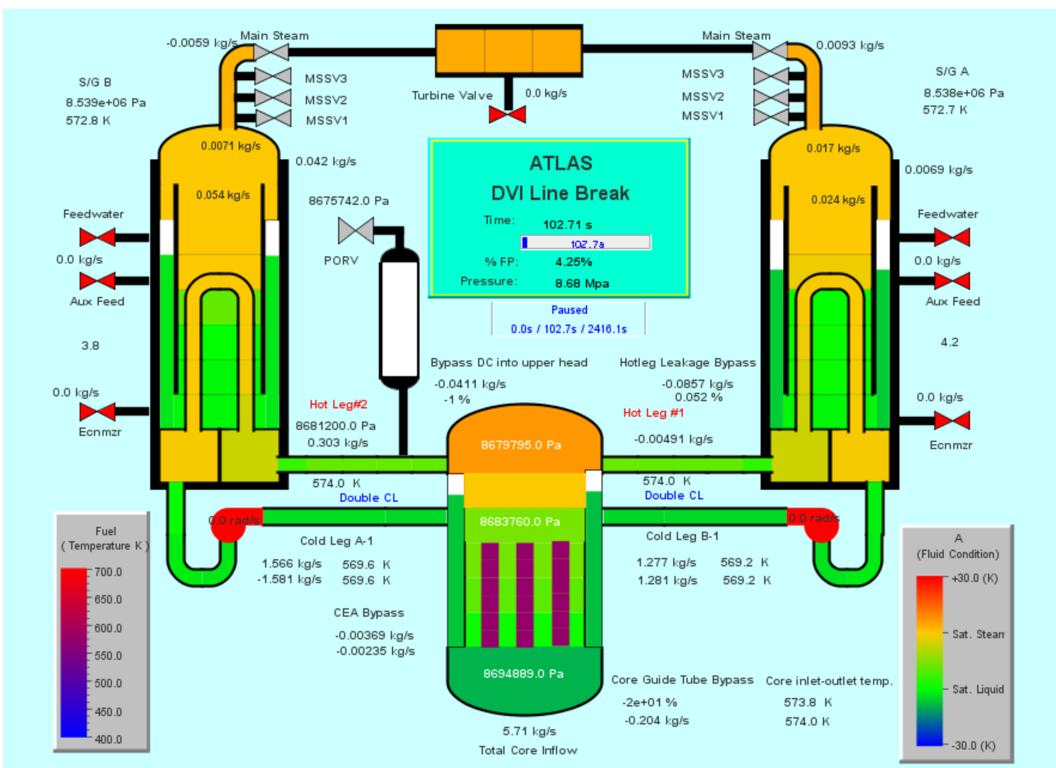


Figure 5.2 Natural Circulation (at 102 seconds after break)

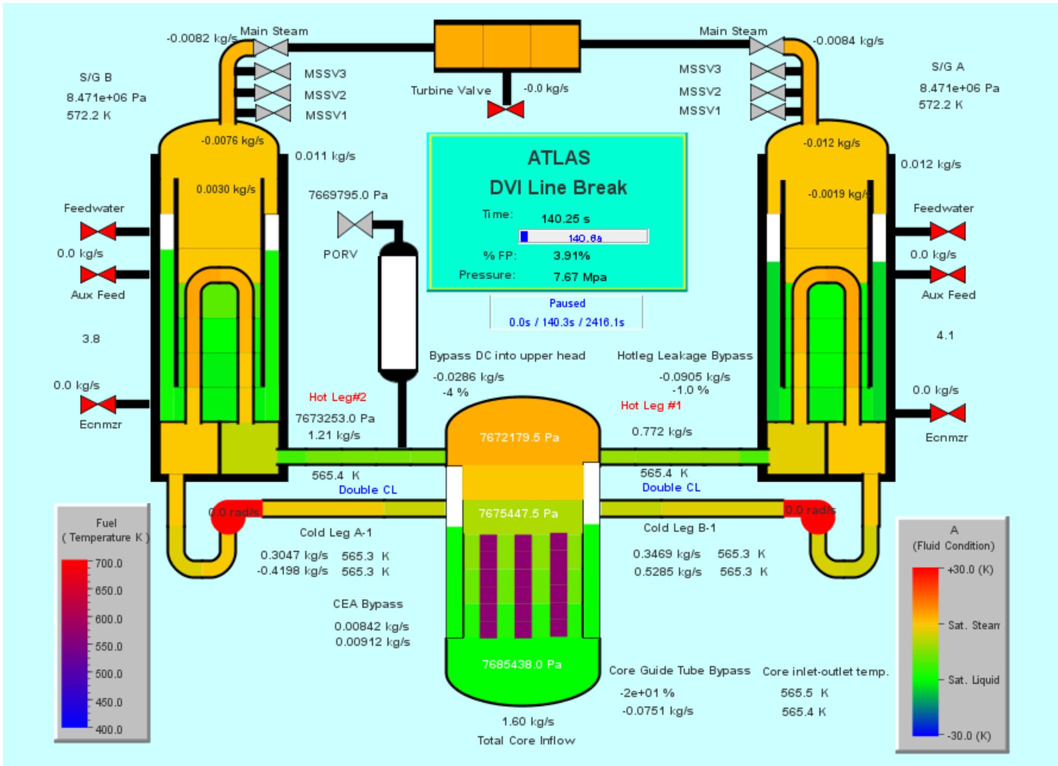


Figure 5.3 RCS Conditions after Loop Seal Clearing (at 140 seconds after break)

The core power is shown in Figure 5.4. The deviations of the core power was regarded as acceptable because we subtracted 5.6% heat loss, which was not measured but estimated from a separate effect test, from the measured core power. [16]. Therefore, the same core power as the experiment was used in this study.

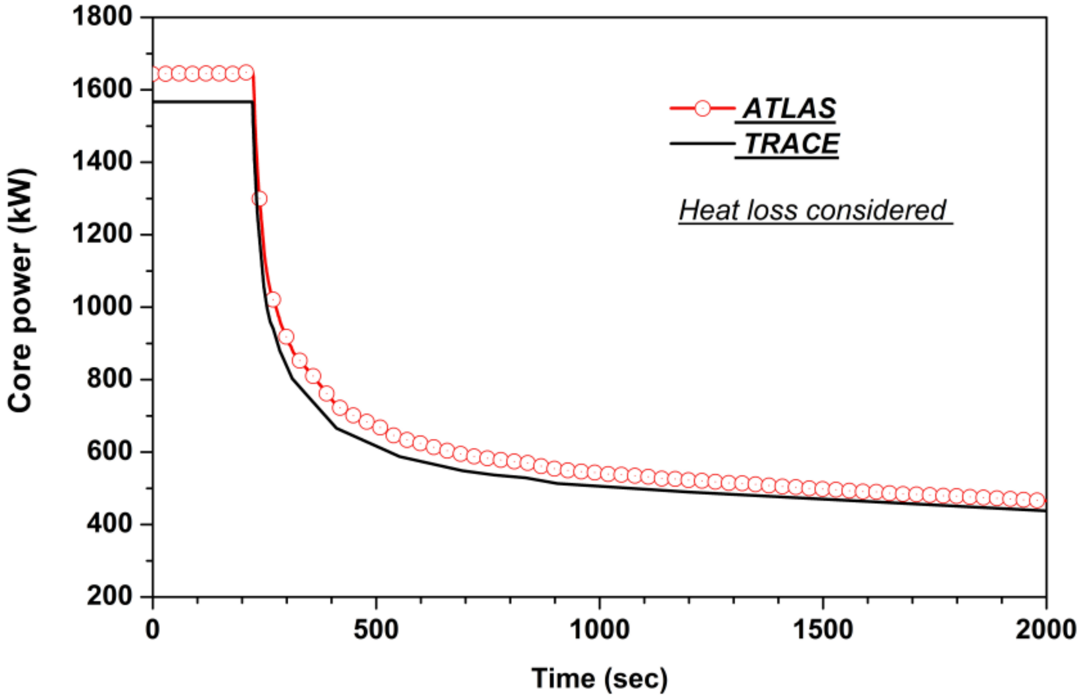


Figure 5.4 Core power

The pressurizer pressure is shown in Figure 5.5. Generally, as soon as the initiation of break occurs at the DVI line, primary pressure rapidly decreases due to the sudden coolant loss and the coolant in the RCS remains in the liquid phase during this blowdown period. As time goes by, the coolant becomes steam by the flashing and the boiling occurs in the core, and steam can be identified in the upper head, upper plenum, and hot legs. After the initial rapid depressurization ends by the flashing and boiling, the primary pressure reaches a plateau just above the saturation pressure of the secondary side between the SIP injection and the loop seal clearing After the loop seal clearing, the primary pressure begins to decrease below the secondary side, and continues to decrease as the break flow continues.

