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Validation of the SPACE Code for Experiments of LOCA

Myung Taek Oh*, Chan Eok Park, and Jong Joo Sohn
KEPCO E&C Company, Inc., 150 Deokjin-dong, Yuseong-gu, Daejeon, 305-353
**Corresponding author:mtoh@kepco-enc.com*

1. Introduction

In order to assess the applicability of SPACE as a system analysis code, various calculations for the phenomenological problems, the separate effects tests (SETs) and the integral effects tests (IETs) have been performed so far. The code has shown quantitatively and qualitatively good agreement with the results from the phenomenological problems [1]. Intensive study including comparison of SPACE predictions on wide variety of events with the results from SETs and IETs has been performed. The simulation results also agreed well with the measured data of each problem. In this paper, LOCA-related SET and IET such as UPTF 4A test and LOFT L2-5 test are simulated using the SPACE and the simulation results are compared to the data for the tests.

2. Test Plan

Code validation is actually a measure of how well the code can analyze problems of interest. The validation of SPACE has been accomplished using three types of problems: phenomenological problems, modeling of separate effects experiments, and modeling of integral experiments. The phenomenological problems, which are quite simple analytical tests mainly dealing with one basic phenomenon, are used to check whether the code is in qualitative agreement with the physics of the problem and in cases where analytical solutions exist. The SETs are used for the various purposes, such as validating each closure relation and defining the best nodalization for each component. After validating the individual models with these tests, more complicated tests such as system effects analyses and analysis of actual plant transients are simulated. Code predictions of integral system parameters, such as pressure, clad temperature, and mass inventory, are used to assess overall accuracy of the code.

The simulations for 12 phenomenological problems, 8 separate effects problems and 6 integral effects problems have been performed by KEPCO E&C. In addition to these problems, lots of problems have been performed by other institutes, as a series of code validation effort.

3. Test Results

3.1UPTF 4A Test

Upper Plenum Test Facility (UPTF) is a full-scale model of a four-loop 1300MWe pressurized water reactor including the reactor vessel, downcomer, lower plenum, core simulation, upper plenum, loop simulation with steam generator simulation [2]. The core, coolant pumps, steam generators and containment of a PWR are replaced by simulators which simulate the boundary and initial conditions during end-of-blowdown, refill and reflood phase following a loss-of-coolant accident (LOCA) with a hot or cold leg break.

The UPTF Test 4A is modeled with the SPACE in order to assess the code capability for predicting the behavior of the two-phase mixture upflow and water downflow in the downcomer and the emergency core coolant (ECC) bypass via the broken cold leg nozzle during end-of-blowdown and refill phase following a LOCA. The simulation is performed under transient conditions starting from 12.2 bar, decreasing to the containment pressure of 2.5 bar. The break valve simulating the vessel side of the broken hot and cold leg is open to the containment simulator. An ECC mass flow rate is injected into each of the three intact loop cold legs. The results of SPACE calculation are compared with the experimental data of UPTF test, particularly the ECC bypass via the broken loop and the liquid level in core.

As shown in the figure 1, the integrated liquid mass through the broken cold leg at the beginning of the transient shows a tendency to increase rapidly. Two phase mixture upflow in the donwcomer is produced by the flashing of saturated water initially stored in the lower plenum. The ECC water delivered to the downcomer is bypassed partially to the broken cold leg by the two phase mixture swelled from the lower plenum to the downcomer. From the figure 1 and the figure 2, the bypass of the ECC water and its penetration into the lower plenum in the early phase of refill seem to be reasonably well predicted. During the final phase of refill, the liquid inventory in the core is sufficiently correct.

