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ASSESSMENT OF SAFETY ANALYSIS CODE ON INTEGRAL EFFECT TEST WITH SNUF

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ABSTRACT

A Safety and Performance Analysis CodE (SPACE) is a thermal-hydraulics computer code under development of South Korea for analysis of complicated phenomena in the nuclear power plants including anticipated transients and postulated accidents. This study assessed the SPACE code capability for prediction of the direct vessel injection (DVI) system, which is adopted as a key safety system in the APR1400 reactor. This study mainly focused on the DVI line break accident, one of the postulated SBLOCA accidents that might result in loss of main coolant water and a quarter of safety injection water simultaneously, for assessment of the code.

In order to evaluate the code prediction capability on the phenomena associated with the downcomer seal clearing at the DVI line break, this study selected the SNUF DVI break experiment as the experimental benchmark. This experiment is a reduced-height and reduced-pressure (RHRP) integral test facility designed for simulation of the primary loop of APR1400 designed in Korea. As a result, the SPACE showed reliable agreement with the experimental data on seal clearing phenomena well predicting both the start point of downcomer seal clearance and loop seal clearance. In the DVI system, downcomer seal clearing appears to be more important than loop seal clearing because the vapor generated from core flows through downcomer to broken DVI line. Therefore, the core collapsed level increases as the vapor pressure decreases in the core.

KEA WODRS

SPACE, IET, SNUF, DVI line break, downcomer seal clearing.

INTRODUCTION

The SPACE is a thermal-hydraulic system analysis code under development of South Korea. [8, 9] This code is focused on analysing complicated phenomena in the nuclear power plants to the anticipated transients or the postulated accidents. The state-of-art SPACE includes multi-dimensional non-equilibrium two-phase flow models, heat transfer models, nuclear fuel kinetics models, and a reflood models based on two phase (liquid and gas) and three field (vapour, continuous liquid, and droplet) governing equations. SPACE code was programmed in an object-oriented manner by using C++ programming language. SPACE code can simulate loss of coolant accidents (LOCA), main steam line break, main feed water pipe rupture, and main steam generator tube rupture accidents required for light water reactor (LWR) safety analysis as well as transient phenomena of loss of offsite power, turbine trip, and nuclear reactor shutdown.

In this study, assessment of the SPACE code capability has been conducted for a direct vessel injection (DVI) system, which is an advanced safety feature adopted in the APR1400 reactor. [7, 10, 11] In the DVI system, the safety injection (SI) water is directly injected into the downcomer of the reactor vessel from the four DVI nozzles. These nozzles are installed about 2.1 m above the center-line of the cold leg instead of installing a cold leg injection (CLI) system. Therefore, the DVI injection requires different safety assessment from the original CLI injection.

One of the postulated accidents considered in the DVI system is a DVI line break categorized as a small break loss of coolant accident (SBLOCA). Since the DVI lines are directly connected to the reactor vessel, the DVI line break results in not only loss of main coolant but also loss of one-quarter of safety injection water. Once the DVI line breaks, the coolant in the primary system can be discharged from the broken DVI line and the safety injection water of one-quarter will be lost. Then, the primary coolant is discharged into the containment through the break, correspondingly, with rapid primary system pressure decrease and the pressure can reach the saturation point of the coolant. Then, the steam could be generated in the reactor core due to the core decay heat and the depressurization of the primary system. Right after the break, the steam generated in the reactor core cannot pass through the loops to the break located in the downcomer until the water in the loops is cleared completely by the steam from the core. Hence, the steam generated in the core would be bound and compressed in the upper plenum of the reactor pressure vessel (RPV). It will enhance the pressure in the reactor core than that in the downcomer. Therefore, the water level in the core goes down than that in the downcomer. Once the loop is cleared by the steam, the pressure difference between the reactor upper plenum and upper downcomer region would be reduced. Then, the coolant in the downcomer could be reflooded back into the reactor core, and the core water level would be considerably recovered. In short, the downcomer seal clearing phenomenon is that the liquid coolant in the upper downcomer is cleared by the steam flow incoming from the cold legs. Thus, when the clearing occurs, a differential pressure between the core and downcomer can be reduced, so that the fuel can be recovered by the liquid coolant. Accordingly, the downcomer seal clearing has an important role on the integrity of fuel during the accident.