The predicted pressurizer pressure agrees relatively well with the experimental data as shown in Figure 5.5; however, before the SIP starts, the predicted pressurizer pressure decreases slowly compared to the experimental results. This is the reason that the break flow is under-estimated as shown in Figure 5.6. Also, the SIP injection and loop seal clearing occurs slightly later than the measured time due to the under-estimation of break flow. After the SIT injection starts, the predicted pressure reduces more rapidly than the experiment due to the over-estimation of steam condensation in the vessel.

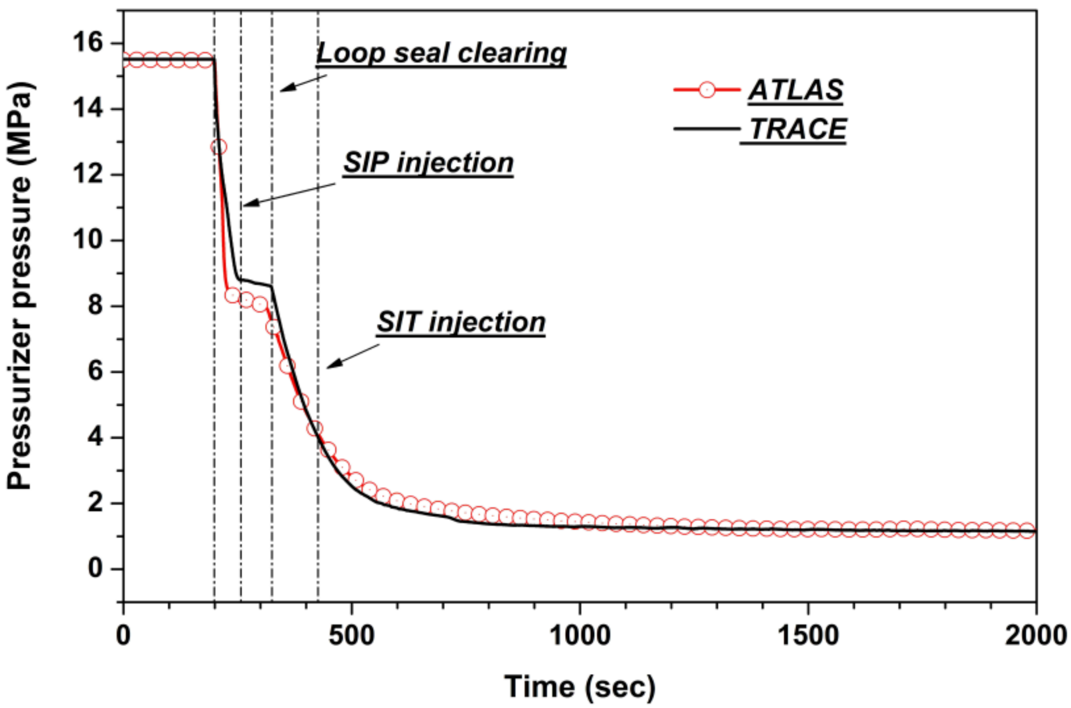


Figure 5.5 Pressurizer Pressure

The predicted break flow rate is compared with the experiment in Figure 5.6. The extension version of the Ransom-Trapp model is used as a default two-phase choked-flow model in TRACE [15]. Therefore, the sensitivity studies for break flow were performed to find the optimal subcooled and two-phase multipliers in the choked-flow model. However, even with the sensitivity study, the predicted pressure reduces more slowly than the experiment and then the predicted break flow shows large difference from measured data before LSC as shown in Figure 5.6. Since the pressure difference between RCS and containment is substantial, the critical flow condition is maintained during the calculation. After the instance of break, the predicted break flow rate grows rapidly like in the experiment, while the value is lower than in the experiment. Actually, measuring of break flow is difficult in the experiment. As shown in Figure 5.6, there are fluctuations before LSC, and it is considered that large uncertainties for measurement of break flow in early period of the transient make these fluctuations [8]. Therefore, it is not proper to say if the break flow before LSC is under-estimated or over-estimated, though a rough guess is possible from the comparison of the integrated break flow.

Figure 5.7 shows the accumulated break mass, and before LSC, predicted break mass is slightly smaller than that in the experiment due to the under-estimation of the break flow. However, because the predicted LSC starts at a higher pressure, the break flow and mass are over-predicted between LSC and SIT injection. After LSC, the code under-estimates the mass discharged, since the predicted break flow rate is slightly smaller than in the experiment by around 420 seconds. As shown in Figure 5.5, the primary pressure suddenly decreases as cold SIT water is injected around 420 seconds, therefore, break flow rate decreases during this period.

