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Analysis of Kori Unit 3 Total Loss of Feedwater using SPACE Code

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1. Introduction

A new regulatory requirement was introduced in Korea, which requires accident management plan(AMP) for each nuclear power plant. As a part of supporting analysis for AMP, multiple failure accidents are analyzed. One of these accident is total loss of feedwater. The total loss of feedwater(TLOFW) assumes loss of main feedwater and auxiliary feedwater. The analysis for AMP used RELAP5 code for thermal hydraulic analysis. In this paper, SPACE code is used for TLOFW analysis for comparison. Kori Unit 3, which is a Westinghouse 3-loop PWR, is the target plant for this analysis. For multiple failure accidents, best estimate analysis is used including operator action for accident mitigation. Sensitivity analysis is performed for operator action.

2. Description of the Event

2.1 Description of Feedwater System

The main feedwater system supplies feedwater to the steam generator secondary side. If the main feedwater system fails and steam generator water level falls below certain value, auxiliary feedwater system supplies emergency feedwater to steam generator secondary side to remove decay heat from the reactor. The auxiliary feedwater system for Kori Unit 3 consists of 2 motor driven pumps and 1 turbine driven pump. The motor driven pump can be powered from emergency diesel generators in case of loss of off-site power. The turbine driven pumps use steam produced from steam generators as power source for the pump. The turbine driven auxiliary feedwater pumps can operate even in the event of station blackout.

2.2 Description of the TLOFW

Because of diversity and redundancy of auxiliary feedwater system, no single failure can result in loss of auxiliary feedwater. However, the TLOFW event assumes loss of both main feedwater and auxiliary feedwater. Since no feedwater is supplied to the steam generator secondary side, the decay heat cannot be removed through the steam generator. To remove heat from the core, feed and bleed operation is used. The operator opens the pressurizer power operated relief valve(PORV)s to decrease RCS pressure and use safety injection pumps to inject cold water into the RCS. Once the primary side pressure and temperature is reduced to the point of shutdown cooling system operation, the shutdown cooling system is used for heat removal and the plant is stabilized. Depending on number of PORVs opened, two cases are analyzed. For normal case, all PORVs open at 30 minutes after steam generator empty. All safety injection pumps are assumed to operate. For single failure case, 1 PORV opens at 5 minutes after steam generator empty. Only one train of safety injection pumps are assumed to operate. In this paper, single failure case is selected for sensitivity analysis.

3. Analysis Method

3.1 Description of the Computer Code

For analysis of Kori Unit 3 TLOFW, SPACE 3.21 code was used. The SPACE code is a system thermal hydraulics analysis code for LOCA and Non-LOCA accident analysis. The SPACE code was developed by Korean nuclear industry. The code uses 2-fluid 3-field 10-equation model. The SPACE code has been approved by KINS for PWR safety analysis.

3.2 SPACE Input

The SPACE nodalization of the Kori Unit 3 consists of 271 cells and 317 faces. Kori Unit 3 is a typical Westinghouse 3-loop PWR with 3 loops. Each loop has one steam generator (SG), one hotleg, one coldleg and one RCP. The core section has two channels: one average channel and one hot channel. Each channel has 8 axial nodes. Safety injection flow rates assumed one train of safety injection pumps. A 10000 second steady state calculation was performed before the transient calculation.

3.3 Operator Actions for Sensitivity Analysis

For sensitivity analysis, 3 cases are analyzed. Case A assumes manual RCP trip and manual safety injection initiation. No manual operation of pressurizer PORV is assumed. One PORV is assumed to automatically open and close at set pressure. Case B assumes manual operation of one pressurizer PORV to depressurize the RCS. This is consistent with current plant operating procedure. As RCS pressure decreases, safety injection is automatically initiated. Case C assumes turning off pressurizer heaters after manual operation of pressurizer PORV. Cases analyzed are shown in table 1.

