LOFT L9-3 ATWS Experiment Simulation using the SPACE Code

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1. Introduction

The Korea nuclear industry has developed a best-estimated two-phase three-filed thermal-hydraulic analysis code, SPACE(Safety and Performance Analysis Code for Nuclear Power Plants), for safety analysis and design of a PWR(Pressurized Water Reactor). As the first phase, the demo version of SPACE code was released on March, 2010. And the code has been verified and improved according to the Validation and Verification (V&V) matrix prepared for the SPACE code as the second phase of the development.

In this study, the LOFT (Loss Of Fluid Test) L9-3 Anticipated Transient Without Scram (ATWS) experiment has been simulated using the SPACE code as one of the V&V work. The results were compared with those of the experiment and other code simulation.

2. Facility and Test Description

2.1 Facility Description

The LOFT facility is a 50 MWt pressurized water reactor (PWR) with 1/60 power-to-volume scale with the Westinghouse 4-loop PWR. It has various instrumentations to measure and to provide data from the thermal-hydraulic and nuclear condition throughout the system. The LOFT facility consists of five major system : reactor system, primary coolant system, blowdown suppression system, emergency core cooling system and secondary coolant system.

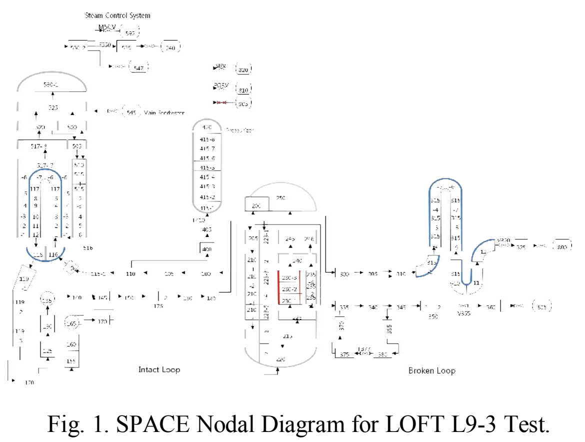

Fig. 1. SPACE Nodal Diagram for LOFT L9-3 Test.

2.2 Test Description

Anticipated operating transients during which the reactor does not scram as designed, ATWS (Anticipated Transient Without Scram), can be occurred by multiple failures. The rapid excursion of the RCS pressure and temperature by loss of feedwater and no scram could result in damaging of the reactor core. To resolve this concern, the system thermal-hydraulic behavior following an ATWS event should be understood, and the capability of the plant safety features to mitigate the event should be assured. For this aspect, a thermal-hydraulic analysis code to be applied to the system response following an ATWS event should be verified for relevant experiment simulating the ATWS event.

The Experiment L9-3 conducted in the LOFT facility was a unique one simulating an ATWS event in pressurized water reactor (PWR). The analysis of the LOFT L9-3 experiment was conducted by several researchers using the RELAP5 code. It was reported that the code could reasonably predict the RCS thermal-hydraulic response, the reactor power response and the secondary system response following the experiment.

The experiment was initiated by turning off the main feedwater pump. The steam generator steam control valve was closed manually at 67.3 s. The experimental PORV opened at 67.3 sec., and the experimental SRV opened at 96.8 sec. at their set-point pressures. The maximum pressure occurred at 17.4 MPa, and the SRV could prevent the further pressure increase as designed. The plant recovery was initiated at about 600 sec. by starting one high pressure injection system (HPSI), starting the secondary coolant system auxiliary feedwater, and opening the PORV. The control rods remained withdrawn. Major initial conditions are summarized in Table 1.

3. SPACE Modeling

3.1 Steady-state of LOFT L9-3 using the SPACE code

At first, SPACE code deck was made using the RELAP5 code deck of LOFT L9-3 for SPACE code capability evaluation. Generally, all initial conditions and assumptions used in REALP5 code were equally adapted to LOFT L9-3 SPACE input deck. LOFT L9-3 SPACE input deck was ran from 0 sec. to 1000 sec. for steady-state confirmation.

Table I : LOFT L9-3 Initial Conditions

Parameter	Measured	Predicted
Primary Coolant System		
Mass flow rate(kg/s)	467.6	493.61
Hot leg pressure(MPa)	14.98	14.96
Core ΔT(K)	19.4	19.1
Intact loop average temperature (K)	566.7	568.3
Cold leg temperature (K)	557.0	558.8
Hot leg temperature (K)	576.4	577.9
Reactor Vessel		
Power level (MWt)	48.7	50.8
Maximum linear heat generation rate (kW/m)	51.6	-
Pressurizer		
Liquid temperature(K)	615.2	614.7
Pressure (MPa)	14.98	14.9
Liquid level (m)	1.00	1.0
Steam Generator Secondary Side		
Liquid level (m)	3.15	3.17
Liquid temperature(K)	544.4	544.5
Pressure (MPa)	5.61	5.60
Mass flow rate(kg/s)	25.7	23.08

Major parameters are presented from Fig.2 to Fig. 5.

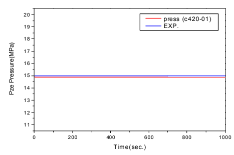

Figure 2. PZR Pressure

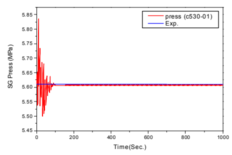

Figure 3. SG Pressure

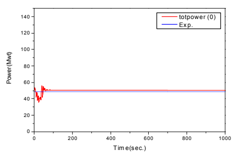

Figure 4. Power

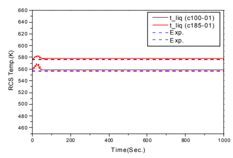

Figure 5. RCS Temperature

3.2 Transient of LOFT L9-3 using the SPACE code

LOFT L9-3 SPACE transient input deck was made based from steady-state deck of SPACE code. For verification of SPACE code, basic assumptions and conditions used in RELAP5 mod3.2.2 gamma code were identically adopted to transient SPACE code deck.

The transient simulation is performed after 300 sec. null transient, and terminated at 500 sec.. So, transient simulation time is 200 sec..

Major parameters are presented from Fig.7 to Fig. 10.

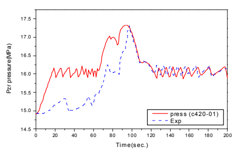

Figure 7. PZR Pressure

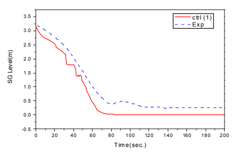

Figure 8. SG water Level

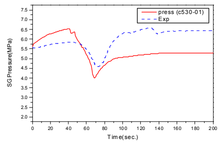

Figure 9. SG Pressure

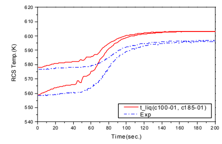

Figure 10. RCS Temp.

3. Conclusions

The Korea nuclear industry has been developing the SPACE code for safety analysis and design of a PWR. The LOFT L9-3 ATWS experiment has been simulated for the SPACE code V&V. The results have been compared with those of the experiment.

Through the evaluation of LOFT L9-3 experiment using the SPACE code, it is concluded that the SPACE code has a capability to predict the system response caused by the ATWS transient. Finally, the further studies were needed on the effect of the MDC feedback and realization of pressurizer model on the system response.

Acknowledgements

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