In order to assess prediction capability of the SPACE code for the phenomena associated with the downcomer seal clearing, the SNUF DVI break experiment was selected as the experimental benchmark case. [6] The following sections describe the details of the SNUF DVI experiment and the assessment results by comparisons of the code analysis with the experimental data as a part of validation program of the SPACE code.

EXPERIMENTAL BENCHMARK

Description of SNUF

The SNUF is an RHRP integral test facility which is scaled down for the APR1400 pressurized water reactor. Figure.1 presents a schematic diagram of the SNUF. The scaling ratios of length and area in the primary system are 1/6.4 and 1/178. The design features of the SNUF are similar to those of the APR1400, which consists of two hot legs and four cold legs. The vessel contains 260 heaters to simulate the core decay heat. The maximum total operation power of the heaters is 200 kW, and the maximum operation pressure is 0.8 MPa. The three non-break DVI lines can supply the SI water into the upper downcomer, as shown in figure 2. A DVI line which assumed to be broken is connected to the discharge tank. The each steam generator contains 16 U-tubes. The scaling ratios of the SNUF are listed in Table 1.

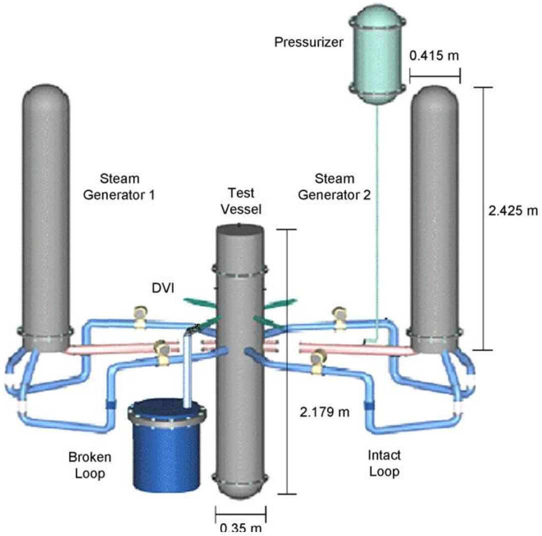

Figure 1 Schematic diagram of the SNUF

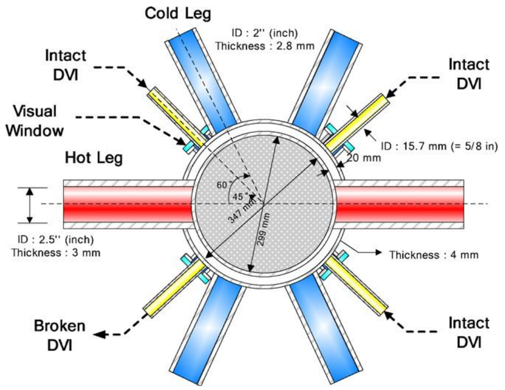

Figure 2 Plane view of the SNUF

Table 1 Scaling ratios of the SNUF

Parameters		APR1400	SNUF	Ratio
Reactor Vessel	Height(m)	13.9	2.18	1/6.4
	Area(m^2)	16.8	0.094	1/179
Hot Leg	Height(m)	4.32	1.02	1/4.24
	Area(m^2)	0.89	0.0032	1/278
Cold Leg	Height(m)	7.25	0.97	1/7.47
	Area(m^2)	0.46	0.00202	1/228
Downcomer Gap Size(m)		0.255	0.02	1/12.8
Fuel Diameter(mm)		9.7	10.0	1/1
Break Area(m^2)		0.0366	1.77×10^{-4}	1/190

SPACE modeling of SNUF

To assess the calculation capability of the SPACE code, nodalization of the integral facility, SNUF was made as shown shown in figure 3. The SNUF has two loops and each loop consists of one hot leg and two cold legs. All nodes are simulated by PIPE component and the injection of safety injection tank (SIT) is simulated by TFBC (Temporal Face Boundary Condition). The initial test condition is shown is Table 2.

