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Simulation Study of Total Loss of Feedwater in OPR1000 using SPACE

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1. Introduction

After Fukushima nuclear accident, simulation of beyond design conditions are important for developing the Reactor Cooling System (RCS) cool down strategy and recovery action. Additional failure of the safety components are also considered in terms of sufficient safety margin with application of proper emergency operating procedures [1].

In this paper, we studied about Total Loss of Feedwater (TLOFW) accident which is used to represent an accident involving the failure of cooling by the secondary cooling system.

This work is focused on the thermal-hydraulic analysis for TLOFW accident in OPR1000 plant using Safety and Performance Analysis Code for Nuclear Power Plant (SPACE) [2]. Based on the thermal hydraulic features, simulation studies can provide physical insight to the system response and confirm the effectiveness of RCS cool down strategy according to the emergency operating guideline.

2. Code Description

The Korea Hydro & Nuclear Power Co. through collaborative works with other Korean nuclear industries and research institutes developed a thermal hydraulic analysis code for the safety analysis of Pressurized-Water Reactors (PWRs) which is named SPACE. The SPACE is a best-estimate two-fluid, three-field analysis code used to analyze the safety and performance of PWR. The version used in this study is SPACE 3.0.

3. Modeling Information

Figure 1 shows the system nodalization of SPACE code for simulation of TLOFW accident. To simulate the TLOFW accident, the steady-state was calculated to confirm the initial conditions and the transient was started using the restart function of the SPACE code. The values of steady-state were represented in Table 1. The calculation for steady-state condition is performed for 3,000 seconds.

When the pressure of Pressurizer (PZR) is higher than 17.3 MPa, Pressurizer Safety Valve (PSV), which is modeled as component C511, is opened. The safety depressurization system (SDS) was designed to prevent severe accident through feed and bleed operation when the TLOFW accident occur. The SDS is modeled using the components C551 and C552. The multiple failure accident which contain the TLOFW accident can be analyzed using best estimation methods. In order to obtain best optimized conditions, Steam Bypass Control System (SBCS) was used. The SBCS is modeled into eight separate lines, C811 ~ C818.

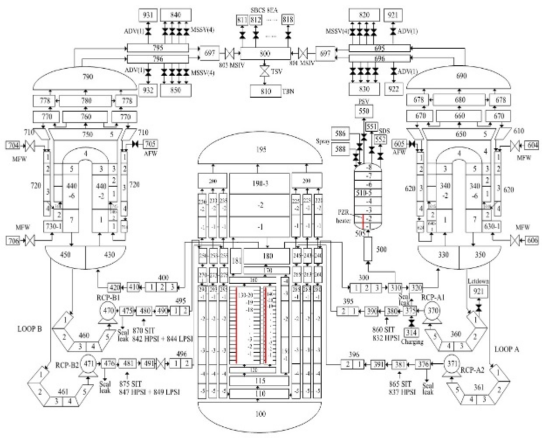

Fig. 1 Nodalization diagram of OPR1000.

Table I: Initial conditions for the TLOFW

Parameters	Design value	Steady state value
Core power (MWt)	2815	2815
Cold-leg Temperature (℃)	295.8	298
Hot-leg Temperature (℃)	327.2	329
RCS flow	15308.7	15336
PZR pressure (MPa)	15.5132	15.5
PZR level (%)	52.6	52.6
Steam Generator Pressure (MPa)	7.5429	7.389
Steam flow rate (kg/s)	802.9	801.5
Feedwater flow rate (kg/s)	802.6	798

4. Simulation Result

4.1 Sequence of events

The simulation scenario of the TLOFW with additional failures considered two different cases, and the descriptions described in final safety analysis report (FSAR) are as follows in [3]. Case 1 is assumed that 2 out of 4 safety injection pump (SIP) and 1 out of 2 SDS are available to simulate additional failures during entire test period and SDS can be used immediately after PSV open. Case 2 is modeled all SIPs and SDS are fully available but the SDS will be opened 30 minutes after PSV open. The safety injection tank (SIT) and the SBCS are assumed to be fully available in both cases. The supply of auxiliary feedwater is unavailable under the TLOFW event. The sequence of events are summarized as shown in the Table II.

