Analysis of Total Loss of Feedwater for APR1400 using SPACE

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1. Introduction

The Total Loss of FeedWater (TLOFW) event is an accident that main feedwater and auxiliary feedwater of secondary side are not supplied to steam generators. After the Three Mile Island accident, the issue of decay heat removal after a TLOFW is brought up, and recent studies have concluded that the Feed and Bleed (F&B) operation can be a viable alternate method of decay heat removal. [1, 2, 3]

APR1400 uses the Safety Depressurization and Vent System (SDVS) for the F&B operation and SDVS is designed to perform the rapid depressurization function of Reactor Coolant System (RCS) through the remote manual operation when TLOFW is occurred. If RCS pressure falls below a Safety Injection Pump (SIP) working pressure, it can be possible to start the F&B operation which injects SIP flow to RCS and releases the RCS vapor and two-phase flow through Pilot Operated Safety Relief Valves (POSRVs) by opening the POSRVs, and then it can be possible to remove the decay heat. The design requirement of SDVS is that the core water level should be maintained at higher than 2 feet from the top of active core during the F&B operation. If the core water level does not meet the above requirement (2 feet level), the maximum fuel cladding temperature would not exceed 2200°F.

TLOFW analysis using newly developed code, Safety and Performance Analysis Code (SPACE), was performed in order to confirm that SDVS meets the design requirement. The reference plant is APR1400. For benchmark purpose, the results are compared with previous results with RELAP5/MOD3 code.

2. Analysis Methodology

2.1 Plant Modeling and Initial Conditions

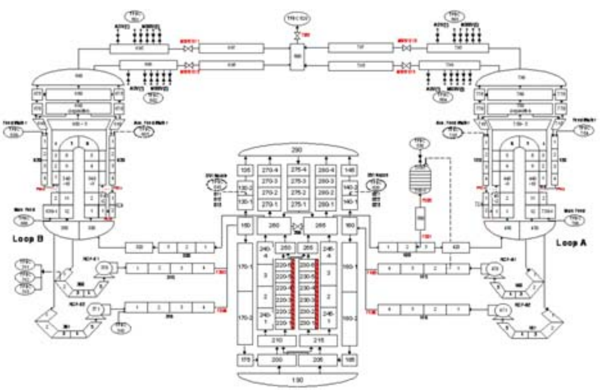

Figure 1. SPACE Nodalization of APR1400

APR1400 plant is two-loop 3983 MWt pressurized water reactor. Figure 1 shows the SPACE nodalization of APR1400. There are 4 POSRVs at the top of pressurizer, and these valves are used for F&B operation. Table 1 provides the initial plant conditions used in this analysis.

Table 1. Initial Conditions for TLOFW Analysis

Parameter	Design Value	RELAP5	SPACE
Core Power, MWt	3983	3983	3983
Pressurizer pressure, MPa(a)	15.51	15.51	15.51
RCS flow rate, kg/s	20,991	20,630	20,363
RCS average temperature, K	580.4	580.8	580.3
RCS total mass, kg	-	299,190	294,926
Secondary pressure, MPa(a)	6.89	7.03	6.89
Secondary total mass, kg/SG	98,911	98,878	99,106
Pressurizer level, %	50	50	50
Steam Generator level, % WR	76.87	76.82	76.87

2.2 Assumptions

TLOFW accident is initiated by the loss of all main and auxiliary feedwater supply to steam generator. The Reactor Coolant Pump (RCP) is assumed to stop at 10 minutes after the reactor trip. Based on the realistic analysis method, nominal design values are used for all reactor trip setpoints.

There are many Nuclear Steam Supply System control systems, such as Reactor Power Control System, Steam Bypass Control System (SBCS), Pressurizer Level Control System, and Pressurizer Pressure Control System. SBCS only is used since the effect of all the other control systems is negligible for TLOFW analysis .

The decay heat model of ANS-5.1-1979 is adopted. Henry-Fauske/Moody critical flow model is used for POSRV discharge. The valve area of a POSRV, 0.0024 m^2, is determined based on the APR1400 design steam flow capacity.

3. Analysis Results

TLOFW analysis is performed for the following three cases in order to ensure that SDVS is designed properly. Table 2 shows the event sequence for each case.

