Preliminary Assessment of SPACE code for Loss of Ultimate Heat Sink in OPR1000

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1. Introduction

After Fukushima nuclear power plant accident, International Atomic Energy Agency (IAEA) has introduced Design Extension Conditions (DECs) which includes Beyond Design Basis Accidents (BDBAs) and Severe Accidents (SAs) with SSR-2/1(2012) and has demanded complementary measures, accident managements and off-site emergency countermeasures for preventing from aggravating accidents, mitigating accidents and influence by radioactive material [1].

The objective of this paper is to analyze the accident for loss of ultimate heat sink (LOUHS) which is included in DEC. This study describes the safety analysis method and result considering the operator action for LOUHS accident in OPR1000 plant using SPACE (Safety and Performance Analysis Code for Nuclear Power Plant) code [2].

2. Methods and Results

2.1 Scenario

The initial event is a multiple failure that proceeds to a more serious condition by failing heat exchanger of the component coolant water system after a loss of ultimate heat sink from the sea water. In this case, the accident is assumed that the first accident that appears in the system is a loss of condenser vacuum (LOCV) accident in LOUHS accident.

When a loss of condenser vacuum accident occurs and the pressure of condenser rises, the turbine bypass valve (TBV) is isolated and the operation of the turbine bypass valve by the steam bypass control system fails. Main turbine and main feedwater pump shutdown occur in succession as the condenser pressure continues to rise. In this analysis, however, it is assumed that turbine stop, main feedwater pump trip, turbine bypass valve closure, and letdown line isolation are equal to 0.0 second after LOUHS accident due to loss of the component coolant system and condenser vacuum.

The operator recognizes the loss of the ultimate heat sink and manually stops the reactor coolant pump 10 minutes after accident. And manually stop one of the two charging pumps that are in operation. Then the operator manually drives the atmospheric dump valves (ADV) to cool and depressurize the reactor coolant system quickly through the secondary side. In the case of a loss of ultimate heat sink accident, the pressure of the reactor coolant system is maintained in a substantially high with the core uncovery if no proper accident response. Thus, the operator must open the steam generator ADV to reduce the pressure and temperature of the indirect reactor coolant system by increasing the heat transfer capacity. The opening of ADV flashes the water in secondary side and reduces the saturation temperature with the steam generator pressure drop. As a result, the pressure and temperature of reactor coolant system are lowered so that the connection of shutdown cooling system (SCS), which is designed to complete the cooldown of the reactor coolant system from 176 ^{0}C (350 ^{0}F) and 28.8 kg/cm^{2}A (409 psia) to 51.67 ^{0}C (125 ^{0}F) within 24 hours following a reactor shutdown.

2.2 Accident Response Operation Strategy

It is primarily necessary to check the operation status of the cooling water system according to the loss of the ultimate heat removal source and shut off the non-safety related heat load to suppress the rise of the cooling water temperature of the equipment. Checking the cooling state of the pump in the shutdown cooling system and charging pump and forming the alternative cooling means should continue from the beginning of the accident. As the coolant temperature of the equipment cooling water system is expected to rise, the physical state of these pumps will gradually deteriorate over time and should be continuously monitored through replacement operation or alternate cooling. Operator can extend the time to core exposure by minimizing inventory losses in reactor coolant system and steam generators and by maintaining secondary source heat removal sources. Operation to reduce primary and secondary inventory, such as RCS drainage, normal letdown flow, and runoff for cooling of the shutdown cooling system, or S/G drainage, shall be maintained and the drainage channel should be isolated to the extent possible. Steam generator pressure is secured using the atmospheric dump valves at a pressure above the available turbine driven auxiliary feedwater pump.

2.3 SPACE Code Modeling

Fig. 1 shows the nodalization scheme of SPACE code for OPR1000 nuclear power plant, which includes primary system, secondary system, and safety injection system. The primary system is composed of reactor pressure vessel which includes average active core, hot channel, hot rod, and down-comer, four reactor coolant pumps, two hot legs, and four cold legs of reactor coolant system. The secondary system is composed of two steam generators, four steam lines, main steam safety valves (MSSV), atmospheric dump valves (ADV), and turbine, etc. The safety injection system is composed of four high pressure injection system, four low pressure injection system, and four safety injection tanks as the emergency core cooling system.