Table II: Sequence of event

Flow	Event	Remark
1	TLOFW occur	
2	SG low level	WR, 42.9%
3	TBN trip	
4	Secondary feedwater injection fail	WR, 23.4%
5	RCP trip	manually
6	SG dry out	
7	PSV open	P_{PZR} > 17.23MPa
8	SDS valve open (manually)	Case 1: PSV open + 0s Case 2: PSV open + 1800s
9	LPP signal	
10	HPSI injection	
11	SIT injection	

4.2 Result: Case 1

Figure 2 shows the pressures of the primary and secondary system. When the TLOFW accident starts, the reactor trip and turbine trip are occurred by the low steam generator level. After the turbine trip, the secondary system pressure increases until SBCS is actuated. As the heat removal capacity of steam generator decrease, the pressure of primary system increase to the opening set point of PSV. After the manual opening the SDS for the feed and bleed operation, the pressure of the primary system decreases. As shown in Fig. 3, the PSV is closed immediately with one of the SDS opened.

The pressure of the primary system decreases as the primary coolant inventory was discharged through the SDS. The SIP is actuated by the low pressurizer pressure signal. When the primary pressure decreases, the SIT is injected in to the RCS. Figure 4 shows the mass flow rates of SIP and SIT.

The expected maximum fuel cladding temperature, acceptance criteria shall not exceed 1,204 ℃ (1,477 K). In Fig. 5, peak cladding temperatures of case 1 is 325 ℃ at 1,130 seconds. It is confirmed that this temperature is sufficiently lower than acceptance criteria. The core cooling capability is sufficient.

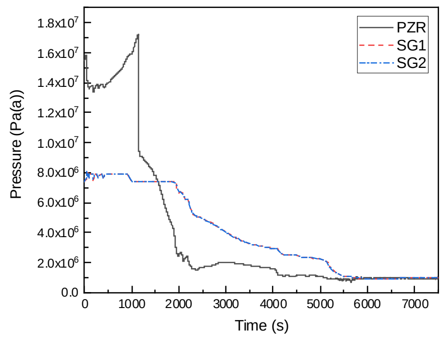

Fig. 2. Pressures of the primary and secondary system (Case 1).

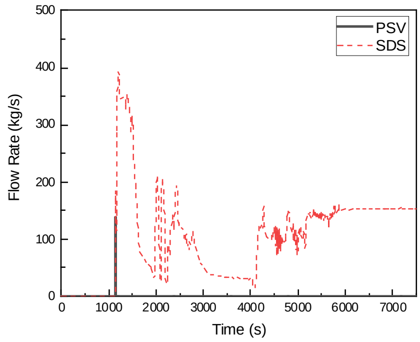

Fig. 3. Mass flow rate of PSV and SDS (Case 1).

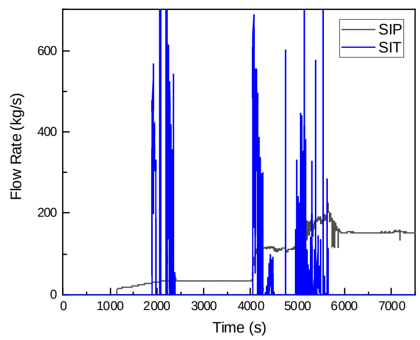

Fig. 4. Mass flow rate of SIP and SIT (Case 1).

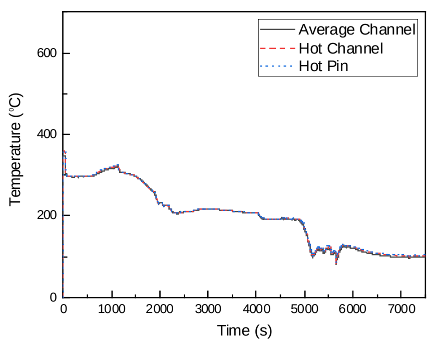

Fig. 5. Peak cladding temperature (Case 1).