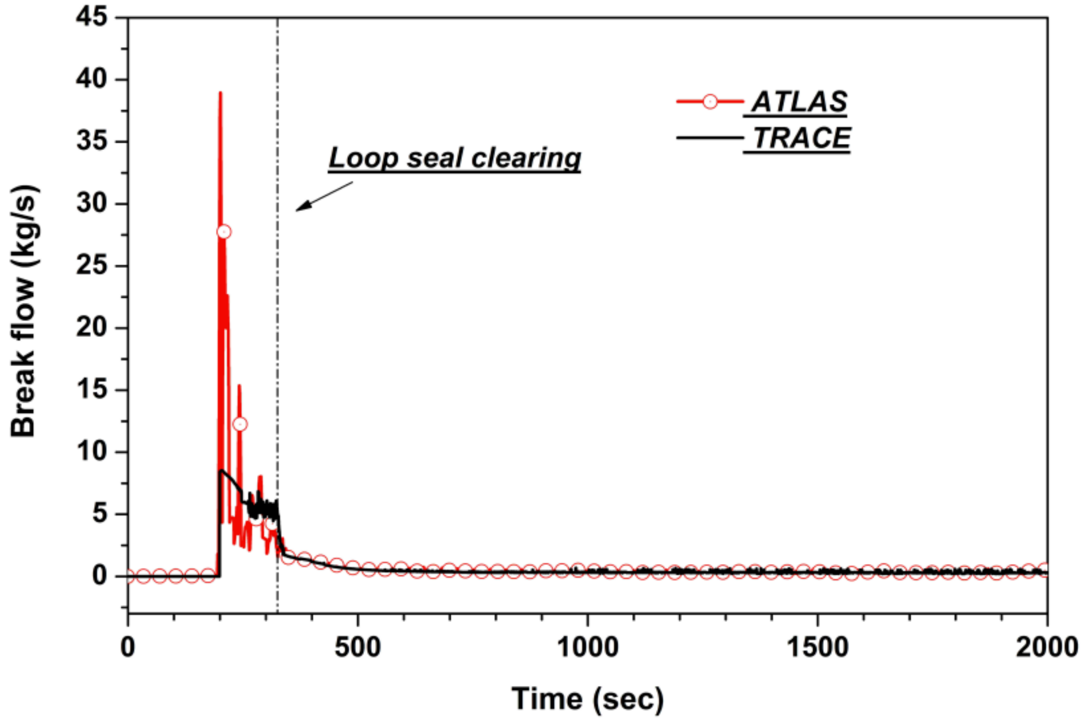


Figure 5.6 Break Flow Rate

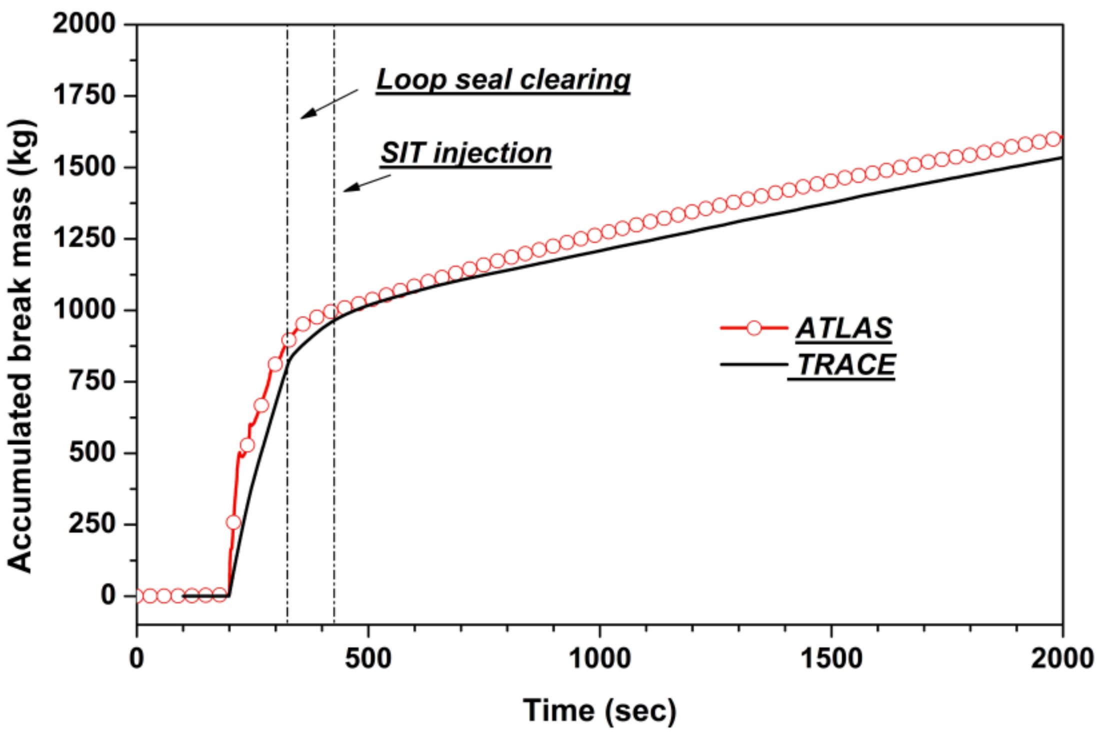


Figure 5.7 Accumulated Break Mass

During a PWR SBLOCA, there are three distinct core heat-up potentials, and the first heat-up is caused by loop seal formation and the manometric core liquid depression.[1] This heat-up is naturally mitigated by the occurring loop seal clearing. Therefore, its timing is very important to core water level and cladding temperature. In this case, liquid pools trapped in the RCP side intermediate leg are the loop seal, and act as plugs for steam flow from SG.

Figure 5.8 and 5.9 show the core and the downcomer collapsed water level in the experiment and TRACE calculation. In general, the core collapsed water level is tightly coupled with the downcomer collapsed water level. In this study, the behavior of the core and downcomer collapsed level can be analyzed better by dividing it into that before the loop seal clearing at ~ 320 seconds and that after the loop seal clearing. Before the loop seal clearing, the code could not predict well the behavior of the core collapsed level. After the break, both the measured and predicted core collapsed water level decrease for the initial ~ 10 seconds due to the coolant loss through the break but the depth of reduction is under-predicted. In the experiment, the collapsed water level increases more than the initial level for ~ 15 seconds after the initial ~ 10 seconds, and it is considered to be a sensor detection error. After beginning of the SIP injection, the injected cold water contracted the downcomer water level in the experiment. This water contraction and the build-up of pressure in the upper region of the vessel continuously reduced the core water level until ~ 290 seconds. However, in the TRACE calculation, the steam generated in the core moves to a cold leg through a hot leg and a steam generator. The moving steam is blocked by the non-saturated water in the loop seal of each loop and then the pressure of the upper plenum slightly increases until the loop seal clearing as shown in Figure 5.10. Therefore, during the period between the SIP injection and the LSC, the downcomer water level shows the stagnation like the pressure behavior because of the interaction between the injected SI water in the downcomer and the build-up pressure in the core. The stagnation of the downcomer water level cannot reduce the core water level to the extent of the measured value and the pressure build-up is also not strong enough to shrink the core water level. Therefore, the predicted water level isn't changed largely like the measured value and maintained as 2.0 m ~ 2.25 m.

At the time of loop seal clearing (~ 290 seconds), the measured core water level dropped rapidly for ~ 15 seconds and then increased for ~ 7 seconds. The measured behavior of core water level around the loop seal clearing time can be regarded as reasonable because a typical loop seal clearing promotes the steam venting to the break and results in a rapid decrease of core water level. In the TRACE calculations, after the LSC, both the downcomer and core water level are reduced rapidly until the SIT injection starts. After the injection of SIT, the core water level is almost identical to the measured value but the downcomer water level is under-estimated because of the low SIT injection water and the large inventory in the intermediate leg.

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[1] The second heat-up occurs following the core quench caused by loop seal clearing and is caused by a simple core boil-off. The third heat-up can occur following depletion of the accumulator tanks and before LPIS injection begins.

