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Analysis of Shin-Kori 2 Reactor Trip and Natural Circulation Using SPACE Code

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1. Introduction

The Korean nuclear industry developed a thermal-hydraulic analysis code for the safety analysis of PWRs, named SPACE(Safety and Performance Analysis Code for Nuclear Power Plant). As a part of code validation effort, Shin-Kori Unit 2 reactor trip during turbine trip and natural circulation test was analyzed with SPACE code. The results were compared with plant data.

2. Description of the Event

2.1 Description of the Plant

The Shin-Kori Unit 2 is a OPR1000 (Optimized Power Reactor 1000) type PWR. It has rated thermal output of 2,775MWt and electric output of 1,000 WMe. It has 2 steam generators, 2 hotlegs, 4 coldlegs, 4 RCP(Reactor Coolant Pump)s and 1 pressurizer.

2.2 Description of the Transient

On March 4th 2012, high pressurizer pressure reactor trip occurred during turbine trip and natural circulation test of Shin-Kori unit 2. The test was carried out as a part of start-up test for new power plant. This test is carried out with RPCS(Reactor Power Cutback System) disabled. The RPCS is designed to avoid reactor trip during load rejection by rapidly inserting control rods and reducing core power. The reactor is supposed to be tripped by DPS(diverse protection system) signal. However, in this case, the reactor was tripped on high pressurizer pressure. The initial condition for the test was plant running at 100% power. The test was initiated by manual turbine trip. Sudden reduction in secondary energy removal resulted in increase of RCS temperature and pressure. Eventually the pressurizer pressure reached reactor trip setpoint and reactor trip occurred. After waiting about 20 minutes, natural circulation test was carried out.

3. Computer Codes

3.1 General Description of the SPACE Code

The SPACE code is an advanced thermal hydraulic analysis code capable of two-fluid, three-field analysis. The SPACE code has many component models required for modeling a PWR, such as reactor coolant pump, safety injection tank, etc. The programming language used in the new code is C++, for new generation of engineers who are more comfortable with C/C++ than old FORTRAN language. The SPACE code can be used in LBLOCA, SBLOCA and Non-LOCA analysis of PWRs. The version used in this study is SPACE 3.0.

4. Calculation Results

4.1 SPACE Input

The SPACE nodalization of the Shin-Kori unit 2 is shown in Fig. 1. Two hot legs, four cold legs, two steam generators are modeled. The core region is modeled using 14 cells and 12 heat structures. Steam generator feedwater is modeled with TFBC boundary condition. To obtain steady state, 1000sec null transient calculation was performed. Turbine trip occurs at t=27sec. In the early portion of the transient, control systems modeled in SPACE is used. After t=800sec, steam pressure and pressurizer level where controlled to match plant data.

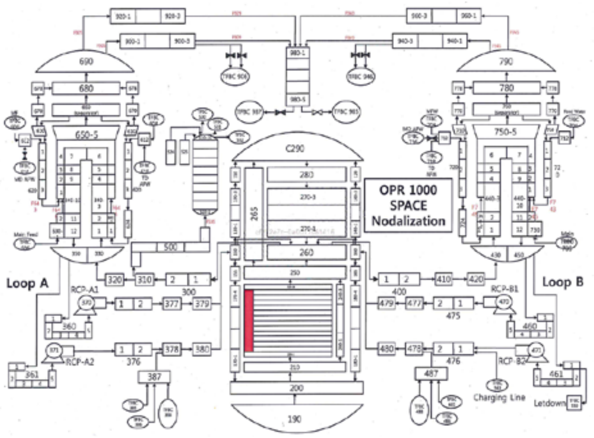

Fig. 1. SPACE Nodalization for Shin-Kori Unit 2

4.2 Calculation Results

The calculation results using SPACE 3.0 were compared with measured data from the plant. Major sequence of events is shown in table 1. Transient calculation starts at t=0sec. Manual turbine trip occurs at 27sec. The reactor is tripped on high pressurizer pressure trip signal at 36sec. After waiting about 20 minutes for plant parameters to stabilize, the reactor coolant pumps are manually tripped at t=1379sec for the natural circulation test. The natural circulation condition is reached at t=1800sec. The calculation ends at t=4000sec.

