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Comparison of Blind and Open Calculation Results for Top-Slot Break LOCA in Fourth ATLAS Domestic Standard Problem

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Abstract: The advanced thermal-hydraulic test loop for accident simulation (ATLAS) was developed and operated at The Korea Atomic Energy Research Institute. The ATLAS is operated to simulate accidents in pressurized water reactors (PWRs). A domestic standard problem (DSP) using the ATLAS was proposed for transferring the database of the integral effect test to Korean nuclear researchers and developing the safety analysis methodology of PWRs. The fourth DSP (DSP-04) exercise was performed during 2015–2017 with 15 participants (13 organizations), that are universities, government, and nuclear industries. In DSP-04, a top-slot break at the cold leg was chosen as the target scenario to resolve an issue about the effect of loop seal clearing and the reformation on the peak cladding temperature during the cold leg top-slot break LOCA for APR1400. The participants performed a code calculation for the experimental simulation and sensitivity studies for the enhancement of the code. This paper includes brief information about the experimental and major code assessment results.

Keywords: ATLAS; DSP; top-slot break; LOCA; loop seal clearing; code assessment

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1. Introduction

The advanced thermal-hydraulic test loop for accident simulation (ATLAS) was developed and operated at The Korea Atomic Energy Research Institute. The ATLAS is operated to simulate accidents in pressurized water reactors (PWRs) for resolving the thermal hydraulic safety issues of PWRs and supporting assessment and validation of safety analysis codes. A domestic standard problem (DSP) using ATLAS was proposed for transferring the database of the integral effect test facility to Korean nuclear researchers and developing the safety analysis methodology of PWRs. The first DSP (DSP-01) was operated with the simulation data for guillotine break loss of coolant accident (LOCA) at a direct vessel injection nozzle [1]. In the second DSP (DSP-02), a 6-inch cold leg-break small-break accident (LOCA) was selected as the target scenario [2]. A double-ended guillotine break of the main steam line at an 8% power without loss of off-site power was selected for the third DSP exercise (DSP-03) [3]. For each DSP exercise, participants performed code calculations and sensitivity studies using thermal-hydraulic safety analysis codes. These activity results were described in references [1–3]. International benchmarking activities using ATLAS were also performed for various scenarios by international joint projects, which are the 50th OECD/NEA/CSN international standard problem [4], OECD/NEA-ATLAS project [5], and OECD/NEA-ATLAS2 project [6].

Loop seal clearing and reformation can induce a core temperature excursion during the cold leg top-slot break LOCA for APR1400 [7]. To resolve this issue, a LOCA experiment with a top-slot break at the cold leg was selected as the target scenario for the fourth DSP exercise (DSP-04). Fifteen organizations, including universities, government, and nuclear industries, participated in DSP-04, and participants performed code calculations for the experimental simulation and sensitivity studies of the enhancement of the code assessment. In particular, they focused on loop seal clearing and reformation in the intermediate legs. In this paper, information about the experimental and major code assessment results is briefly described. Table 1 shows the list of organizations and code names used in DSP-04. The thermal-hydraulic safety analysis codes used in DSP-04 are SPACE, RELAP5, MARS-KS, and TRACE. These codes are introduced as "The SPACE code adopts advanced physical modeling of two-phase flows, mainly two-fluid three-field models which comprise gas, continuous liquid, and droplet fields and has the capability to simulate 3D effects by the use of structured and/or nonstructured meshes." [8], "The RELAP5 code has been developed for best-estimate transient simulation of light water reactor coolant systems during postulated accidents. The code models the coupled behavior of the reactor coolant system and the core for loss-of-coolant accidents and operational transients such as anticipated transient without scram, loss of offsite power, loss of feedwater, and loss of flow. A generic modeling approach is used that permits simulating a variety of thermal-hydraulic systems." [9], "Korea Advanced Energy Research Institute (KAERI) conceived and started the development of MARS code with the main objective of producing a state-of-the-art realistic thermal-hydraulic systems analysis code with multi-dimensional analysis capability. MARS achieves this objective by very tightly integrating the one dimensional RELAP5/MOD3 with the multi-dimensional COBRA-TF codes." [10], "The TRACE (TRAC/RELAP Advanced Computational Engine) is the latest in a series of advanced, best-estimate reactor systems codes developed by the U.S. Nuclear Regulatory Commission in the frame of CAMP (Code Application and Maintenance Program). The TRACE code is a component-oriented reactor systems analysis code designed to analyze light water reactor transients up to the point of significant fuel damage." [11].

Table 1. List of organizations and code names.

Organizations	Code
Korea Atomic Energy Research Institute	SPACE
Korea Electric Power Corporation Engineering and Construction	SPACE
Korea Hydro & Nuclear Power	SPACE, RELAP5
FNC technology	MARS-KS
Hanyang University	MARS-KS
Korea Advanced Institute of Science and Technology	MARS-KS
Korea Institute of Nuclear Safety	MARS-KS
Pusan National University	MARS-KS
Sen Tech.	MARS-KS
Ulsan National Institute of Science and Technology	MARS-KS
EN2T	MARS-KS, TRACE
DOOSAN heavy industries	RELAP5
Korea Electric Power Corporation Nuclear Fuel	RELAP5

2. ATLAS Facility and Test Description

2.1. Overview of the ATLAS Facility

ATLAS was developed in 2006. The ATLAS has geometrical similarity with the APR1400, which has a rated thermal power of 4000 MWth and two hot legs and four cold legs of the reactor coolant system (RCS). ATLAS has scale ratios of half-height and 1/288 volume with respect to the APR1400. The ATLAS can simulate high-pressure scenarios because the designed operating pressure is 18.7 MPa. The ATLAS has a primary system, a secondary system, a simulated break system, and a simulated containment system. The primary system has two hot and four cold legs, one pressurizer, and two steam generators [12,13]. ATLAS has about 1500 measurement points to investigate multi-dimensional phenomena for accidents [14]. Figure 1 shows a schematic diagram view of ATLAS. The detailed information on the ATLAS is described in the literature [14].

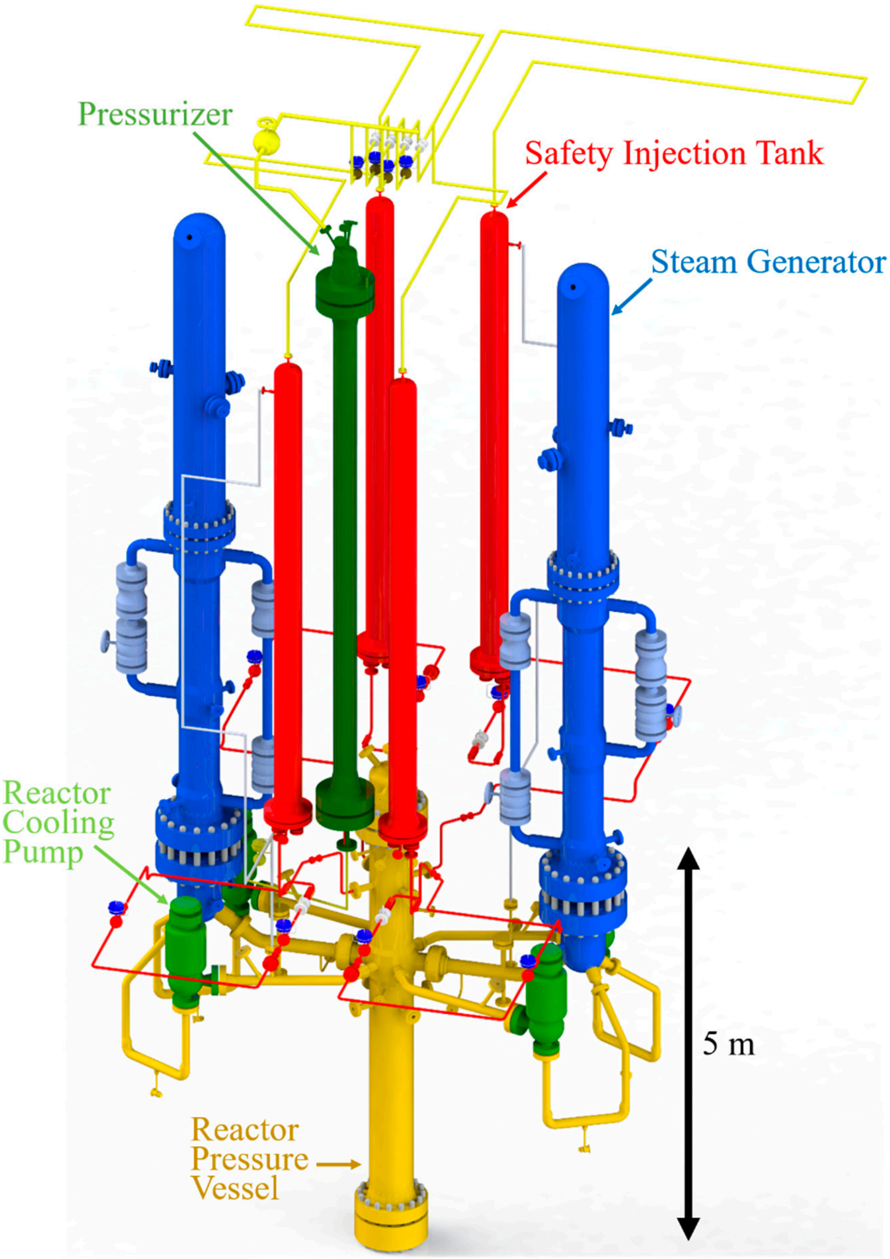

Figure 1. Schematic diagram view of ATLAS [14].

