OECD/NEA Main Steam Line Break Benchmark Problem Exercise I Simulation Using the SPACE Code with the Point Kinetics Model

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1. Introduction

The Safety and Performance Analysis Code for Nuclear Power Plants (SPACE) has been developed in recent years by the Korea Nuclear Hydro & Nuclear Power Co. (KHNP) through collaborative works with other Korean nuclear industries [1]. The SPACE is a best-estimated two-phase three-field thermal-hydraulic analysis code to analyze the safety and performance of pressurized water reactors (PWRs). The SPACE code has sufficient features to replace outdated vendor supplied codes and to be used for the safety analysis of operating PWRs and the design of advanced reactors.

As a result of the second phase of the development, the 2.14 version of the code was released through the successive various V&V works. The topical reports on the code and related safety analysis methodologies have been prepared for license works.

In this study, the OECD/NEA Main Steam Line Break (MSLB) Benchmark Problem Exercise I [2] was simulated as a V&V work. The results were compared with those of the participants in the benchmark project.

2. Benchmark Description

The OECD/NEA MSLB benchmark project was established in 1996 [2], and various institutions participated in the project to assess the capability of their own code systems, especially thermal-hydraulics and neutronics coupled codes. The postulated MSLB transient based on the TMI-1 plant was been developed as the benchmark problem. The TMI-1 plant is a 2,772MWt two loop PWR with two vertical once-through steam generator (OTSG) designed by the Babcock & Wilcox Company.

The MSLB accident is characterized by a rupture in one of the main steam lines, leading to an overcooling of the corresponding primary loop. The overcooled moderator inserts a positive reactivity caused by the negative feedback effects. The MSLB is one of the representative asymmetry accidents, so to reflect the overcooling and reactivity inserting to the primary system, the separate loop model between break and intact loops should be considered.

The benchmark problem was separated into three exercises: an integral one-dimensional (1D) plant simulation using the point kinetics model, a three-dimensional (3D) simulation of the core neutronic response to time dependent core thermal-hydraulic boundary conditions over core inlet and outlet, and an integral 1D plant simulation with a 3D kinetics core model using coupled best-estimated codes.

3. Modeling & Simulation

The best-estimated modeling and simulation of the benchmark problem require the 3D kinetics core model to appropriately reflect the space-time variations of the core neutronic behavior or power distribution caused by the asymmetric cooling of the core and the various reactivity effects.

The SPACE code has been expanded to have the capability to simulate the 3D kinetics effects such as the benchmark problem through the coupling to the 3D kinetics code system, such as the RASK-K. As the first stage to verify the capability, the Exercise I of the benchmark problem was simulated using the point kinetics model in the SPACE code.

2.1 SPACE Code Modeling

For the simulation, the TMI-1 was modeled by 199 hydraulic components, 157 cells, 186 faces and 40 heat structures, *etc*. (Fig. 1).

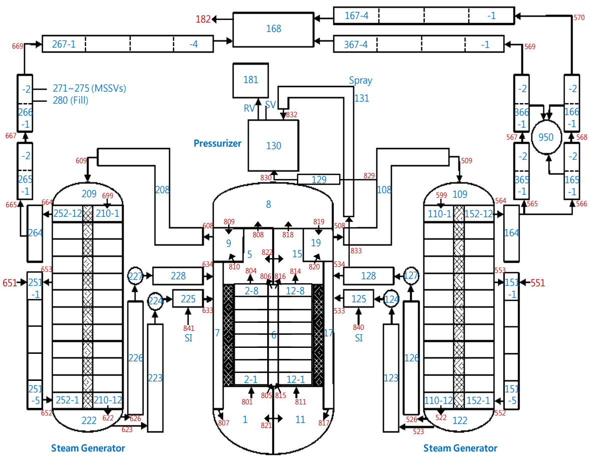

Fig. 1 SPACE Model for MSLB Benchmark Problem

The core was modeled as two vertical channels of 8 sub-cells and corresponding heat structures to represent the core asymmetry during the transient. The flow paths in vessel, such as downcomer, bypass, *etc*., were appropriately set up to represent the flow directions in the core. The cross flows between the core channels were modeled in the lower and upper plena, and the cross flows in the core were not considered.

The four cold-legs were modeled separately with a pumps and a cross-over leg, respectively. The two hot-legs were modeled to connect to the top of the OTSGs, respectively, and the hot-leg in break loop (BL) was connected to the pressurizer (PZR).

The tube or shell sides of an OTSG were split into 12 vertically staked heat structured and cells, respectively. The two steam lines of BL were separately modeled to implement the break scenario, and the steam lines of intact loop (IL) were modeled as one steam line. The feed lines modeled using the downcomer PIPE and feeding TFBC. The 6 main steam safety valves were modeled in the IL steam line using TFBC models.

The breaks were modeled as the abrupt opening of the valves connected to cell 165, 166 and 366.

2.2 MSLB Simulation

To simulate the problem, the steady-state was pre-calculated to confirm the initial conditions and the transient was started using the restart feature of the SPACE code. The results calculated by the SPACE code were compared with those presented in the problem [2].