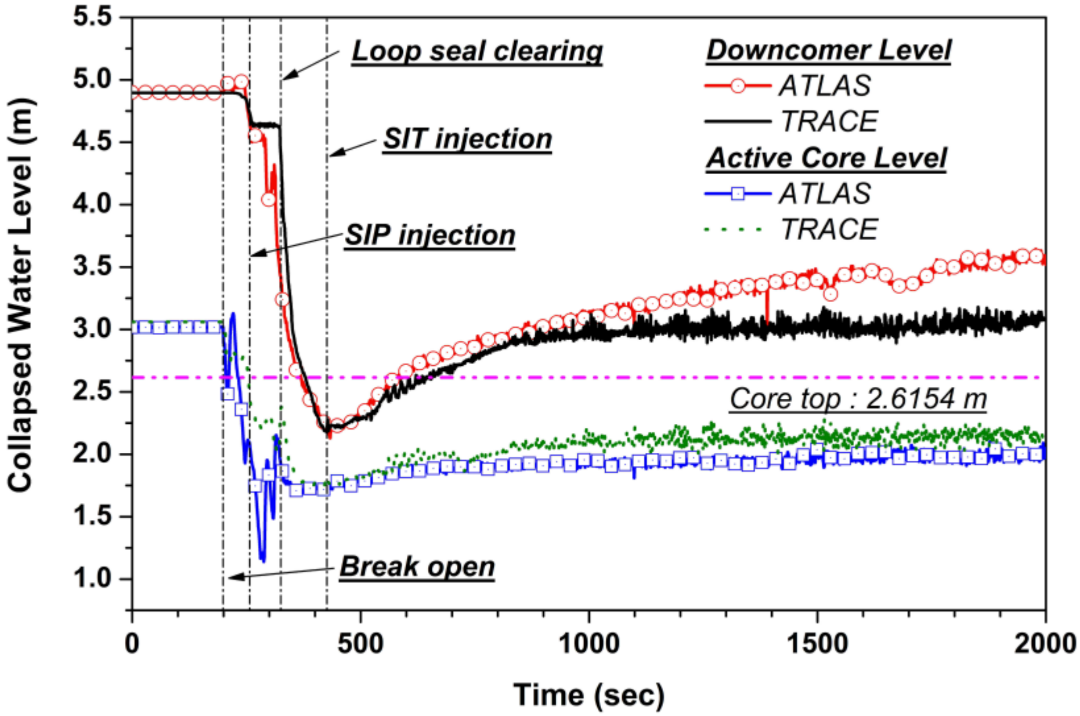

Figure 5.8 Core & Downcomer Collapsed Water Level

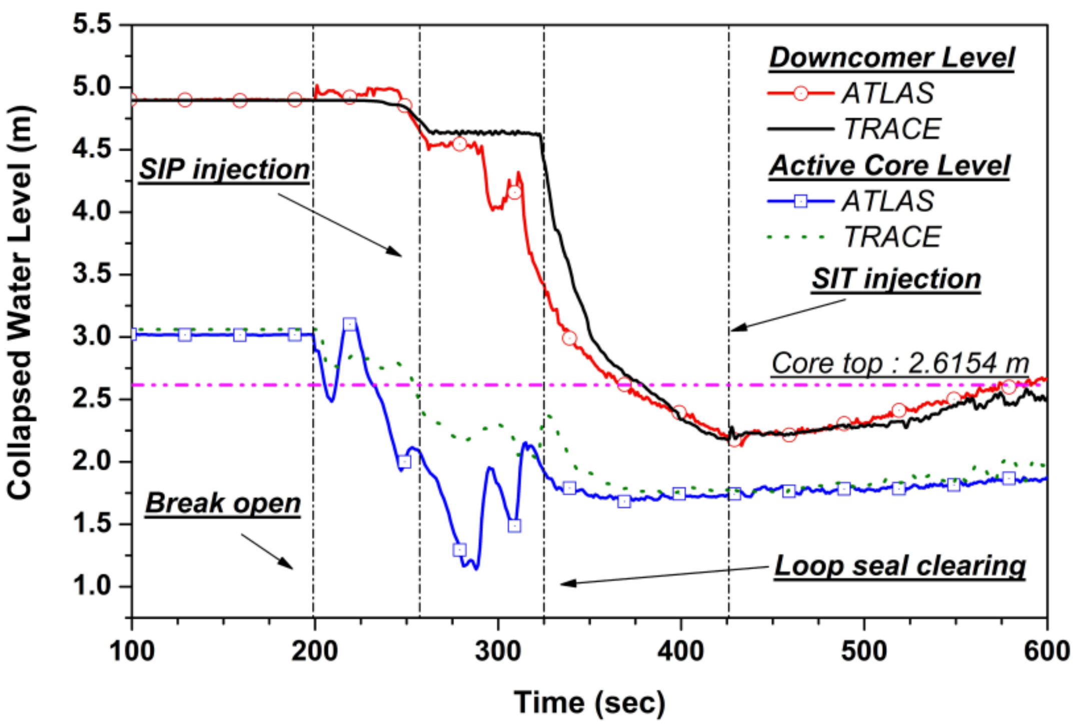

Figure 5.9 Core & Downcomer Collapsed Water Level in the Initial Period

As mentioned before, the accurate prediction of loop seal clearing is very significant in the SBLOCA analysis since it can influence the core water level and cladding temperature. The loop seal clearing occurs when the pressure in the upper region of vessel has enough force to overcome the hydrostatic pressure of liquid pools in the loop seals.

The collapsed water level of the RCP suction leg is shown in Figure 5.10. As shown in this figure, all 4 pump suction legs or loop seals are cleared completely and suddenly at ~ 300 seconds in the experiment. The predicted water levels in the 4 pump suction legs also drop rapidly and all loop seals are cleared completely ~ 25 seconds later than the measured value due to the difference of depressurization behavior. It is known that the sequence and location of loop seal clearing are influenced by the location and size of the break. Figure 2.3 shows the arrangement of the primary loop of the ATLAS including DVI lines. As shown in this figure, the break occurs at the DVI-4 nozzle, and the DVI-3 nozzle is the closest the DVI nozzle from the break and the DVI-1 nozzle is the next closest. However, in the sensitivity study calculations for the location of the break, it is seen that the location and the number of loop seal clearing has no trends and LSC appears like chaotic phenomena, not depending on the flow resistance [11]. Therefore, comprehensive studies to investigate the mechanism of loop seal clearing should be performed. However, the timing of LSC is more important than the location and number of that in a safety point of view.

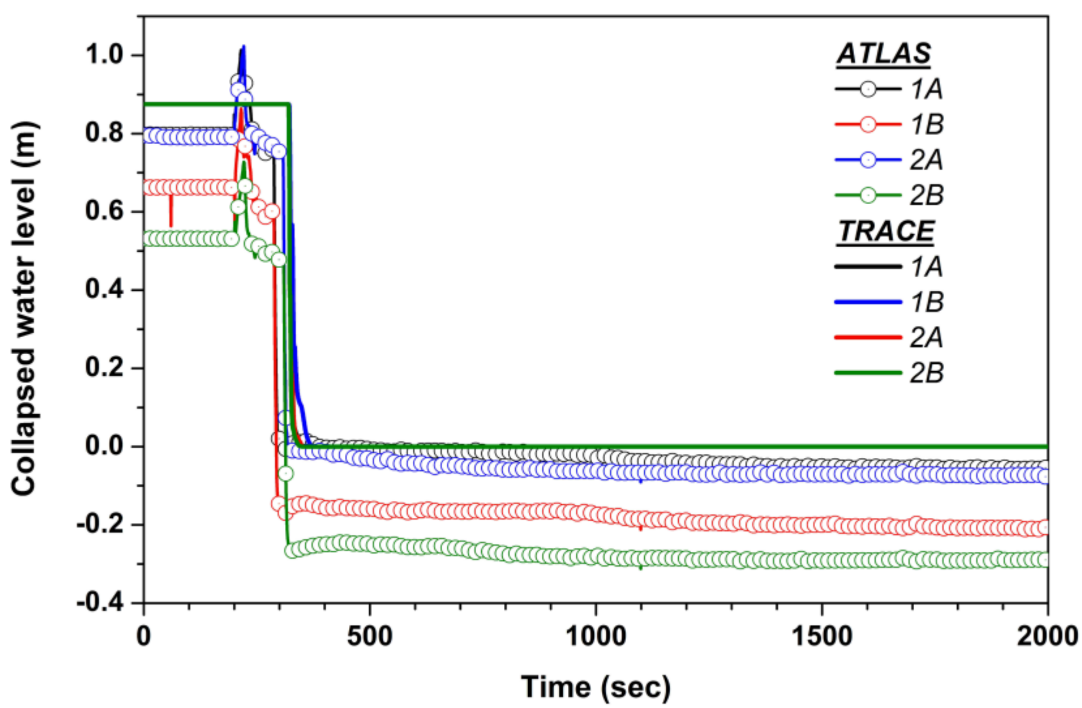

Figure 5.10 Pump Suction Leg Collapsed Water Level

The predicted safety injection flow rate is compared with the experiment in Figure 5.11 and 5.12. For the safety injection pump (SIP) flow, code calculation result is in good agreement with the experiment results before the SIT injection. After the SIT injection, predicted values are slightly larger than the experimental data since the measured flow is reduced due to the interaction of SIT and SIP flows injected through the same DVI nozzle as shown in circle mark of Figure 5.11.

For the SIT flow, injection starting time agrees well with that of the test, and the prediction shows much higher oscillations than in the experiment. We are not sure, but highly oscillating interfacial heat transfer in the downcomer after the SIT injection may result in some oscillations in the downcomer pressure and subsequent oscillations of SIT flows. Also, high oscillations of SIT flows can occur due to some numerical problem. Differently from the experiment, at some regions after ~ 1,000 seconds, SIT water is not injected. From previous studies, it is reported that if SIT is modeled by the PIPE component (not ACCUM component), SIT injection flow rate does not have any oscillations [17]. Therefore, models of the ACCUM component, especially numerical scheme should be improved to predict a more accurate SIT injection flow rate. As mentioned in Figure 5.8, the predicted SIT flow is under-estimated and this results in the under-prediction of the downcomer water level after the SIT injection.

