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OPR1000 ATWS Analysis using the SPACE code

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1. Introduction

Since 1998, the U.S NRC has conducted an international joint study on the embrittlement of cladding in LOCA under high-burnup conditions. Based on the results of an experimental study of Zirconium Based Alloy, which is widely used as a fuel rod cladding material worldwide, NRC raised the need to change the LOCA acceptance criteria in 2003. KHNP is currently developing SPACE-based LOCA/Non-LOCA safety analysis methodology for OPR1000 and WH-type nuclear power plants that meet the requirement of new ECCS. In this study, preliminary calculations were performed for ATWS for verification of non-LOCA preliminary calculations based on SPACE.

2. Methods and Results

The ATWS scenario selected was the loss of main feedwater, for the most limiting case in terms of overpressure during the ATWS accident. The target power plant were selected for the OPR1000 model. The computer code used the SPACE 3.22.

2.1 Steady State

The SPACE input model is based on the RELAP5[1] model. The SPACE modeling for steady-state simulations of Shin-Kori Units 1 and 2 in shown in Fig. 1. The core is divided into a total of 20 cells in the axial direction and consists of two channels: average channel to check the overall behavior of the core and hot channel to analyze the hot thermal behavior. The heat structure of the core is also divided into 20 cells in axial direction and 8 in radial direction. It consists of 195 thermal hydraulic components, including valves, pumps, and control, and the primary system includes a reactor vessel, two hot legs, four cold legs, a pressurizer, and four reactor coolant pumps. The secondary system consists of the steam generator, the main steam safety valve, and the turbine. The safety injection system consists of four safety injection tanks and a high-pressure safety injection pump, which can simulate charging, letdown, auxiliary spray and safety injection. To simulate ATWS, systems and devices that can affect the sequence of event, such as the pressurizer safety valve, the safety decompression system, the turbine bypass system, and the main steam isolation valve are configured. The pressurizer safety valves were simulated into one unit with the combined area of three units, and the turbine bypass system is controlled by the steam bypass control system and consists of eight valves. The results of the steady state analysis are shown in Table I.

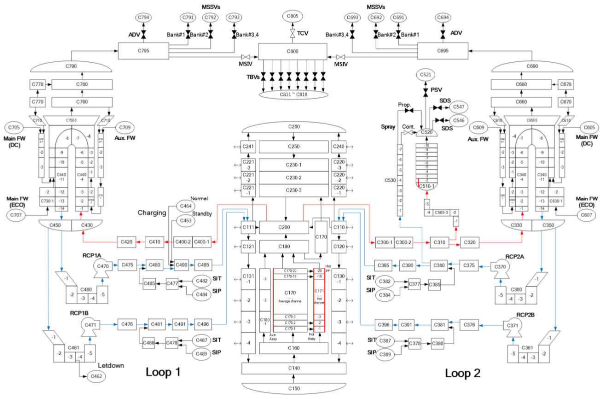

Fig. 1. Shin-Kori Units 1&2 SPACE Modeling.

Table I: SPACE Steady State Analysis Results

	Design Value[2]	SPACE
Core Power (MW t)	2,815	2,815
PZR pressure (MPa)	15.5201	15.0974
PZR Water Level (%)	52.51	50.71
Hot Leg Temp. (℃)	327.2	327.2
Cold Leg Temp. (℃)	295.89	295.83
RCS Flow (kg/s)	15,298.1	15,207.2
SG Pressure (MPa)	7.515	7.507
Total FW Flow (kg/s)	798.3	798.9
SG Level (%)	79	79
Circulation Ratio	3.704	3.554

2.2 Shin-Kori Units 1&2 DPS

Shin-Kori Units 1 and 2 are equipped with DPS(Diverse Protection System) that are independent of the reactor protection system in preparation for possible occurrence of ATWS.

The DPS includes the following three functions:
- Reactor trip by pressurizer high pressure signal
- Turbine trip by Reactor trip
- Auxiliary Feedwater injection by steam generator low level signal

The reactor trip function by high pressure of the pressurizer is included to prevent overpressure of the pressure boundary of the reactor coolant system.

2.3 Analysis Results

ATWS by loss of the main feedwater, which is a typical example of ATWS, starts when feedwater is tripped during normal operation. The pressure of the reactor coolant system is increased by the reduction of heat removal through the steam generator, causing the reactor trip, reaching the opening of the pressurizer safety valve, and supplementing the inventory of the lowered steam generator, the auxiliary feedwater is injected to remove the heat form the primary side. The sequence of the event is as shown in Table II.

Table II: Sequence of the Event

Time (sec)	Event	Set Point
0.0	Event start	
45.4	Aux. Feedwater Injection	22.1 WR%
74.0	PZR high pressure signal	16.396 MPa
74.6	Rx. Trip by PZR high pressure (DPS)	16.527 MPa
77.8	Max. PZR pressure	17.227 MPa
78.0	PSV open	17.24 MPa
85.0	PSV close	14.065 MPa
93.0	MSIV close	6.17 MPa
145.2	Max. PZR water level	78.2 WR%

If the main feedwater to the steam generator is lost, the level of the steam generator starts to decrease and the auxiliary feedwater operation set point is reached by the low level of the steam generator. Since this is ATWS, reactor is not tripped by low steam generator level. And then, the pressurizer high-pressure signal is occurred at 74.0 seconds, but the reactor is not tripped according to the ATWS assumption. The pressure is continuously increased, reaching the DPS high pressure set point and causing reactor trip at 74.6 seconds. The pressure of the reactor coolant system decreases when the pressurizer safety vale is opened after the reactor trip. The pressure of the pressurizer is reduced after reaching the maximum pressure of 17.227MPa. The pressure of the steam generator rises when the secondary main steam isolation valve is closed from 93.0 seconds due to the low pressure of the steam generator. The steam generator level is recovered up to 37.2 WR% by injection of the auxiliary feedwater, and the pressure of the reactor coolant system stabilized by the reactor trip and insertion of negative reactivity. The main hydrothermal behavior is shown in Fig. 2~6.