2.2. Experimental Conditions and Procedures

A 4-inch cold leg top-slot break LOCA test (ID: LTC-CL-04R) was performed for DSP-04. The main aims of the test were to investigate the effect of loop seal clearing and reformation on core temperature excursions and to produce experimental data to validate the safety analysis codes in DSP-04.

As shown in Figure 2, a break system was connected at the upper side of the cold leg (1A) to simulate a cold leg top-lot break in the LTC-CL-04R test. The cold emergency core cooling (ECC) water temperature and rated safety injection pump (SIP) flow rate were assumed to promote repeatable loop seal clearing and reformation. The rated SIP flow rate was 0.32 kg/s, which was according to the scaling ratio of loop seal clearing and reformation in the intermediate legs.

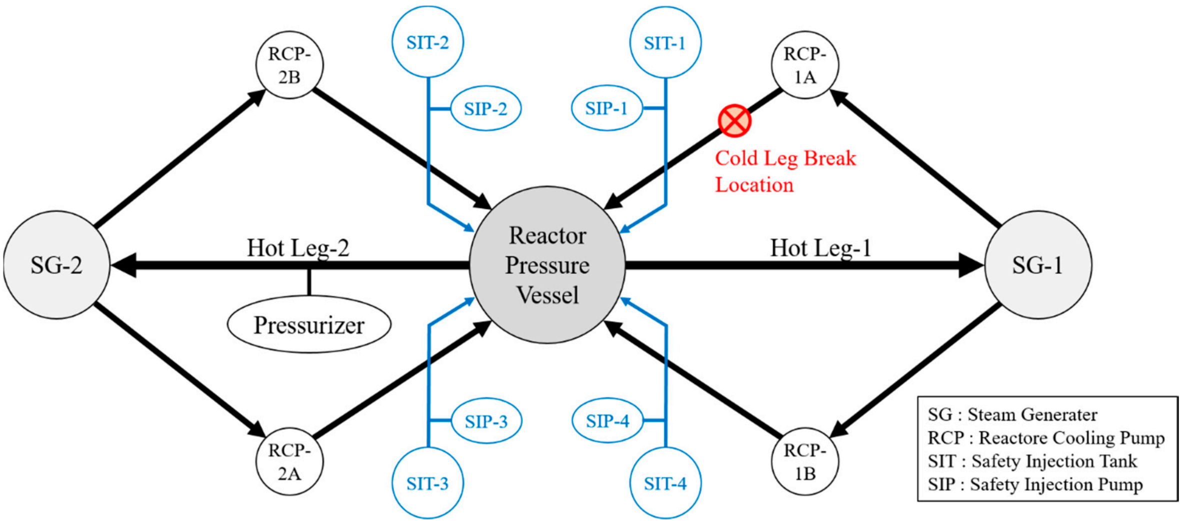

Figure 2. Arrangement of the primary loop of ATLAS and break location.

1.2 times of the ANS-73 decay heat curve was applied for the conservative test. The heater power at the beginning of the transient test was 1.64 MW including the core power 1.567 MW, and the heat loss rate at the primary system 88 kW. Four bypass valves are installed at the downcomer at ATLAS. Two bypass valves connect the downcomer and upper head, and the other valves connect the downcomer and hot legs. All bypass valves were closed to provide a conservative condition for loop seal clearing and reformation.

In Table 2, the major control logic setpoints are summarized. For the confidentiality of the test data, the test results were normalized using an arbitrary value including the time frame. The steady-state conditions of ATLAS were maintained for more than 30 min. And then, the test was begun with a break simulation valve opening.

Table 2. Control logic set point for LTC-CL-04R test.

	Set Point (Non-Dimensional Units)
Primary system logic	
By-pass rate	0%
Cold leg-break time	0.033
LPP	Pressurizer pressure < 0.78
Reactor scram/RCP trip/Turbine trip/MFIV and MSIV close	LPP + 0.0 s
SIP on time	(Pressurizer pressure < 0.67) + 0.003
SIT on SIT low-flow conversion SIT stop	Downcomer pressure < 0.25 SIT level 1,2,3,4 < 72.8, 72.6, 72.0, 72.2% SIT level 1,2,3,4 < 47.4, 47.2, 46.6, 47.0%
Secondary system logic	
MSSV1,2-01 open MSSV1,2-01 close	SG secondary pressure > 0.51 SG secondary pressure < 0.48
MSSV1,2-02 open MSSV1,2-02 close	PT-SGSD1,2-01 > 0.52 PT-SGSD1,2-02 < 0.49
MSSV1,2-03 open MSSV1,2-03 close	PT-SGSD1,2-01 > 0.53 PT-SGSD1,2-03 < 0.50

LPP: Low Pressurizer Pressure. RCP: Reactor Cooling Pump. MFIV: Main Feedwater Isolation Valve. MSIV: Main Steam Isolation Valve. SIP: Safety Injection Pump. SIT: Safety Injection Tank. MSSV: Main Steam Safety Valve. SG: Steam Generator. PT: Pressure Transmitter. SGSD: Steam Generator Steam Dome.

2.3. Major Experimental Results

Figure 3 shows the pressure of the pressurizer and steam generators. The red regions in the graph represent the duration of the loop seal clearance. After the break, the pressurizer pressure, which represents primary system pressure, was decreased. Then, the pressure plateaued for a short time. The plateau of pressure ended at the first loop seal clearing. The pressurizer pressure decreased during the whole period except during the loop seal reformation. In particular, pressure sharply decreased at the beginning of the loop seal clearing. During the loop seal reformation, the loop seals were filled with liquid water and the pressure gradually increased because of steam that had accumulated in the upper head of the reactor pressure vessel (RPV). The pressure of the steam generators increased after the isolation of the feed and steam lines when the valves were closed by the low pressurizer pressure (LPP) signal (Table 2). Subsequently, the steam in the steam generators (SGs) was vented by the main steam safety valves (MSSVs). The pressure of steam generators gradually decreased after the first loop seal clearing owing to heat removal of the secondary side to the primary side at the SGs because the saturated temperature at the secondary side was higher than the primary side.

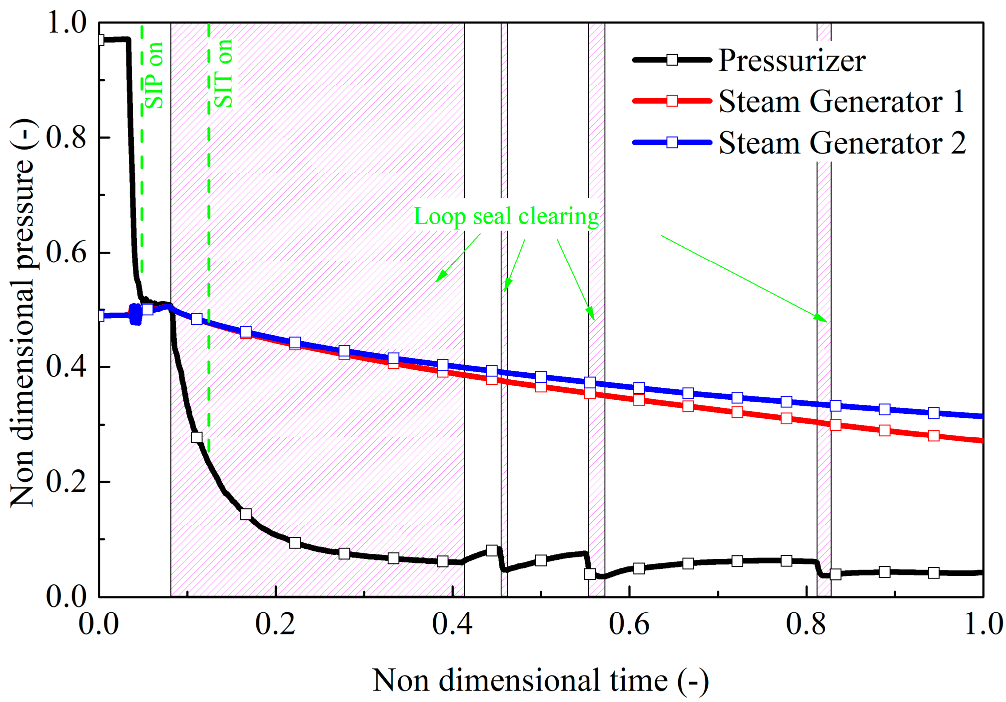

Figure 3. Experimental result—system pressure.