Table 1: Cases for Sensitivity Analysis

Case	Description
A	Manual RCP trip, manual SI
B	Manual RCP trip, manual PORV open
C	Case B + Pressurizer heater off

4. Calculation Results

4.1 Base Case Transient Calculation Result

At t=0 sec, the main feedwater to the steam generator secondary side is lost. This results in reduction of steam generator water level and eventual to reactor trip on low steam generator water level. As SG level continues to drop, U-tubes are uncovered and heat transfer area is reduced. Due to decrease in heat removal, RCS pressure increases. 10 minutes after reactor trip, the RCPs are manually stopped. The RCS pressure continues increase and reach pressurizer PORV opening setpoint. The PORV open/close as pressure changes and pressurizer pressure remains close to PORV setpoint as shown in Fig. 1. The steam generator is considered empty when wide range level reaches 6%. After 5 min. delay from SG empty, safety injection is manually started. Kori Unit 3 HPSI pumps have high design pressure, so SI flow can be injected against high RCS pressure. However SI flowrate is low when RCS pressure is high. The core collapsed level decreases because safety injection flow not sufficient to maintain core water level as shown in Fig. 2.

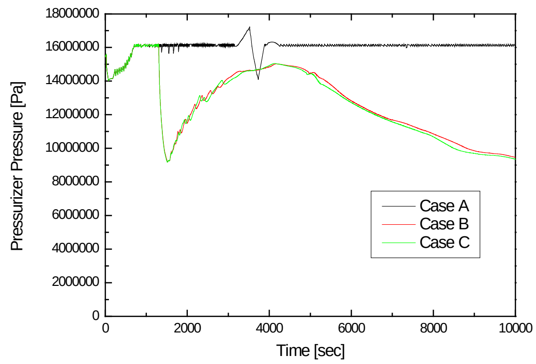

Fig. 1. Sensitivity analysis results for pressurizer pressure

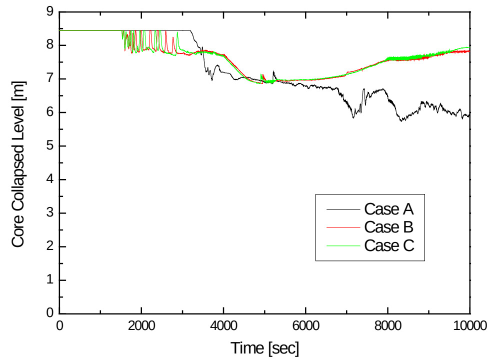

Fig. 2. Sensitivity analysis results for core collapsed level

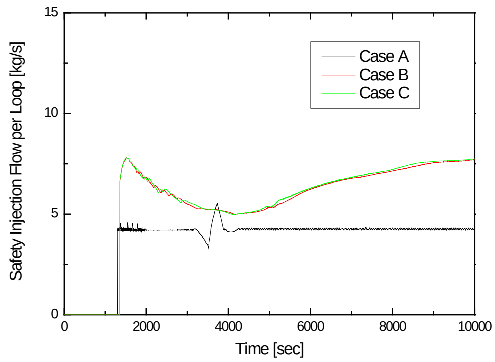

Fig. 3. Sensitivity analysis results for safety injection flow rate

4.2 Sensitivity Analysis Results

For Case B, the pressurizer PORV was manually open to depressurize the RCS. This resulted in lower RCS pressure than case A. Safety injection is started by low pressurizer pressure. Lower RCS pressure results in higher safety injection flowrate than case A as shown in Fig. 3. With higher safety injection flowrate, the core collapsed level begins to recover after about 5000 sec.

For Case C, the pressurizer heater was turn off after manual PORV opening. This resulted in slightly lower pressurizer pressure, but the difference was very small.

5. Conclusion

The total loss of feedwater accident was analyzed and sensitivity analysis performed with SPACE 3.21 code. The reference plant was Kori Unit 3 which is Westinghouse 3-loop plant. Sensitivity analysis of operator action was performed. The results show that manual operation of pressurizer PORV to depressurize the RCS was effective in cooling the core and maintaining core liquid inventory. Therefore current operating procedures are appropriate in mitigating the TLOFW event.

REFERENCES

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