The integral modelling of the SNUF consisted of a vessel, two hot legs, two steam generators, and four cold legs. The vessel consisted of the lower plenum (C160 and C180), core (C190), upper plenum (C210 and C220), upper downcomer (from C115 to C120), and lower downcomer (from C145 to C150). The upper and lower downcomers were divided into six regions to simulate the multi-dimensional phenomena in the downcomer. In the vertical direction, the upper and lower downcomers had seven nodes. The FACEs were connected with the neighbour cells.

The RCPs (Reactor Coolant Pumps) acted as a resistance that increased the differential pressure between the upper plenum and downcomer during the accident in the prototype. To simulate the RCPs' resistance, the pressure loss coefficients were implemented to consider a pressure drop through the RCPs (C370, C390, C470, and C490). As a flow boundary, the intact DVI lines (C910, C920, and C940) were connected to the upper downcomer to supply the SI water as TFBC. The break valve (C913) simulated the DVI line guillotine break. The five pipes (C586, C590, C592, C596, and C598) and the single cell (C954) were utilized to simulate the broken pipeline and the discharge tank.

The core barrel was modelled as a heat structure to transfer the heat between the core and downcomer. The vessel and pipes were modelled as a heat structure to simulate the heat loss to the surroundings. The primary side components of the steam generator included the inlet plenum (C340 and C440), tube region (C350 and C450), and the outlet plenum (C360 and C460). The secondary side consisted of the downcomer (C602 and C702), riser (C610 and C710), separator (C620 and C720), and dome region (C630 and C730).

Since the SNUF doesn't have a turbine, it removes the decay heat at the secondary system by connecting the steam generator to a chiller. However, when the temperature and pressure exceeds the design values, the steady state cannot be maintained. Therefore, to set the initial test conditions, about 0.6MPa and 428K, reactor core is heated from room temperature and normal pressure. When the primary pressure and secondary temperature reach to the aimed initial condition, a transient accident scenario with DVI break will be simulated.

Among the four DVI lines, it is assumed that one is broken and the others are intact. For the critical flow simulation, Henry-Fauske model is used as the critical flow model. [4] The downcomer consists of six channels to concern the crossflow effect between each downcomer. The heat loss of secondary system is not simulated because it is not significant.