Figure 4 shows the maximum core heater temperatures and saturated temperature at the upper head of RPV. During loop seal reformation, the core temperatures increased. The reason for this increase can be an increase in the saturated temperatures owing to the increased pressure due to the accumulated steam in the upper head of RPV. Therefore, the heater temperature increases were not a critical phenomenon that affected the overall cooling ability of the RCS during the transient test.

Loop seal clearing and reformation were measured with a collapsed water level of a vertical pipe on the intermediate leg (IL) of the primary loop. Figure 5 shows the location of the collapsed water level at IL (RPV side). Figure 6 shows the collapsed water level at the RPV side of the intermediate leg. During loop seal clearing, the collapsed water level is lower than the top of the horizontal intermediate leg in this graph. On the other hand, loop seal reformation is confirmed if the collapsed water level becomes higher than the top of the horizontal intermediate leg.

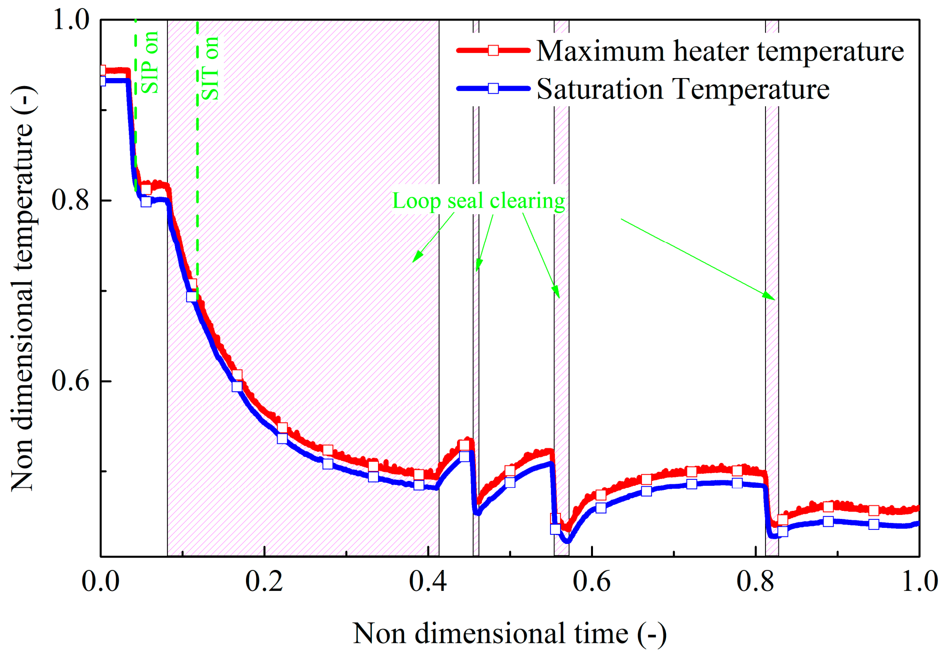

Figure 4. Experimental result—maximum heater temperature and saturation temperature.

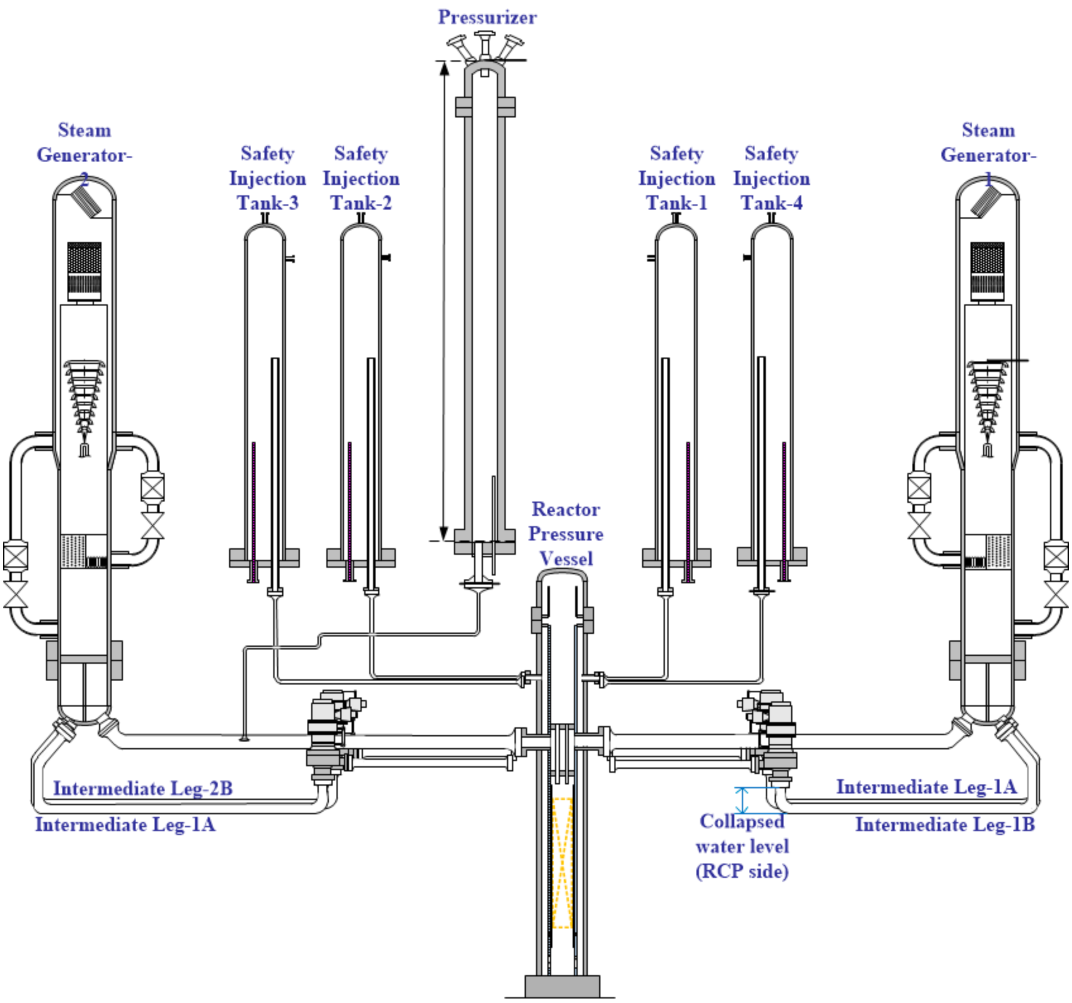

Figure 5. Schematic diagram of ATLAS.

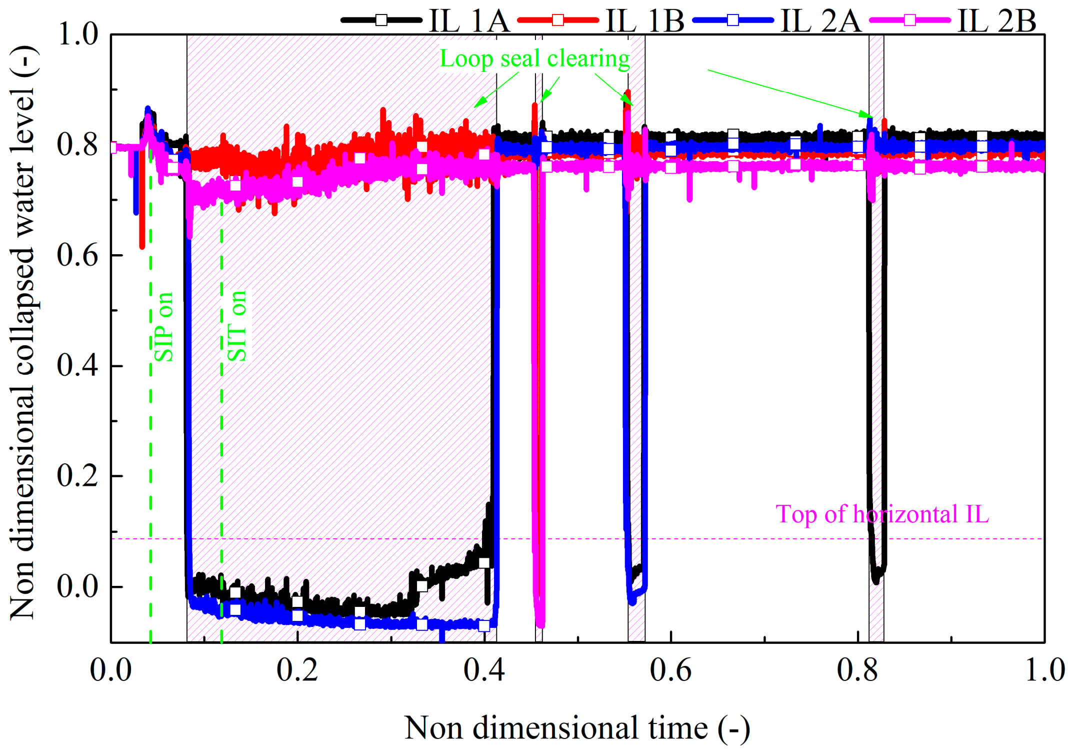

Figure 6. Experimental result—collapsed water level in intermediate leg (RCP side).

