NURETH14-181

Experimental and MARS Code Simulation of DVI Line 25% and 50% Break LOCAs using SNUF for APR1400

XinGuo Yu[1,2], KeoHyoung Lee[1] and GoonCherl Park[1]
[1]Seoul National University (SNU), Seoul, South Korea
[2]Korea Atomic Energy Research Institute (KAERI), Daejeon, South Korea

Introduction

APR1400, an evolutionary PWR (Pressurized Water Reactor) based on the well-proven OPR1000 (Optimized Power Reactor 1000MWe) design, adopted DVI (Direct Vessel Injection) system instead of CLI (Cold Leg Injection) system as advanced safety features of ECCS (Emergency Core Cooling System) [1]. The configuration of the improved ECCS in the APR1400 is completely different from that in the OPR1000 in which pipes for the safety water injection are connected to the cold leg as shown in **Fig. 1**. Whereas, in APR1400, the safety water injection pipes are directly connected to the RPV (Reactor Pressure Vessel). Thus, the safety water injection system in the APR1400 is called the DVI (Direct Vessel Injection) system. Moreover, safety water injections by the HPSI (High Pressure Safety Injection) pumps are mechanically separately in the APR1400.

The DVI system of APR1400 consists of four SITs (Safety Injection Tanks) which passively inject water into primary system, four SIPs (Safety Injection Pumps) which draw cooling water from the IRWST (In-containment Refueling Water Storage Tank), and four DVI lines which deliver the safety injection water into the reactor upper downcomer through the reactor vessel. Each DVI line, which is directly connected to the reactor vessel of upper downcomer, is attached with one SIT and one SIP. Two emergency diesel generators power the four SIPs with each powering two SIPs. APR1400 has adopted a fluidic device in the SIT which is installed in the discharge line of the SIT [2]. This device passively controls the flow of SIT water into the primary system such that a high flow is delivered in the early stage of transient and a lower flow is delivered in the late stage for a longer period of time.

Since a DVI line can rupture in any size, the DVI line break LOCA (Loss-Of-Coolant Accident) would occur with different break sizes. Once DVI line breaks, the coolant in the primary system can be discharged from the broken DVI line and a portion of safety water injection is lost. Therefore, the safety of the nuclear reactor may be threatened. In view of this, understanding of the thermal-hydraulic behaviour of the APR1400 under DVI line break is essential to ensure the safety of APR1400.

To investigate the thermal-hydraulic transients with different break sizes of DVI line, the experiment and computer code simulation are performed with SNUF and MARS code, respectively. In the MARS code simulation, two different models — the APR1400 model (the prototype model) and the SNUF model (corresponding to the test facility SNUF) — are investigated. Since SNUF is a scaled-down test facility in terms of both dimensions and operational conditions with respect to APR1400, the scaling methodology proposed by Jose

N. Reyes Jr. [6] is adopted to obtain the test conditions of SNUF experiments by scaling down the conditions of APR1400. The transient conditions of the APR1400 are obtained by conducting the MARS code simulation of DVI line break LOCA with APR1400 model. Then, the conditions of APR1400 are scaled down to obtain the test conditions of SNUF experiments by applying the scaling methodology. After that, the scaled-down test conditions are implemented in the MARS code with the SNUF model to validate the test conditions for the SNUF experiment.

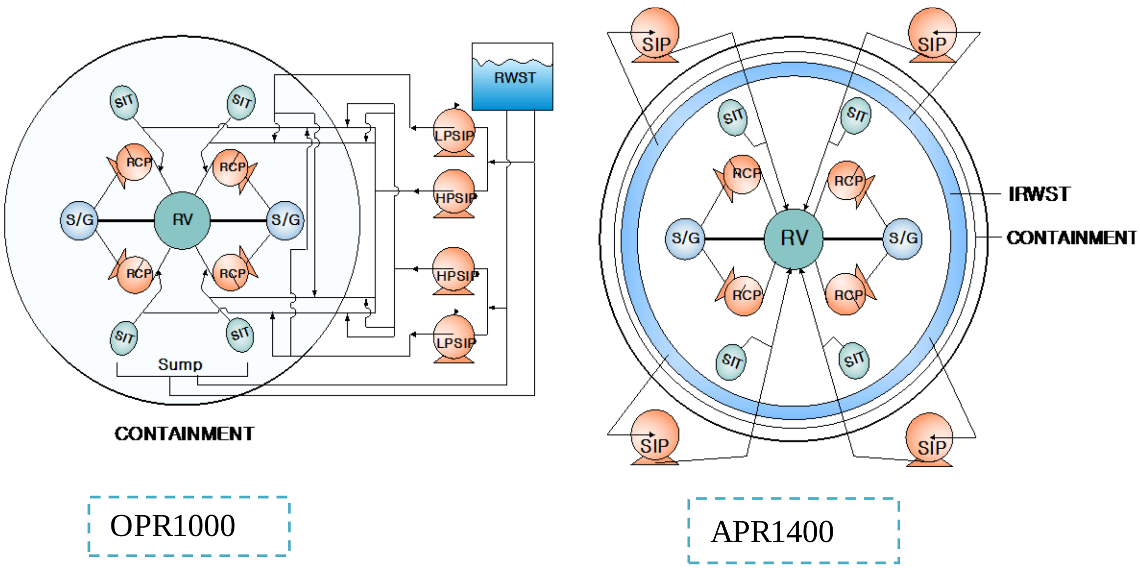

Figure 1 ECCS Configurations of OPR1000 and APR1400.

1. Integral Test Facility SNUF

1.1 Description of SNUF

SNUF (Seoul National University Facility) is a scaled-down test facility from APR1400. Its dimensional scaling ratios with respect to APR1400 as given in **Table 1** are 1/6.4 in length and 1/178 in cross-sectional area, respectively [3]. SNUF has similar geometrical configurations to those of APR1400: a typical 2x4 loops (2 hot legs and 4 cold legs), two steam generators, four DVI lines with one connected to the storage tank for simulating the broken DVI line, RPV (Reactor Pressure Vessel) with an annular barrel separating the RPV into core and downcomer region as shown in **Fig. 2**.

For storing the fluid discharged from the break and measuring the break flow rate, a storage tank which is connected by the simulated broken DVI line with RPV upper downcomer is installed as shown in **Fig. 3**. A spray device is also equipped with the storage tank to condense the discharged two phases from the break. In the simulated broken DVI line, an orifice with different inner diameter can be installed to simulate different break sizes, and an electrically controlled butterfly valve is installed to initiate the break of the DVI line.

Many systems such as the core power supply system and the DVI system in SNUF are only able to be controlled manually. Therefore, the core power supply and SI flow rate for the SNUF experiment can only be implemented with some constant values for certain time duration. The safety injection water flow rate can be changed by adjusting the opening of the valve installed in the DVI line; the core power supply can be controlled by setting up the core power supply rate ranging from 0 to 150 KW.

