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ATLAS MSLB Accident Analysis Using the SPACE Code

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Abstract: An integral effect test for a MSLB (main stem line break) was performed with the ATLAS (advanced thermal-hydraulic test loop for accident simulation) by KAERI (Korea Atomic Energy Research Institute). A MSLB is defined as a pipe break in the main steam system. This data was used to validate the safety analysis code SPACE (safety and performance analysis code for nuclear power plants). In the test, a double-ended guillotined break of the main steam line was simulated. After steady-state was reached, the test was started by opening the break simulation valves. With the start of the test, the pressure of the secondary system decreased rapidly, and reached the set-point of the LSGP (low steam generator pressure) signal. With the occurrence of the LSGP signal, the main steam isolation valves were closed. The SIPs (safety injection pumps) were started by the LPP (low pressurizer pressure) signal. In order to validate the SPACE code, a double-ended guillotine break of the main steam line at ATLAS was simulated. Most of the results show good agreement between the experiment data and the code calculated values.

Key words: SPACE, MSLB, ATLAS.

I. Introduction

In order to validate the ability of the SPACE (safety and performance analysis code for nuclear power plants) code to simulate MSLB (main stem line break) accident in nuclear power plants, steam line break experiment SLB-GB-02T, performed with the ATLAS (advanced thermal-hydraulic test loop for accident simulation) experiment facility, was analyzed and the results were compared with experimental data.

2. Description of SPACE Code

The SPACE code is an advanced thermal hydraulic analysis code capable of two-fluid, three-field analysis. The SPACE code can be used in LBLOCA (large break loss of coolant accident), SBLOCA (small break Loss of coolant accident) and in Non-LOCA analysis of PWRs. The SPACE code is composed of input and output packages, hydrodynamic model package, heat structure model, control system model and reactor kinetic model, etc.. Hydrodynamic model package is composed of hydraulic solver, constitutive models, special process models and component models. The input and output packages performs read of input file and restart file, input error check, storage space setup, as well as variable initialization, preparation of main output file, plot files, and restart file. Unlike the major best-estimate nuclear reactor system analysis codes in Refs. [1-4], which consider only liquid and vapor phases in their governing equations, SPACE incorporates a dispersed liquid field in addition to vapor and continuous liquid fields: two-fluid, three-field formulation. A dispersed liquid field is expected to be important in annular-mist and post-dryout conditions since a dispersed liquid field behaves differently with a continuous liquid field. This is the major reason to incorporate a dispersed liquid field as an additional liquid field. The three fields are allowed to be at non-homogeneous and non-equilibrium state, while the gas field is assumed to be a homogenous equilibrium mixture of vapor and non-condensable gas. The governing equations also involve porosity to take into account the structural material impact on the fluid flow. Each field equation is discretized by applying finite volume method to the unique SPACE mesh

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system which naturally encompasses various three dimensional structured and/or unstructured mesh systems, as well as one-dimensional pipe meshes. Non-linear terms appeared in the temporal and source terms of the phasic mass and energy equations are linearized by using Taylor expansion technique. Semi-implicit scheme is chosen as the basic numerical time advancement scheme.

The proper physical models can significantly improve the accuracy of the prediction of a nuclear reactor system behavior under many different transient conditions because those models are composed of the source terms for the governing equations. To develop the physical models and correlations for the SPACE code, various models currently used in major nuclear reactor system analysis codes, such as RELAP5, TRAC-M, COBRA-TF, and MARS, have been reviewed. In addition, a literature survey of recent studies has been performed in order to incorporate the up-to-date models into the SPACE code. Unlike RELAP5, TRACE and CATHARE which are major best-estimate nuclear reactor system analysis codes that only consider liquid and vapor phases, the SPACE code incorporates a dispersed liquid field in addition to vapor and continuous liquid fields; interfacial interaction models between continuous, dispersed liquid phases and vapor phase have to be developed separately. Constitutive models of the SPACE code is composed of the surface area and surface heat transmission correlation, surface-wall friction correlation, droplet separation and adhesion correlations, wall-fluid heat transmission mode and correlations, all correlations required for governing equations, and the type of the correlation is determined by the flow-form map.