The major sequence of events of the experiment is summarized in Table 3. Major experimental results are described in this section, and the detailed results of the experiment are described in reference [15].

Table 3. Sequence of events for LTC-CL-04R Test.

Events	Timing (Non-Dimensional Time)	Remarks
Break start	0.033	MFW terminated
MSSV	0.037/0.038	SG pressure
LPP trip	0.037	Low pressurizer pressure
SIP on	0.042	
SIT on	0.118	Low downcomer pressure
Loop seal clearing	0.081–0.410 (1A) 0.084–0.413 (2A)	
	0.455–0.461 (1A,1B) 0.455–0.462 (2B)	
	0.554–0.571 (1A) 0.553–0.572 (2A)	
	0.814–0.828 (1A)	

MFW: Main Feed Water.

3. Calculation Result and Findings

3.1. Blind and Open Calculation Results and Discussions

The participants of the DSP-04 exercise performed blind and open code calculations for the LTC-CL-04R test data. For the blind calculation phase, the calculations were performed with only the initial and boundary conditions of the experiment. After that, the experimental results were distributed to participants and they modified the calculation input deck and performed sensitivity studies.

3.1.1. Blind Calculation Results

Figures 7–17 show blind calculation results. Some participants did not submit blind phase results and their results are not included in these graphs. The predicted primary system pressures are shown in Figures 7–9. The MARS-KS calculation results were more variable than the results of other codes. This variability indicates that the MARS-KS code is sensitive to users, which is related to the selection of models for a complex scenario that includes the loop seal phenomena. Figures 10–12 show the predicted secondary system pressures. Most calculations had predictions that were high for the secondary system pressures. These high predictions are caused by the heat-loss model of the steam generator. Many participants created an input deck without using the heat-loss model for the secondary system and their codes calculated a larger steam generation at the secondary side of the steam generators. Therefore, the codes predicted higher pressure for the secondary side of the steam generator. In the case of TRACE, however, the secondary pressure was under-predicted. The participant for TRACE applied a heat-loss model for the secondary system and this model shows lower pressure because it predicted higher heat loss than the experiment. The maximum core temperatures are plotted in Figures 13–15. All results included no temperature excursion in the core, as did the experimental results. However, the calculations showed a different core temperature at the end of the test and different core temperature peaks. These differences in the predictions are due to the difference of primary pressure because the core temperature was dependent on the primary pressure, as mentioned in the previous section. Figures 16 and 17 compare the loop seal clearing times of the calculations and experiments. The colored boxes represent the loop seal clearing time. Every code had a different loop seal clearing period and numbers of loop seal clearings/reformations. Therefore, the peaks of the maximum core temperature in Figures 13–15 were also different from the experimental result because the core temperature is affected by loop seal clearing and reformation. Additionally, the plateau times of the primary system pressure were also different because the plateau time of the primary system pressure is dependent on first loop seal clearing, as mentioned in Section 2.3. Figure 18 shows the maximum and minimum difference for pressures and the maximum cladding temperature. MCT, P, pri, SG1, and SG2—mean maximum cladding temperature, pressure, primary, steam generator one, and steam generator two, respectively. Figure 18 is not a representative result for the code prediction ability but one of the supplementary data to understand calculation results, because this experiment is a transient test with complex events, and a conclusion with only simple variables is not reasonable. In Figure 18, each group shows different discrepancies though they use same code. The thermal-hydraulic safety analysis codes have many models for various phenomena and the user selects combinations of them. Therefore, there are many combinations and each combination calculates a different result.

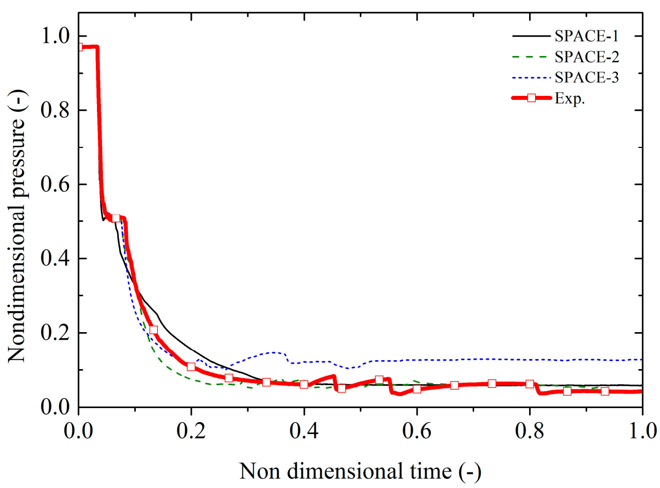

Figure 7. Blind calculation results of the primary pressure (SPACE).

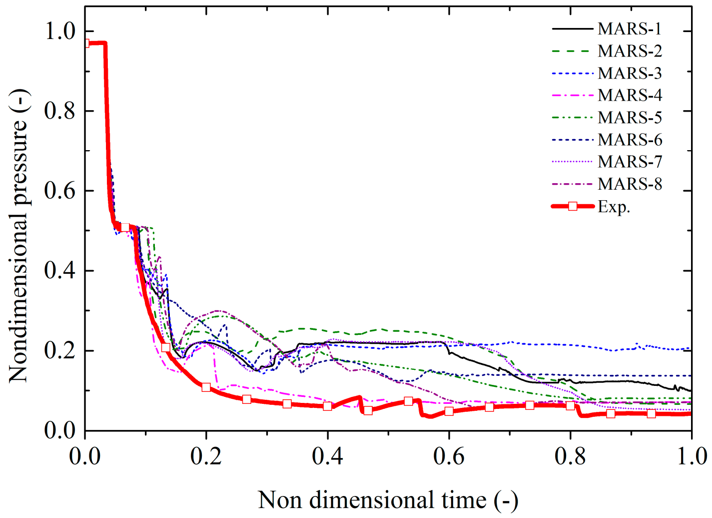

Figure 8. Blind calculation results of the primary pressure (MARS-KS).

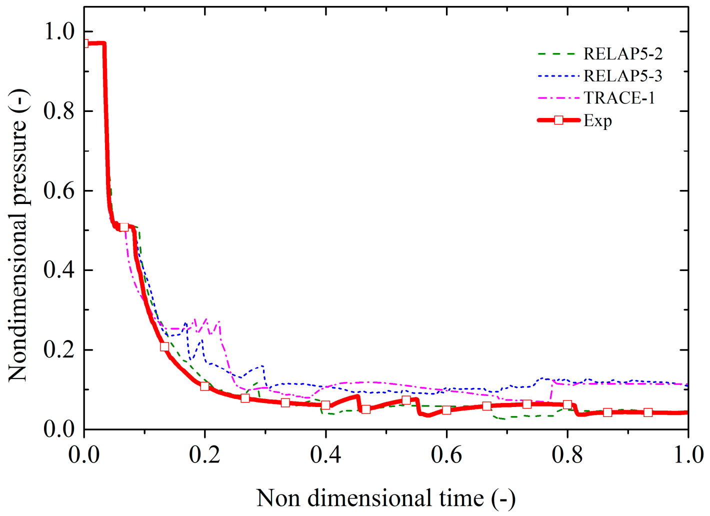

Figure 9. Blind calculation results of the primary pressure (RELAP5 and TRACE).

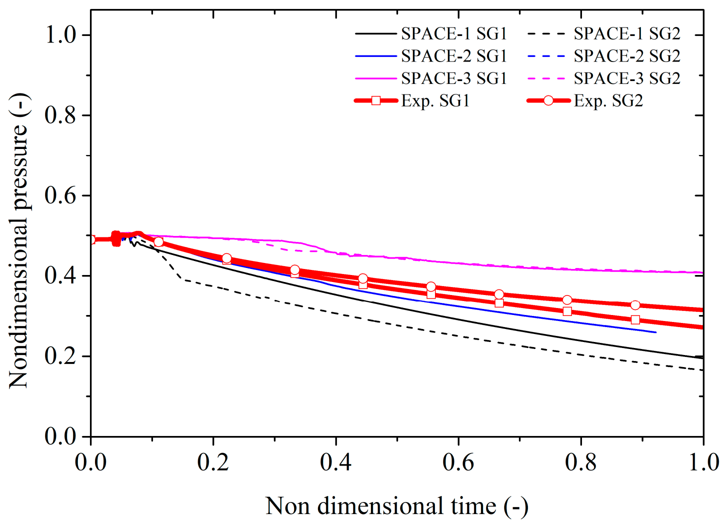

Figure 10. Blind calculation results of the secondary pressure (SPACE).