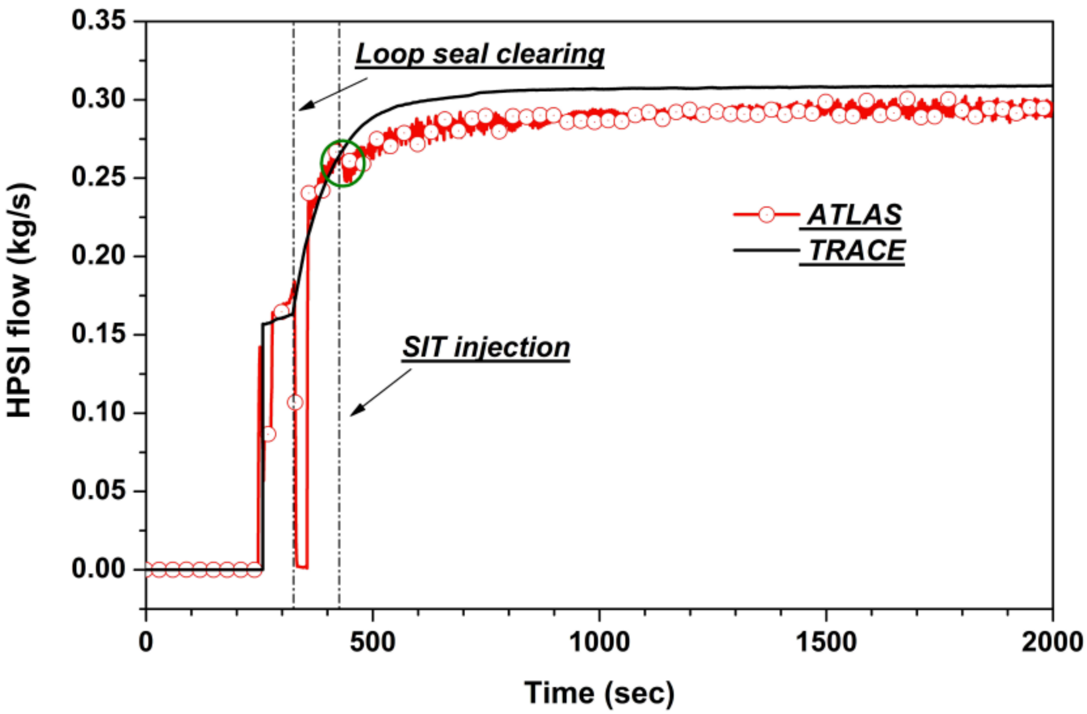

Figure 5.11 Safety Injection Pump (HPSI) Flowrate

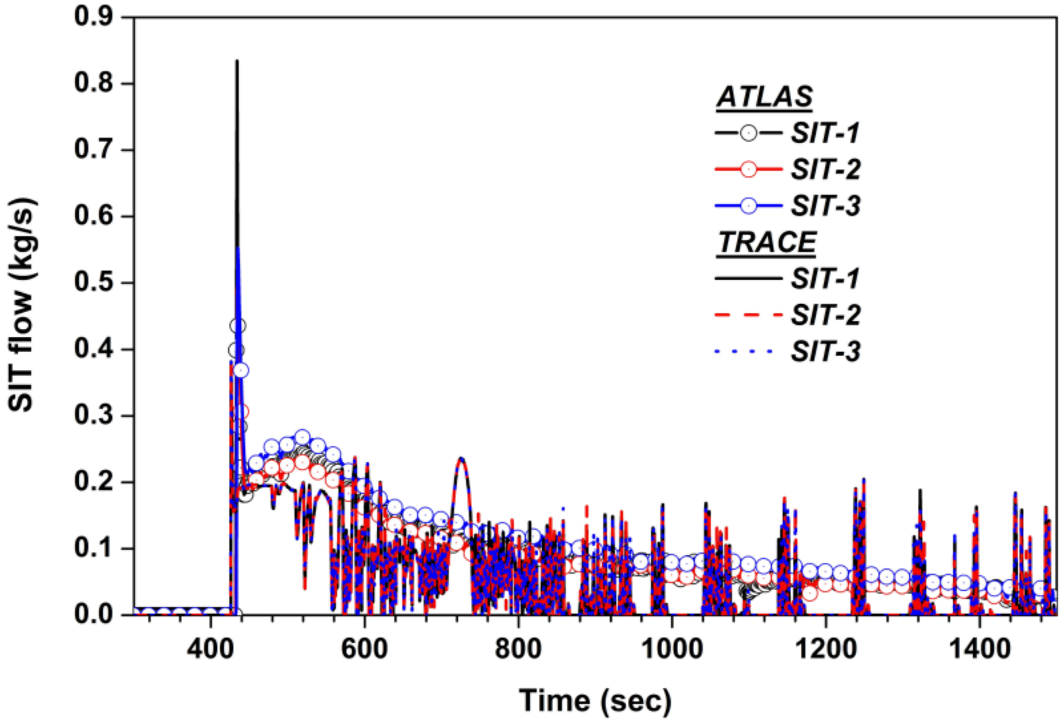

Figure 5.12 Safety Injection Tank (SIT) Flowrate

Figure 5.13 shows the comparison of hot leg flows in the experiment and calculation. The code significantly over-predicts the hot leg flow before ~ 250 seconds. It is considered that this over-prediction of hot leg flow results in the under-prediction of the core collapsed liquid level before ~ 230 seconds. After ~ 250 seconds, a large amount of water is predicted to flow from the steam generators to the core for ~ 75 seconds as shown in the circle mark of Figure 5.13. However, in the experiment, the water continuously flows from the core to SG. This negative flow from the steam generators to the core results in over-prediction of the core water level in a period between SIP injection and LSC. As in a previous study [18], it is reported that if the CCFL (Count-Current Flow Limitation) option is applied in a FAP (Fuel Alignment Plate), the backward flow from SG to core disappears when RELAP5 and MARS are used. Therefore, in the sensitivity study of Chapter 6, we identify whether or not the CCFL model used at FAP can restrict the flow from the steam generators to the core. After the LSC, the increase of flow to hot leg can cause the reduction of the core water level.

The predicted and measured cold leg mass flow rates are shown in Figure 5.14. At the beginning of the transient, the code shows a tendency to over-predict cold leg flows, but the difference between the experiment and actual calculation is not as large as the case of hot leg flows. At the time of loop seal clearing (~ 320 seconds), the mass flow of each loop increases rapidly in the calculation.

