Application of SPACE code for the OPR1000 SLB analysis

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1. Introduction

The Safety and Performance Analysis CodE (SPACE) has been developed and finally approved in 2017 for the Advanced Power Reactor 1400 (APR1400) type Nuclear Power Plant (NPP) Non-Loss of Coolant Accident (LOCA) [1] safety analysis. Considering similarity of the APR1400 and the OPR1000 design, it is expected that SPACE code could be applied to the OPR1000 Non-LOCA analyses. To make sure the code's applicability and to support the licensing of the methodology, many efforts have been made, such as comparing the analysis results with those from the current licensing Non-LOCA system code of Combustion Engineering Simulation Exclusion Code-III (CESEC-III) [2].

This paper shows the analysis of the Steam Line Break (SLB) without loss of off-site power among the Non-LOCA accidents of the Optimized Power Reactor 1000 (OPR1000) type NPP.

2. Event Analysis

2.1 Identification of event and causes

A steam line break is defined as a pipe break in the main steam system, which results in excessive Reactor Cooldown System (RCS) cooldown and causes the core reactivity. Degradation in fuel cladding performance may result from this accident. The largest possible size of SLB is the double ended rupture of a steam line upstream of the Main Steam Isolation Valve (MSIV). An integral flow restrictor exists in each steam generator (SG) outlet nozzle. The largest effective steam blowdown area for each steam line, which is limited by the flow restrictor throat area.

2.2 Sequence of the events

The summarized sequence of accident is as follows:

- SLB accident occurs
- RCS temperature and pressure decrease
- Positive reactivity insertion by feedback effects
- Increase in core power
- Reactor trip due to high power or low SG pressure
- Departure from Nucleate Boiling Ratio (DNBR) decreases
- Operator action at 1,800 seconds after event initiation

2.3 Assumptions and initial conditions

The major initial condition parameters are core power, core inlet temperature, core flow rate, pressurizer pressure, pressurizer level, and steam generator level. The initial condition parameters each are selected to maximize the post-trip degradation in fuel performance. Maximizing the core power and core inlet temperature and minimizing the core flow have increased the RCS average temperature and outlet temperature affecting the post trip Return To Power (RTP) adversely. Maximizing initial RCS average temperature maximizes the rate of cooldown since it maximizes SG pressure.

Use of the most negative moderator and Doppler coefficients maximizes the reactivity feedbacks obtained during the cooldown, thereby maximizing the possibility of post-trip RTP.

Assumptions and initial conditions used in SPACE analysis and CESEC-III analysis are the same as summarized in Table I.

Table I: Assumptions and initial conditions for SLB

Parameters	CESEC-III	SPACE
Core Power	Max.	Same as left
Core Inlet Temperature	Max.	
Core Flow Rate	Min.	
Pressurizer Pressure	Max.	
Pressurizer Level	Max.	
SG Level	Max.	
MTC	Most Negetive	
FTC	Most Negative	
Break Size	Double-ended	
LOOP	Not assumed	

3. Analysis Results

Fig. 1 shows the comparison of core power. Since it is assumed that the reactor is tripped at the same time as the accident occurs, the core power decreases as soon as the accident occurs. As shown in Fig. 1, the core power behavior of the SPACE and CESEC-III is similar.

Fig. 2 shows the comparison of break flow. When the SLB accident occurs, the steam is released through the breaking part, and the Main Steam Isolation Signal (MSIS) is generated by the low SG pressure, and the MSIV is isolated. As shown in Fig. 2, the break flow behavior of the SPACE and CESEC-III is similar. Fig. 3 shows the comparison of RCS pressure. SLB is characterized as cooldown events due to the increased steam flow rate, which cause excessive energy removal from the SGs and the RCS. This results in a decrease in temperature and pressure in RCS and SG pressure. As shown in Fig. 3, the RCS pressure behaviors of the SPACE and CESEC-III are similar.