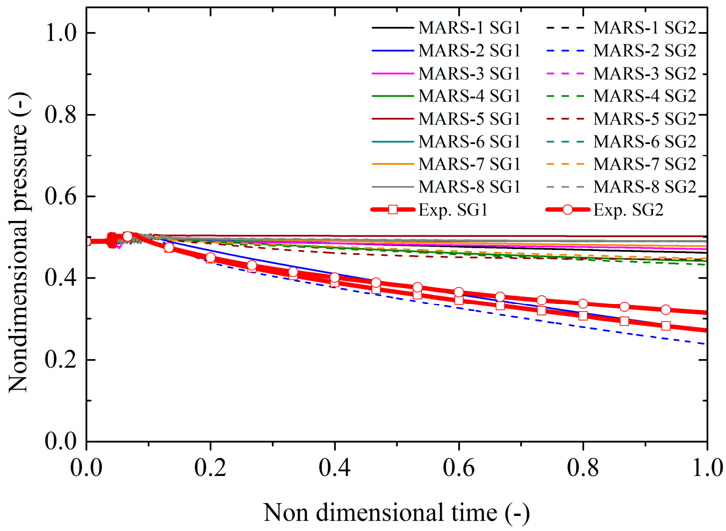

Figure 11. Blind calculation results of the secondary pressure (MARS-KS).

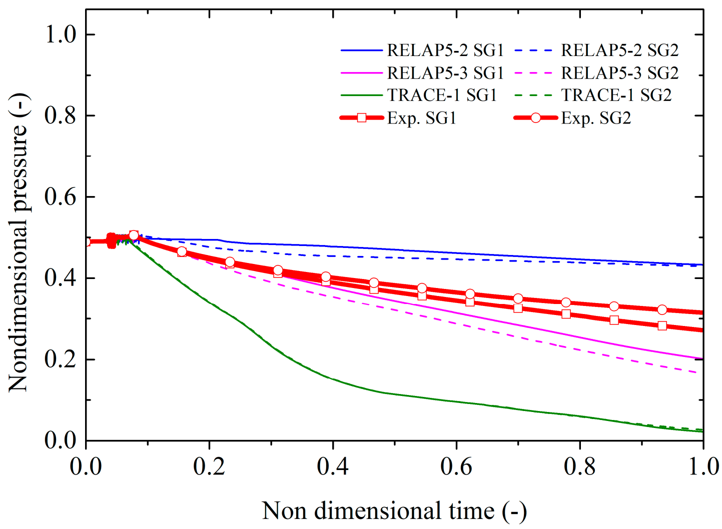

Figure 12. Blind calculation results of the secondary pressure (RELAP5 and TRACE).

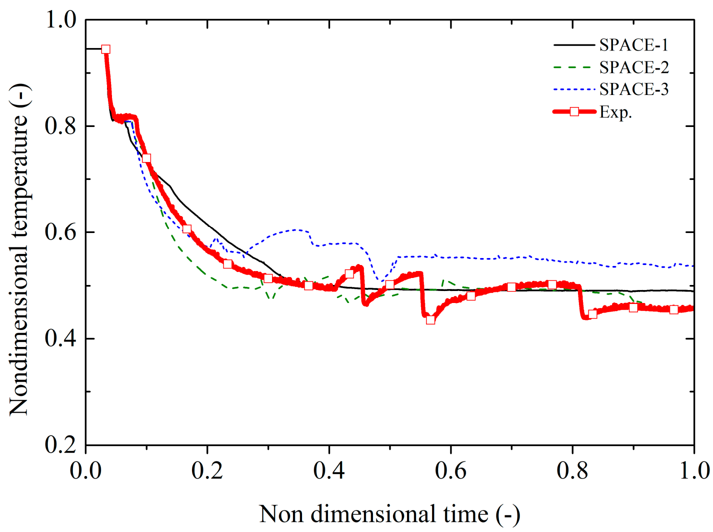

Figure 13. Blind calculation results of the maximum core temperature (SPACE).

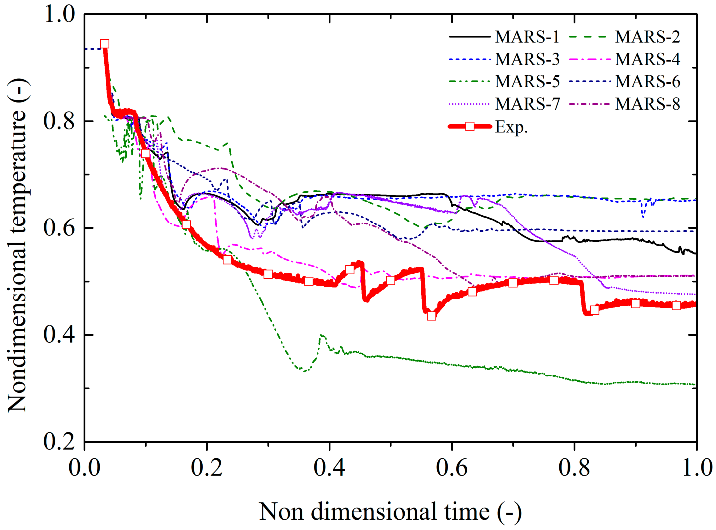

Figure 14. Blind calculation results of the maximum core temperature (MARS-KS).

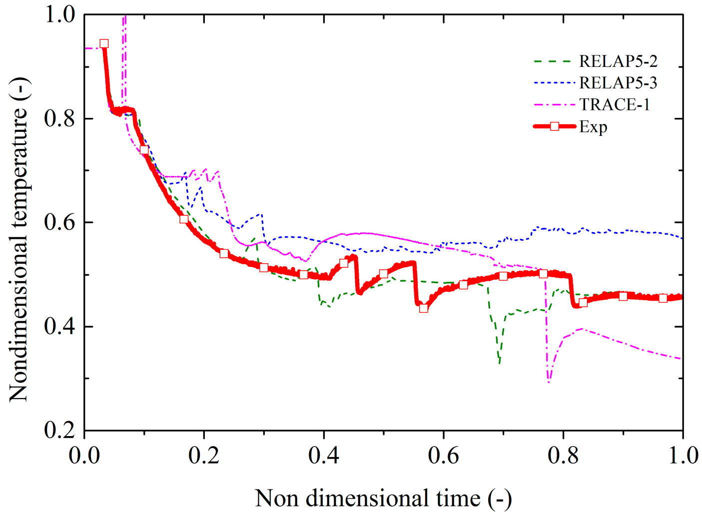

Figure 15. Blind calculation results of the maximum core temperature (RELAP5 and TRACE).

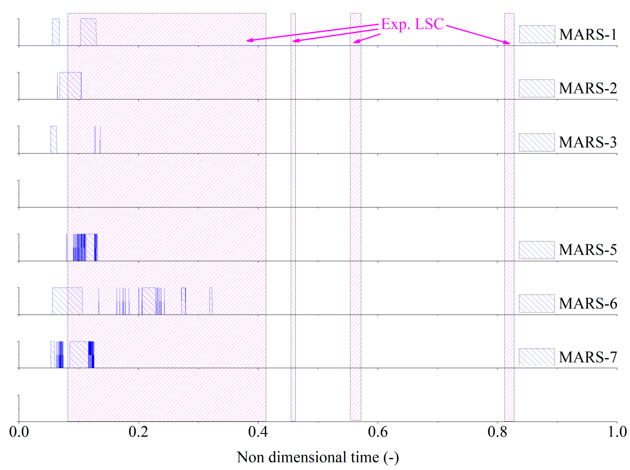

Figure 16. Blind calculation results of loop seal clearing (MARS-KS).

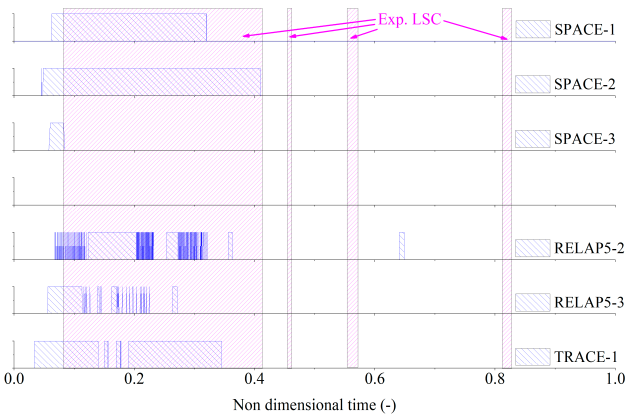

Figure 17. Blind calculation results of loop seal clearing (SPACE, RELAP5, and TRACE).