The SPACE code contains special process and system component models to limit or modify the solution of the basic governing equations reflecting the physical phenomena and provide the capability to simulate the systems of nuclear power plant. Major special process and component models are critical flow model, counter current flow limit model, off-take model, abrupt area change model, two-phase level tracking model, pump model, safety injection tank model, valve model, pressurizer model, separator model, heat structure model, neutron kinetics model, etc. The critical flow model calculates the mass flux of fluid when choked flow condition occurs. The critical flow model is important in calculating break flow rate in a LOCA (loss of coolant accident) or MSLB, etc.. For predicting two-phase critical flow, Ransom Trapp model is used as a default model. The CCFL (counter current flow limit) model is used when liquid phase and gas phase flow in the opposite direction. When gas phase flowrate is small, two phases flow in the opposite direction without much interaction. When gas phase flowrate becomes large, the gas phase and liquid phase becomes mixed and limits the amount of liquid flowing in the opposite direction. The CCFL model is used in LOCA analysis. The off-take model is needed when fluid is being removed from a volume where liquid and gas separated and forms a liquid level. Without off-take model, the thermodynamic properties of fluid being removed assume the average value of the volume where it is being removed. However, when liquid/gas phase are separated and liquid forms a level, the thermodynamic properties of the fluid being removed is not the same as the volume average value. The off-take model is used when this condition occurs. The abrupt area change model calculates the pressure loss when there is a sudden change in flow area. The abrupt area change model includes abrupt area expansion, abrupt area contraction and area change with orifice. The two-phase level tracking model is used to predict presence of liquid or mixture level separate from gas phase in two-phase mixture. The pump model is used to model the centrifugal-type reactor coolant pumps in nuclear power plants. This model is required to provide the driving force for primary coolant flow. A positive displace pump model is also being considered for SPACE code. The SIT (safety injection tank) model is used to predict safety injection flow from SIT (or accumulator) in the early stage of LOCA analysis. The

SIT model in SPACE code will be able to model the behavior of fluid device used in APR1400. The valve model is used to stop fluid flow or to control fluid flowrate. The valve models in SPACE include on/off switch controlled valve, pressure controlled valve and time controlled valve. The pressurizer model is used to predict pressurizer behavior during transient conditions. The separator model is used to simulate the steam separation process in SG (steam generator) secondary side. The heat structure model is used to calculate heat addition/removal from a control volume. The heat structure model includes transient heat conduction in rectangular or cylindrical geometry. The boundary conditions at heat structures include convection, radiation, user specified temperature, user specified heat flux or user specified heat transfer coefficient. The neutron kinetics model is used to calculate core power. The neutron kinetics model in SPACE code is point kinetics with ANS decay heat models. If 1D or 3D neutron kinetics calculation is needed, a separate neutronics code will be linked with SPACE code to provide core power. The main features of SPACE are described in Refs. [5-7].

3. Experiment Facility and Condition

ATLAS is an experiment facility that was set up at KAERI for estimations, analyses and evaluations of various integral phenomena which can occur with the APR1400 and OPR1000 designs through experiments [8]. The APR1400, a reference reactor of ATLAS, is a 4,000 MW PWR type reactor consisting of two hot legs, four RCPs (reactor coolant pumps) and cold legs, and two steam generators. Therefore, ATLAS uses DVI (direct vessel injection) as its safety injection facility to simulate a reference reactor. It also uses a CLI (cold-leg injection) facility to simulate the OPR1000 design. The geometric scale of ATLAS is 1/288 for the volume and 1/2 for the height scale as compared to the APR1400. In addition, the transient proceeding time is known to be faster than that of the APR1400 by about 1.414 times. The wall roughness equals the APR1400 level, and the hydraulic diameter is also of the same scale. The main initial conditions of ATLAS and the APR1400 design are listed in Table 1.