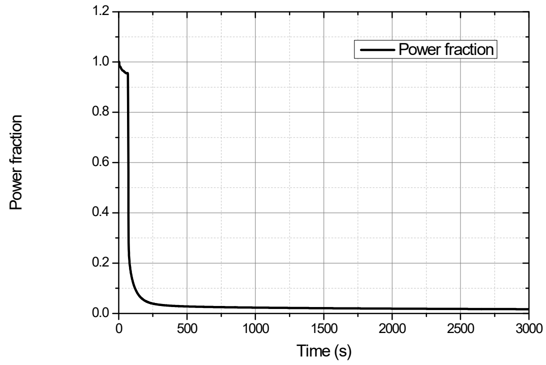

Fig. 2. Reactor Power Trend.

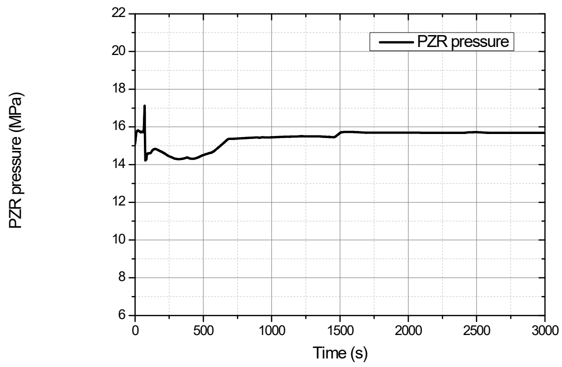

Fig. 3. Pressurizer Pressure Trend.

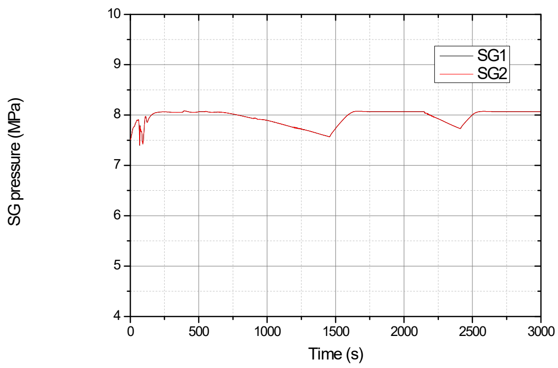

Fig. 4. Steam Generator Pressure Trend.

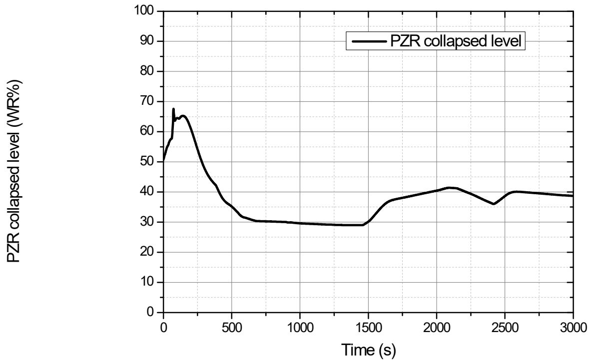

Fig. 5. Pressurizer Water Level Trend.

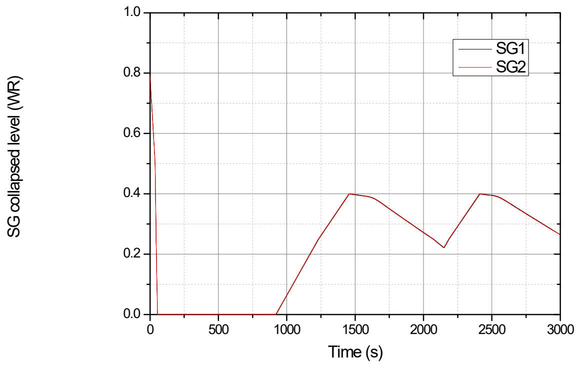

Fig. 6. Steam Generator Water Level Trend.

3. Conclusions

In the process of developing OPR1000 SPACE safety analysis methodology to satisfy new ECCS regulatory requirements, transient input of SPACE code was developed to analyze Shin-Kori Units 1 and 2 ATWS. The accident analysis also confirmed that the developed input is appropriate for the ATWS analysis of Shin-Kori Units 1 and 2. In the future, we will complement the completion of the SPACE input by comparing it with the previously analyzed RELAP5 results.

REFERENCES

[1] K. H. NAM, B. S. Youn, RELAP5 Code Input Calculation Sheet for the Anticipated Transient Without Scram(ATWS) Analysis of Shin-Kori units 1&2, 2018, KHNP.
[2] Final Safety Analysis Report of Shin-Kori units 1&2, KHNP.