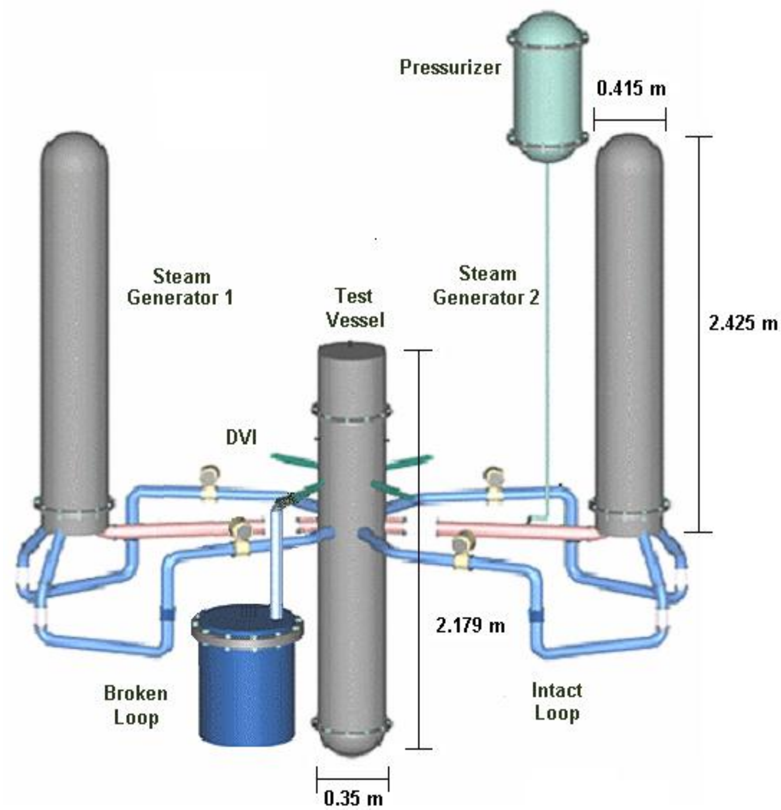

Figure 2 Schematic of SNUF

Figure 3 Installed storage tank

1.2 Instrumentation of SNUF

In order to adequately understand the thermal-hydraulic transients undergoing during the experiments, various instrumentation is installed in SNUF to measure various thermal-hydraulic parameters. Forty-two thermocouples are installed in the core and downcomer, along the loop to measure the temperature distribution in the primary system. Differential pressure transmitters are also installed in primary system to measure the core and downcomer water levels, the coolant flow rate in the four cold legs. The storage tank connected to the broken DVI nozzle has the functions of storing the discharged coolant through the simulated broken DVI line, and measuring the mass flow rate of the discharged coolant. For measuring the coolant discharging flow rate, a differential pressure transmitter is installed to measure the water level change rate in the storage tank, and a steam flowmeter is installed to measure the steam flow rate through the steam venting pipe connected to the top of the storage tank. The sum of the coolant accumulating rate in the tank and steam venting rate out of the tank equals the coolant discharging rate through the simulated broken DVI line.

Table 1 Geometrical and Operational Condition of APR1400 and SNUF

Parameters	**APR1400**	**SNUF**	**Scaling Ratio**
Cross-section area	1	0.005618	1/178
Length	1	0.1563	1/6.4
Operational Conditions			
Primary-side pressure (MPa)	15.5132	0.60624	
Secondary-side pressure (MPa)	6.8947	0.3	
Core outlet temperature (^{o}C)	324.5	149.8	
Core inlet temperature (^{o}C)	290.8	132.6	

2. Scaling Analyses and Test Conditions

2.1 Research Methodology

The procedures for the study of DVI line break LOCAs are depicted in **Fig. 4**. For the study of DVI line break LOCAs in APR1400, two different approaches are deployed: the MARS code simulation and SNUF experiment. For MARS code simulation, two different models — APR1400 and SNUF model — are utilized. For the experimental study, SNUF is used along with test conditions to perform the experiments. Because SNUF is a scaled-down test facility with respect to APR1400, the test conditions for SNUF experiments should be obtained by scaling down the conditions of APR1400 which are obtained by the MARS code simulation with the APR1400 model. The obtained test conditions for SNUF experiment then are applied for the code simulation with SNUF model. Results obtained with the SNUF model are then normalized with the initial reference values and compared with those obtained with the APR1400 model. If the transient results agree well between the SNUF model and the APR1400 model, the test conditions are then applied to the SNUF experiments. The results obtained by simulations with

APR1400 model and SNUF model, and SNUF experiment can be compared with one another. Comparison of results between APR1400 model and SNUF model can verify the scaled-down test conditions for the SNUF experiment and scaling methodology adopted for this study; the comparison of results between SNUF model and SNUF experiment can validate that the MARS code is capable of predicting the thermal-hydraulic transients of DVI line break LOCA under low-pressure conditions.

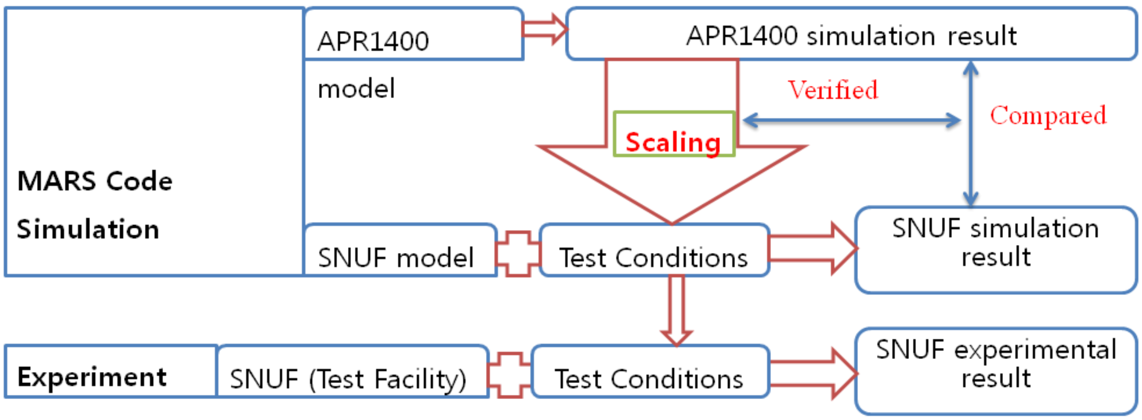

Figure 4   Procedures for the study of DVI line break LOCA

2.2 Scaling Analyses

2.2.1 Similitude Issues between SNUF and APR1400

It is much cheaper to conduct LOCA experiments with a reduced-size test facility at reduced pressure and temperature conditions, such as the case with SNUF. However, when the experiments are performed in a scaled-down test facility operated at reduced-pressure condition, the fluid properties at reduced pressure are different from those at prototypic conditions, thus, it is essential to make sure that the thermal-hydraulic transients of interest at reduced pressure in SNUF are similar to those in APR1400.

The similarity of thermal-hydraulic transients between SNUF and APR1400 could be preserved by scaling analyses in which the mass, momentum and energy conservation laws along with some thermodynamic relations are deployed, and some major similarity groups describing physical phenomena of LOCA transients are derived and are preserved the same between the SNUF and APR1400 [4].

Therefore, the purpose of scaling analysis is to obtain the test conditions for the scaled-down test facility and to extrapolate the test results obtained from the small-scale test facility to the prototypical conditions. In other words, by scaling analyses, two questions should be answered: how the thermal-hydraulic transients in a small-scale test facility can be kept similar to those in the prototype, how the test data obtained in a small-scale test facility can be scaled up to conditions of the prototype.

2.2.2 Scaling Analyses

It is impossible to preserve all the scaling criteria for a complicated thermal-hydraulic system [5], thus, some dominant factors such as the core decay heat, break size and safety injection flow rate that affect the thermal-hydraulic transient the most are considered.

In this study, the scaling methodology proposed by Jose N. Reyes, Jr [6] is adopted for the scaling analyses. Governing equations for two-phase natural circulation and two-phase system depressurization are deployed to derive the similarity groups for the thermal-hydraulic similarities between SNUF and APR1400. The obtained similarity groups are given as follows:

Similarity groups obtained from governing equations for two-phase natural circulation:

$$\left(q_c\right)_R = (a_c L_c^{1/2})_R \left(\frac{\rho_{ls}\rho_{gs}h_{\lg}}{\Delta\rho}\right)_R \tag{1}$$

$$u_R = (L_c^{1/2})_R \tag{2}$$

$$\tau_R = \frac{L_R}{u_R} = \frac{L_R}{(L_c^{1/2})_R} \tag{3}$$

Where subscript R represents the parameter ratio of the model to the prototype; q_c means core thermal power; a_c the coolant flow area in the core; L_c the actively heated length in the core; ρ_{ls} water saturation density; ρ_{gs} vapor density; $h_{\lg}$ latent heat; $\Delta\rho$ differential density; τ time, u fluid velocity and L flow path length.

Similarity groups obtained from governing equations for system depressurization:

$$\tau_R = \left(\frac{M_o}{\sum \dot{m}_{out,o}}\right)_R \tag{4}$$

$$\left(\sum \dot{m}_{out,o}\right)_R = \left(\frac{M_o}{\tau_{Depress.}}\right)_R = \left(\frac{\rho_{ls,o}V_R}{\sum C_d G_{e,o} a_e}\right)_R \tag{5}$$

$$\Pi_m = \frac{\sum \dot{m}_{in,0}}{\sum \dot{m}_{out,0}} \tag{6}$$

By setting $\left(\Pi_m\right)_{\text{model}} = \left(\Pi_m\right)_{\text{prototype}}$, we get

$$\left(\sum m_{in,o}\right)_R = \left(\sum m_{out,o}\right)_R \tag{7}$$

Where subscript 0 represents the initial conditions; M is the total mass within the control volume; and $\dot{m}$ the mass flow rate entering or leaving the control volume.