Table I: Initial & Boundary Conditions

Parameters	Measured	Calculated
Core power, MW	2,772	2,772
PZR pressure, MPa	14.96	14.96
Cold-leg Temperature, K	563.76	565.72
Hot-leg temperature, K	591.43	593.87
Core flow, kg/sec	16,052.4	15,929.2
Total RCS flow, kg/sec	17,602.2	17,600.1
Feedwater per OTSG, kg/sec	761.59	761.10
OTSG outlet pressure, MPa	6.41	6.28
OTSG outlet temp., K	572.63	579.812
Delayed N. fraction (β_{eff})	0.005211	0.005211
Prompt N. lifetime	1.8445e-4	1.8445e-4
HFP EOC MTC, pcm/K	-62.35	-62.35
HFP EOC DTC, pcm/K	-2.57	-2.57
Total SCRAM worth, %dk/k	-4.526	-4.526

The results of the transient simulation were compared with those mentioned in the Phase I summary report [3].

The occurrence of three main steam line breaks was caused to the discharging of main steam in BL (Fig. 2) and the decrease of OTSG pressures (Fig. 3). To simulate the break, the Ransom-Trap critical flow model, the default model in the SPACE code, was used using *Cd* = 1.0 for all fields. In the case of Henry-Fauske critical flow model, the break flow showed similar trends to the default model when the *Cd* was 0.85.

As depicted in the figure, the break flow was dropped rapidly following the closure of main steam isolation valves (MSIVs) by the vanishing of steam in the lines, and increased by the steam efflux from the BL OTSG. Following the closure of MSIVs, the decrease of IL OTSG pressure was stopped and rebounded to increase while the decrease of BL OTSG pressure was continued.

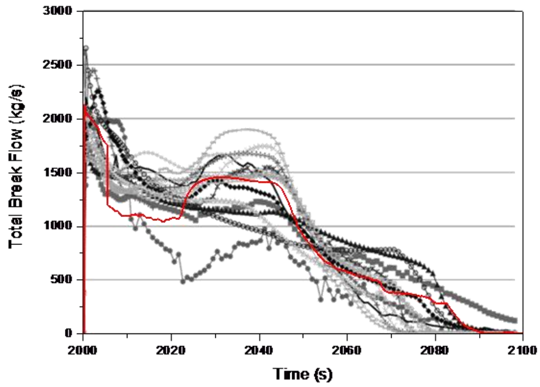

Fig. 2 Total Break Flow

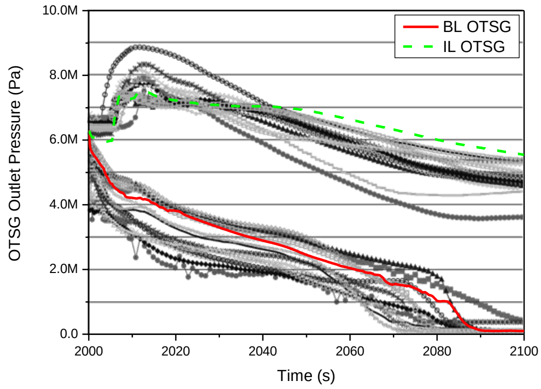

Fig. 3 OTSG Outlet Pressure

The primary loops were overcooled by the increased steam flows in the secondary sides caused by the breaks (Fig. 4 & 5). The pressure decrease in both OTSGs showed similar trends by the MSIV closure. The decrease of BL temperatures showed more rapid trends after the closure than those of IL. The pressurizer pressure was also decreased according to the coolant temperature (Fig. 6).