1) Case 1: No F&B operation.

2) Case 2: F&B operation with 2 POSRVs and 2 SIPs. Operator manually opens 2 POSRVs at the first automatic POSRV opening setpoint, and 2 SIPs are credited.

3) Case 3: F&B operation with 4 POSRVs and 4 SIPs. Operator manually opens all 4 POSRVs at 30 minutes after reaching the first automatic POSRV opening setpoint, and all 4 SIPs are credited.

Table 2. Event Sequences of TLOFW for APR1400

Time (sec) / Sequence	Case 1	Case 2	Case 3
Event Start	0	0	0
Reactor Trip	27.3	27.3	27.3
RCP Trip	627.3	627.3	627.3
SG Depletion	1369.5	1369.5	1369.5
POSRV 1st automatic open	1389.7	1389.7	1389.7
POSRV manual open	-	1389.7	3189.7
SI injection	-	1438.1	3354.7

3.1 Case 1 : No F&B Operation

Figures 2 and 3 show the pressurizer pressure and collapsed core level, respectively. The transient is initiated by the instantaneous loss of all main and auxiliary feedwater. The reactor trip occurs at 27.3 seconds due to low steam generator level. After reactor trip, pressurizer pressure decreases by the RCS shrinkage due to sudden decrease of core power. However, the pressurizer pressure starts to increase again soon since the heat removal capacity of the SG is decreasing due to loss of SG inventory through SBCS. SG inventory is depleted at 1369.5 seconds and pressurizer pressure reaches the POSRV open setpoint at 1389.7 seconds.

After reaching the POSRV setpoint, the pressure of the pressurizer fluctuates between POSRV opening and closing setpoints. RCS inventory is gradually decreasing but SIPs can't provide any makeup since the RCS pressure is higher than shut-off head of SIP. As RCS inventory is reduced, the core uncovery begins and temperature of the fuel cladding increases rapidly as shown in Figure 4.

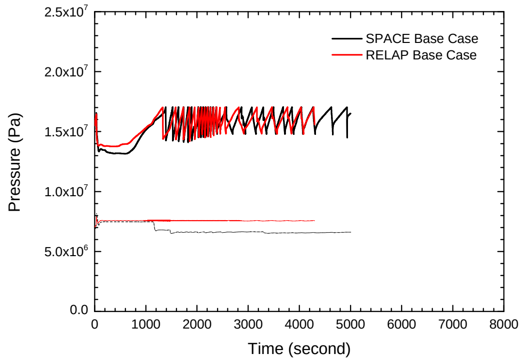

Figure 2. Pressurizer Pressure (Case 1)

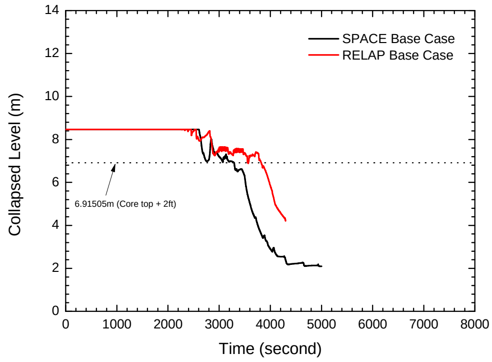

Figure 3. Collapsed Core Level (Case 1)

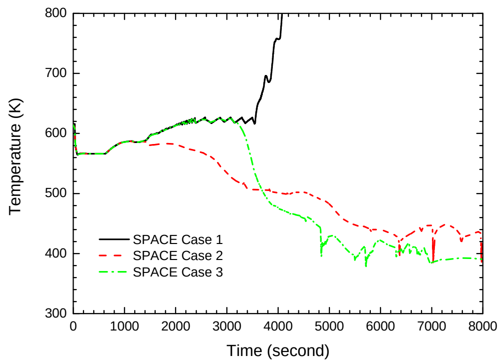

Figure 4. Fuel Cladding Temperature

3.2 Case 2 : F&B Operation with 2 POSRVs and 2 SIPs

Figures 5 and 6 show the pressurizer pressure and collapsed core level, respectively. When POSRVs automatically open, the operator action for manual opening of two POSRVs is assumed. Then pressurizer pressure decreases rapidly and SIP flow is injected to the RCS.