For comparing the data obtained from the scaled-down test facility with those of the prototype, the results should be normalized with some reference values so that the data at different scales can be compared. The reference value should be well defined for the normalization by considering the physical meaning of parameters. For example, for comparing the primary-system pressure of APR1400 with that of SNUF, because the two-phase critical flow and hence the depressurization is related to the pressure difference between the primary system and the atmosphere, the pressure difference between the primary-system and the atmosphere should be utilized to normalize the primary-system pressure [7].

2.3 Critical Flow Model and Calculation of Break Size in SNUF

The initial flow through the break under LOCA conditions is generally critical flow which is mainly a function of fluid conditions in the primary system but not affected by the downstream pressure. At the initiation of LOCA, the fluid in the primary system is subcooled, thus, the critical flow through the break is initially subcooled critical flow. The model proposed by Fauske as given in **Eq. (8)** [8], which considers the effect of subcooling on critical mass flux, is adopted to calculate the initial critical mass flux of APR1400 and SNUF through the break.

$$G = c_d \sqrt{2[P_0 - P_v(T_0)]\rho_f + \frac{h_{fg}^2}{v_{fg}^2}\frac{1}{Tc}} \tag{8}$$

where c_d is the discharge coefficient; P_0 the stagnation pressure; $P_v(T_0)$ the vapor pressure at the stagnation temperature T_0; ρ_f the liquid density; h_{fg} latent heat at the stagnation pressure; v_{fg} specific volume difference; T temperature; and c specific heat.

As initial mass inventory in the primary systems of APR1400 and SNUF are known and critical mass flow through the break in APR1400 can be calculated from the critical mass flux model and break size, according to similarity group in **Eq. (5)**, the initial critical mass flow rate through the break in NUF can be calculated. Provided that the initial fluid conditions in SNUF prior to the initiation of the break are given, the initial critical mass flux can be calculated from critical flow model. The break area in SNUF can then be obtained through dividing the critical mass flow by critical mass flux.

2.4 Steady State and Transient Conditions in APR1400

To obtain transient conditions in APR1400, a pretest simulation of the DVI line break LOCAs in APR1400 with MARS code is performed. In this simulation, the safety actions in response to the LOCAs in APR1400 as given in **Table 2** should be implemented. In addition, the conservative conditions as shown in **Table 3** are also applied. The nodalization of APR1400 for the simulation is shown in **Fig. 9**. The steady state under normal operational mode of the APR1400 is maintained for 1200s, and then the break of DVI line is initiated. Following the break, the actions of reactor systems for the reactor safety are activated according to safety setpoints and logic of APR1400 during LOCAs. Evolution of major parameters in APR1400 during LOCA is simulated, and several transient conditions can be obtained from these simulations. For example, timings of safety actions shown in **Table 4**, core decay power in **Fig. 5** and total safety injection water flow rate in **Fig. 6** are obtained for the DVI line 25% break LOCA.

Table 2 Safety actions in response to LOCAs in APR1400

Events	Trip Conditions
Break initiated	0 s
Low pressurizer pressure (LPP)	If pressurizer pressure<10.72MPa
Pressurizer heater trip	LPP+0.0 s
Turbine and feedwater isolation	LPP+0.1 s
Reactor scram & RCP trip	LPP+0.5 s
HPSI pump initiation	LPP+40.0 s
SIT valve open	Low pressure of SIT outlet (P < 4.03MPa)

Table 3 Conservative conditions for the safety analysis of APR1400

Parameters	Conservative Conditions
Thermal power	102% normal power
Core decay heat	120% decay heat (ANS 79 model [9])
SI system	Single Failure (failure of two HPSI pumps)

Table 4 Sequence of events for DVI line 25% break LOCA in the APR1400

Events	Time(s)	Conditions
Break initiated	0	
Low pressurizer pressure (LPP)	52	If pressurizer pressure<10.72MPa
Pressurizer heater trip	52	LPP+0.0 s
Turbine and feedwater isolation	52	LPP+0.1 s
Reactor scram & RCP trip	52	LPP+0.5 s
HPSI pump initiation	90	LPP+40.0 s
SIT Valve Open	1446	Low pressure of SIT inlet (P_{SIT} <4.03MPa)

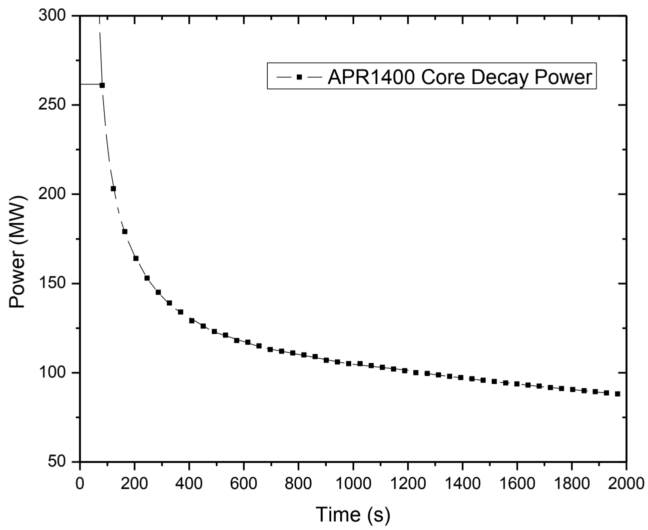

Figure 5 Core decay power (DVI line 25% break in the APR1400)

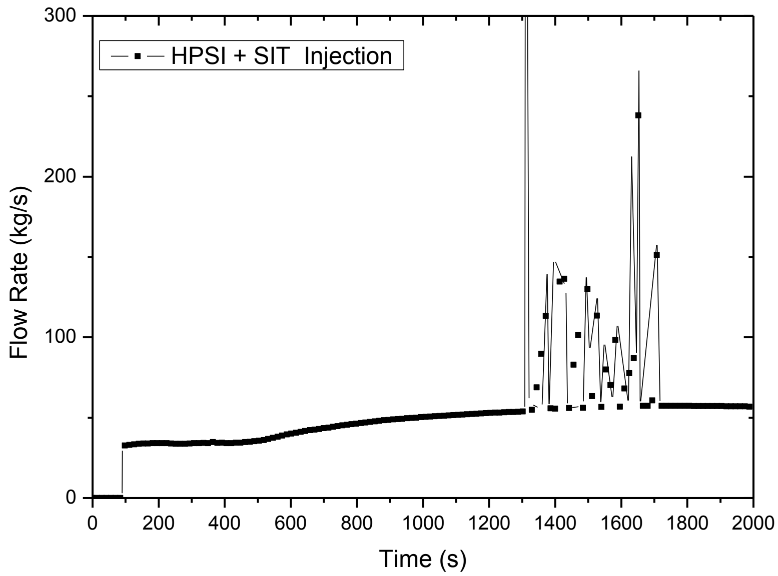

Figure 6 Total SI water flow rate (DVI Line 25% break in the APR1400)

2.5 Calculation of Test Conditions for the SNUF Experiments

For calculating the temperature distribution in the primary system of SNUF, the ratios of the saturation pressures corresponding to the core inlet and outlet temperature to the steady-state pressure in SNUF are set equal to those in APR1400 to calculate the steady-state core outlet and inlet temperatures for the SNUF experiment. Calculated steady state conditions for the SNUF experiments are given in **Table 1**.

As introduced before, SNUF is a scaled-down test facility with respect to APR1400, thus, it is necessary to calculate the test conditions of SNUF by scaling down the conditions of APR1400 through scaling analyses. In the scaling analyses, some similarity groups, which are obtained in section 2.2, should be kept same between SNUF and APR1400 so that the similar thermal-hydraulic transients between SNUF and APR1400 are ensured. Since the dimensions and steady state conditions of SNUF and APR1400 are known as given in **Table 1**, the dimensional ratios and fluid property ratios of SNUF to APR1400 can be obtained, and hence the scaling ratios for power scaling, the SI (Safety Injection) water flow rate scaling and break flow rate scaling could be calculated through the similarity groups as shown in **Table 5** and **Table 6.** After the scaling ratios are obtained, the test conditions for SNUF experiments are calculated by multiplying the scaling ratio with the corresponding conditions of the APR1400 — $X_{SNUF} = X_R X_{APR1400}$ as given in **Table 7**, **Fig. 7** and **Fig. 8** for each break size of DVI line.

In SNUF the core heater power and SI water flow rate can only be controlled manually. The core decay power and SI water flow rate in APR1400 are scaled down and then averaged for different time duration so that the scaled-down core decay power and SI water flow rate for the SNUF experiments can be some constant values in separate time durations and be practical to implement as shown in **Fig. 7** and **Fig. 8**.

