Simulation of LOFT LP-02-6 test using SPACE code

Kwi-Seok Ha, and Kyung-Doo Kim
Korea Atomic Energy Research Institute, 1045 Daedeok-daero, Yuseong-gu, Daejeon, Republic of Korea
E-mail:ksha@kaeri.re.kr

1. Introduction

For the assessment of analysis capability of the SPACE (Safety & Performance Analysis CodE) code the LP-02-6 of a large break LOCA (Loss-of-coolant-accident) test which had been conducted at LOFT (Loss-of-fluid-test) was simulated. Various models of the SPACE could be investigated through the modeling for the phenomena of a heatup and quenching during blowdown and reflood, a critical flow, CCFL(Counter-current-flow-Limitation), and so on. The calculation results by SPACE are compared with the test data herein.

2. SPACE model for LOFT facility

LOFT facility is a 50 MWt PWR (Pressurized water reactor) system that was designed to simulate the major components and system responses of a commercial PWR during LOCAs and anticipated transients. The volume ratio of 1/60th was used to scale-down a commercial 4-loop PWR[1].

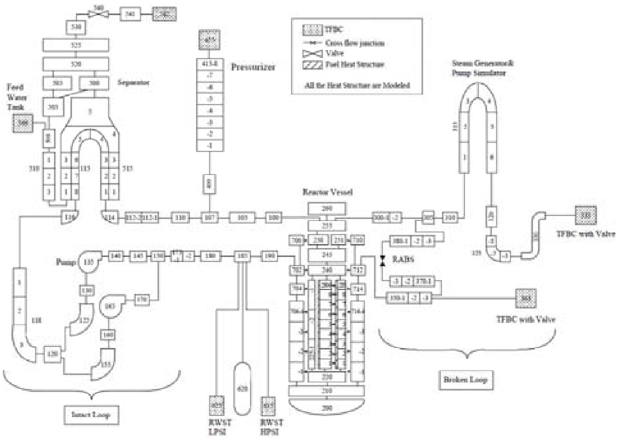

Fig. 1 SPACE Nodalizations for LOFT

The major components are the reactor with nuclear core, primary coolant system, blowdown suppression system, emergency core cooling system (ECCS), and secondary coolant system. The reactor core was composed of 8 fuel assemblies and the 1300 fuel rods of 1.68 m length was equipped. The primary coolant system consists of two coolant loops (intact and broken loops) connected to the reactor vessel. The intact loop includes following principal components; steam generator, two pumps, and pressurizer. The broken loop includes piping, steam generator simulator, pump simulator, and quick-openning valves. The ECCS which provides a core cooling under an accident situation contains 2 high-pressure injection systems, 2 accumulators, and 2 low-pressure injection systems. The coolant flow of 270 kg/s circulates through intact loop hot leg pipe, steam generator, pumps, cold leg, downcomer of reator vessel, lower plenum, reactor core, and upper plenum. The initial core power is 46 MW and the maximum linear heat generation is 49 kW/m.

As shown in Fig. 1[2], the core was modeled as hot and average channels. The hot channel represents 19 fuel rods and subchannels with highest power in the central assembly. The remaining part of the facility was modeled as it was. In the broken loop pressure boundary conditions were applied at the hot and cold leg breaks. The Ransom-Trapp model was used for the critical flow calculation. Also, the CCFL model of the Kutateladze's correlation was applied in the vertical nodes of reactor vessel.

3. Transient Simulation Results

The test of the LP-02-6 was initiated by open two quick-opening valves located at hot and cold legs of broken loop. The reactor was scrammed when the hot leg pressure reached 14.8 MPa (0.1 seconds) and the primary coolant pumps experienced coastdown by 16.5 seconds due to the loss of offsite power.