Table 1. Major Sequence of Events

Event	Time [sec]
Turbine trip (manual)	27
Reactor trip (high pressurizer pressure)	36
Reactor Coolant Pump trip (Manual)	1379
Reached Natural Circulation Test Condition	1800
End of calculation	4000

Fig. 2 shows hotleg and coldleg temperatures as a function of time. With RCPs in operation, the hotleg and coldleg temperature difference is proportional to the core thermal power. Before reactor trip, hotleg and coldleg temperatures show large difference. After turbine trip, hotleg and coldleg temperature increases as heat removal through turbine is stopped. As RCS average temperature increases, the coolant expands and RCS pressure increases as shown in Fig. 3. Reactor trip occurs on high pressurizer pressure. After reactor trip, core power is reduced to decay heat level and temperature difference is also reduced.

For natural circulation test, RCPs are manually tripped at t=1379sec. With loss of forced flow, the RCS flow is driven by density differences between cold side and hot side. Natural circulation condition is reached at about 1800 sec. During natural circulation cooling period, temperature difference between hotleg and coldleg remain constant and average temperature is slowly decreasing. The SPACE calculation results and plant data show good agreement.

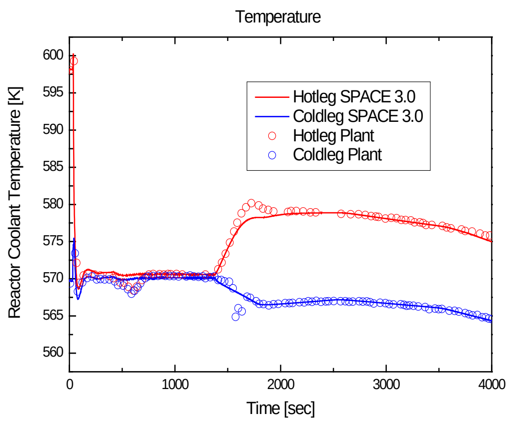

Fig. 2. Comparison of Hotleg and Coldleg Temperature

Fig. 3 shows pressurizer pressure as a function of time. During 26sec ~ 36sec period, heat removal path to the turbine is closed without reactor trip, resulting in increase of RCS average temperature and pressure. High pressurizer pressure trip occurred at t=36sec. After reactor trip, the RCS average temperature drops due to reduction of core power. The RCS temperature decrease results in shrinking of the coolant and decrease of pressurizer pressure. As secondary heat removal is reduced and pressurizer heaters add heat to the pressurizer, the pressurizer pressure slowly recovers to normal operating range. During natural circulation, the pressure slowly decreases as RCS temperature decreases. The SPACE calculation result follows plant pressure trends well.

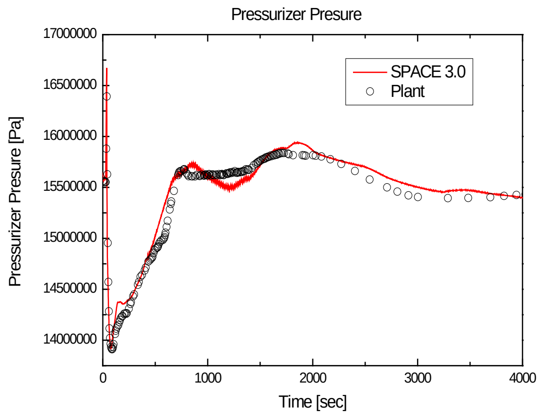

Fig. 3. Comparison of Pressurizer Pressure

5. Conclusions

As a part of SPACE code validation effort, Shin-Kori Unit 2 reactor trip during turbine trip and natural circulation test was analyzed with SPACE code. The results were compared with plant data. Comparison of hotleg temperature, coldleg temperature and pressurizer pressure show that SPACE calculations agree with plant data with reasonable accuracy. Thus, SPACE code can be used to predict plant response to a PWR reactor trip and natural circulation test.

Acknowledgement

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REFERENCES

[1] SPACE Users Manual, KHNP, 2016

[2] Nuclear Power Plant Accident Report 2012-03, Shin-Kori Unit 2 High Pressurizer Pressure Reactor Trip During Start-up Test, KINS, 2012

[3] D. H. Lee, Y. H. Kim and B. S. Youn, Analysis of Shin-Kori 2 High Pressurizer Pressure Trip Event for Assessment of SPACE Pressurizer, Transactions of the Korean Nuclear Society Autumn Meeting, Gyeongju, Korea, 2015