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Open Calculation Result of DSP-05 Activity utilizing ATLAS Test Facility with Multiple Steam Generator Tube Rupture under PAFS Operation Scenario

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1. Introduction

A domestic collaboration using an integral effect test facility, ATLAS (Advanced Thermal-Hydraulic Test Loop for Accident Simulation) [1], which was named as domestic standard problem (DSP) exercise was started in 2009 first. The DSP exercise program has contributed to enhancing the safety analysis technology and to keeping the human networking among nuclear safety experts in Korea.

The 5th DSP exercise was launched on September 11, 2018. Among various accident scenarios, a multiple steam generator tube rupture (SGTR) with operation of passive auxiliary feedwater system (PAFS) was proposed as the test item of the DSP-05 based on the technical discussion between the participants [2]. The SGTR accident is one of the design basis accidents having a significant impact on safety in a viewpoint of radiological release. The system might show an asymmetric behavior with the multiple SGTR at one steam generator and the PAFS operation at another generator. This scenario can be a challenge for evaluation of the system code prediction capability in this point of view including the PAFS modeling.

As the first step of DSP-05, the blind phase analysis was conducted [3]. Only the initial and boundary conditions of the test were provided to the participants. After finalizing the blind phase, the test result was opened to the participants and they performed the open phase analysis. In the open phase, total 15 organizations participated as listed in Table I.

In this paper, the open phase calculation results were compared and discussed not only between the participants but also with the blind phase analysis results. All the test data and analysis results which are shown in this paper were normalized by an arbitrary value for the confidential problem of test data.

Table I: Participants of DSP-05

Participants	Code
DOOSAN	RELAP5 MOD3.3 Patch4
EN2T-A	MARS-KS 1.5
EN2T-B	TRACE V5 patch4 / SNAP 2.4.1
FNC	SPACE 3.2
KAERI	SPACE 3.2
KAIST	MARS-KS 1.5
KEPCO E&C	SPACE 3.2
KHNP-A	SPACE 3.2
KHNP-B	MARS-KS 1.4
KHU	MARS-KS 1.5
KINS	MARS-KS 1.5
KNF	SPACE 3.12
PNU	MARS-KS 1.5
SENTECH	MARS-KS 1.4
UNIST	MARS-KS 1.5

2. Description of the Experiment

The sequence of major events of the test was shown in Table II. The initial and boundary conditions were determined from the scaling analysis result of APR1400 (Advanced Power Reactor 1400 MWe) plant condition which was obtained from MARS-KS (Multi-dimensional Analysis of Reactor Safety – KINS Standard) calculation result. [4] The detailed test conditions and procedures are described in the literature [1, 2].

Table II: Sequence of events

Description	Remark(Set-point)
SGTR initiation	OV-BS-04 Open
RCP trip	Coincidence with break
PRZ heater off	LT-PZR-01 < 1.2 m
HSGL signal	SG-1 level > 5.05 m
Reactor trip	Coincidence with HSGL
Decay Power	Reactor trip + 12.07 sec delay
MSCV close	Coincidence with HSGL
MFIV close	Coincidence with HSGL
MSIV1/2 close	Coincidence with HSGL
MSSV operation	7.7 MPa < PT-SGSD1/2-01 < 8.1 MPa
SIP injection	PT-PZR-01 < 10.72 MPa + 28.28 s delay
PAFS actuation	SG-2 wide level<25 % (2.78 m)
SIT injection	PT-PZR-01 < 4.03 MPa

3. Evaluation of Open Phase Calculation

Total 58 major thermal-hydraulic parameters were submitted from 15 organization participants for quantitative comparison. The whole test was divided into three phase: initial steady state, transient from a break valve opening to PAFS operation, and transient after PAFS operation.

3.1 Steady State Analysis Result Evaluation

To evaluate the steady state analysis result quantitatively, the global acceptability factor, Q_B, was utilized. The global acceptability factor is the sum of whole single acceptability factor, Q_i, which is calculated based on the error between an experimental value and a calculated value of each thermal hydraulic parameter with its weighting factor. As the Q_B is close to 1.0, it means that calculation result close to the test result.