The hydraulic system of ATLAS is composed of a primary system, a secondary system, a safety injection system, a break system, a containment system and an auxiliary system. The primary system include a reactor vessel, two hot legs, four cold legs, a pressurizer, four RCPs and two steam generators. The secondary system is a simple circulation loop. The steam from the two steam generators is condensed by a direct condensation tank with a subsequent injection into the steam generator. As previously stated, the safety injection system has a DVI and a CLI to simulate the system of the APR1400 and the OPR1000. These systems include four SITs (safety injection tanks) and SIPs (safety injection pumps), to simulate a safety injection and, long-term cooling. Also, ATLAS can charge its auxiliary feed by using the charging pump and simulate a low-pressure safety injection, stop cooling, and engage a recirculation operation through the stop cooling pump and the stop heat exchanger.

The break system has various pipes and break facilities to simulate a large loss of coolant accident, a DVI line break, a steam generator tube rupture, a MSLB and a main feed line break. Each break pipe

Table 1 ATLAS SLB-GB-02 experiment main initial conditions.

Design parameters	APR1400	ATLAS
Reactor vessel		
Normal power, MWt	3,983	1.634
Pressurizer pressure, MPa	15.5	15.56
Core exit temp., ^{o}C	324.2	294.6
Core inlet temp., ^{o}C	291.3	289.6
Core flow, kg/s	20275	65.6
Steam generator		
SG-1 steam flow, kg/s	1,152.4	0.388
SG-2 steam flow, kg/s	1,152.4	0.420
Saturated steam press., MPa	6.9	7.33
Primary piping		
Hot leg flow, kg/s	10,496	32.8
Cold leg flow, kg/s	5,540.1	16.4
Hot leg temp., ^{o}C	323.3	294.5
Cold leg temp., ^{o}C	291.3	289.5

consists of a rapid-open valve, a break nozzle and an instrumentation panel. Containment system can press to system using discharged steam at break for simulate a phenomenon in the containment. Also, ATLAS has auxiliary systems, which include a supplementation water system, an instrument cooling system, a nitrogen/air/steam supply system, a vacuum system, and a heat trace system.

The break system for the SLB-GB-02T MSLB is composed of two rapid-open valves (OV-BS-09, 10), a break flow discharge pipe, a flow restrictor, and instrumentation facilities in the form of orifice flow meters (QV-MS1-03, QV-MS2-03) [9]. A flow restrictor is installed in the steam generator upper-head steam discharge section, the function of which is to restrict steam discharge at the MSLB as in an actual real plant. The flow restrictor follows the ATLAS scaling method, with an inside diameter of 38.6 mm (area: 0.001170211 m^2) under critical conditions and, 45.9 mm (area: 0.001654684 m^2) under non-critical conditions. The break flow goes to the atmosphere through a silencer.

The MSLB experiment was simulated to open a rapid-open valve installed at the main steam line between the SG-01 outlet nozzle and the MSIV (main steam isolation valve) component (OV-MSIV1-01). Because of suppose one diesel generator shutdown as single accident condition, safety injection into reactor pressure vessel is minimum. Safety injection from the SIP is injected to DVI-1 and DVI-3 among the four DVIs; therefore, there is no safety injection from the SIT.

The initial conditions of the experiment and the boundary conditions are set to record estimated results in the case of a main steam line guillotine break in the reference reactor, an APR1400-type reactor, using the optimized hydraulic analysis code MARS 3.1. For the safety injection flow, it is assumed to be a maximum flow of 0.32 kg/s injected through each DVI. In this case, the injection water temperature is 50 ^oC, equal to the reloading water temperature.