Table 5 Thermal-hydraulic scaling ratios of SNUF to APR1400

	APR1400	SNUF	Scaling Ratios ($X_R = \frac{X_{SNUF}}{X_{APR1400}}$)	
Core thermal power (MW)	see **Fig. 5** and **Fig. 7**		$(q_c)_R = (a_c L_c^{1/2})_R \left(\frac{\rho_{ls}\rho_{gs}h_{lg}}{\Delta\rho}\right)_R$	0.00025
Core coolant velocity			$(u_c)_R = (L_c^{1/2})_R$	0.4344
Core coolant flow rate (kg/s)	20992	78.30	$(\dot{m}_c)_R = (\rho_{ls} u_c a_c)_R$	0.00373
Time (s)	1	0.36	$\tau_R = \frac{L_R}{u_R} = \frac{L_R}{(L_c^{1/2})_R}$	0.3597
Initial break flow rate (kg/s)	840.0	1.23	$\left(\sum m_{out,o}\right)_R = \left(\frac{M_o}{\tau_{Depress.}}\right)_R$	0.00146
Initial critical mass flux (kg/m^2/s)	91825.860	17494.005	Fauske model with C_d=0.7	
Break area(m^2)	0.00915	0.00007		
Safety injection flow rate (kg/s)	see **Fig. 6** and **Fig. 8**		$\left(\sum m_{in,o}\right)_R = \left(\sum m_{out,o}\right)_R$	0.00146

Table 6 Sequence of events for DVI Line 25% Break LOCA

Events	APR1400	SNUF	Remarks
	Time (s)	**Time* (s)**	
Break initiation	0	0	reduced time scale: $\tau_R = \frac{t_{SNUF}}{t_{APR1400}} = 0.36$
Low pressurizer pressure (LPP)	52	19	
Pressurizer heater trip	52	19	
Turbine and feedwater isolation	52	19	
Reactor scram & RCP trip	52	19	
Safety water injection initiation	90	32	

* applying the reduced time scale for the SNUF experiment

Table 7 Break sizes and sequence of events for each case in the SNUF

	25% Break	50% Break
Break area (m^2)	0.000070	0.00014
	Time* (s)	
Events	**25% Break**	**50% Break**
Break initiation	0	0
Secondary system isolation	19	10
Core decay power initiation	30	22
Safety injection initiation	32	24

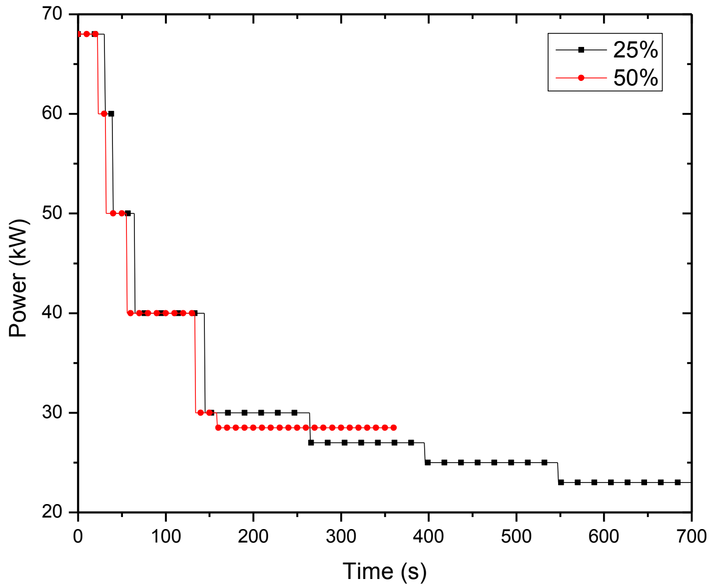

Figure 7 Scaled-down core decay power transients in the SNUF (DVI line 25% and 50% Break)

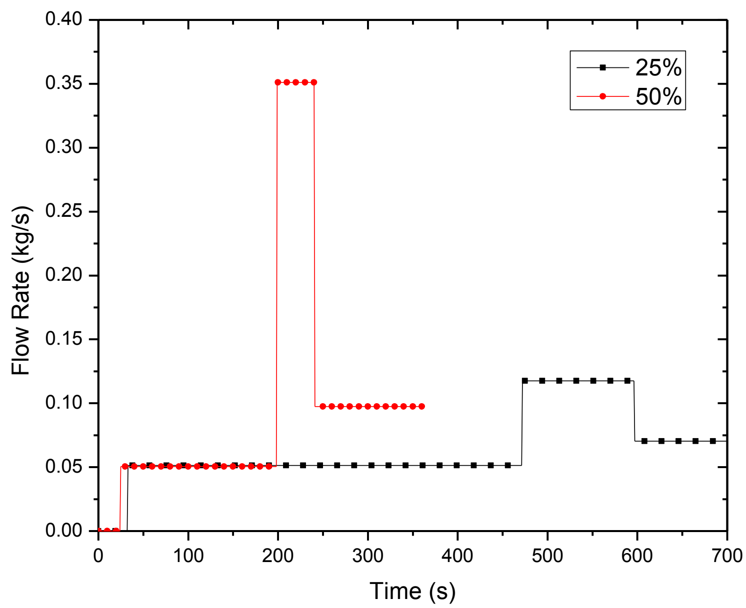

Figure 8 Scaled-down SI water flow rates in the SNUF (DVI line 25% and 50% Break)

2.6 MARS Code Calculation Results with APR1400 Model and SNUF Model

2.6.1 MARS Code Simulation of APR1400 and SNUF

In this study, the MARS code is utilized to simulate the DVI line break LOCAs with both APR1400 model and SNUF model. In these simulations, the geometries of both APR1400 and SNUF are discretized into one-dimensional models as shown in **Fig. 9** and **Fig. 10**. Thermal-hydraulic dynamics of APR1400 and SNUF systems are simulated separately by solving one-dimensional two-phase governing equations along with several thermal-hydraulic models in the MARS code.

The break of DVI line with different break area can be easily implemented in the MARS code by specifying the break area to the prescribed value. In the simulation of DVI line break LOCAs with the APR1400 model, various trips could be implemented by the code according to the safety logic as given in **Table 2** to activate safety systems such as the HPSI system and SIT. The test conditions for different break sizes of DVI line in SNUF as given in **Table 7**, **Fig. 7** and **Fig. 8** are implemented in the MARS code for each case so that the thermal-hydraulic transients for different break sizes of DVI line with SNUF model could be obtained.

In order to compare transient results obtained from the SNUF model with those obtained from the APR1400 model, values of different parameters are normalized over some reference values such as the initial values or the initial physical values. For example, the primary-system pressure is normalized through dividing the pressure difference between the primary system and the atmosphere by the initial steady-state pressure difference between the primary system and the the atmosphere. In the following section, the obtained transient results with the APR1400 model and SNUF model are compared and discussed for DVI line 25% and 50% break LOCAs. If the transients obtained with the SNUF model agree well with those obtained with the APR1400 model, it indicates that the test conditions for the SNUF are well obtained by scaling analyses and they can be applied to the SNUF experiments.