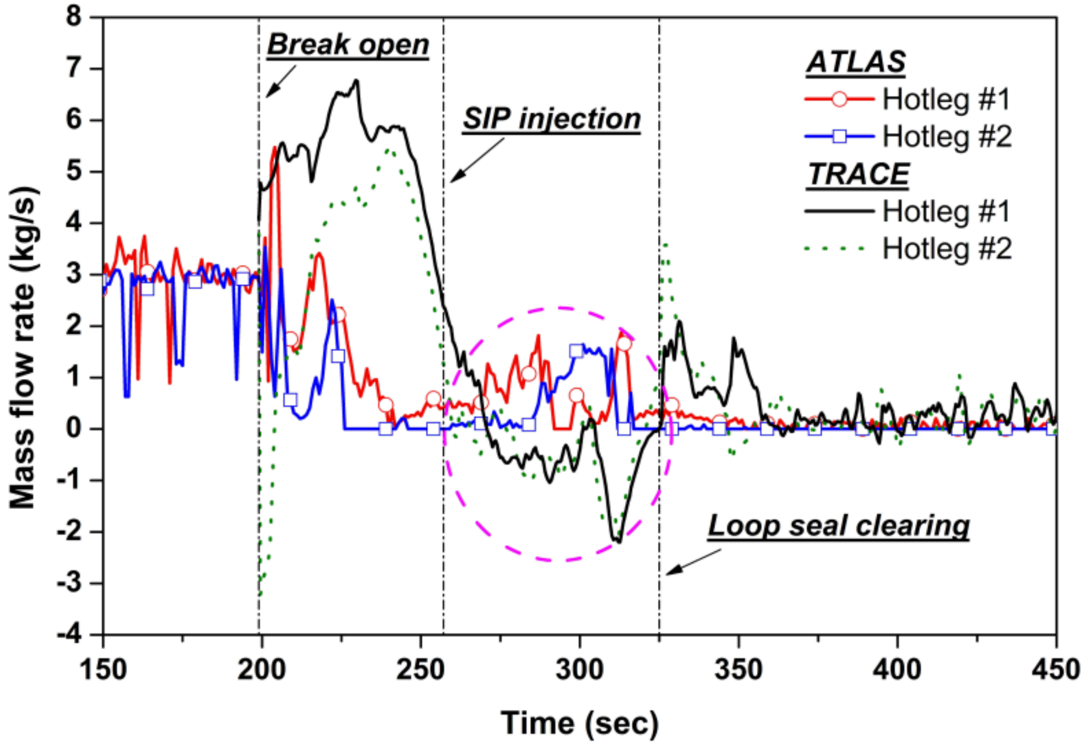

Figure 5.13 Hot Leg Massflow Rate

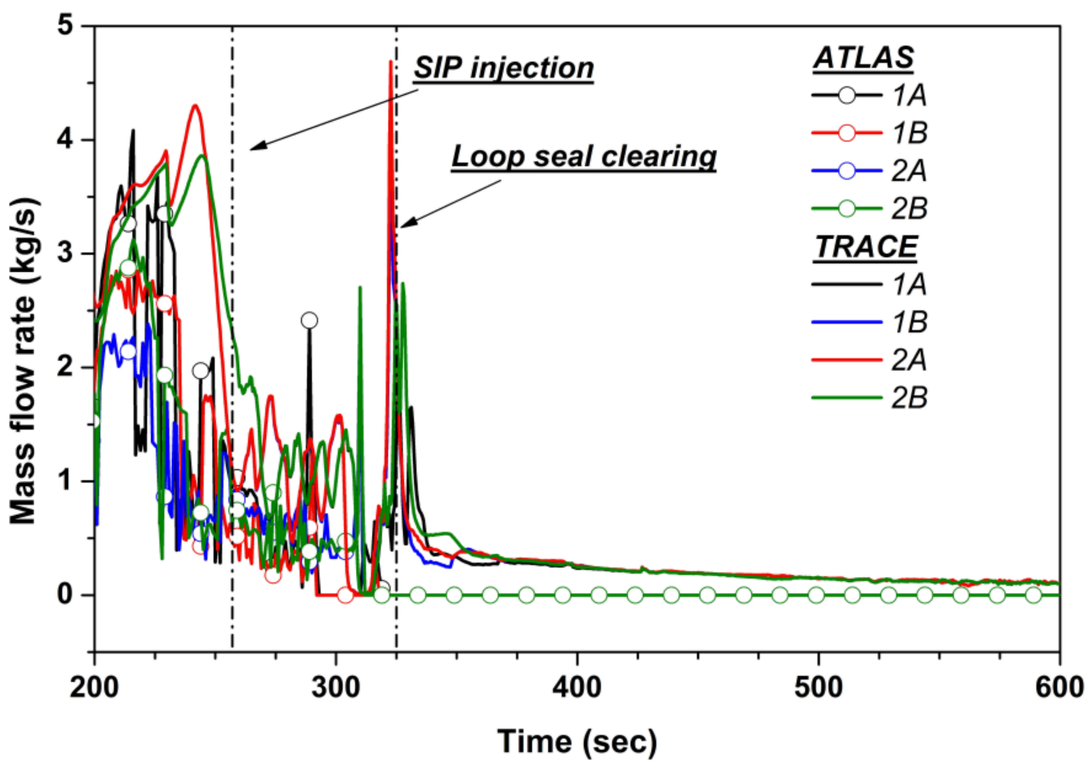

Figure 5.14 Cold Leg Massflow Rate

Figure 5.15 shows the measured heater rod surface temperature and predicted one. The measured peak cladding temperature is 632 K at ~ 290 seconds while the peaking for predicted cladding temperature is not shown in this study. The predicted cladding temperature generally agrees well with the experiment results except in the peaking behavior. This peaking behavior of cladding temperature is directly coupled with the core collapsed water level. The measured core collapsed water level decreases up to ~ 1.1 m but the predicted water level is over-estimated between the SIP injection and the LSC as shown in Figure 5.9. The heater rod surface temperature has a similar trend of saturation temperature and primary pressure. Better predictions of discharging flow and core water level should be made first to predict peak cladding temperature more accurately

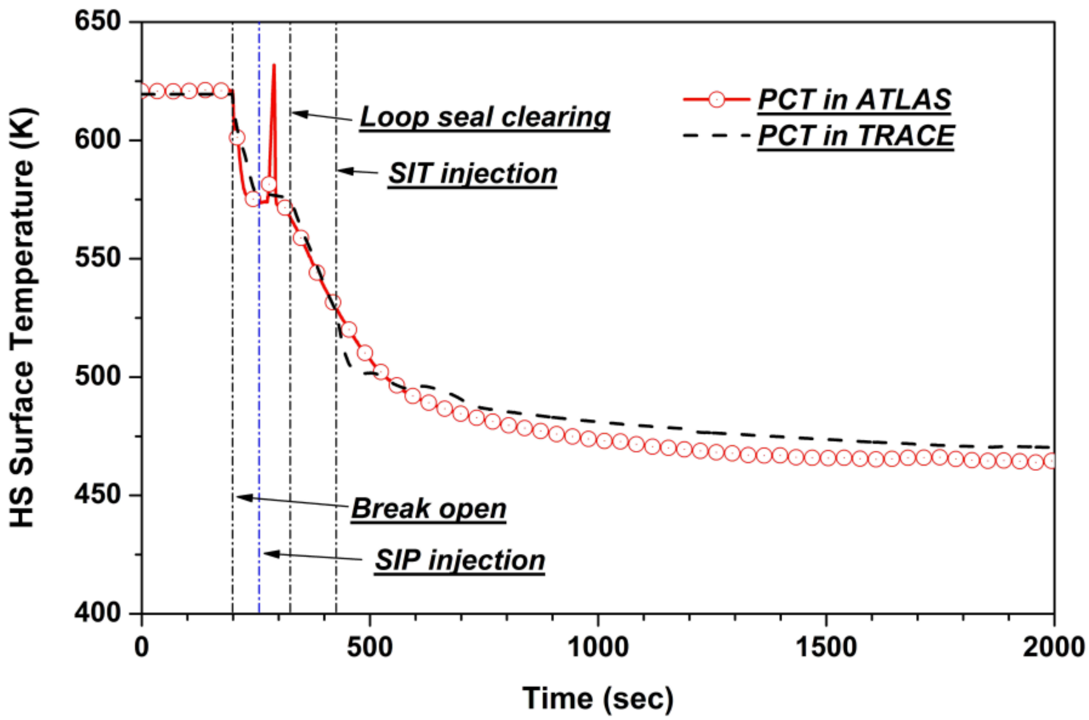

Figure 5.15 Heater Rod Surface Temperature

6. SENSITIVITY STUDY

6.1 Increase of Break Flow

As we explained in the prior chapter, after initiating the DVI line break, the predicted pressurizer pressure decreases slowly compared to the experimental results until the SIP starts. This is the reason that the break flow is under-estimated as shown in Figure 5.3. Low discharged mass at break also results in the over-prediction of the downcomer and core water levels during this period. Therefore, the sensitivity study for the break mass is performed to identify the effect of depressurization. The break area is arbitrarily modified to simulate a similar accumulated break mass with experimental data as shown in Figure 6.1.

The downcomer and core water level predicted in this sensitivity run are compared to those of the base case calculation in Figure 6.2 and 6.3, respectively. Shown in the figure, if we increase the break flow, the downcomer water level decreases more rapidly and the stagnation period of the downcomer water level becomes short since the loop seal clearing occurs earlier around ~ 300 seconds compared to the base case. Before the loop seal clearing, the downcomer water level shows the better prediction results with the measured values. After break, the predicted core collapsed water level in this sensitivity run decreases initially ~ 10 seconds and the depth of reduction is closer to the measurement due to the assumption of more coolant loss. The predicted core water level generally agrees well with the experiment results until ~ 260 seconds except for the abnormal increment of measured value at ~ 225 seconds that is considered as the sensor detection error. However, in the sensitivity run, the core water level still cannot be reduced to the extent of the measured value around 290 seconds due to the stagnation of the downcomer water level and weak pressure build-up in the core. Also, except for the periods before LSC, the overall behavior for main parameters shows the worse prediction results compared to the base case.

Generally, there are the Henry-Fauske model and the Ransom-Trapp model as the choked-flow model of RELAP5 code [19] but the Ransom-Trapp model is only the two-phase choked-flow model in the TRACE code. The Henry-Fauske model has two input parameters that have to be given by the users. They are a discharge coefficient (C_D) and a non-equilibrium factor (N_{eq}). In order to find out the optimal C_D and N_{eq}, the sensitivity study is performed usually for the pressure behaviors. Both models in RELAP5 have been used widely in the real-plant and experiment calculations. Therefore, if the Henry-Fauske model would be implemented in the TRACE code, the diversity for the choked-flow model can be increased for a better prediction of break flow.