The decay heat uses a value that is 1.2 times the ANS-73 decay heat curve. The initial core power is 1.634 MW, which is calculated as 8% of the power of the APR1400 1.574 MW plus primary system whit a heat loss of 60 kW. When all systems are at their initial conditions, this is considered to be a steady state. After the steady state operates for 30 minutes, an accident starts and the rapid-open valve opens. At this time, due to the discharging steam, steam generator pressure is depressurized to 5.9 MPa and the MSIV and the MFIV (main feed-water isolation valve) are isolated by the low water level of the steam generator.

4. Space Modeling

This study used generic SPACE steady state input for ATLAS and added steam line break specific nodes to simulate ATLAS main steam line break. This model is shown in Fig. 1. The reactor vessel is separated to simulate the core, bypass flow, reactor vessel lower plenum, and the reactor vessel upper dome. Specifically, the core model includes the top and bottom inactive core regions (C210 and C250) and the vertical-node average channel core region (C220). The connection directions of the cell and the pipes are set up to show the bypass and flow characteristics.

The SIP modeled face models C561~C564. The DVI safety injection flow modeled C385, C386, C485, and C486.

The steam generator u-tubes modeled 12 volumes each (C340 and C440). The secondary system is separated into 19 volumes, which includes five nodes for the evaporator (C650 and C660) and, two nodes for the economizer (C630 and C730). The u-tube horizontal region is modeled as a simple horizontal pipe under the assumption that the curved pipe and the straight pipe do not differ. The auxiliary feed water system is modeled as injected into SG downcomer. The steam generator water level does not use a mixed level, using instead a steam generator down-comer collapsed liquid level owing to the use of actual measurements using the geometric shape and pressure difference.