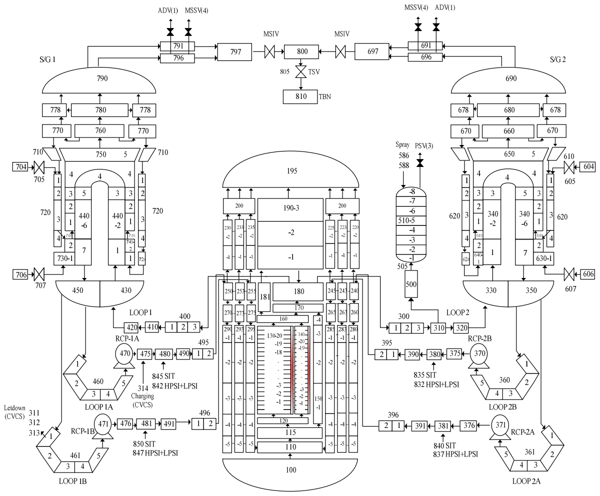

Fig. 1. SPACE nodalization scheme for OPR1000

2.4 Results of Steady-state Condition

Table 1 shows the results of the steady-state calculation using SPACE code (version 3.2) for OPR1000 nuclear power plant.

Table 1. The results of steady-state condition

	Position	Calculation	Error (%)
	Primary System		
Power (MWth)	100%	2815.0	0.0
Pressure (MPa)	Pressurizer	15.40	0.7
Level (%)	Pressurizer	52.61	0.0
Temperature (K)	Hot leg	602.11	0.3
	Cold leg	571.34	0.4
Flowrate (kg/s)	Average channel	14772.50	0.1
	Hot channel	82.77	1.3
	Hot leg A	7655.55	0.0
	Hot leg B	7652.70	0.0
	Cold leg A1	3825.75	0.0
	Cold leg A2	3825.78	0.0
	Cold leg B1	3826.41	0.0
	Cold leg B2	3826.41	0.0
	Secondary System		
Pressure (MPa)	Steam dome A	7.38	0.0
	Steam dome B	7.39	0.1
Level (%)	S/G A, NR-level	42.98	2.3
	S/G B, NR-level	43.14	1.9
Flowrate (kg/sec)	S/G A	398.25 ×2	0.6
	S/G B	402.49 ×2	0.4
	MFW A	801.34	0.0
	MFW B	801.34	0.0

2.5 Results of Transient Condition

We simulated a loss of ultimate heat sink accident that is assumed that the first accident that appears in the system is a loss of condenser vacuum accident in LOUHS accident. In the beginning of the event, the following trip signal is generated; turbine stop, main feedwater pump trip, turbine bypass valve closure, and letdown line isolation.

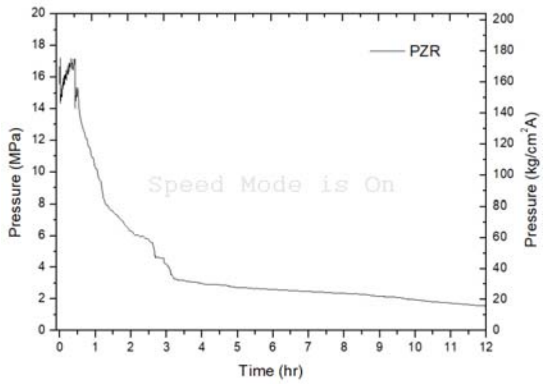

Fig. 2. The trend of pressurizer pressure

Fig. 2 shows the trend of pressurizer pressure. The pressure is rapidly increased due to the heat unbalance of between the primary side and secondary side in the steam generators. When the pressurizer pressure is 16.437 MPa, the reactor trip signal occurs with the high pressurizer pressure (HPP) signal. After 30 minutes, the opening of ADV flashes the water in secondary side and reduces the saturation temperature with the steam generator pressure drop. As a result, the pressure of reactor coolant system is lowered gradually such as the trend of the steam generator.

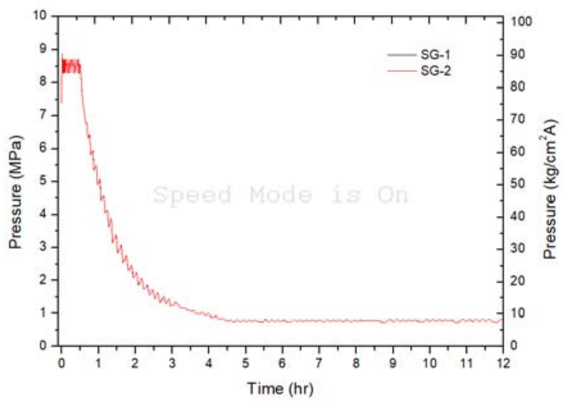

Fig. 3. The trend of steam generator pressure

Fig. 3 shows the pressure of the secondary system with two steam generator. Up to 30 minutes, the secondary side pressure increases and continues until it reaches the operational set points of main steam safety valves and by opening these valves and dumping steam in to atmosphere. Thereafter, the pressure of the steam generator was decreased gradually because the operator opens the steam generator ADV.