Fig. 4 shows the comparison of reactivities. Even after reactor trips, the positive reactivity is inserting by the moderator and Doppler feedback, and when SI is injected by the low pressurizer pressure, the negative reactivity begins to insert due to the boron acid. As shown in Fig. 4, the reactivity behaviors of the SPACE and CESEC-III are similar.

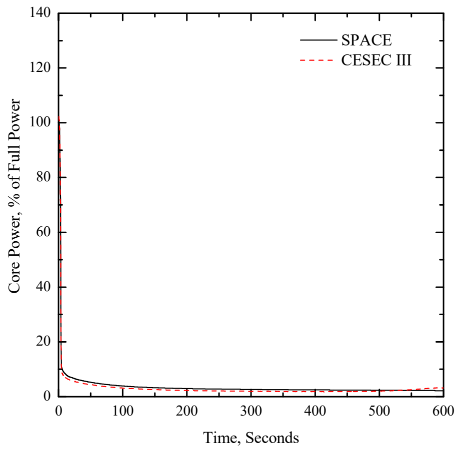

Fig. 1. Simulation result for core power

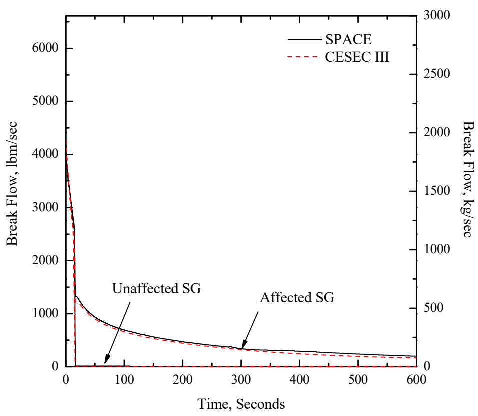

Fig. 2. Simulation result for break flow

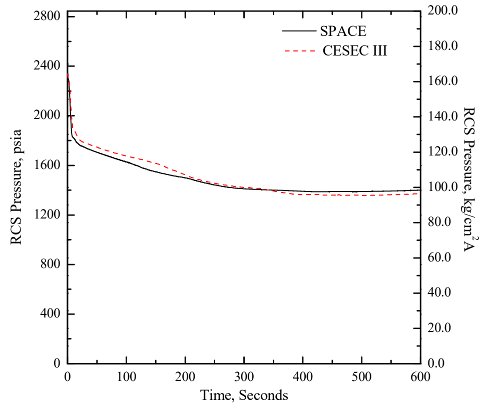

Fig. 3. Simulation result for RCS pressure

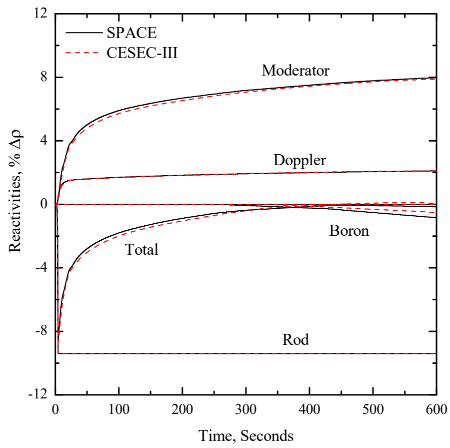

Fig. 4. Simulation result for reactivities

4. Conclusions

Non-LOCA methodology has been successfully developed for the OPR1000 using SPACE code.

This paper shows the comparison of the results between CESEC-III and SPACE for the SLB event as an example. As expected, good agreement between them has been found and it is concluded that SPACE code is applicable to the OPR1000 type NPP.

REFERENCES

[1] KHNP, "Non-LOCA Safety Analysis Methodology for Typical APR1400 with the SPACE Code," TR-KHNP-0029, March 2017.

[2] S. Y. Ro, Analysis on a Letdown Line Break Event for OPR1000 Plant Using SPACE. Korean Nuclear Society Autumn Meeting, 2021.