4.3 Result: Case 2

Figure 6 shows the pressures of the primary and secondary system in case 2. Until the SDS is opened, the pressure of primary system is increased and decreased according to the PSV setting. As shown in Fig. 7, the PSV is operated consistently open and close. 30 minutes after the first PSV opening, the SDS were opened manually. At that time, the pressure of the primary and secondary system decreases as the primary coolant inventory was discharged through the SDS. The SIP is actuated by the low pressurizer pressure signal. When the primary pressure decreases, the SIT is injected in to the RCS. Figure 8 shows the mass flow rates of SIP and SIT. In Fig. 9, peak cladding temperatures of case 2 is 350 ℃ at 2,930 seconds. It is also confirmed that the temperature is sufficiently lower than acceptance criteria. The core cooling capability is sufficient.

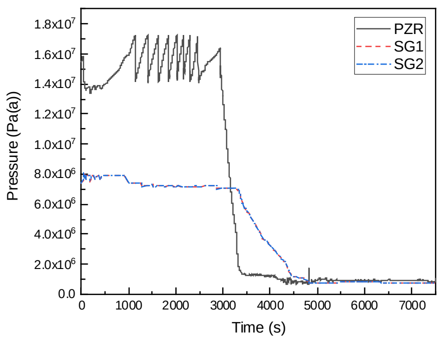

Fig. 6. Pressures of the primary and secondary system (Case 2).

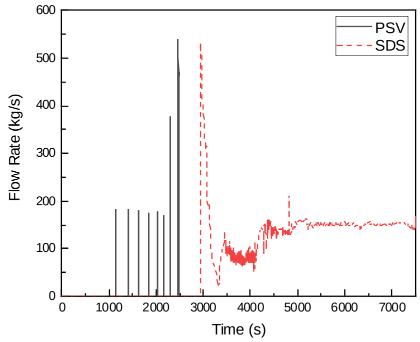

Fig. 7. Mass flow rate of PSV and SDS (Case 2).

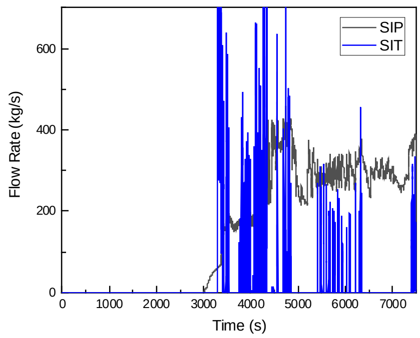

Fig. 8. Mass flow rate of SIP and SIT (Case 2).

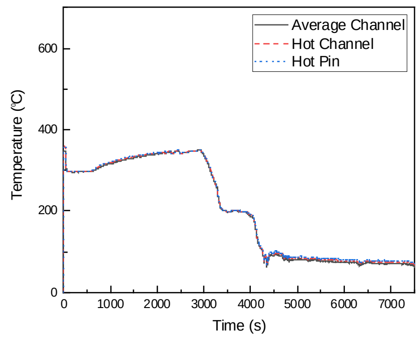

Fig. 9. Peak cladding temperature (Case 2).

4. Conclusions

In this study, the TLOFW accident in OPR1000 plant were assessed using SPACE 3.0. Two different cases were simulated to confirm safety criteria. In both cases, it is verified that the core cooling capability is sufficient. In case 2, it is expected that fuel rod is heated up in a short time. If the TLOFW accident occur, it is require to start feed and bleed operation as soon as possible. Also, the SPACE code can be useful for modeling the TLOFW accident.

ACKNOWLEDGEMENT

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[3] FSAR, Hanul 3&4, Chapter 5.