3.1.2. Open Calculation Results

Figures 19–30 show the open calculation results. The participants performed sensitivity studies of the calculation results so that they could approach the experiment results. They considered several models such as the heat-loss model for the steam generator, the off-take model, critical flow model, fine/coarse nodalization, countercurrent flow limitation (CCFL) model, and detailed break-line modeling. The effects of some of these models on the calculations were analyzed in a previous study [15]. Most open calculations better predicted the experimental results compared to the blind calculations owing to the sensitivity studies. According to the previous analysis and sensitivity studies, the selection of coefficients in the critical flow model, detail break-line modeling, CCFL modeling, and the heat-loss model of the steam generator were important for this scenario. The Ransom–Trapp model for critical flow showed better prediction than the Henry–Fiske models. Detailed break-line modeling simulated realistic flow resistance at the break system and CCFL modeling simulated realistic flow in vertical pipes at the intermediate leg and break line. In the case of the fine nodalization of the loop seal (intermediate leg), some participants reported that the smaller break flow rate during the two-phase break flow condition reduced the oscillation of the loop seal clearing, but the effect was minor for the overall trend of the scenario. Figures 19–21 show the primary system pressures. All the participants had better predictions than the blind phase calculations, showing an improvement in the prediction of the break flow rate. In particular, the MARS-KS group had a significantly reduced difference between the calculation and experimental results compared with the difference of the blind calculation results. Figures 22–24 show the secondary side pressures. Many participants used a heat-loss model on the steam generator for the open calculation. When the heat-loss model was used, the secondary pressures approached the experimental results. From this, we recognized that heat-loss modeling is important during the code analysis of a thermal-hydraulic system. The calculation results of the maximum core temperature are plotted in Figures 25–27. There were still gaps for many participants, though most of their results were better predictions than the blind calculation results. As mentioned in Section 2.3, the core temperature is dependent on the primary pressure and the primary pressure is dependent on loop seal clearing and reformation. The participants tried to match the loop seal clearing time and obtained better results than those of the blind calculations. However, it was still difficult to obtain results that were similar to the experimental results (see Figures 28 and 29). The difficulty of predicting the loop seal clearing was discussed in DSP-04, and it was concluded that the difficulty is due to the lack of understanding and modeling of the loop seal clearing phenomena, such as the initiation and termination conditions of the loop seal clearing. Research has been studied to understand the loop seal clearing phenomena [16], but the knowledge of these phenomena is not enough to understand them. If the understanding of loop seal clearing is improved and more modeling code is developed, a code analysis of loop seal clearing can be improved. Figure 30 shows the maximum and minimum difference for pressures and the maximum cladding temperature.

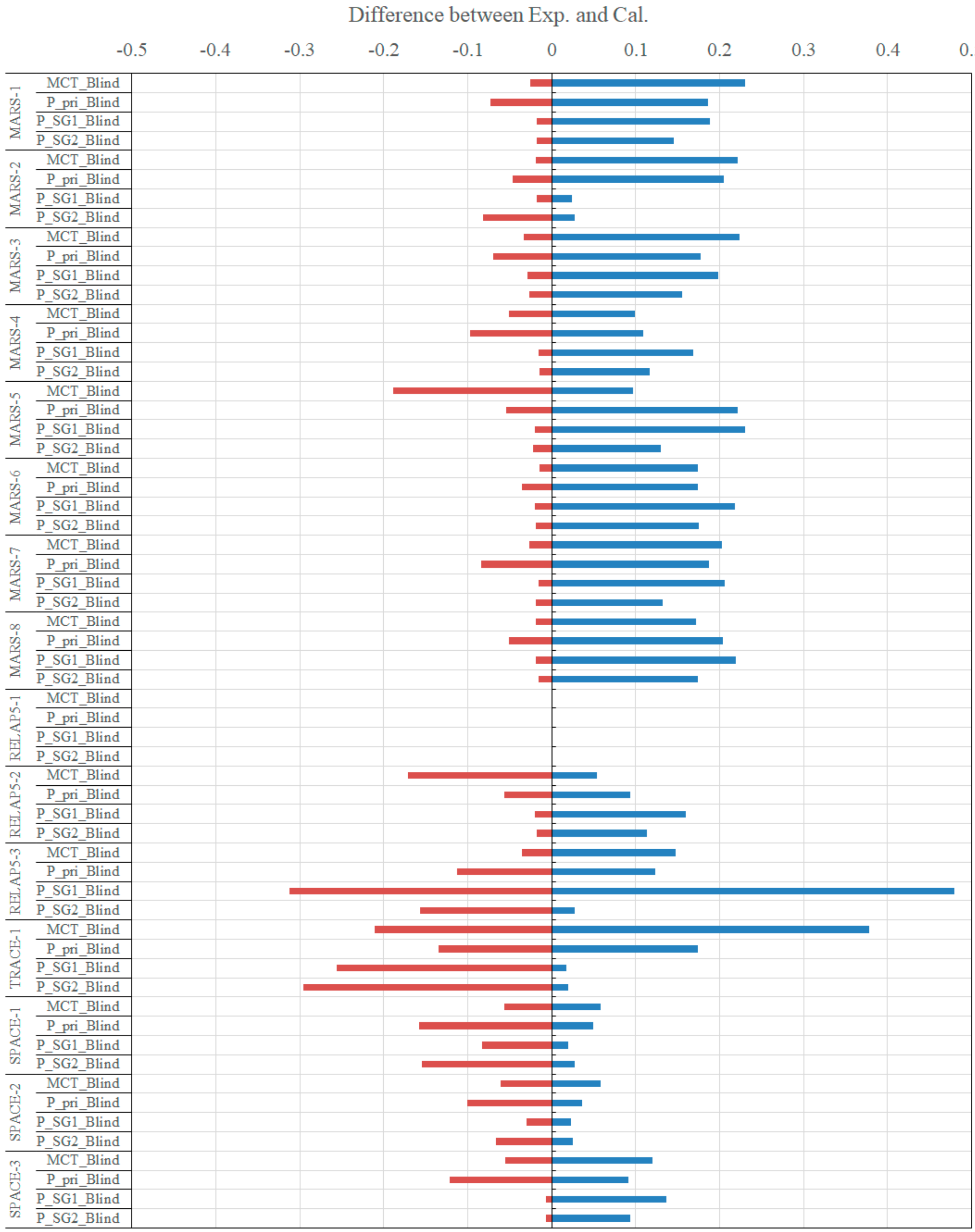

Figure 18. Maximum and minimum differences between experiment and blind calculation results.

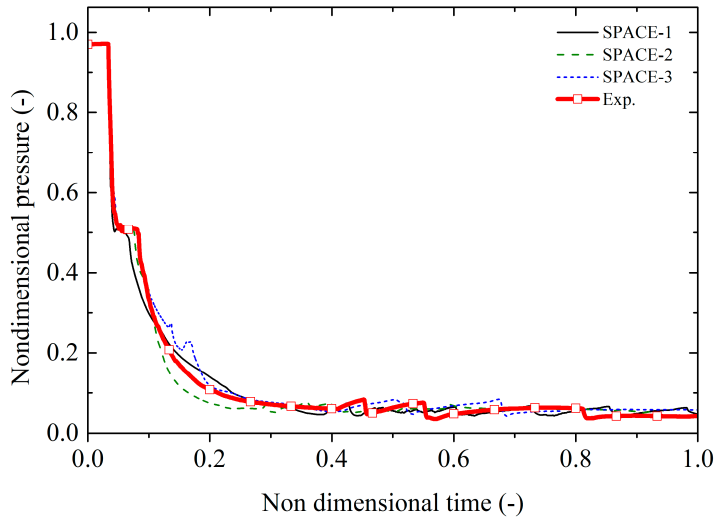

Figure 19. Open calculation results for primary pressure (SPACE).

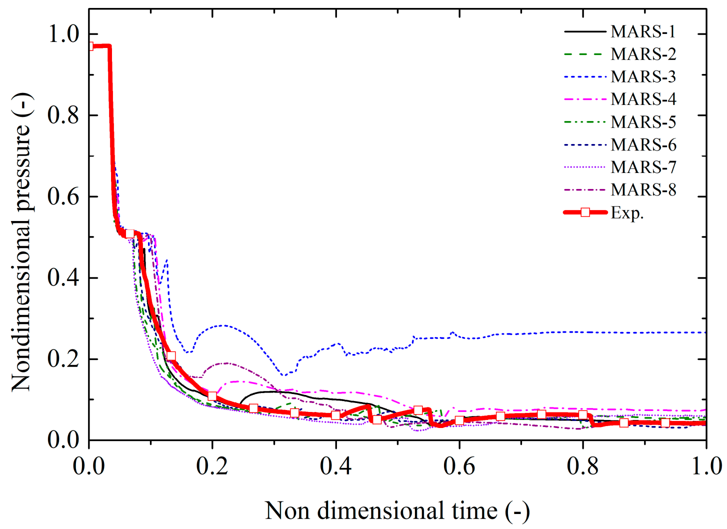

Figure 20. Open calculation results for primary pressure (MARS-KS).