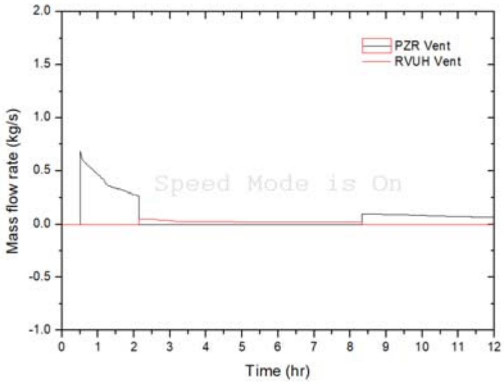

Fig. 4. The trend of PZR & RVUH vent

Fig. 4 shows the mass flow rate of the pressurizer vent and reactor vessel upper head (RVUH) vent system. When the pressurizer vent valve is opened for depressurization of the reactor coolant system, the reactor coolant system pressure begins to decrease. Releasing the reactor coolant inventory through the pressurizer vent valve causes steam bubbles in the reactor vessel upper head, which increases the water level of the pressurizer.

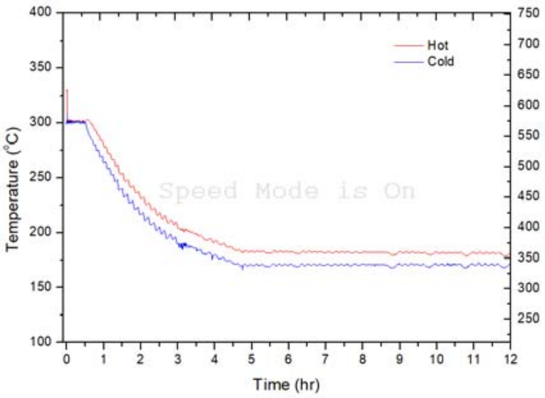

Fig. 5. The trend of hot and cold legs temperature

Fig. 5 shows the temperature of the primary system with the average temperature of hot-legs and cold-legs. The cooling and depressurization of the reactor coolant system were assumed to start 30 minutes after the accident. At the beginning of the accident, the temperature is maintained and the control operation is performed so that the cooling rate becomes less than 56 degrees per hour after 30 minutes. Fig. 6 shows the trend of ADV discharge flow rate to atmosphere. As the steam generator pressure decreases, the flow rate of the ADV decreases with time. When the temperature of the reactor coolant system reaches 176 degrees, the steam generator pressure is maintained by repeatedly opening and closing the valve to maintain the temperature of the reactor coolant system.

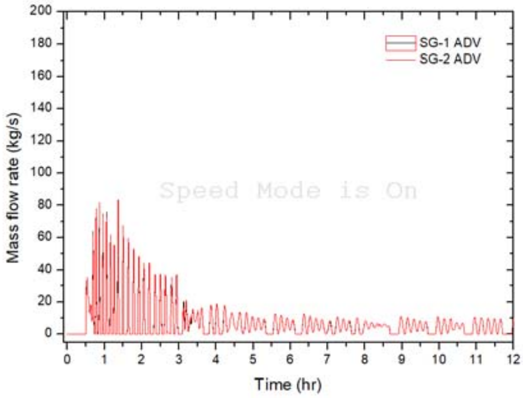

Fig. 6. The trend of ADV discharge flow rate

3. Conclusions

Recently, in order to simulate multiple failure accident with design extension conditions using the license and safety analysis code, we developed the SPACE input deck of the OPR1000 nuclear power plant and analyzed the accident for a loss of ultimate heat sink according to the strengthened domestic and foreign nuclear reactor regulatory requirements.

A preliminary analysis of the LOUHS accident in OPR1000 was performed using the SPACE code to identify the core cooling capability and operator action. In this study, it has been confirmed that an accident can be mitigated if the steam generator ADV opening time is proper. Thus, the operator must open the steam generator ADV to reduce the pressure and temperature of the indirect reactor cooling system by increasing the heat transfer capacity.

ACKNOWLEDGEMENT

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REFERENCES

[1] IAEA, SAFETY STANDARDS SERIES No. SSR-2/1, International Atomic Energy Agency VIENNA, 2012.
[2] SPACE 3.2 User's Manual, KHNP, 2018.