Since the collapsed core level is maintained at higher than 6.915 m (corresponding to 2 feet above from the top of active core), the fuel in the core is cooled well, and it is confirmed that SDVS is designed properly.

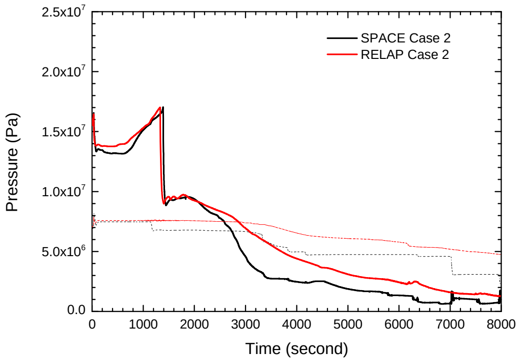

Figure 5. Pressurizer Pressure (Case 2)

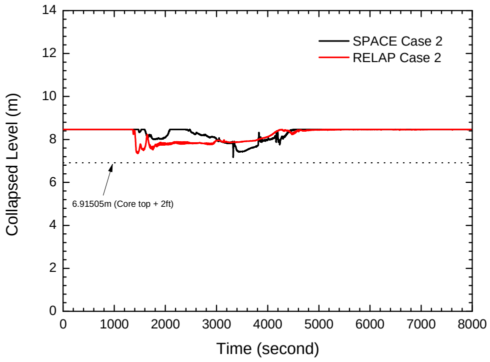

Figure 6. Collapsed Core Level (Case 2)

3.3 Case 3 : F&B Operation with 4 POSRVs and 4 SIPs

Figures 7 and 8 show the pressurizer pressure and collapsed core level, respectively. When the operator performs the F&B operation, the operator open all 4 POSRVs manually after 30 minutes from the first automatic POSRV open. Pressurizer pressure decreases more rapidly than Case 2, and 4 SIP flow is injected to the RCS. The collapsed core level falls below 6.915 m, but there is no fuel heatup as shown in Figure 4.

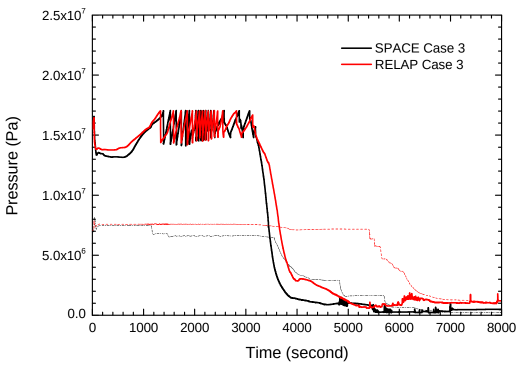

Figure 7. Pressurizer Pressure (Case 3)

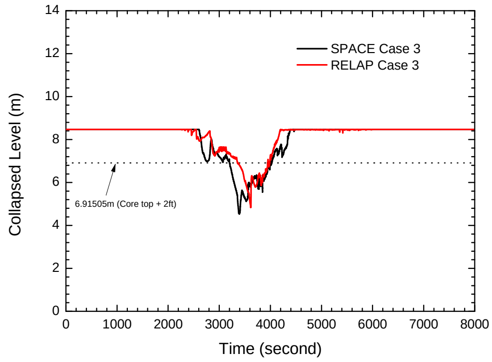

Figure 8. Collapsed Core Level (Case 3)

As shown in Figure 2 through Figure 8, the pressurizer pressure and collapsed core level behaviors calculated by SPACE and RELAP5/MOD3 are similar each other.

4. Conclusions

The TLOFW analysis was carried out to evaluate the capability of decay heat removal for APR1400 using newly developed SPACE code.

The analysis results show that the F&B operation with 2 POSRVs and 2 SIPs and the F&B operation with 4 POSRVs and 4 SIPs meet the SDVS design requirement for the fuel cladding temperature as shown in Figure 4.

The comparison with RELAP5 shows good agreement and it validates the applicability of SPACE code for this type of accident analysis.

ACKNOWLEDGEMENTS

This work was supported by the Nuclear Research & Development of the Korea Institute of Energy Technology and Planning (KETEP) grant funded by the Korea government Ministry of Trade, Industry and Energy. (No. 20151510101780)

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