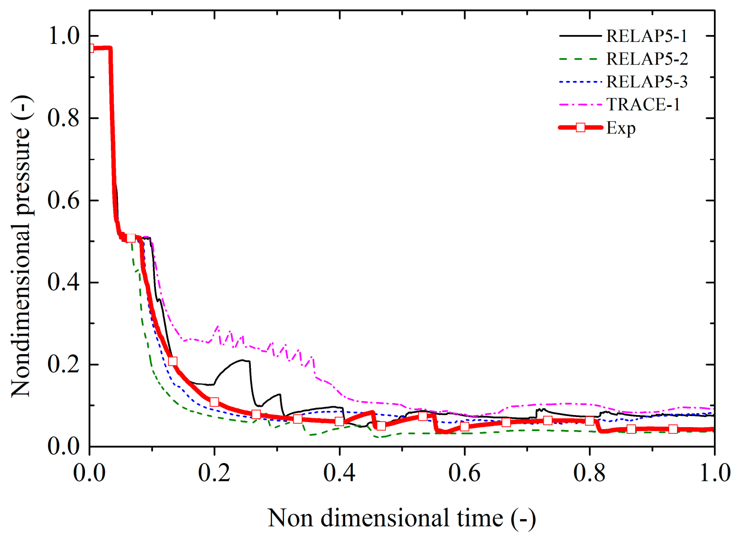

Figure 21. Open calculation results for primary pressure (RELAP5 and TRACE).

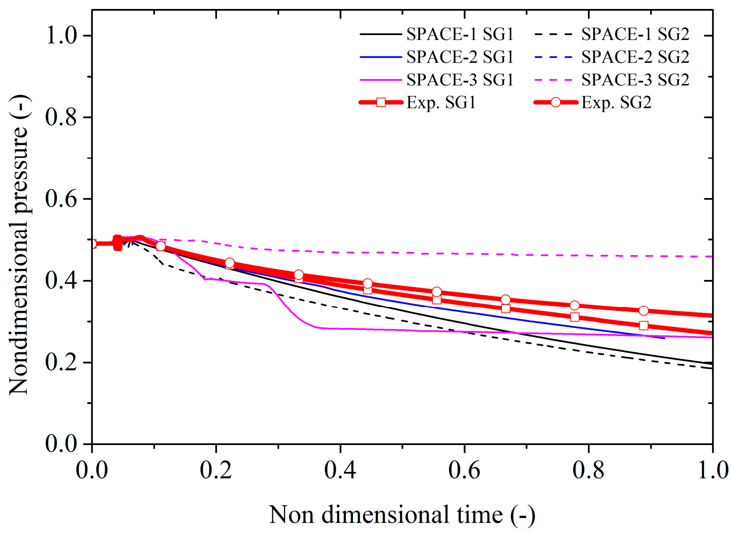

Figure 22. Open calculation results for secondary pressure (SPACE).

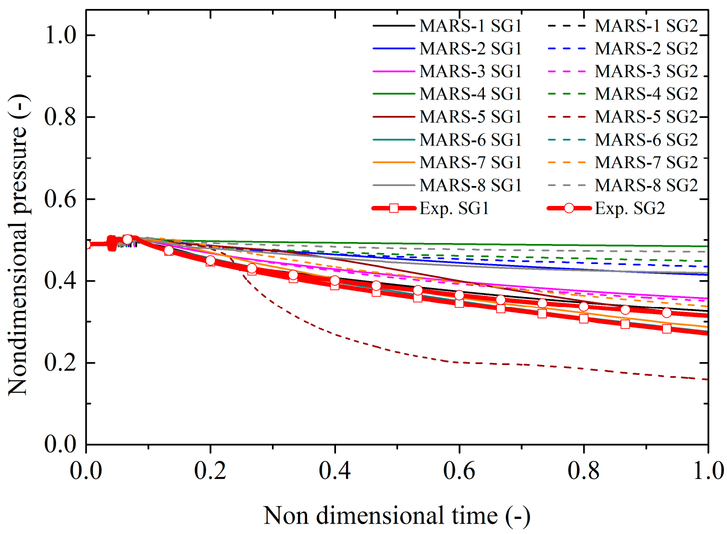

Figure 23. Open calculation results for secondary pressure (MARS-KS).

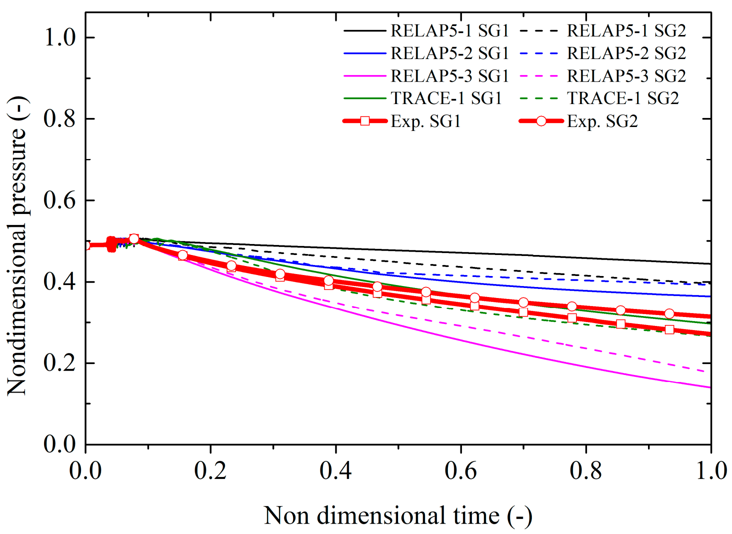

Figure 24. Open calculation results for secondary pressure (RELAP5 and TRACE).

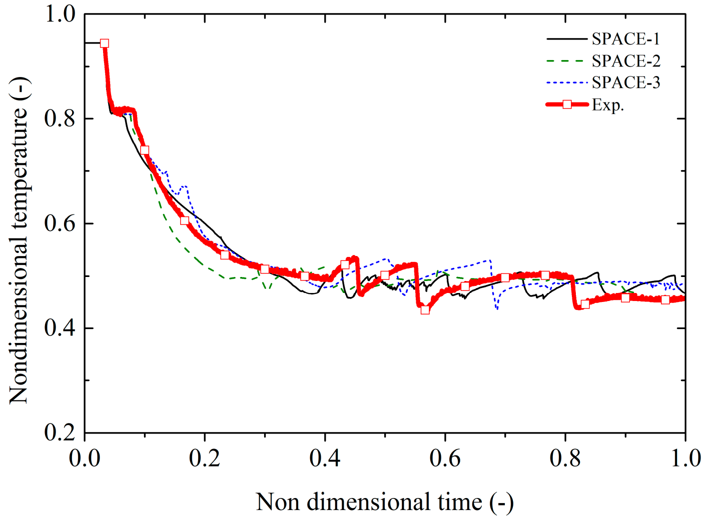

Figure 25. Open calculation results for maximum core temperature (SPACE).

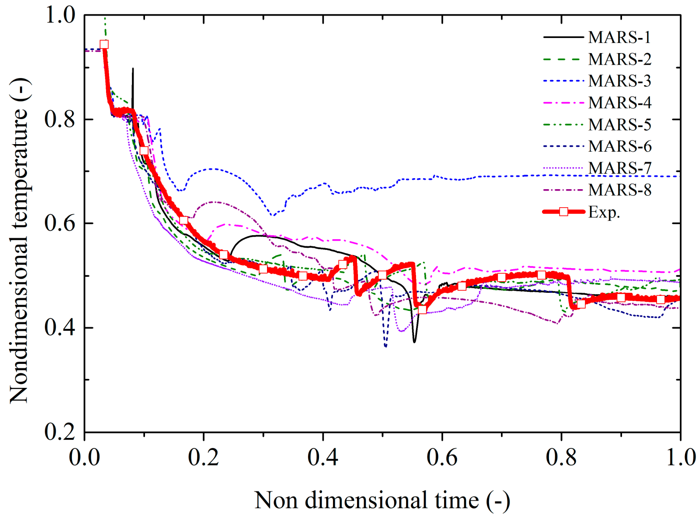

Figure 26. Open calculation results for maximum core temperature (MARS-KS).

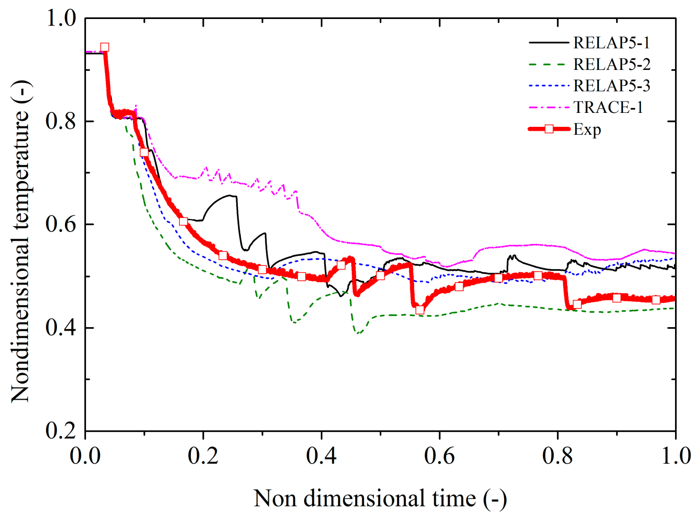

Figure 27. Open calculation results for maximum core temperature (RELAP5 and TRACE).