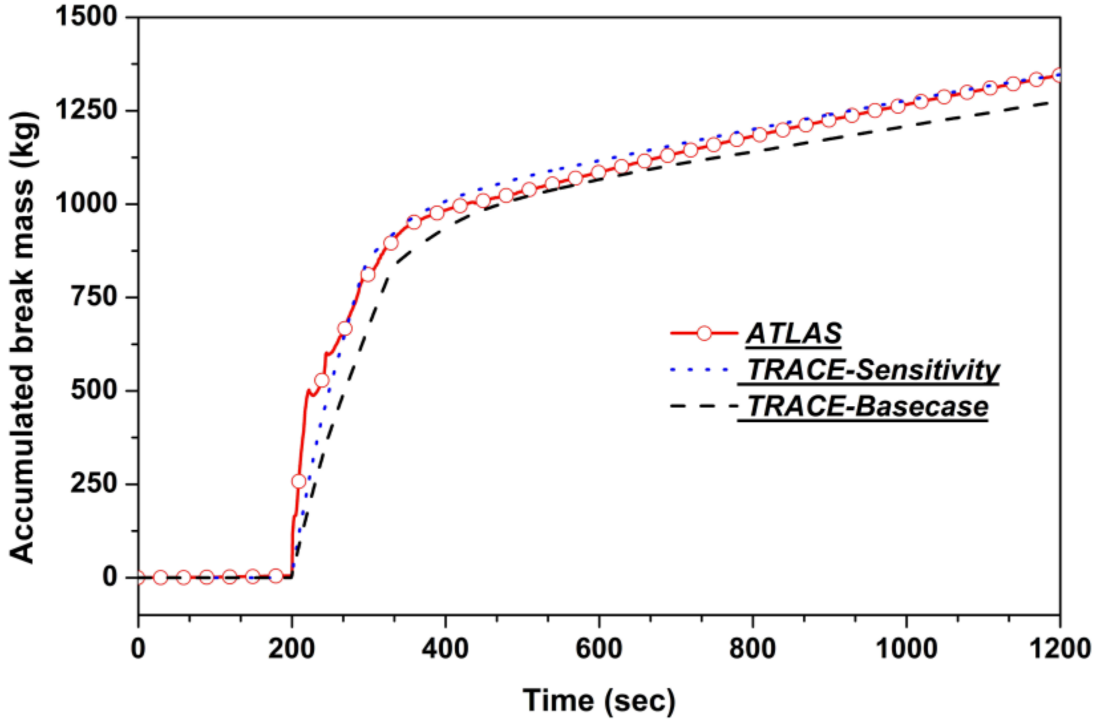

Figure 6.1 Accumulated Break Mass

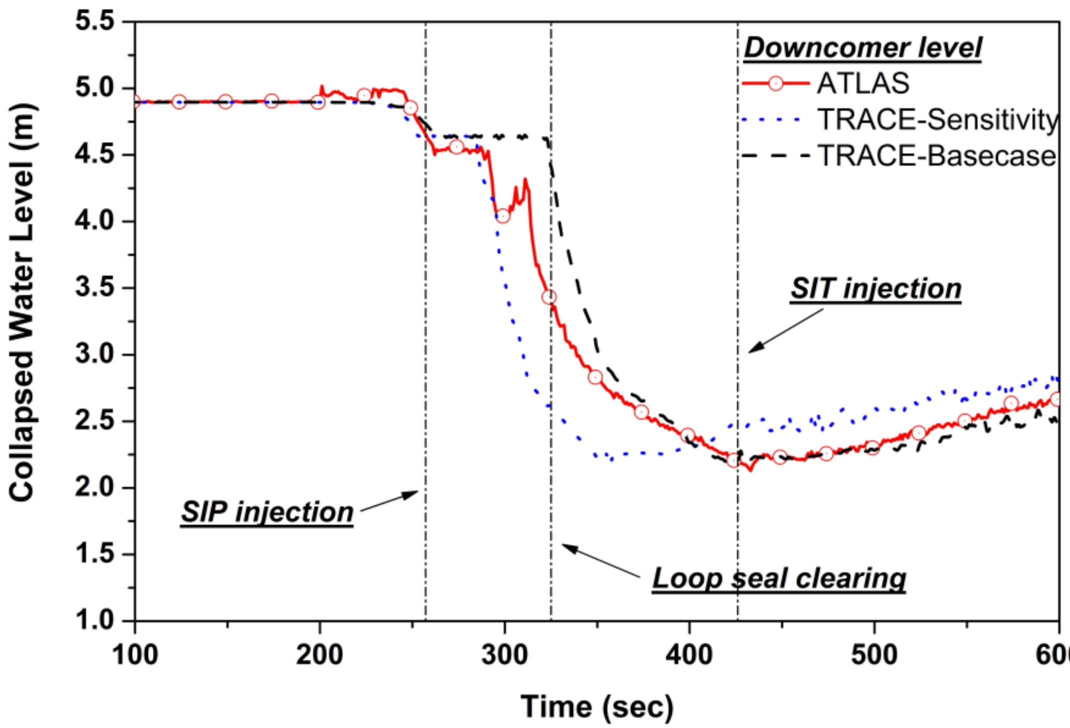

Figure 6.2 Downcomer Collapsed Water Level

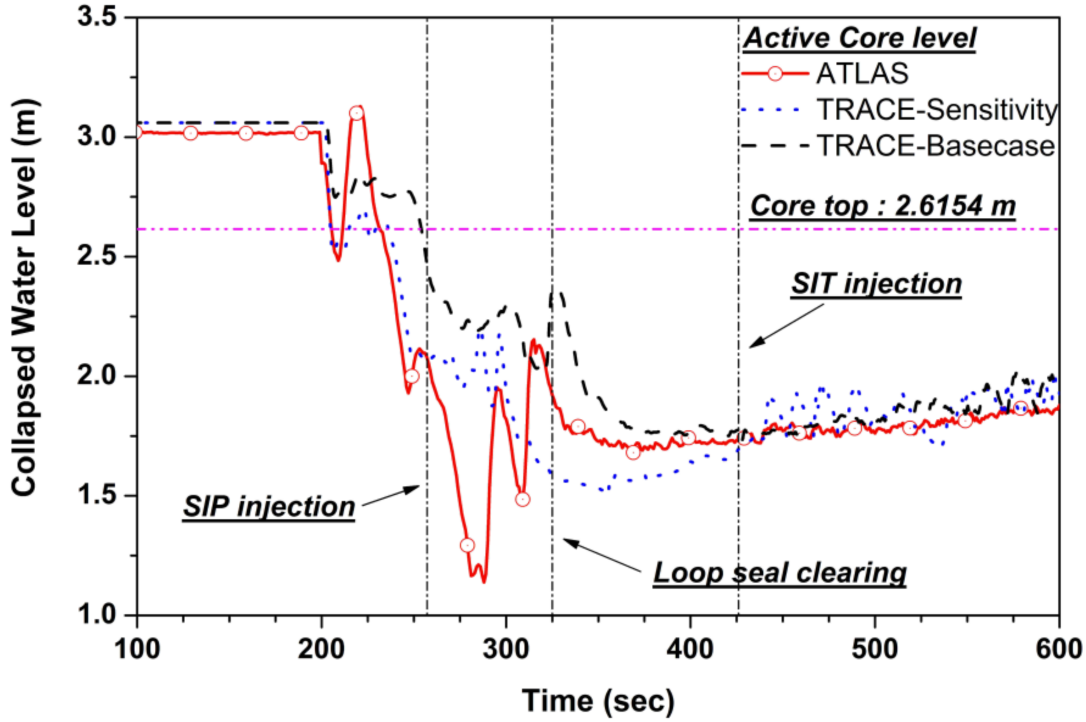

Figure 6.3 Core Collapsed Water Level

6.2 CCFL Model in the Upper Plenum

In the base case calculation, relatively large amount of water is predicted to flow from the steam generators to the core from ~ 250 seconds for ~ 75 seconds as shown in Figure 5.10 and it is believed that such a large liquid flow to the core is the main reason why the predicted core collapsed water level is not predicted to decrease much during the same period. The measured core water level reduced continuously during this period to reach its minimum at ~ 290 seconds (Figure 6.3). During this period, the steam generated in the core should flow out to the hot legs through the upper plenum. Therefore, there must be a steam/water counter-current flow in the upper plenum when the water flows from the hot legs to the core. At that time, the water flow from the upper plenum to the core can be restricted by the CCFL (Counter-Current Flow Limitation).

In the base case, the CCFL model in the upper plenum is not considered at all. Therefore, we change the axial edge data of the upper inactive core volume (V2A20 in Figure 3.1) connecting the fuel alignement plate (FAP) to turn on the CCFL model. The Wallis model and Kutateladze model are used by adjusting the 'bankoff interpolation' input.

If the CCFL model, as expected, restricted the liquid flow from the upper plenum to the core, the predicted core water level has to agree better with the measured value. In this study, despite using the CCFL model, the predicted core water level reduces a little more than that of the basecase and the significant decrease for the core water level isn't shown from ~ 250 seconds to ~ 300 seconds. This shows the different results compared to that of the RELAP5 code. Figure

6.5 shows the sensitivity results for the CCFL model of RELAP5/MOD3.3 which was performed in DSP2 [20]. Though the 6-inch cold leg break is analyzed in DSP2, we can find out that the overall thermal hydraulic behaviors are similar to those of the DVI line break. The Wallis model, which is the default of the CCFL model of RELAP5/MOD3.3, is applied to the upper core in Figure 6.5. In this case, since the CCFL model restricts the water flow to the core, the predicted core water level follows the measured level much better and it decrease continuously to reach its minimum value as shown in Figure 6.5. So, from more detailed studies, we need to identify if the CCFL model is operated properly or not in the upper core of the vessel component.

