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ATLAS Steam Generator Tube Rupture Accident Simulation using SPACE Code

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1. Introduction

The Korean nuclear industry is developing a thermal-hydraulic analysis code for safety analysis of pressurized water reactors (PWRs). The new code is called the **S**afety and **P**erformance **A**nalysis **C**ode for Nuclear Power Plants (SPACE). The SPACE code adopts advanced physical modeling of two-phase flows, mainly two-fluid, three-field models that comprise gas, continuous liquid, and droplet fields and has the capability to simulate 3D effects by the use of structured and/or non-structured meshes. The programming language for the SPACE code is C++ for object-oriented code architecture. The SPACE code will replace outdated vendor supplied codes and will be used for the safety analysis of operating PWRs and the design of advanced reactors.

In this paper, the SGTR (Steam Generator Tube Rupture) experiment data (double-ended guillotine break of a single U-tube was simulated) in ATLAS have been simulated using the SPACE code as part of the V&V work. The results were compared with those of experiments and other code simulations.

2. Facility and Test Description

2.1 Facility Description

The reference plant of ATLAS is the APR1400 (Advanced Power Reactor 1400 MWe), which has a rated thermal power of 4000 MW and a loop arrangement of 2 hot legs and 4 cold legs for the reactor coolant system. ATLAS also incorporates some specific design features of the Korean standard nuclear power plant, the OPR1000 (Optimized Power Reactor 1000 MWe), such as a cold-leg injection mode for a high pressure and a low pressure safety injection modes. ATLAS can be used to investigate the multiple responses between the systems for a whole plant or between the subcomponents in a specific system during anticipated transients and postulated accidents. Furthermore, ATLAS can be used to provide unique test data for the 2(hot legs) x 4(cold legs) reactor coolant system with a DVI of emergency core cooling (ECC); this will significantly expand the currently available data bases for code validation.

2.2 Test Description

The SGTR accident test in ATLAS was performed to simulate a double-ended rupture of a single U-tube at the hot side of an affected steam generator. In the present test, considering the safety analysis results for the SGTR accident of the APR1400, a reactor trip was assumed to occur by an increase in the steam generator level, i.e., a high steam generator level (HSGL) trip signal. In addition, a single-failure of a loss of a diesel generator, which results in the minimum safety injection flow to the RPV, was assumed to occur in concurrence with the reactor trip. Therefore, the safety injection water from the SIP was only available through the DVI-1 and -3 nozzles, and the safety injection water from the SIT was available through all of the DVI nozzles. Since the primary system pressure was maintained above the set-point of the SIT, 4.03 MPa during the present SGTR test period, the SIT water was not supplied.

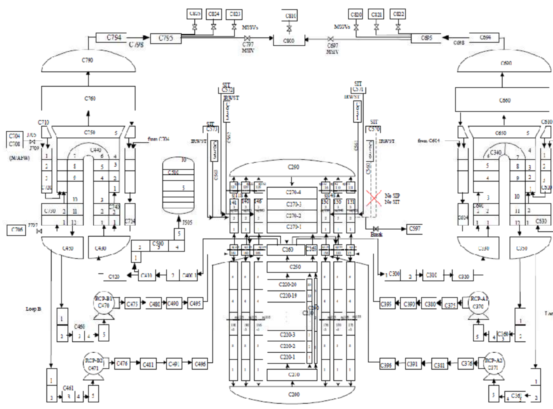

Figure 1. ATLAS Nodalization

The SGTR accident is one of the design basis accidents, which has a unique feature of penetration of the barrier between the reactor coolant system (RCS) and the secondary system resulting from failure of steam generator U-tubes. The SGTR has importance with respect to safety due to concern of a containment bypass of radioactive inventory. In the course of the SGTR, the radioactive coolant passing through broken steam generator U-tubes mixes with the shell-side water in the affected steam generator. The break flow from the ruptured U-tubes can increase the water level and the pressure of the affected steam generator. Following reactor and turbine trips, the main steam safety valves (MSSVs) can be opened to mitigate an increase in the secondary system pressure. Meanwhile, the SGTR can provide a direct flow path from the primary to the secondary systems, resulting in the release of fission products into the atmosphere.

It is generally known that the break flow rate from the primary to the secondary sides is the most important factor affecting the overall thermal-hydraulic behaviors such as the depressurization rate of the RCS, the water level increase and the pressurization rate of the secondary system, the consequent MSSV opening time, etc. The break flow rate through the broken U-tubes depends on the primary-to-secondary system differential pressure, the primary coolant subcooling, and the break area and the break location along the U-tubes.

3. SGTR Modeling using the SPACE code

First, the SPACE code deck was made using the RELAP5 code deck of SGTR for SPACE code capability evaluation. Generally, all initial conditions and assumptions used for experimental data were equally adapted to the SGTR SPACE input deck. Initial conditions are presented in Table 1

Table I : SGTR Initial Conditions

Parameter	Measured Predicted
Primary Coolant System	
Normal Power (Mwt)	1.631
Pzr pressure (MPa)	15.39
Core inlet temperature (C)	289.9
Core outlet temperature (C)	324.7
Cold leg flow (kg/s)	2.24
Hot leg temperature (C)	325.1
Cold Leg temperature (C)	291.9
Secondary System	
Steam flow rate (Kg/s)	0.397
Feed water flow rate (kg/s)	0.419
Steam pressure (MPa)	7.83

A comparison of the calculated results with experiments for SGTR accident in ATLAS is presented from Fig. 6 to Fig. 7.

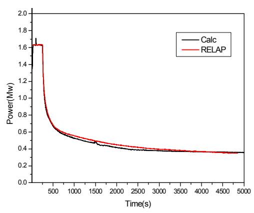

Figure 2. Core Power

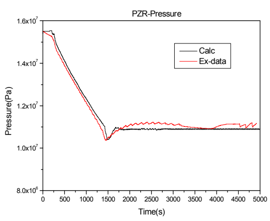

Figure 3. Pzr Pressure

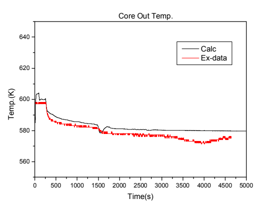

Figure 4. Core In Temp.

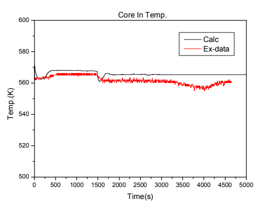

Figure 5. . Core Out Temp.

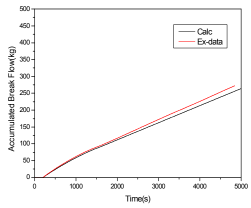

Figure 6. Accumulated Break Flow

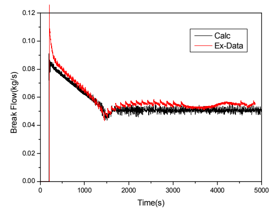

Figure 7. Break Flow.

3. Conclusions

The Korea nuclear industry has been developing the SPACE code for safety analysis and design of PWRs. The SGTR accident experiment in ATLAS has been simulated for V&V of the SPACE code. The results have been compared with those of an experiment.

Through evaluation of the SGTR experiment in ATLAS using the SPACE code, it is concluded that the SPACE code has capability to predict the system response.

Acknowledgements

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