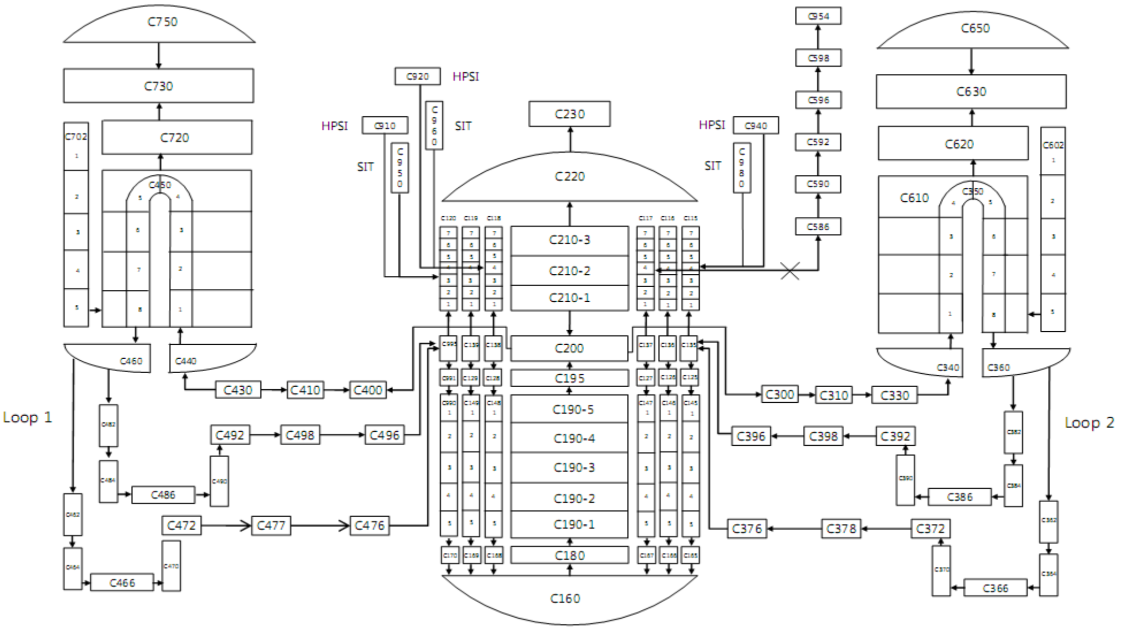

Figure 3 Schematic nodalization of the SNUF

Test Condition of SNUF

To simulate the accident scenario, an appropriate scaling method should be applied. Considering that the SNUF is a RHRP facility, an energy scaling method was proposed to determine the test conditions [3]. The transient results of the thermal-hydraulic phenomena for simulating in a small scale facility ware determined by energy scaling method [6]. For conservative conditions, we assumed that the core power was 102% of the normal power and 120% of the decay heat according to the ANS73 model [1]. A Guillotine-break of the DVI line was postulated for the most severe case of the DVI line break LOCA. From the energy scaling method, the initial conditions of the analysis are listed in Table 2.

The energy scaling method conserved the total energy as well as the mass inventory of the coolant in the primary system with a prototype. Therefore, the differences of the coolant properties such as the specific volume and the specific enthalpy between the reduced pressure test facility and the prototype were conserved by the scaling method. This scaling method was derived by taking a non-dimensional formulation of the coolant mass and energy in the system. The scaling factors with respect to the safety injection flow, the size of the broken DVI line, and thermal power in the core were determined to satisfy following relations: [6]

$$\left(\frac{\tau \dot{m}_{in}}{M_0}\right)_R = 1, \quad \left(\frac{\tau \dot{m}_{out}}{M_0}\right)_R = 1, \quad \left(\frac{\tau Q}{M_0 h_c}\right)_R = 1 \qquad (1)$$

Table 2 Initial condition for DVI break simulation

Variables	Initial Condition
Test Time(s)	Break after 30 ~ 530
Primary Pressure(MPa)	0.6 (pressure ratio 1/13)
Secondary Temp.(K)	428
Coolant Temp.(K)	423 ~ 433
Core Power(kW)	110 (0 ~ 60s) 70(60 ~ 300s) 60(300 ~ 500s)
HPSI Flow rate(kg/s)	0.13
SIT Flow rate(kg/s)	0.11
SI Temp.(K)	300.4
Break Area (m^2)	0.000177
Discharge Coefficient	0.62 (Henry-Fauske critical flow model)
L/D of broken DVI	5.9
# of HPSI injection	3

VERIFICATON AND VALIDATION OF SPACE FOR INTEGRAL MODELING

Preparation of Initial Condition for Transient Simulation

As described in the previous section, the SNUF experiment does not have a sufficient heat sink in the secondary side to maintain the steady-state continuous operation as it was designed for simulating accident scenarios. Therefore, the initial condition at the beginning of the test should be first obtained by preliminary calculations. To obtain the initial condition for the DVI line break experiment, the SPACE modeling for SNUF was simulated to following the increasing pressure and temperature with induced power. To confirm and verify the SPACE input, MARS code was also used for benchmark simulation and compared to the SPACE results. If the modelling of SNUF is well designed, MARS and SPACE can approach the same range of pressure and temperature value by same induced power after all. Target values of primary pressure and secondary temperature were about 0.6MPa and 428k, respectively. The procedures of calculating the initial condition are shown in Figures 4 and 5. From the room temperature and atmospheric pressure, after 2300s in MARS and 2400s in SPACE, the temperature and pressure reached at the target condition, 0.6MPa and 420K. In this figure, the reference pressure was taken from the upper plenum (C220) and the reference temperature was taken from the secondary coolant of riser (C610).

Small deviation of the pressure increase rate between two code results is anticipated to be originated from different sets of constitutive models including interfacial heat transfer. However, in overall, the two codes showed good agreement with each other.