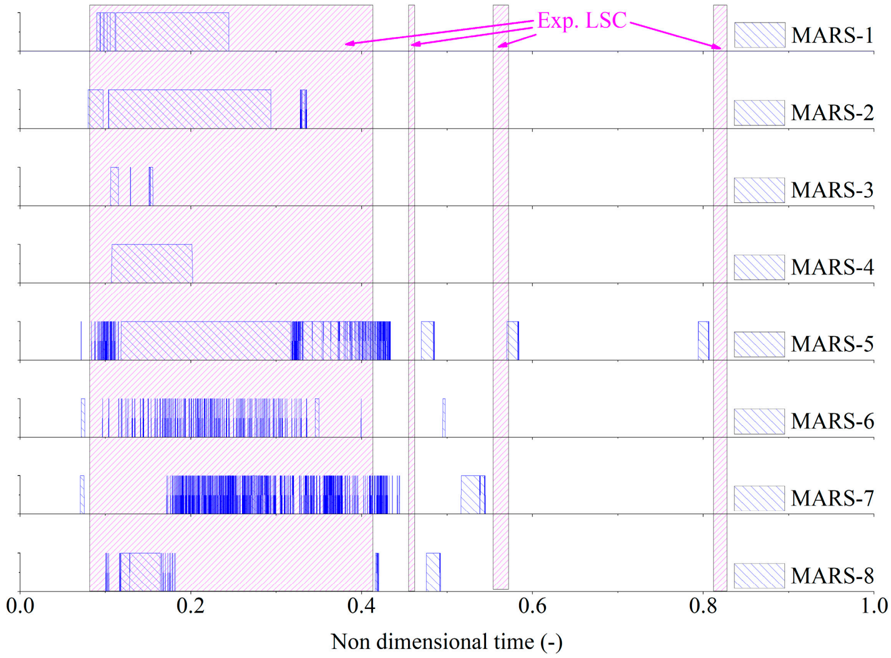

Figure 28. Open calculation results for loop seal clearing (MARS-KS).

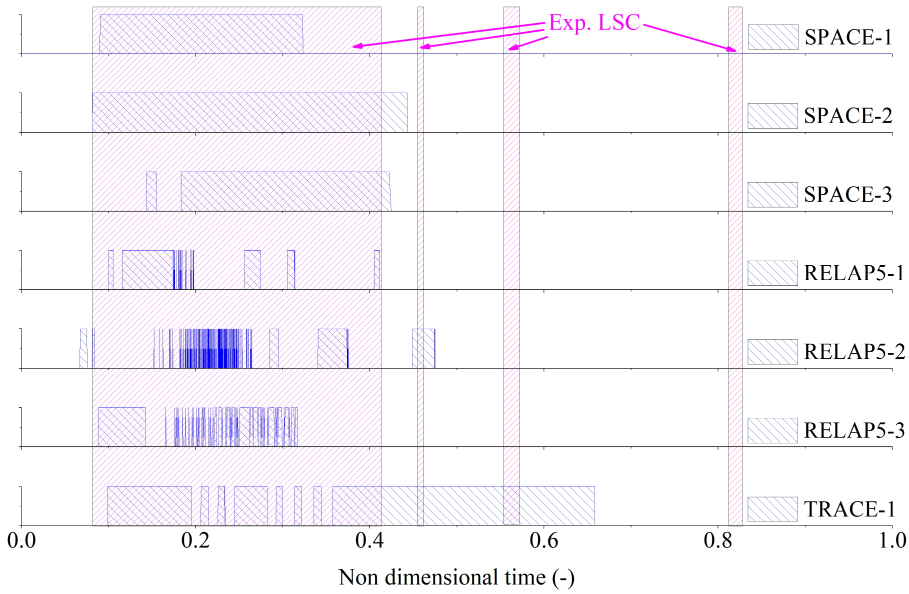

Figure 29. Open calculation results for loop seal clearing (SPACE, RELAP5, and TRACE).

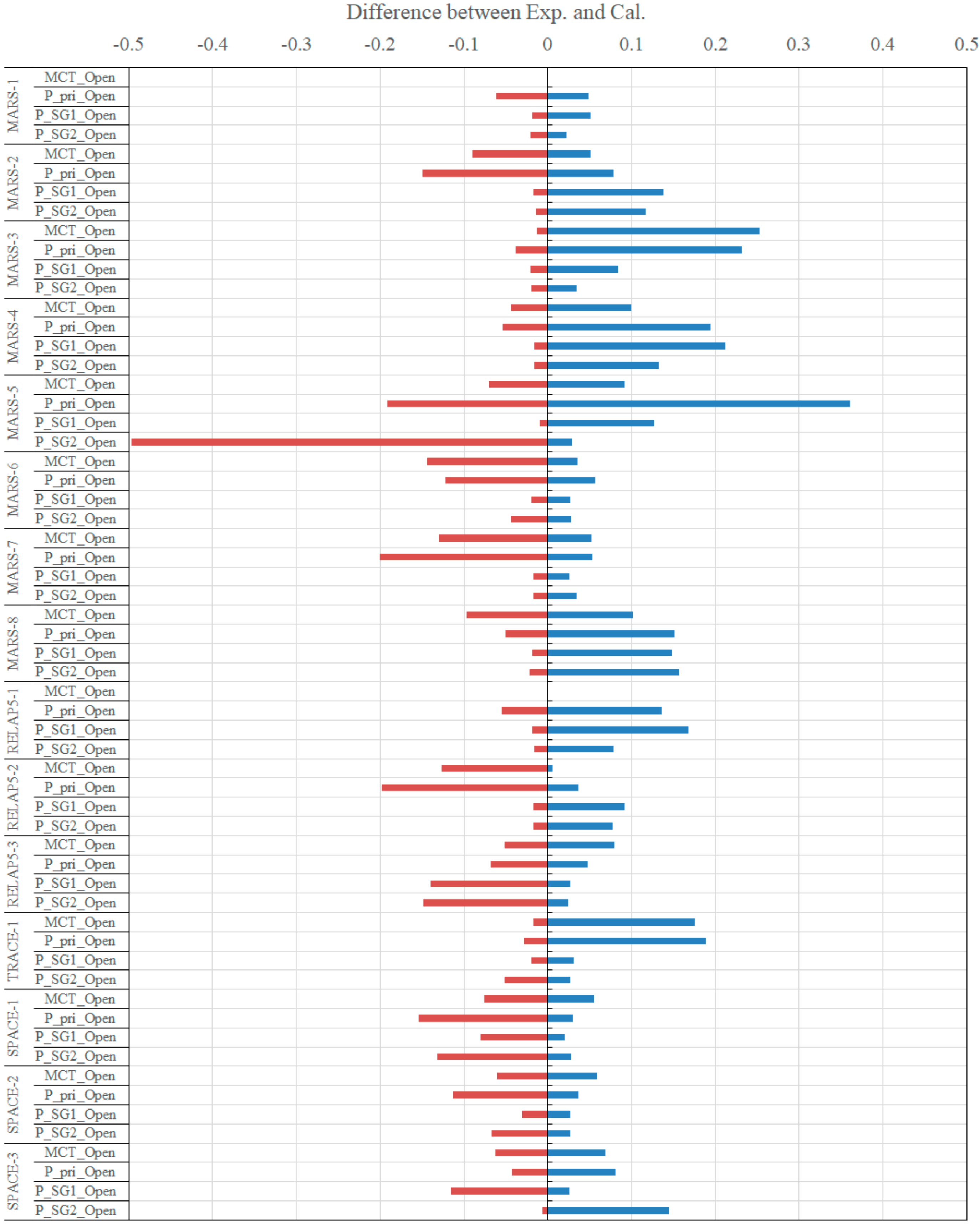

Figure 30. Maximum and minimum differences between experiment and open calculation results.

4. Conclusions

A LOCA experiment with a top-slot break at the cold leg was performed as the activity of DSP-04. One of the aims of this experiment was to investigate the effect of loop seal clearing and reformation on the peak cladding temperature during the cold leg top-slot break LOCA for APR1400. The participants of DSP-04 performed blind and open code calculations for the experiment and sensitivity studies to enhance the code assessment. The open calculation results showed better predictions than the blind calculation results. The participants modified several models to improve their calculation results. For the calculations, the heat-loss model for steam generators, off-take model, critical flow model, fine/coarse nodalization, countercurrent flow limitation model, and detailed break line modeling were considered, and critical calculation options for the models were discussed through sensitivity analysis.

Though the prediction results were improved, it was still difficult to predict the loop seal clearing and reformation, which have a major impact on thermal-hydraulic phenomena in PWR. Hence, limitations in the understanding of the loop seal clearing phenomena and the need to improve the code analysis of this phenomena were recognized. To improve code predictions for loop seal clearing, additional studies to improve the understanding of the loop seal clearing phenomena must be conducted and models for development of models for safety analysis code must be developed.

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