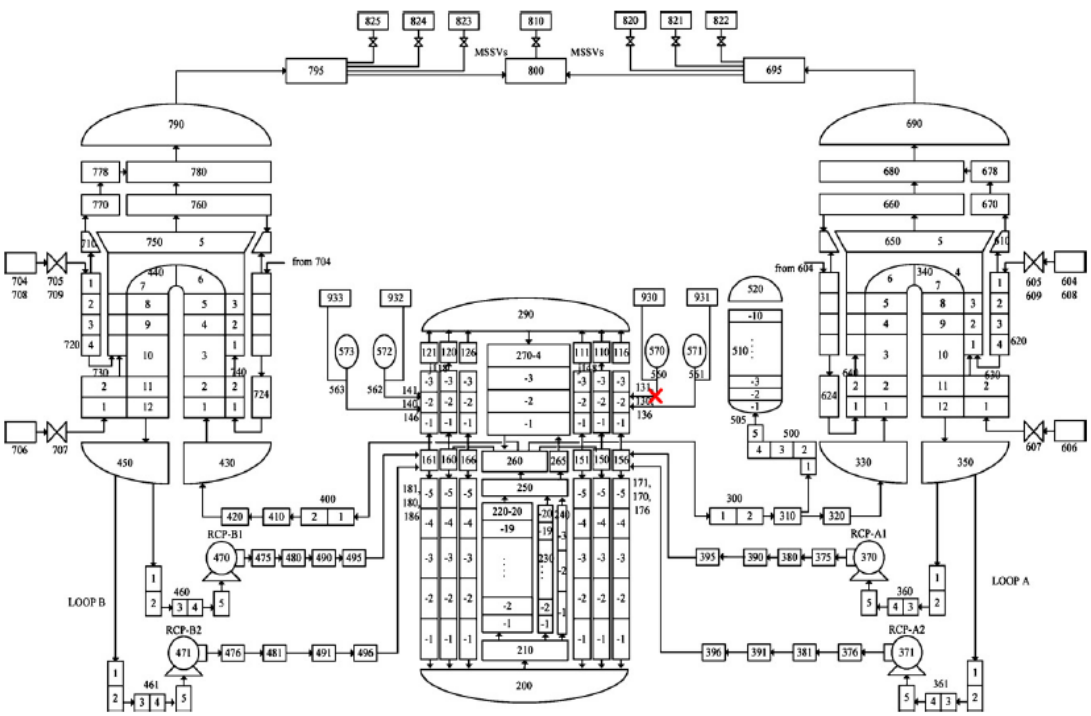

Figure 9 Nodalization of APR1400 for MARS code simulation

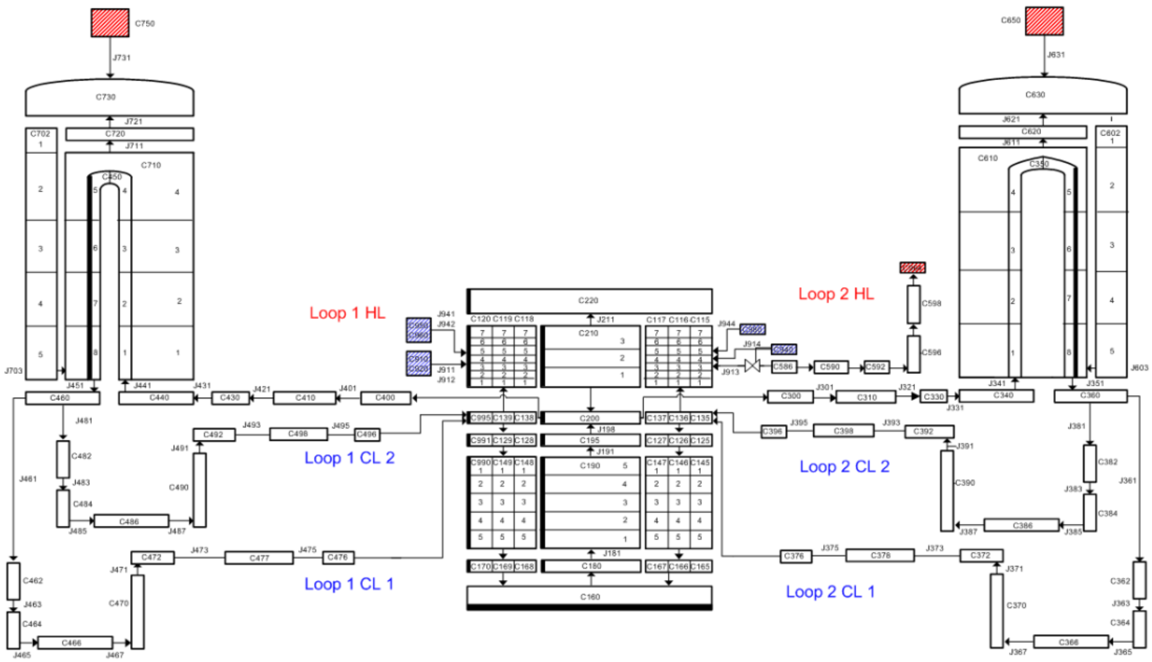

Figure 10 Nodalization of SNUF for MARS code simulation

2.6.2 Comparison of DVI Line 25% and 50% break LOCA Transients

Results of the APR1400 model and the SNUF model for the 25% break of DVI line are discussed here. Initial transients of primary system pressure before 300s agree well between the APR1400 and the SNUF simulation results as shown in **Fig. 11**. Large deviation of primary-system pressure after 300s is observed. This might result from the fact that there is no pressurizer in the SNUF model which plays an important role in stabilizing the primary-system pressure for the smaller-size DVI line break. The break flow transients between the APR1400 simulation and SNUF simulation are quite similar as seen from **Fig. 12**. Although there is some quantitative deviation of break flow rate in two-phase critical flow region, the transition from subcooled critical flow to two-phase critical flow and the transition from two-phase critical flow to vapor flow in the APR1400 can be well preserved in the SNUF as shown in **Fig. 12**.

The transients of primary-system pressure and break flow rate between APR1400 and SNUF are also quite similar for DVI line 50% break LOCA as shown in **Fig. 13 and Fig. 14**. Because the maximum operational pressure of SNUF is 8 bar, for DVI line 50% break LOCA in SNUF, the primary-system pressure decreases to 1 bar at around 600 s, after which the break flow is not critical flow. Whereas, the primary-system pressure in APR1400 is still well above 1 bar at around 600 s, after which the break flow is still critical flow. It means the SNUF cannot simulate the later LOCA transients of APR1400 for larger break sizes. In view of this, it is better to design a scaled-down test facility operated at a higher pressure to investigate the later LOCA transients of PWRs.

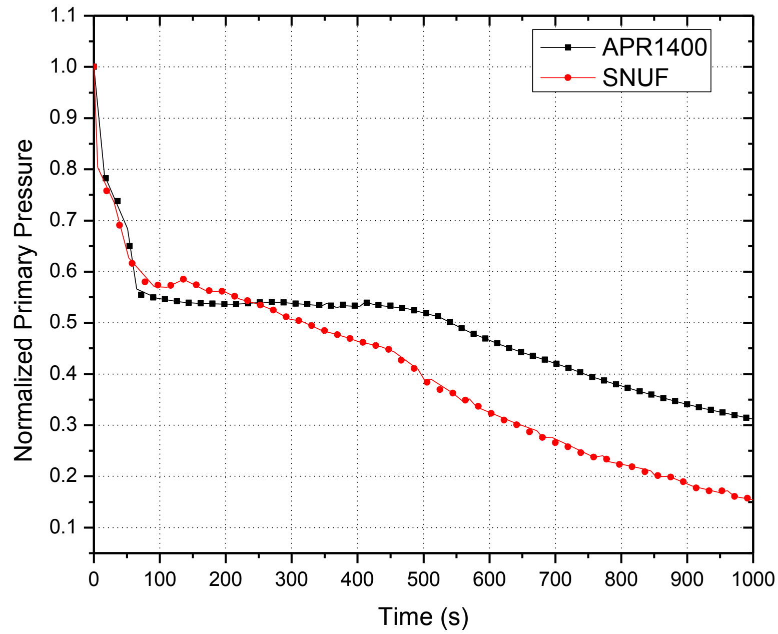

Figure 11 Core water level transient (25% break)

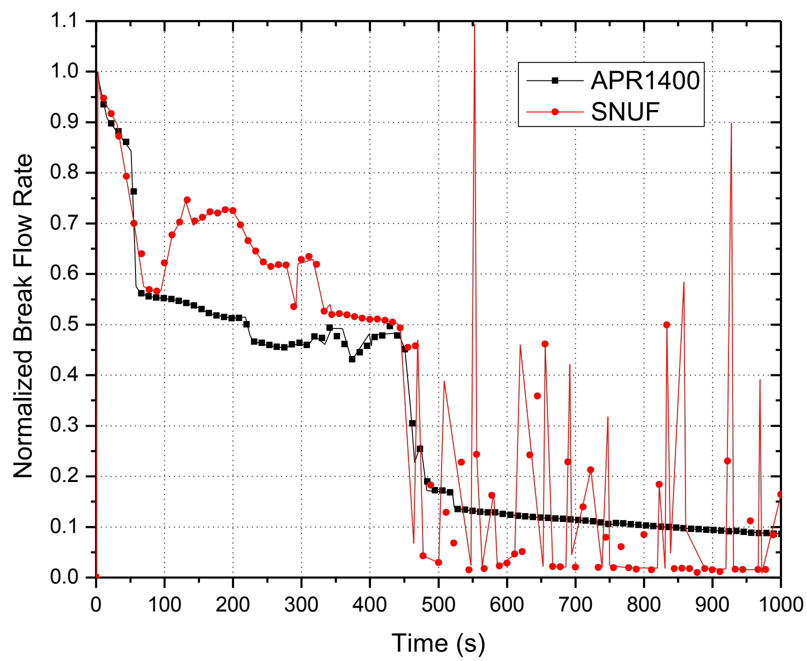

Figure 12 Normalized break flow rate transient (25% break)