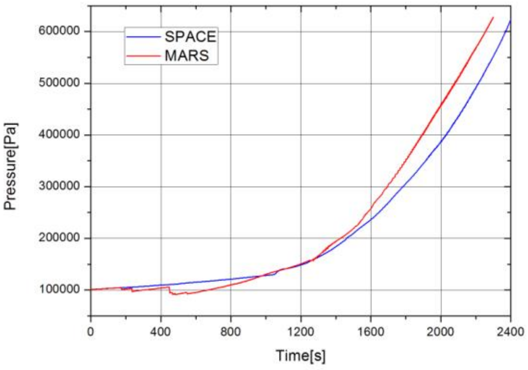

Figure 4 Primary pressure increase (C220)

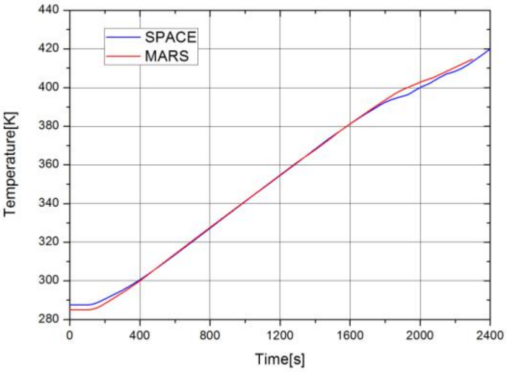

Figure 5 Secondary temperature increase (C610)

Comparisons of the SPACE Results with SNUF DVI Line Break Experimental Data

After verification of the SPACE based on the code-to-code benchmark tests, validation with experimental data is feasible to be discussed. The aim of the validation using the SNUF DVI break experiment data is to confirm prediction capability of the SPACE for the downcomer seal clearing phenomena.

Prior to discuss about downcomer seal clearing phenomena, we should first check the fundamental parameters which describes the conservative maintenance. In the figure 6, the primary pressure from the SPACE tends to decrease more rapidly than that from experiment. The early decrease of the primary pressure predicted by the SPACE appears to be originated from underestimated total energy of the system. The reduced energy inventory calculated by the SPACE means the asymmetric energy distribution of the system via experimental results. The calculated temperatures of the secondary loop, cold leg and hot leg by the SPACE are shown in figures 7, 8, and 9. It also means that the SPACE predicted reduced energy inventory. However, to confirm this, it would need more investigations on the test results, input models, and the code itself for further study. The differnece of the teperature between secondary, cold-leg and hot leg temperature represent the asymmetric energy distribution of the system.

In preparation of the initial condition of this DVI break simulation (shown in the previous section), the two parameters (the initial value of primary pressure (C220) and secondary temperature (C610)) were checked and matched to the experimental data. However, those in the other locations were not confirmed by comparisons with the experiment due to lack of data. Thus, there is a possibility that the initial energy distribution by the SPACE could be different from the experimental one. For this reasion, it is expected that the modeling and the calculation by the SPACE could be improved further if the initial conditions are improved more. As shown in the figures, the SPACE predictions getting closer to the experimental data as the time increases, and eventually become very close each other. It also supports that the initial conditions could be the main reason for the deviation in the early time stages.

Another reason of rapid drop in the primary pressure is considered to be overestimated discharged break flow rate. In the SNUF DVI experiment, the broken DVI line was simulated with a series of pipeline, so that the streamline contraction occurred. As a result, a discharge coefficient of 0.62 was determined by experiment in spite of the general discharge coefficient ranges from 0.8 to 1.2. [6] The discharge coefficient, 0.62, was inappropriate for predicting the phenomenon with a compressible gas flow like steam. After the downcomer seal clearing, a large amount of steam from the cold leg was discharged. As a consequence, the ratio of the streamline contraction may have increased gradually. Thus, a different discharge coefficient, which may be larger than 0.62, should be applied to simulate the break flow during the later period of the accident. Therfore, in case with SPACE calculation, the default discharged coefficient of 1.00 was applied for the entire period of the accident. Thus, the break flow rate could be over-discharged to the broken DVI line in the SPACE analysis than the experiment. As a result, the primary system pressure of the analysis was underestimated than that of experimental result.