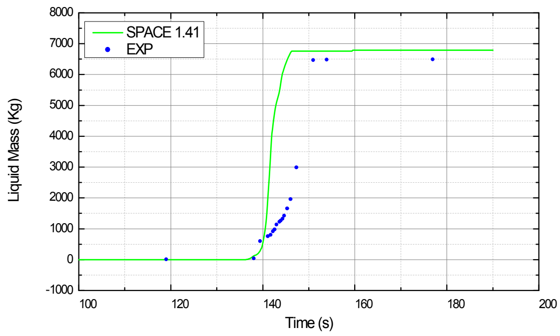

Figure 1. Measured and Calculated Liquid Mass

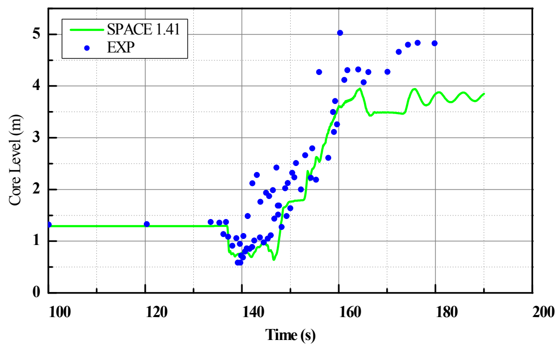

Figure 2. Measured and Calculated Core Liquid level

3.2 LOFT L2-5 Test

The LOFT Integral Test facility is a 50 MW pressurized water reactor system designed to simulate the major components and system responses of a commercial PWR during postulated LOCAs and anticipated transients. The detailed descriptions on the LOFT system are given in Reference [3]. The LOFT Test L2-5 was simulated with the SPACE. The LOFT L2-5 simulated a 200% double-ended cold leg break with an immediate primary coolant pump trip. The core steady state power level was 36 MW. The experiment was initiated by opening the quick-opening blowdown valves. The reactor was scrammed on a low pressure signal at 0.34s. Following the scram, the operators tripped the primary coolant pumps at 0.94s. The pumps were not connected to the flywheels in this experiment. This was done in order to provide an early rapid pump coastdown which would prevent the early core rewet phenomena and result in higher fuel cladding temperatures.

Figure 3 shows the calculated mass flow rates in the broken loop cold leg and broken loop hot leg, each compared to the data. The code calculations are in good agreement with the measured data. Figure 4 and 5 show the predicted and measured fuel rod cladding temperatures at 0.66 and 1.57m elevations. At the lower elevations, the code predicts the rod clad temperature and the time of quench reasonably well. At the 1.57m elevation, the code predicts the first rod heat-up and top-down quench, but over-predicts the second quench time.

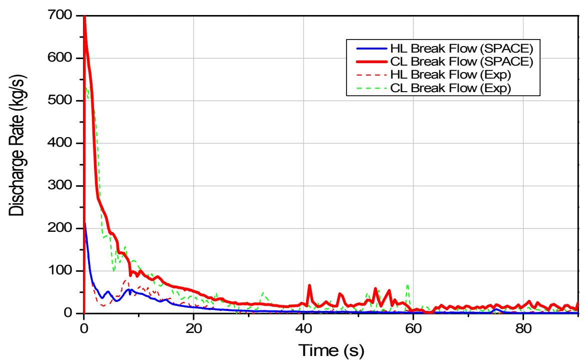

Figure 3. Measured and Calculated Mass Flow Rate

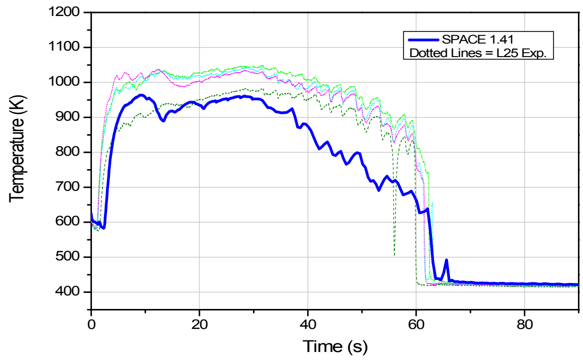

Figure 4. Measured and Calculated Fuel Cladding Temperature (0.66m above the Bottom of the Core)

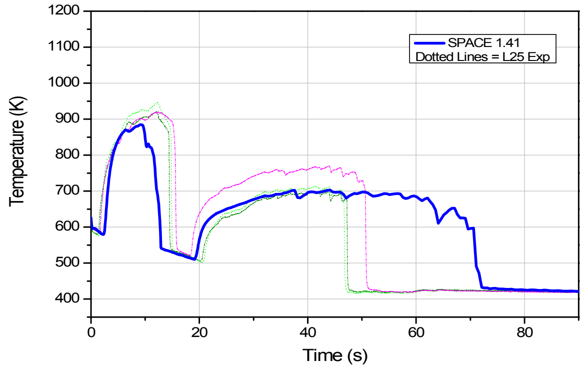

Figure 5. Measured and Calculated Fuel Cladding Temperature (1.57m above the Bottom of the Core)

4. Conclusions

As an effort for validation, simulations for experiments of LOCA have been performed. Though only some results are shown here, most test results agreed well with the measured data of each problem. These results demonstrate that the SPACE code can provide a reasonable prediction of thermal hydraulic phenomena that may be encountered during the nuclear power plant system transient events such as LOCA.

Acknowledgment

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