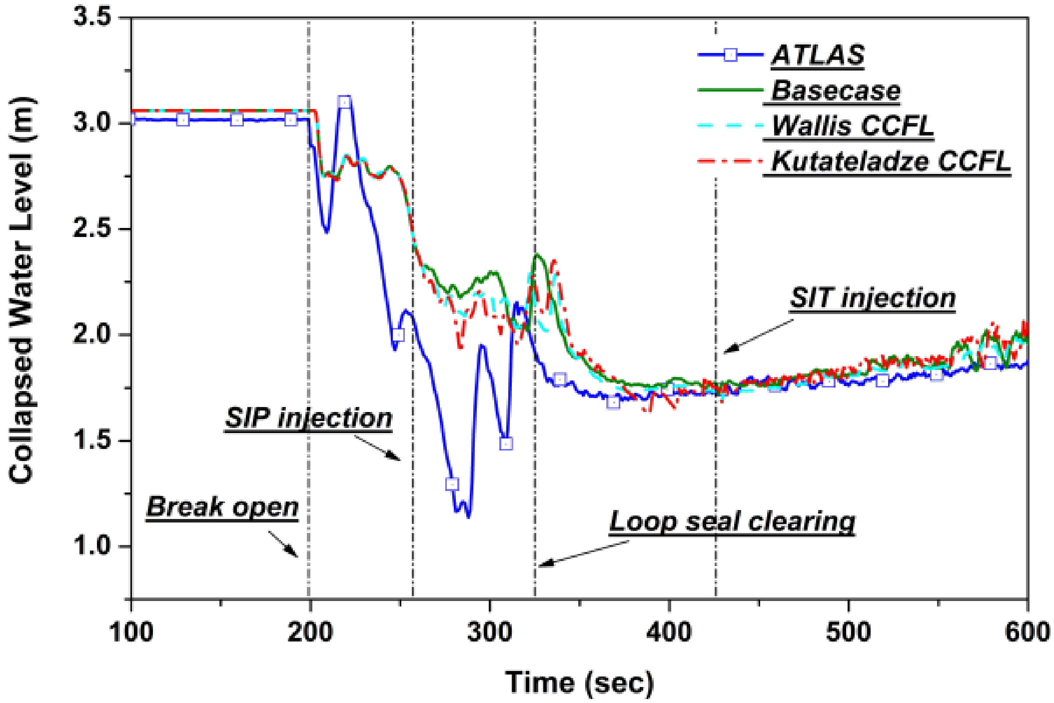

Figure 6.4 Effect of CCFL Model on the Core Collapsed Water Level

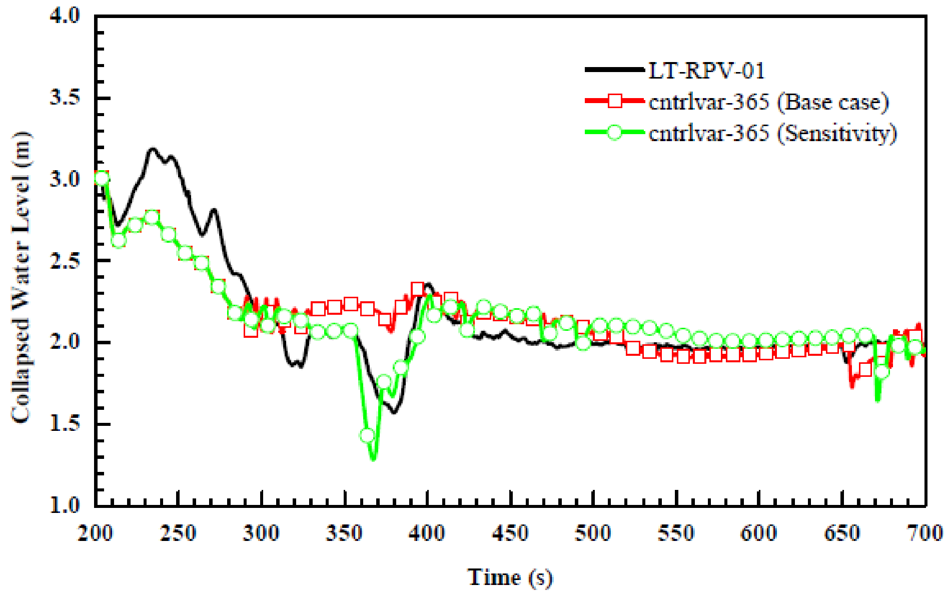

Figure 6.5 Effect of CCFL Model in RELAP5 on the Core Collapsed Water Level [Ref. 20]

7. CONCLUSIONS

Calculations using the TRACE V5.0 patch2 code were conducted for 100% DVI line break test of the ATLAS which is the first domestic standard problem (DSP-01). For the modeling of ATLAS facility, the reactor vessel was modeled in three-dimension, using the VESSEL component and flows within a coolant loop of the primary and secondary sides were modeled in one dimension using the main hydraulic components such as PIPE and TEE in the TRACE codes. Also, the emergency core cooling system, the heater rods and main control logics were properly modeled in order to consider the various conditions of this experiment. The steady state was determined by conducting a null transient calculation and the errors between the calculated and measured values are acceptable for most primary/secondary system parameters.

In the experiment, after the initial steady-state conditions were reached, the DVI line break test was initiated by opening a break simulation valve at 199 seconds, and transient calculation was also conducted by setting up 199 seconds as the initial time. The predicted sequence of events occurred some seconds later than that of experiment due to slow depressurization of the primary side after the break. Before the SIP starts, the predicted pressurizer pressure decreases slowly compared to the experimental results because of the under-estimation of break flow according to the characteristic of the choked flow model. Before the LSC, the code could not predict well the behavior of the downcomer and core collapsed water levels due to the under-prediction of the discharged mass and high flow rate returned to core from hot leg. After the LSC, the downcomer and core collapsed water levels drop rapidly and the code predicts relatively well the trend of the downcomer and core water level. In this study, all RCPs suction legs or loop seals are cleared completely and suddenly, and the code predicted the LSC ~ 25 seconds later than the measured value due to the slow depressurization rate. The predicted cladding temperature generally agrees well with the experiment results except in the peaking behavior. This peak of cladding temperature is directly related to core collapsed water level dip before the LSC.

The sensitivity studies were performed to identify 1) the effect of discharging mass for the depressurization and 2) the effect of flow restriction from hot leg for the core collapsed water level. If the discharging mass increases arbitrarily, the downcomer and core water level before SIP injection shows better prediction results than that of the base case. However, except for this, the overall behavior for the main parameters shows the bad prediction results. For the effect of flow restriction from hot leg using the CCFL model, the predicted core water level was not decreased significantly compared to the base case. The results were different in comparison with that of the RELAP5 code and we need to do further studies to better understand this difference.

In conclusion, the TRACE code has good capabilities to simulate the 100% DVI line break test of the ATLAS. However, there are some discrepancies in quantitatively predicting the primary pressure, break flow, downcomer and core collapsed liquid level, and so on. Therefore, the TRACE code including the choked flow and CCFL models needs to be improved and more detailed modeling is needed to predict more accurate results.

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A. Calvo, NRC Project Manager

11. ABSTRACT *(200 words or less)*

The calculations using TRACE V5.0 patch2 code were conducted for 100% DVI line break test of ATLAS which is the first domestic standard problem (DSP-01). In the steady state conditions, the errors between the calculated and measured values are acceptable for most primary/secondary system parameters. From the transient calculations, the predicted sequence of events occurred some seconds later than that of experiment due to slow depressurization of the primary side after break. Before the LSC, the code could not predict well the behavior of downcomer and core collapsed water levels due to the under-prediction of the discharged mass and high flow rate returned to core from hot leg. After the LSC, the downcomer and core collapsed water levels drop rapidly and the code predicts relatively well the trend of downcomer and core water level. All RCPs suction legs or loop seals are cleared completely and suddenly. The predicted cladding temperature generally agrees well with the experiment except the peaking behavior. In the sensitivity studies, if the discharging mass increases arbitrarily, the downcomer and core water level before SIP injection shows better prediction result than that of base case. However, except for this, the overall behavior for main parameters shows the bad prediction results. For the effect of flow restriction from hot leg using the CCFL model, the predicted core water level was not significantly decreased compared to the base case. In conclusions, TRACE code has good capabilities to simulate the 100% DVI line break test of ATLAS. However, TRACE code including the choked flow and CCFL models needs to be improved and more detailed modeling is needed to predict more accurate results.

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ATLAS (Advanced Thermal-Hydraulic Test Loop for Accident Simulation)
TRAC/RELAP Advanced Computational Engine (TRACE) Code
OPR-1000
APR-1400
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SBLOCA
RELAP5

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