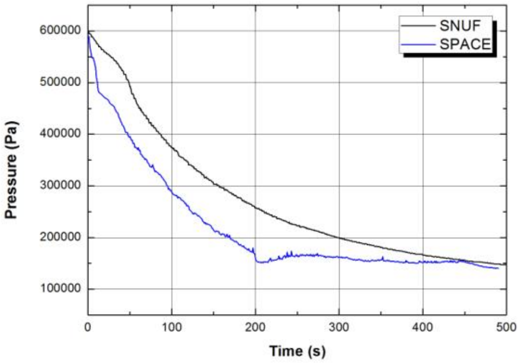

Figure 6 Primary pressure decrease (C220)

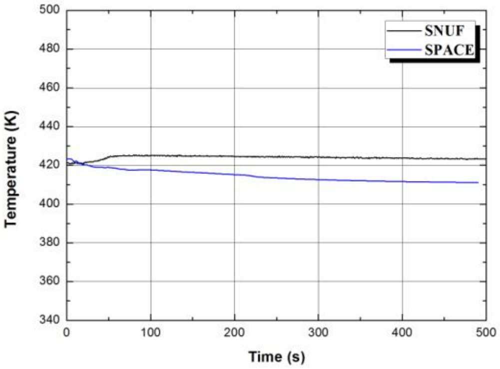

Figure 7 Secondary temperature decrease (C620)

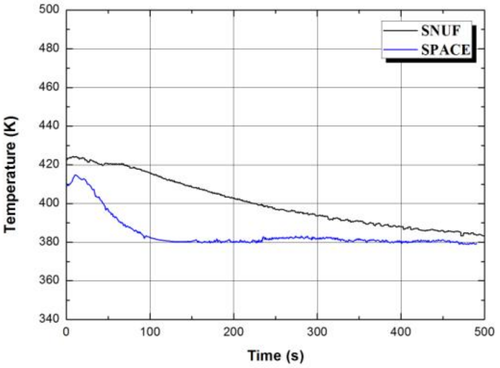

Figure 8 Cold leg temperature decrease

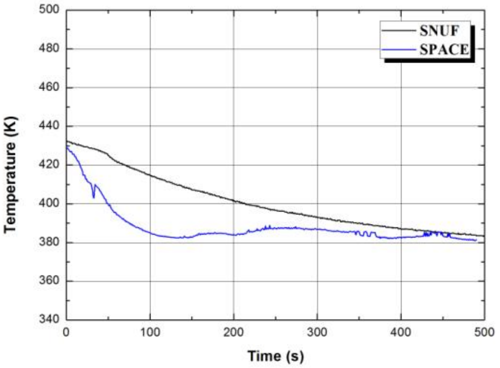

Figure 9 Hot leg temperature decrease

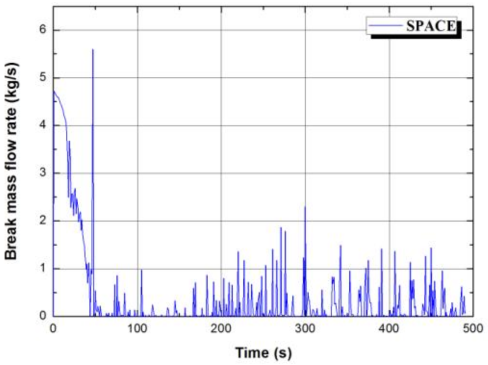

Figure 10 Break flow rate

The downcomer seal clearing phenomenon can be seen in the figure 11 and 12. From the figure the sequence of clearing phenomenon of each part could be observed. At first, the hot leg was cleared and then the loop seal was cleared rapidly. As soon as the loop seal was cleared, the upper downcomer started to be cleared. After all of the parts were cleared, the steam generated in the core could be vigorously discharged to the broken DVI line. The SPACE well predicts the seal clearing phenomenon. However, the collapsed water level of core and downcomer were predicted lower than experiment as shown in figure 13 and 14. It is also because the SPACE overestimate the break flow.