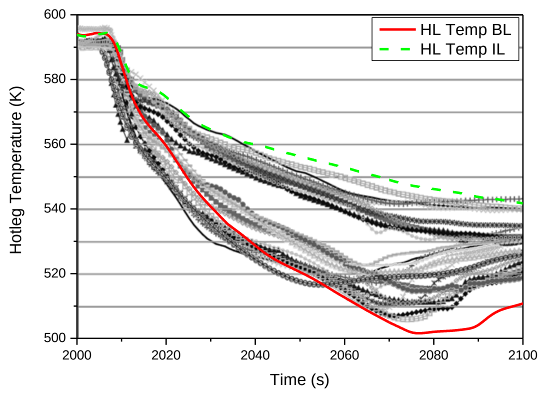

Fig. 4 Hotleg Temperature

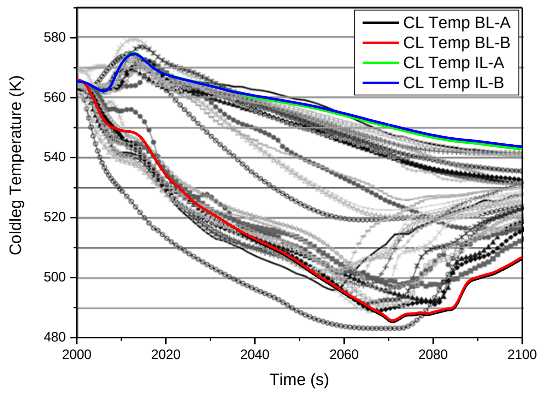

Fig. 5 Coldleg Temperature

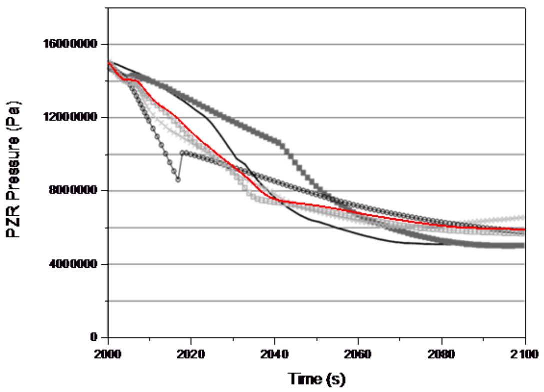

Fig. 6 Pressurizer Pressure

The reactivity was rapidly decreased by the SCRAM at about 6 seconds caused by the high power signal (114% to the rated power) (Fig. 7). The overcooled coolant was caused the increase of reactivity by the negative moderator temperature and Doppler coefficients. The total power was varied with the reactivity (Fig. 9). The rapidly decreased power was kept around 10% caused by the decreased decay heat and increased reactivity. After about 60 seconds, the power was gradually increased to about 18% (500MW).

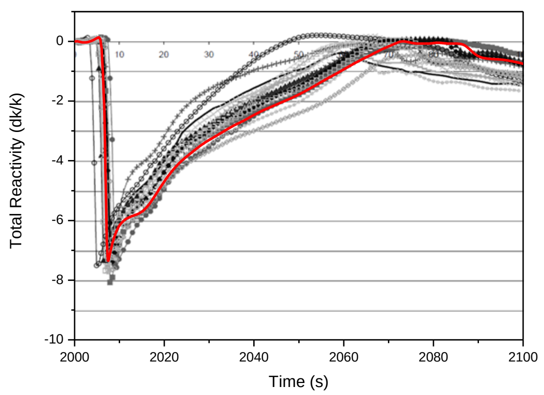

Fig. 7 Total Reactivity

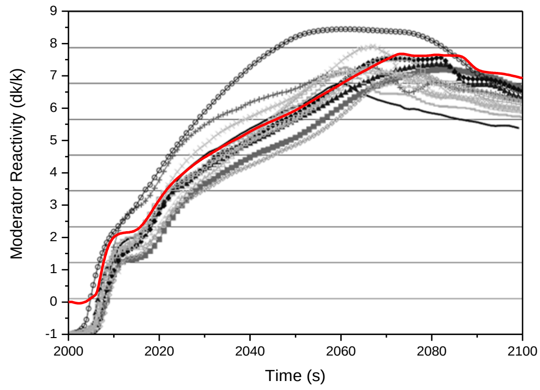

Fig. 8 Moderator Reactivity

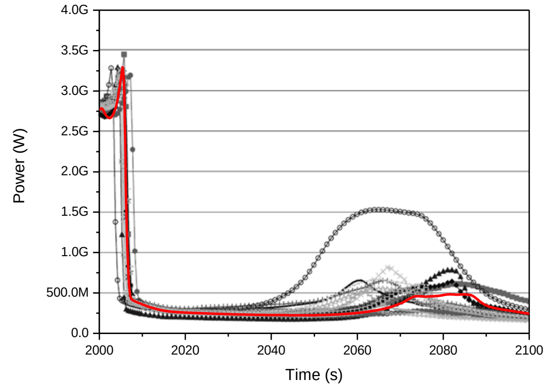

Fig. 9 Total Power

Through the result comparison to those mentioned in the summary report prepared by various participants to the benchmark project, it was concluded that the SPACE code can effectively simulate benchmark problem except some trends. The heat transfer balance would be counted as one of the most effective reasons to the exceptions. As mentioned in Tab. 1, the shell side of OTSG was more superheated than the specifications as suggested [2], which could lead to less inventories in the steam line. To exclude this dissonance, the heat loss could be considered to remove excess heat from the primary loop in the next step study.

4. Conclusions

The OECD/NEA MSLB Benchmark Problem Exercise I was simulated using the SPACE code. The results were compared with those of the participants in the benchmark project. Through the simulation, it was concluded that the SPACE code can effectively simulate PWR MSLB accidents.

REFERENCES

[1] S. J. Ha, *et al*., Development of the SPACE Code for Nuclear Power Plants, Nuclear Engineering and Technology, Vol.43(1), p.45, 2011.
[2] Ivanov, *et al*., Pressurized Water Reactor Main Steam Line Break (MSLB) Benchmark: Vol. I Final Specifications, NEA/NSC/DOC(99)8, OECD/NEA, 1999.
[3] Ivanov, *et al*., Pressurized Water Reactor Main Steam Line Break (MSLB) Benchmark: Vol. II Summary Results of Phase I (Point Kinetics), NEA/NSC/DOC(2000)21, OECD/NEA, 2000.