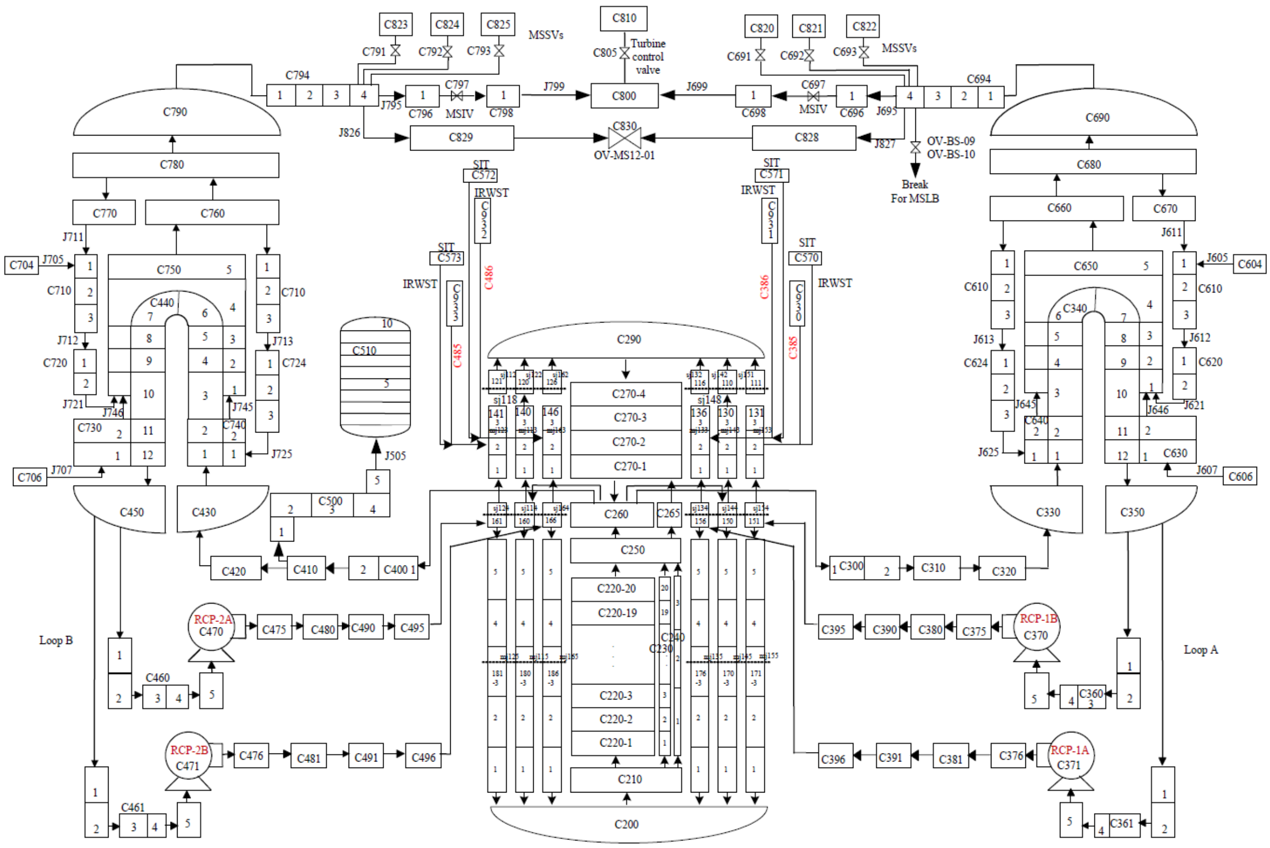

Fig. 1 SPACE model for ATLAS SLB-GB-02 analysis.

The pressurizer is modeled as a single cell (C510) with ten vertical nodes. The lower part was connected to an intact loop hot leg (C410) through a surge line (C510) separated into five nodes.

The hot legs and the cold legs are modeled with three volumes and four nodes each, and the middle loop between the steam generator and the RCP is modeled with three volumes and five nodes. Each loop is modeled in the pump and can therefore be controlled individually.

Both main steam lines are modeled with three volumes each and all are connected to a turbine (C810) after merging through a common head. The MSSV (main steam safety valve) is modeled into three separate groups C691~C693, and C791~C793. The main steam line break is modeled as an open FACE (C914 and C915) for the trip valve, which is simulated as undergoing a break at the fourth node of C694.

The ATLAS MSLB experiment main variable initial values and steady state SPACE code values are listed in Table 2.

Table 2 ATLAS SLB-GB-02 main variable initial value difference.

Design parameters	ATLAS	SPACE
Reactor vessel		
Normal power, MWt	1.634	1.634
Pressurizer pressure, MPa	15.56	15.56
Core exit temp., °C	294.6	294.8
Core inlet temp., °C	289.6	290.0
Core flow, kg/s	65.6	65.6
Steam generator		
SG-1 steam flow, kg/s	0.388	0.426
SG-2 steam flow, kg/s	0.420	0.426
Saturated steam press., MPa	7.33	7.32
Primary piping		
Hot leg flow, kg/s	32.8	32.8
Cold leg flow, kg/s	16.4	16.4
Hot leg temp., °C	294.5	294.7
Cold leg temp., °C	289.5	290.1
T_H-T_C, °C	5.0	4.6

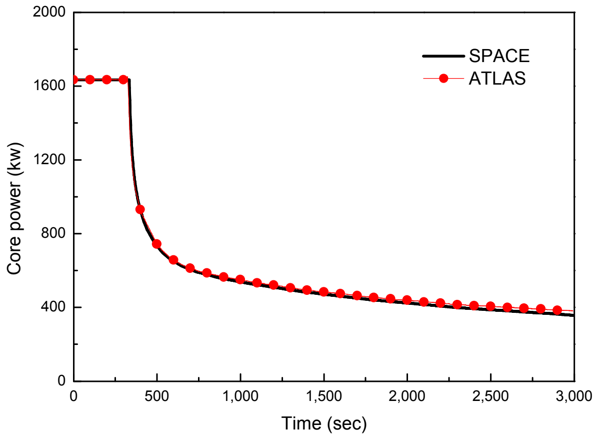

Fig. 2 Core power.