The detailed procedure for calculation of the global acceptability factor is explained in the literature [3]. According to the evaluation result of the steady state analysis result as shown in Figure 1, the analysis result from the most participants showed very good agreement with the test result with 1.12 of averaged Q_B .

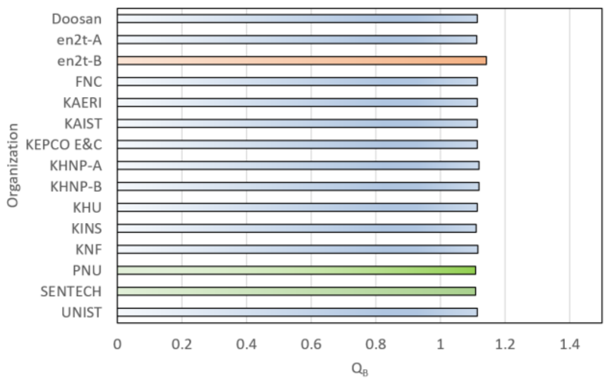

Fig. 1. Evaluation result of steady state analysis

3.2 Transient Analysis Result Evaluation

The Fast Fourier Transform Based Method (FFTBM) was used to evaluate the transient analysis result quantitatively. [5] The overall accuracy of a code calculation result can be evaluated by the total weighted average amplitude (AA). If the AA value is 0, it means that the code analysis result is exactly the same with the test result. If the AA value is smaller than 0.3, it means a very good prediction capability. And it can be considered as a fairly good prediction result between 0.3 and 0.5 for AA values. If the AA is larger than 0.5, the code analysis result can be considered as a poor prediction. The FFTBM evaluation was conducted for 58 major thermal hydraulic parameters which were calculated from 15 organization participants.

The actuation of PAFS is a major event in this scenario and the system thermal hydraulic phenomena were significantly different before and after PAFS actuation. Most participants predicted earlier actuation time of PAFS than that of the test result. Due to the different prediction result of PAFS actuation time, the AA values of whole transient period showed poor prediction result from most participants. However, prediction results were good in the transient period after PAFS operation except several organizations.

Figure 2 shows the primary system pressure behavior. Most participants predicted the primary system pressure behavior in a good agreement (~0.2<AA) with the pressure decrease tendency after PAFS operation.

Figure 3 shows the secondary system pressure variation after PAFS operation. Most of the code analysis results showed lower pressure behavior than the test result. It means that the system cooling through the steam generators was estimated more active scale than that of the test result even though the higher decay heat power due to the early actuation time of PAFS.

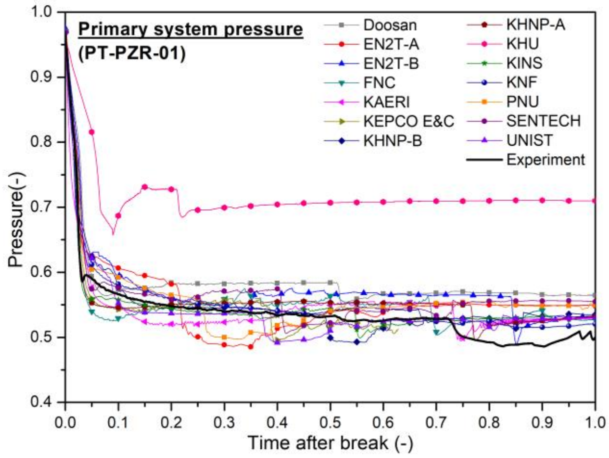

Fig. 2. Primary system pressure calculation results

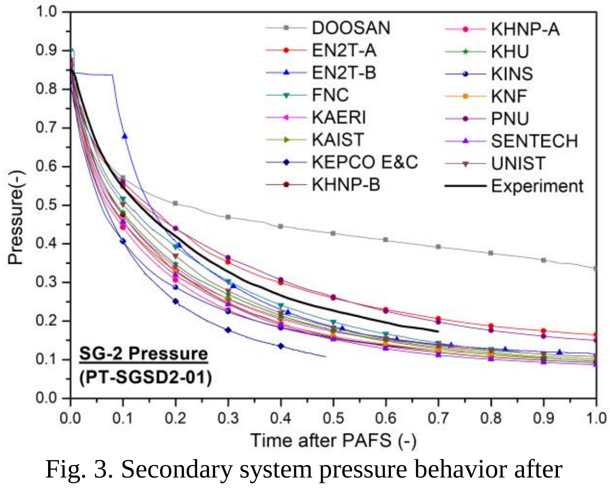

Fig. 3. Secondary system pressure behavior after

In the test, the break flow rate from the SGTR was directly measured utilizing the orifice flow meter which was installed on the break spool [2]. The integrated mass of the break flow was compared with the code analysis results as shown in Figure 4. Total integrated mass of break flow was estimated larger than the test result as the transient went on. Nevertheless, the collapsed water level in the reactor pressure vessel