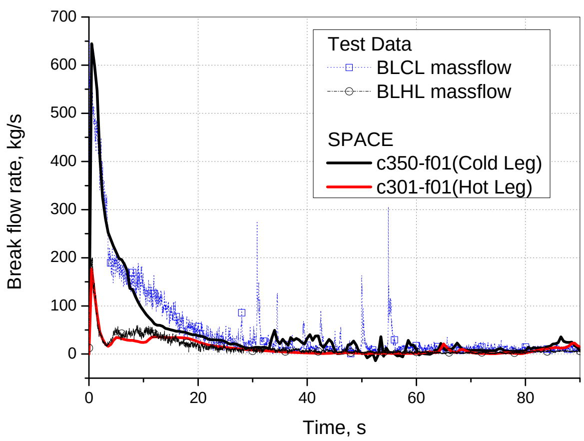

Fig. 2 Break flow comparison

The single phase liquid was released from the both break valves of hot and cold legs. The Ransom-Trapp model well predicted the initial break flow behaviors. After the pump trip the intact loop hot leg pressure was decreased to 7

MPa and two-phase flow dischage was started due to the flashing. The initial two phase critical flow discharge was large compared with the test data, which caused the reflood heatup to be delayed. While on the other, from 10 second to 20 second the break flow at the cold leg was significantly small, so the system pressure was slowly decreased compared with the experiment.

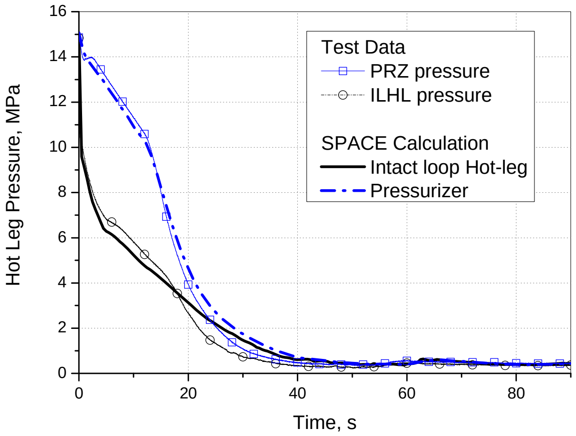

Fig. 3 Pressure behaviors at hot leg and pressurizer

In the core region the flow was initialy splited into the both up and down and then the flow reentered into the core due to the pump coastdown. So the clad temperature was heated up and cooled during the period of blowdown as shown in the Fig.4 which represented the peak clad temperature behavior. However the fuel rods were again heated due to the decrease of the pump velocity and the coolant discharge.

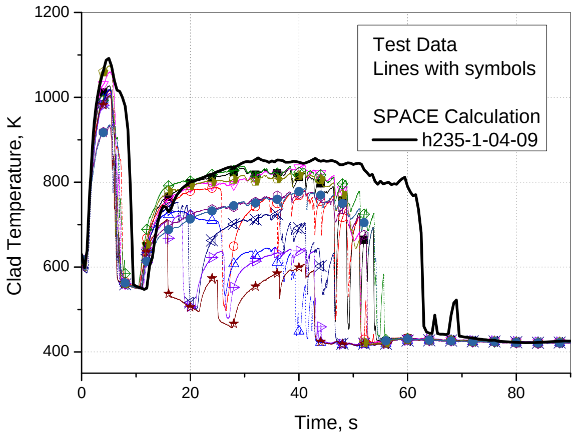

Fig. 4 The comparison of the peak clad temperatures

The calculated maximum temperature during the blowdown was 1092 K which is very similar to the test data of 1074.5 K. The downcomer water level which was empty due to the break flow started to increase after the injection of the emergency core cooling flows. The recovery of the downcomer water level occurred the gravity head difference between downcomer and core and the fuel rods were finally quenched. The maximum temperature during reflood was calculated to be 857.3 K and in the test 838.4 K was recorded. The calculated reflood quenching time was delayed about 10 seconds compared with the test because the more break flow was claculated than the test. However, the overall results were very well agreed with the test data.

4. Conclusion

The large break LOCA test of LP-02-6 was simulated using the SPACE code. The single phase liquid discharge was well agreed with the test data, however the calculated two phase flow discharge was small. So the depressurization of the system was predicted to be slightly delayed. Overall clad temperature behaviors were also very well agreed with the test data. Conclusively it can be said that the SPACE code has enough capability to analyze the system behavior under the large break LOCA.

ACKNOWLEDGMENTS

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[2] Large Break LOCA Realistic Evaluation Methodology, Volume 1 Model Description and Validation, KAERI and KEPRI, 1995.