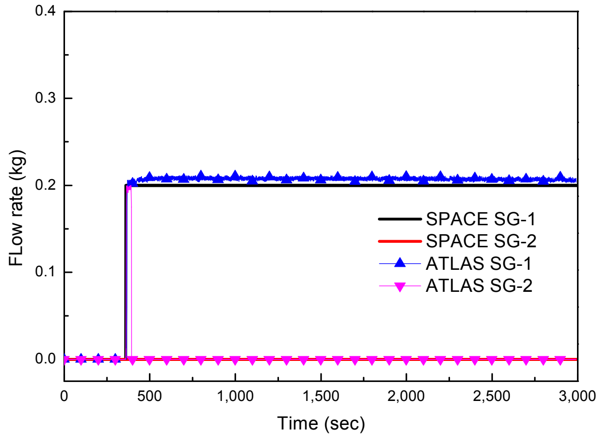

Fig. 4 Auxiliary feed water flow rate.

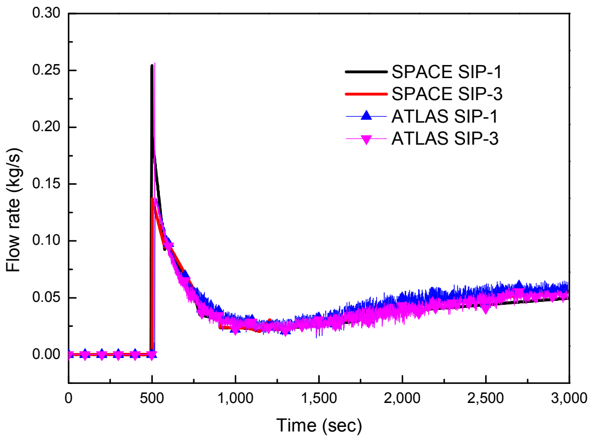

Fig. 3 High pressure safety injection flow rate.

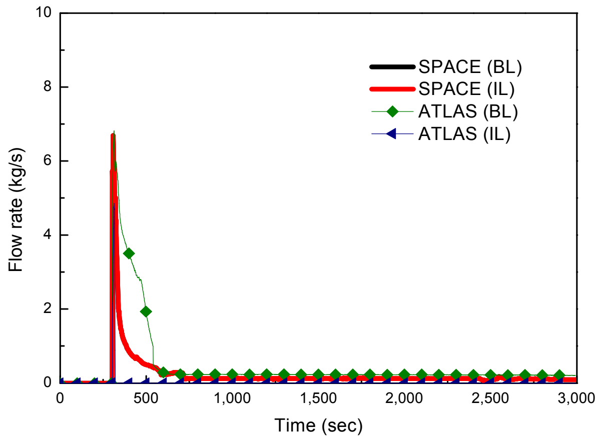

Fig. 5 Break flow rate.

5. Analysis Results

For the analysis, the boundary conditions used the experiment measurement values of the core power (Fig. 2), the safety injection time and flow rate (Fig. 3), and the auxiliary feed water flow rate (Fig. 4). The injection of the auxiliary feed water was started by the steam generator low-low water level signal.

For the break flow (Fig. 5), the initial break flow is different according to the Cd (discharge coefficient) value, but the total flow is similar. For the Ransom-Trapp model, the basic critical flow model of the SPACE code, using a Cd value of 0.5, discharges much steam early on but generally shows a trend similar to that in the experiment values. The rapid discharging at an early time decreases as the steam generator rapidly depressurizes and the inventory is reduced in the steam generator. After depressurization, the break flow rate becomes balanced with the auxiliary feed water flow rate.

With the start of a transient, the secondary system pressure decreases rapidly until the MSIV isolation set point of 5.9 MPa. After isolating the main steam, the intact loop recovers the pressure quickly, tending to decrease slowly (Fig. 6). At the area of the break, the pressure decreases continually to nearly atmospheric pressure.