(RPV) did not decrease significantly before safety injection pump (SIP) actuation in the most code analysis results.

As PAFS was actuated, the system cooling was conducted mainly through the steam generator number 2 (SG-2) which is connected with PAFS. Thus, primary loop flow rates showed asymmetric behavior after PAFS operation. The flow rate of the loop-1 decreased and the flow rate of loop-2 increased. The decrease tendency of loop-1 flow rate after PAFS operation was well predicted from all participants. However, as shown in Figure 5, the calculated flow rates of loop-1 decreased to almost zero during the late phase of test period (AA>0.5).

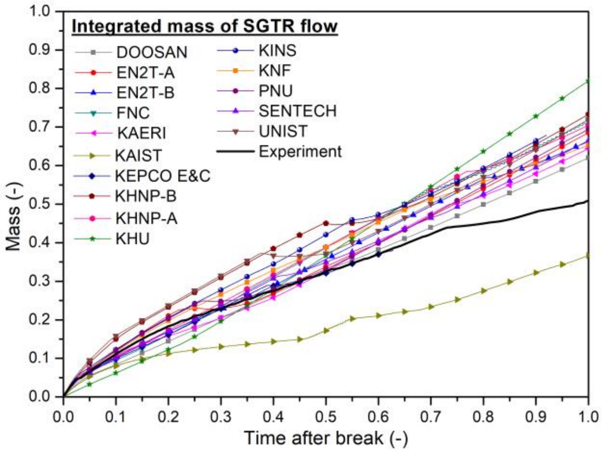

Fig. 4. Integrated mass of break flow from SGTR

Not only the loop-1 flow rate but also loop-2 flow rate were predicted smaller than the test result from the most participants as shown in Figure 6 (0.3<AA<0.5). In conclusion, the total loop flow rate of the primary system was predicted smaller after PAFS operation. Since the secondary system pressure decreased faster than the test result in spite of the smaller loop flow rate, the heat removal rate from the primary system by steam generators needs to be modified with a heat loss modeling analysis, as a further study.