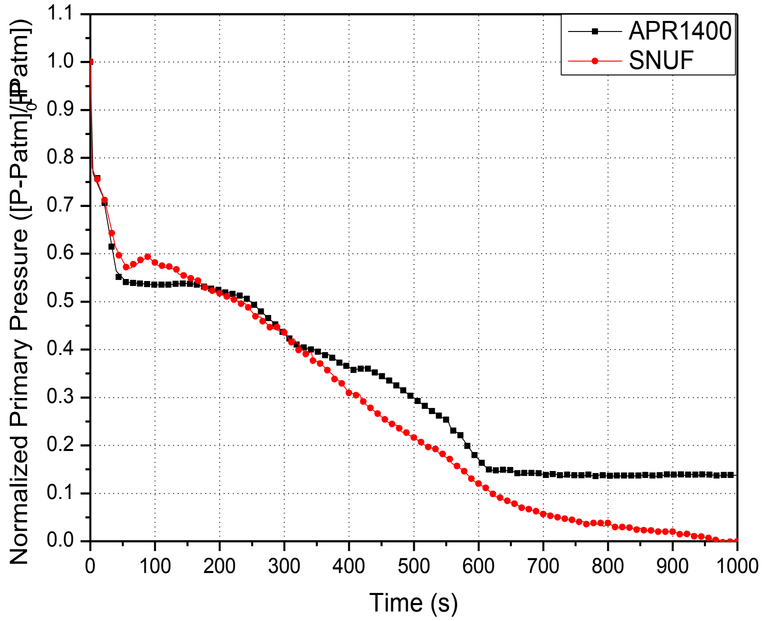

Figure 13 Primary-system pressure transient (50% break)

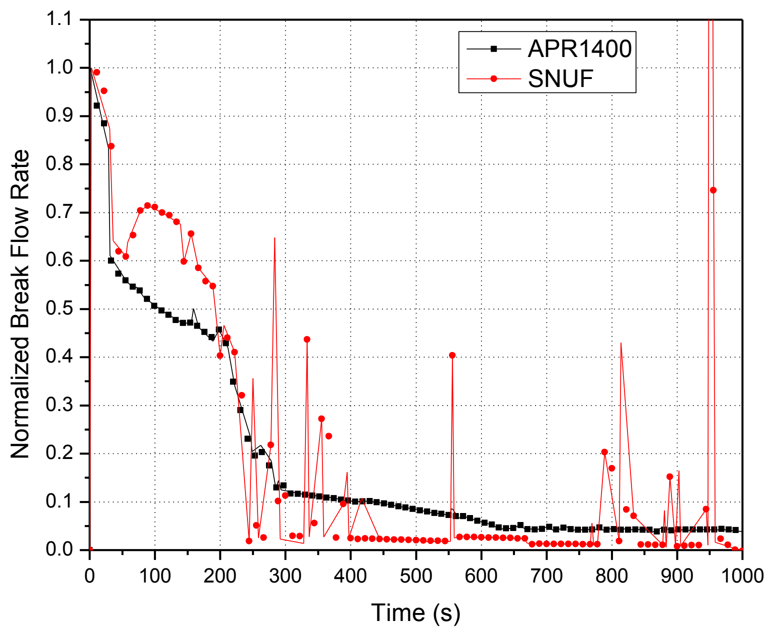

Figure 14 Normalized break flow rate transient (50% break) Test Procedures and Results

3. SNUF Experiments

3.1 Test Procedures of Steady State and Transients

The steady-state conditions for the SNUF experiments as given in **Table 1** are determined by the scaling analyses. As discussed in section 2.2 for scaling analyses, the important steady-state conditions are primary pressure, the core inlet and outlet temperatures, the secondary-side pressure and temperature. All these parameters should be maintained close to the scaled-down values during the steady state. The procedures to reach steady state in SNUF are as follows. Due to the relative low power supply of the SNUF heaters, the pressure increase rate along with the temperature increase rate is relatively low. Therefore, the rising of the primary-system temperature is realized without turning on the heat removal system of the secondary side. However, when the primary-system temperatures are close to the targeted values, small amount of feedwater is supplied to the SGs to remove the generated heat in the core so that steady state can be reached. The amount of feedwater supplied is controlled by adjusting the opening of the valve in the feedwater supply line so that the amount of heat removal from the primary system can be adjusted and thus the exact steady state in the primary system could be maintained. If the primary-system temperature and pressure still increase rapidly after the feedwater is supplied to the SGs, some heaters are turned off until the primary-system pressure keeps unchanged. The steady-state power in the reactor core for the SNUF is found to be 68 kW.

After the targeted values such as the primary-system temperature distribution and pressure are reached, the break is initiated by opening the butterfly valve installed in the simulated broken DVI line — the opening time duration of the butterfly valve used to initiate the break is around 15 s. Thereafter, various trips are activated according to the trips of the LOCA as shown in **Table 7**. Because all the safety systems in SNUF are manually rather than automatically controlled, the trips are activated manually by following the timings of various trips as shown in **Table 7**. In addition, the core decay power and SI flow rate as given in **Fig. 7** and **Fig. 8** are supplied in the SNUF experiments for each case.

3.2 Results and Discussions

3.2.1 Comparison of DVI Line 25% Break LOCA Transients

In this section, the transient results obtained by SNUF experiment and MARS code simulation with SNUF model for 25% break of DVI line are compared and discussed. Furthermore, deficiencies of MARS code are addressed.

For the transients of primary-system pressure shown in **Fig. 15**, the pressure plateau where pressure stays almost constant is at around 5 bar by SNUF experiment, whereas at around 4 bar by SNUF simulation. Because just one thermal couple and a mechanical pressure gauge is equipped to measure the fluid temperature and pressure on one SG's top plenum, not enough information of SGs' secondary side was known and thus the design values were used in the MARS code simulation. Moreover, the break unit, which consists of orifice simulation the break area and two valves, were not modeled in the code. These might contribute to the inconsistence of pressure plateau. This inconsistence of pressure plateau might also result from the code deficiency that the non-homogeneous effects such as flashing are not well modeled by the MARS code for the low pressure systems. In addition, primary-system pressure obtained by code simulation decreases more rapidly than that by the SNUF experiment after 400s. This is because the two phase flow rate through the break predicted by the MARS code is higher than the experimental value after 400s as shown by the accumulated break flow curve in **Fig. 18**.

For the transients of core water level in **Fig. 16**, the results agree quite well between SNUF experiment and simulation. In the transient of SNUF experiment, the values of water level change smoothly, whereas, in the transient of SNUF simulation, the core water level fluctuates a lot, which indicates that the simulation of thermal-hydraulic dynamics by MARS code with one-dimensional two-phase models exhibit some instabilities. Thus, some numerical solution schemes in the MARS code might need to be improved.

The transients of downcomer water levels also are very similar between the SNUF experiment and the SNUF simulation as shown in **Fig. 17**. However, the values of downcomer water level predicted by the SNUF simulation are lower than those by the SNUF experiment after 500s, which results from the fact the MARS code overestimates the break flow rate after 400s as shown in **Fig. 18**.

Although MARS code overestimates the accumulated break flow, the break flow transients of the MARS code simulation and the SNUF experiment are similar as shown in **Fig. 19**. A peak of break flow rate in the beginning of the break is observed in the SNUF experiment due to the opening of break initiation valve, whereas, it is not present in the results of code simulation in which the opening time 15 s of break initiation valve is not modeled in code simulation.

From this discussion, it indicates non-homogeneous effects such as flashing might not be well modeled by the MARS code, which results in the initially lower pressure plateau value and different transients of primary-system pressure in simulation results, and that the MARS code predicted the higher break flow rate which contributes to the more rapid depressurization rate of primary system after 400s. Hence, it is considerably important to simulate the break flow rate as accurately as possible for the thermal-hydraulic code used in the nuclear reactor safety analysis. There is very large fluctuation of core and downcomer water level values predicted by the MARS code, which indicates that the instability of numerical solution scheme might need to be improved in the MARS code.