Regarding the primary system behavior, the pressurizer pressure decreases rapidly at an early stage due to the increase in the heat transient with a rapid increase of the inventory leakage flow rate caused by the secondary system break. However, the pressurizer pressure is stabilized as broken side SG is emptied and the break flow decrease to auxiliary feedwater flowrate

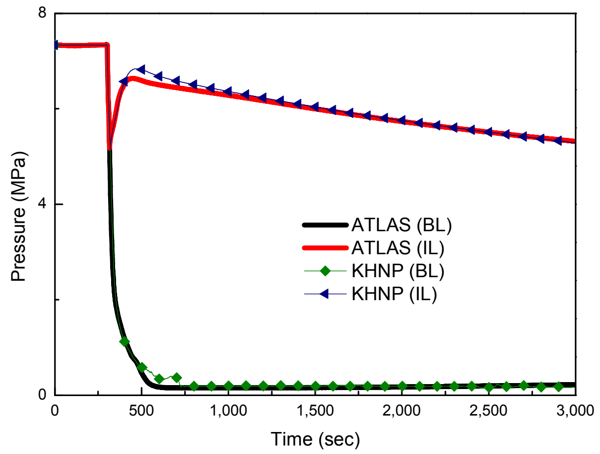

Fig. 6 Steam generator pressure.

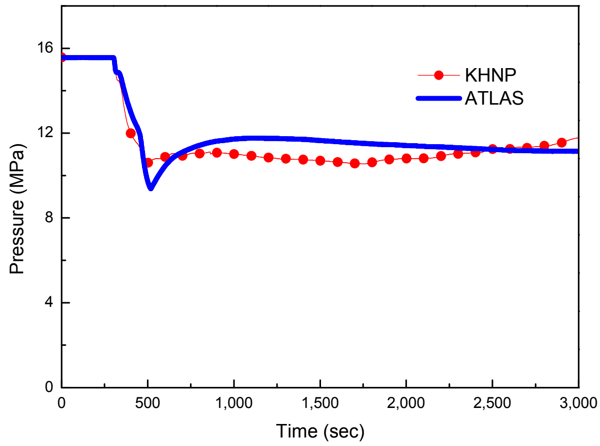

Fig. 7 Pressurizer pressure.

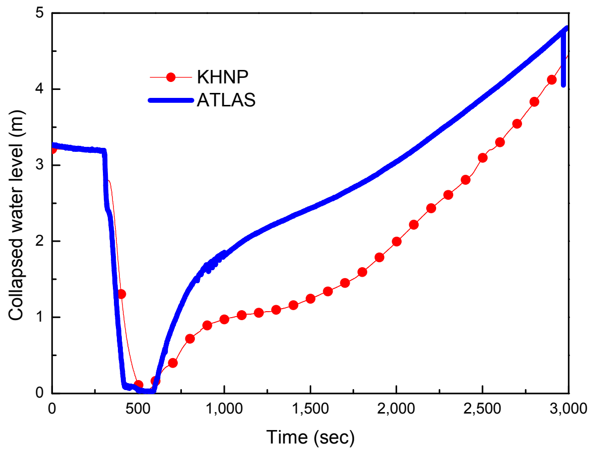

Fig. 8 Pressurizer water level.

(Fig. 7). Also, after the pressurizer water level decreases rapidly at an early stage, the heat transient increase, like the pressurizer pressure, showing a trend in which the water level recovers slowly with the safety injection (Fig. 8). The recovery time is slightly different from only the experiment, but the general trend verifies a proper simulation.

6. Conclusions

In the analysis results, although the main steam line guillotine break is a rapid transient of the secondary system, the SPACE code was able to simulate the secondary system pressure behavior well. With regard to the primary system behavior, the pressurizer water level was slightly different. Overall, the SPACE code was able to reasonably simulate ATLAS main steam line break experiment.

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