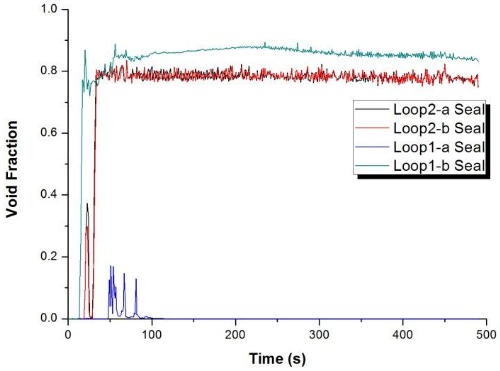

Figure 11 Loop seal void fraction

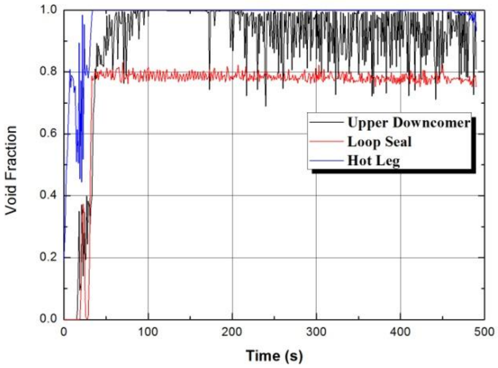

Figure 12 Void fraction

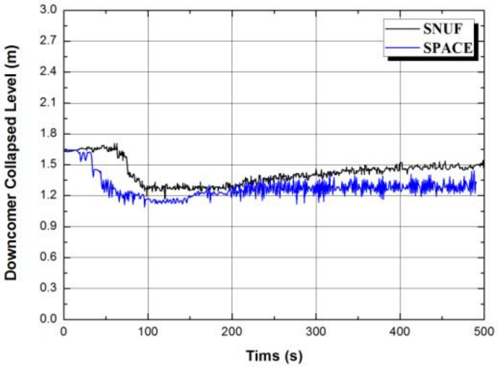

Figure 13 Downcomer collapsed water level

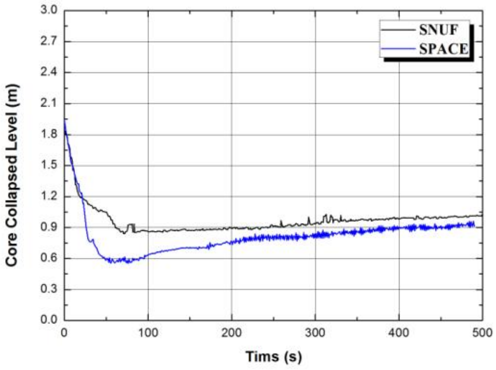

Figure 14 Core collapsed water level

Even though the fundamental parameters were underestimated due to the unbalanced initial conditions and overestimated discharged coefficient, the downcomer seal clearing phenomenon, which is main purpose of DVI LOCA simulation, was well predicted. As a result, it could be concluded that the SPACE could make good predictions for the experiments related to downcomer seal clearing phenomenon if some proper perparations are made on the initial conditions.

CONCLUSION

This study analysed the DVI line break LOCA of SNUF using the SPACE. As a result, it was evaluated that the SPACE has large potential to make good predictions on the phenomena related to the downcomer seal clearing phenomenon if the initial conditions are properly defined. The following summarizes the final results:

- The SPACE predicted the change of primary system pressure more rapidly than the experimental results. It appears to be because of underestimated energy inventory of the system and overestimated break flow rate.

- In case with the downcomer seal clearing phenomenon, SPACE could predict it quite well but it needs some improvement on the initial conditions.

ACKNOWLEDGMENTS

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NOMENCLATURE

M	[kg]	Mass
$\dot{m}$	[kg/s]	Mass flow
h	[W/kg]	Enthalpy
Q	[W]	Power
Special characters		
τ	[s]	Time scale
Subscripts		
0		Prototype
c		Coolant
in		Inlet value
out		Outlet value
R		Ratio

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