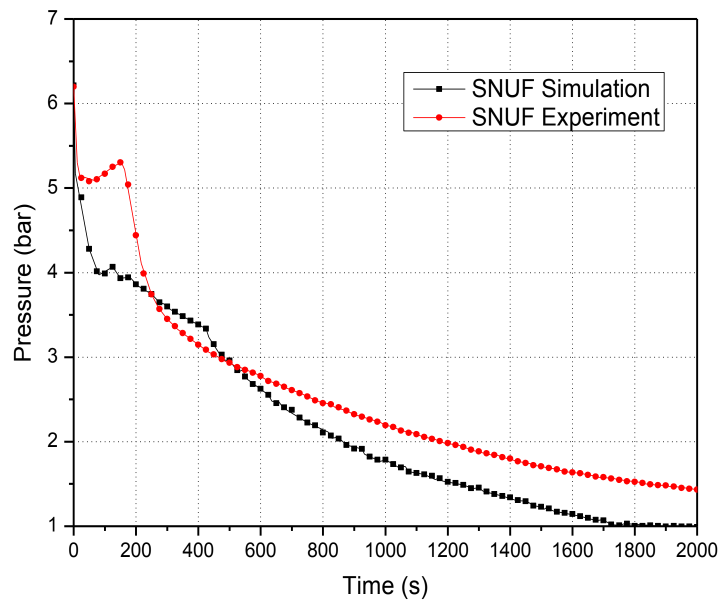

Figure 15 Primary-system pressure transients (25% break)

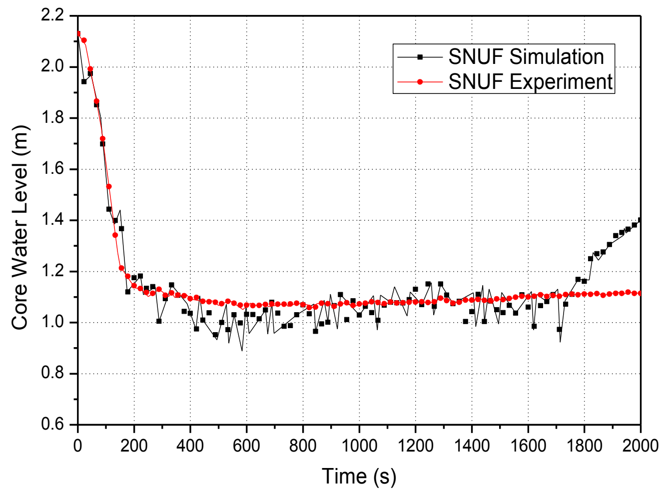

Figure 16 Core water level transients (25% break)

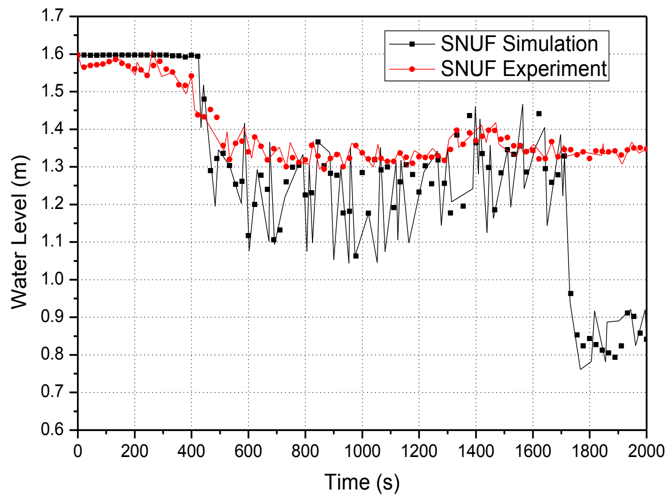

Figure 17 Downcomer water level transients (25% break)

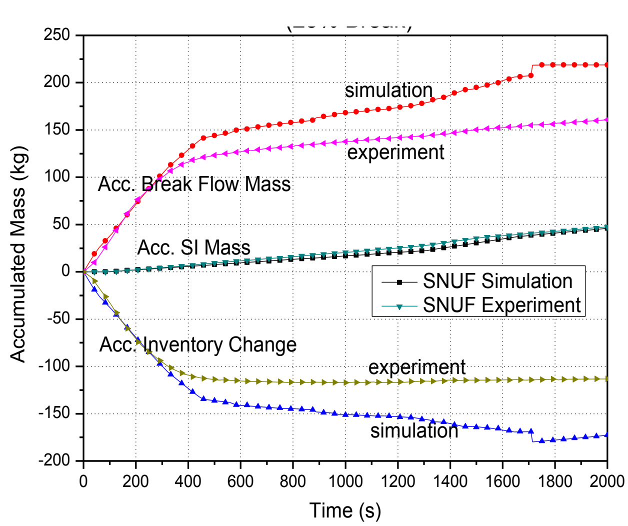

Figure 18 Accumulated mass transients (25% break)

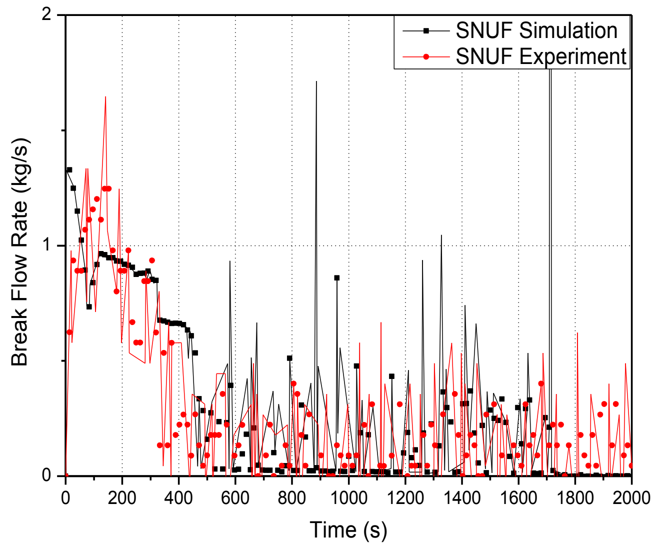

Figure 19 Break flow transients (25% break)

3.2.2 Comparison of DVI Line 50% Break LOCA Transients

In this section, the transient results for the 50% break of DVI line by both SNUF experiment and code simulation are compared and discussed.

For the pressure transients seen from the **Fig. 20**, the transients of the SNUF simulation and the SNUF experiment after 150s agrees well. However, there is again a large deviation of values between the code simulation and SNUF experiment before 150s. Again, pressure plateau at higher pressure around 5.5 bar is observed in the SNUF experiment, but, pressure plateau predicted by the MARS code is at a much lower value 4 bar. The reasons for this deviation are the same as those for DVI line 25% break LOCA as explained in Section 3.2.1.

For the core water level transients as shown in **Fig. 5-21**, results by the MARS code simulation agree well with those by experiment, except that there is a large deviation of the results at around 200s when the water level obtained by the SNUF experiment stays almost unchanged between 150s and 250s, whereas the core water level predicted by the MARS code first decreases at the initial decreasing rate after 150s and thereafter increases rapidly to the value of SNUF experiment at around 250s when the downcomer water level predicted by the MARS code decreases rapidly as shown in **Fig. 22**. This large deviation results from that the loop seal clearing occurs in MARS code simulation, whereas, not occurred in the SNUF experiment.

For downcomer water level transients as shown in **Fig. 22**, there is a large deviation in values between the MARS code simulation and the SNUF experiment. The overestimated break flow by the MARS code for 50% break is much lower than those for the 25% break as compared between **Fig. 18** and **Fig. 23**, but the deviation of downcomer water level between the SNUF experiment and the MARS code simulation for the 50% break is much larger than that for the 25% break. It is because the downcomer water level range taken in code simulation is not same as that in the experiment and some more SI water was injected into the primary system by mistake in code simulation, which was found after paper review. It gives us the lesson that the code should match the experimental conditions as closely as possible. Otherwise, the code would not reproduce the experimental results.

As shown in **Fig. 23**, the accumulated SI mass is preserved same between the SNUF experiment and MARS code simulation, and the MARS code overestimated break flow as observed from the curves of accumulated break flow. As shown in **Fig. 24**, the MARS code quantitatively predicts the break flow rate.