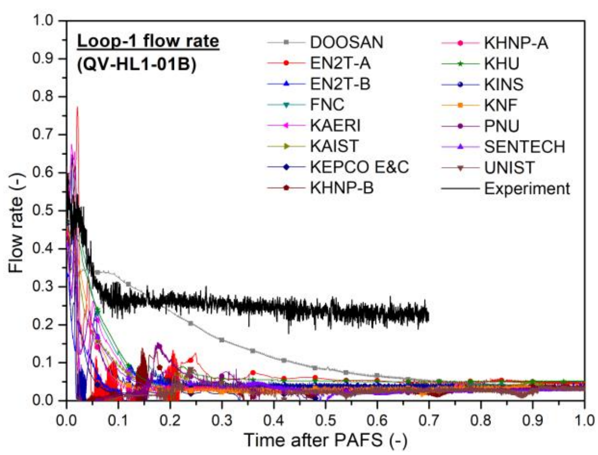

Fig. 5. Hot leg flow rate of loop-1

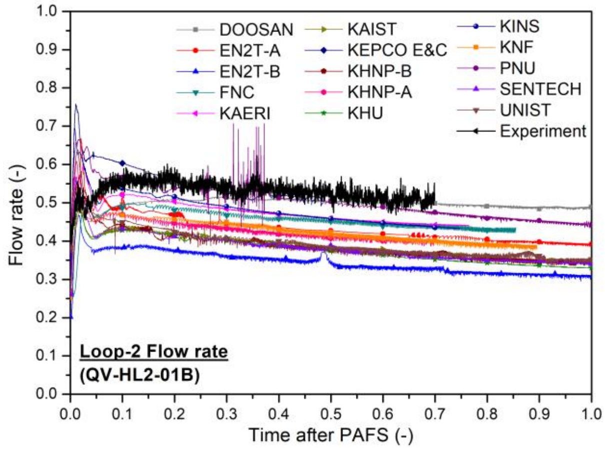

Fig. 6. Hot leg flow rate of loop-2

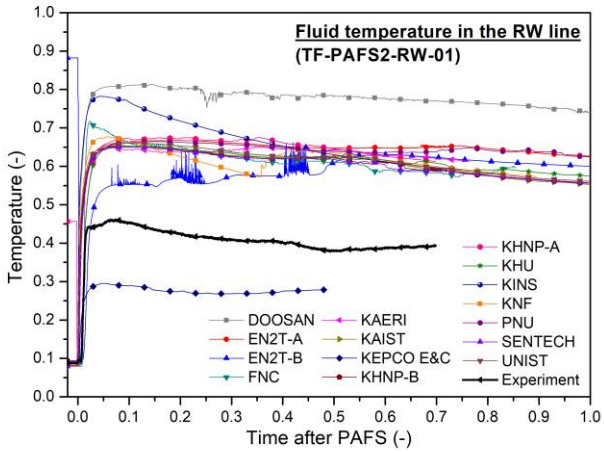

Fig. 7. Fluid temperature in RW line of PAFS

The calculated flow rates of the PAFS steam supply (SS) line and returned water (RW) line show a good agreement with the test result from the most participants in spite of the fluctuation behavior. The fluid temperatures in the RW line, however, was estimated higher than the test result as shown in Figure 7 (AA>0.5) while the fluid temperatures in the SS line was predicted reasonably (0.3<AA<0.5). It can be concluded from this result that the heat removal from PAFS was underestimated in the code calculation analysis.

In the preceding blind phase analysis result, most participants predicted the late actuation time of PAFS compared with the test result. In addition to that, the flow rate of the primary system was estimated smaller and the fluid temperature in the RW line of PAFS was estimated higher than the test result.

In the open phase calculation, the major different analysis strategy adopted by most participants was that they added or modified the heat loss modeling of the system, i.e. primary system (especially for a RPV), SGs, or pressurizer. In the open phase analysis result, similar mismatch point with the blind phase analysis result were also found. However, the differences with the test result were reduced and the AA values were enhanced approximately 14% based on the averaged AA value for the transient period after PAFS operation in the open phase analysis result. First of all, the prediction of the PAFS actuation time was enhanced in the open phase analysis compared with the blind phase analysis from the most participants

4. Conclusion

The open phase calculation of DSP-05 activity was conducted for an accident scenario of a multiple SGTR accident under the PAFS operation condition. Total 15 organizations participated in the open calculation and their analysis results were compared quantitatively.

Compared with the preceding blind phase analysis result, most participants added or modified the heat loss modeling in their system analysis code. So prediction capability of the thermal hydraulic system behaviors including the PAFS actuation time was enhanced in the most calculations. So it can be concluded that the proper modeling of the heat loss in the system analysis code can have a significant effect on the prediction capability for this kind of accident scenario.

Based on the insights from the open phase analysis result, major system behaviors such as the actuation time of PAFS, behaviors of the coolant inventory including the break flow rate, the collapsed water level in the RPV and the loop flow rate can be improved as a further study.

List of Abbreviations

AA	Average Amplitude
APR1400	Advanced Power Reactor 1400 MWe
ATLAS	Advanced Thermal-Hydrulic Test Loop for Accident Simulation
DSP	Domestic Standard Problem
FFTBM	Fast Fourier Transform Based Method
HSGL	High Steam Generator Level
MARS-KS	Multi-dimensional Analysis of Reactor Safety-KINS
MFIV	Main Feedwater Isolation Valve
MSIV	Main Steam Isolation Valve
MSCV	Main Steam Control Valve
MSSV	Main Steam Safety Valve
PAFS	Passive Auxiliary Feedwater System
PRZ	Pressurizer
RCP	Reactor Coolant Pump
RPV	Reactor Pressure Vessel
RW	Returned water
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SIP	Safety Injection Pump
SIT	Safety Injection Tank
SS	Steam Supply

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