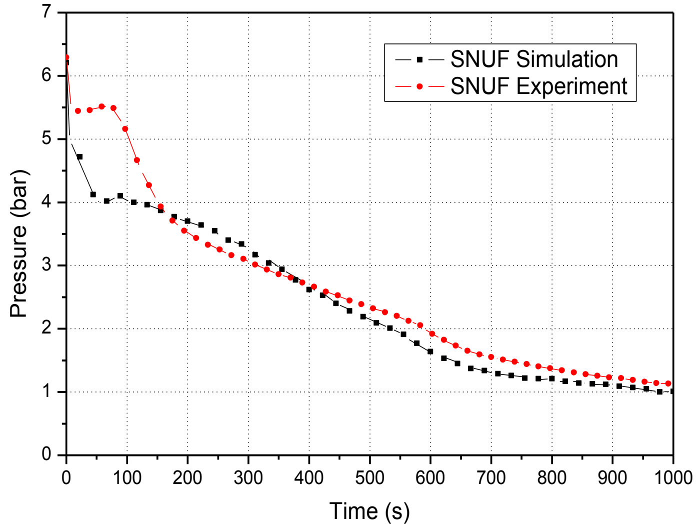

Figure 20 Primary-system pressure transients (50% break)

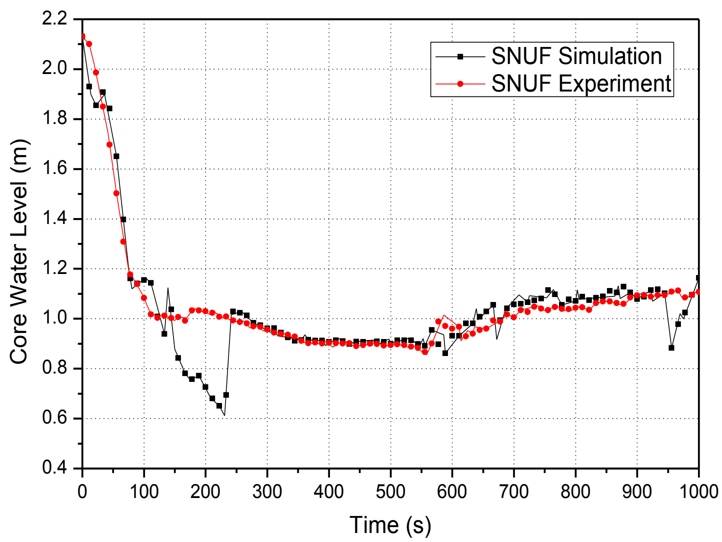

Figure 21 Core water level transients (50% break)

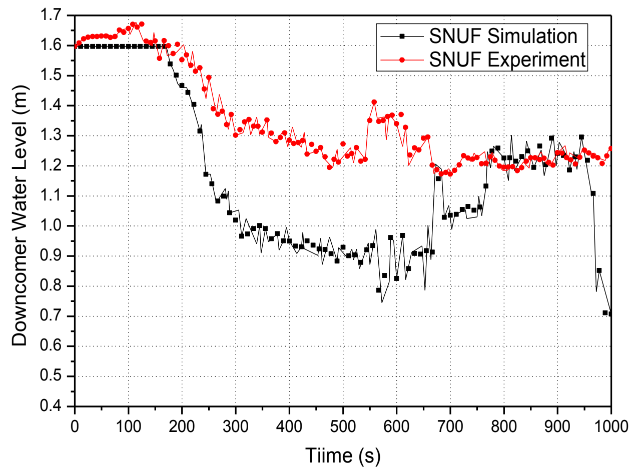

Figure 22 Downcomer water level transients (50% break)

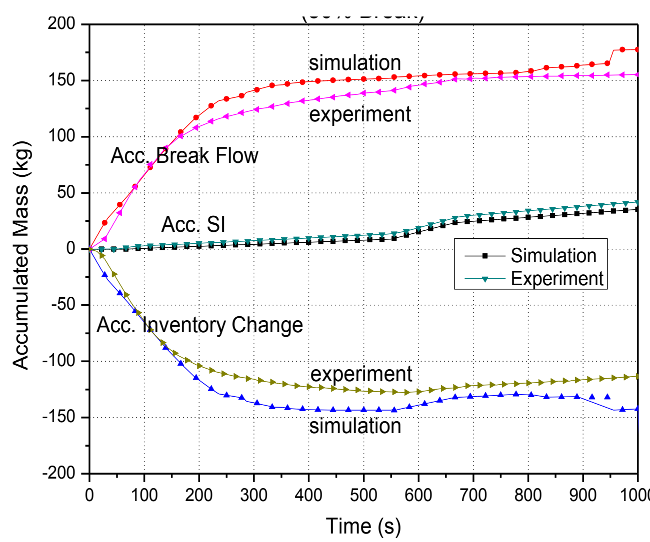

Figure 23 Accumulated mass transients (50% break)

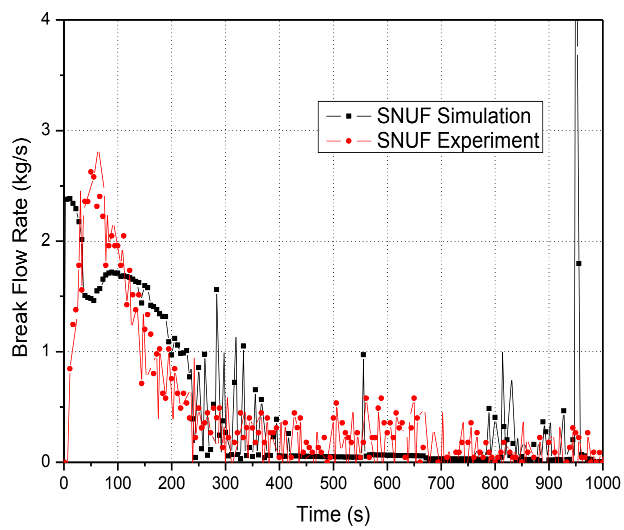

Figure 24 Break flow transients (50% break)

4. Conclusion

MARS code simulations of LOCAs with different break sizes of DVI line using APR1400 model and SNUF model are performed. Resorting to a critical flow model for subcooling coolant proposed by Fauske and a rational scaling methodology proposed by Dr Jose N. Reyes, Jr, the test conditions of SNUF are well obtained by scaling down those of the APR1400 which are calculated by MARS code. After that, the test conditions are applied to both MARS code simulation of SNUF and SNUF experiments.

The code calculated transients of APR1400 and SNUF are similar, which indicates that some major thermal-hydraulic transients of APR1400 are preserved in the scaled-down test facility SNUF, and verifies the scaling methodology adopted. It shows that the scaled-down test facility is capable of preserving the similar thermal-hydraulic transients of large-scale system.

Test data from SNUF experiment are compared with MARS code calculations with SNUF model for DVI line 25% and 50% break LOCAs. From these comparisons, the transients of some major parameters between the SNUF experiments and the code calculations are quite consistent. However, there are some deviations in values of some major parameters such as primary-system pressure and break flow rate. By analysing these deviations, some deficiencies of MARS code for simulating transients of low-pressure thermal-hydraulic systems are addressed. From the comparison of break flow rates, it shows the MARS code generally overestimates the break flow, but quantitatively predicts break flow rate. MARS code under-predicts the values of pressure plateau in primary-system pressure transients for all the two cases, which might come from the fact that the non-equilibrium effects are not well modelled in the MARS code. In general, the MARS code appropriately predicts the transients of DVI line break LOCAs in SNUF.

5. References

[1] Bae, K. H., Lim, H. S., et al., "Design Options for the Safety Injection System of Korean Next Generation Reactor", Annals of Nuclear Energy, Vol. 27, pp. 1011-1028 (2000).

[2] Chu, I. C., Song, C. H., et al., "Development of Passive Flow Controlling Safety Injection Tank for APR1400", Nuclear Engineering and Design, Vol. 238, pp. 200-206 (2008).

[3] Lee, K. H., et al., "An Integral Loop Test and MARS code Analysis for a DVI Line Break LOCA in the APR1400," Nuclear Engineering and Design, Vol.238, pp. 3336-3347 (2008).

[4] Ishii, M., Kataoka, I., "Scaling Laws for Thermal-Hydraulic System under Single Phase and Two-Phase Natural Circulation," Nuclear Engineering and Design, Vol. 81, pp. 411-425 (1984).

[5] Levy, S., "Two-phase flow in complex systems," John Willey & Sons, Chapter 2 (1999).

[6] Reyes, J. N., Hochreiter, L., "Scaling Analysis for the OSU AP600 Test Facility (APEX)", Nuclear Engineering and Design, Vol. 186, pp. 53-109 (1998).

[7] Wulff, W., "Scaling of Thermohydraulic Systems", Nuclear Engineering and Design, Vol. 163, pp. 359-395 (1996).

[8] Fauske, H. K., and Epstein, M., "Source Term Considerations in connection with Chemical Accidents and Vapour Cloud Modeling", J. Loss Prev. Process In., Vol. 1, (1988).

[9] American Nuclear Society, "Decay Heat Power in Light Water Reactors", ANSI/ANS 5.1-1979, LaGrange Park, IL, USA (1979).