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Evaluation of the SPACE code using the SB-LOCA in Zion unit 1

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1. Introduction

The SPACE (Safety & Performance Analysis Code for Nuclear Power Plants) has been developing by KHNP through the cooperation with KEPCO E&C and KAERI[1]. Unlike existing safety analysis code, the SPACE is based on the three-field governing equations (vapor, continuous liquid and droplet). It improves the accuracy by solving the mass, energy and momentum conservation equations for each phase and adopt the proven numerical methods as well as the models for various thermals hydraulic phenomena.

To evaluate the SPACE code analysis capability for the transient phenomena, a SB-LOCA in Zion unit 1 was simulated and compared with RELAP5 code as a reference.

2. Modeling

The Zion unit 1 plant is a Westinghouse 4-loop 3,400 MWth PWR and consisted of four steam generators, four RCPs and one pressurizer as in Fig. 1.

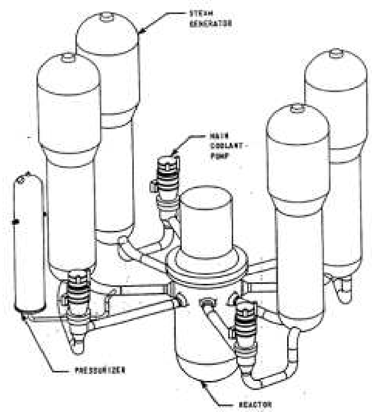

Fig. 1. Westinghouse 4-loop NSSS

In this study, the 2% cold leg break was modeled under the loss of offsite power condition.

SPACE model for Zion unit 1 is prepared on the basis of the RELAP5 model[2] and nodalization diagram is shown in Fig. 2. The model contains 125 fluid cells, 142 faces and 83 heat structures. Two primary coolant loops were modeled. One loop, called the broken loop, represented a single primary coolant loop. The break was modeled in the pump discharge piping of the broken loop cold leg. The other loop, called the intact loop, represents three primary coolant loops lumped as one. The pressurizer was connected to the intact loop. The intact and broken loops were modeled symmetrically except for differences due to the existence of the break and pressurizer.

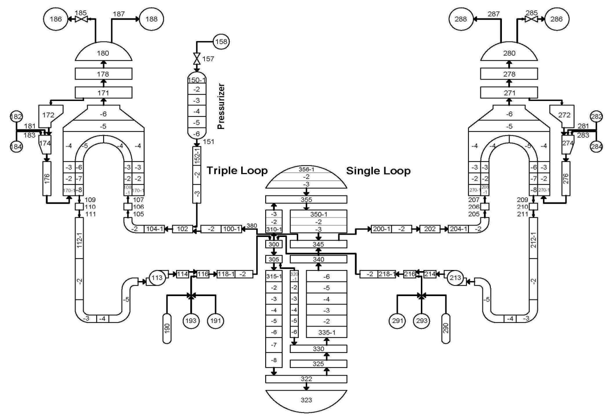

Fig. 2. Nodalization for the Zion unit 1 PWR

3. Analysis

The transient behavior was calculated with the SPACE code and the results were compared those of RELAP5 (MOD3.3) code. The transient was initiated by an instantaneous opening of the break and performed from 0 to 200 seconds. This was accomplished by using a trip valve, which was open for times greater than 0.01 seconds. A scram signal was generated when the pressurizer pressure decreased to 12.82 MPa (1860 psia). Scram occurred 3.4 seconds after the scram signal was generated. The reactor coolant pumps began to coast down simultaneously with the scram signal. Valves in the steam generator feed water and steam lines began to close simultaneously with the scram signal. The steam line and feedwater valves were linearly closed within 1 second and 10 seconds, respectively, following the scram signal. Safety injection and charging began 5 seconds after the pressurizer pressure decreased to 12.62 MPa (1830 psia). Secondary side auxiliary feedwater flow was initiated 14 seconds after the scram signal was generated. Automatic control of the feedwater flow based on steam generator downcomer liquid level was simulated.

Fig. 3. shows the pressure in the upper plenum (Volume 345) and is representative of the primary system pressure.

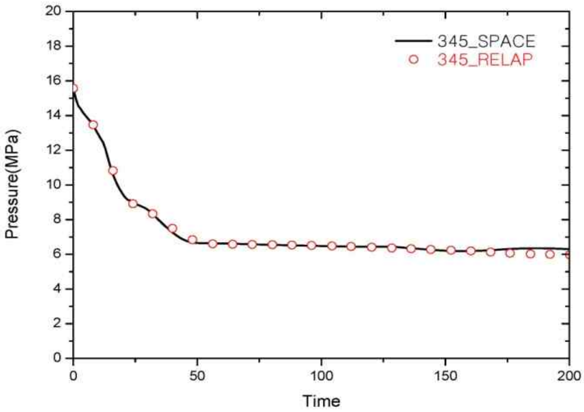

Fig. 3. Primary System Pressure of Core Outlet

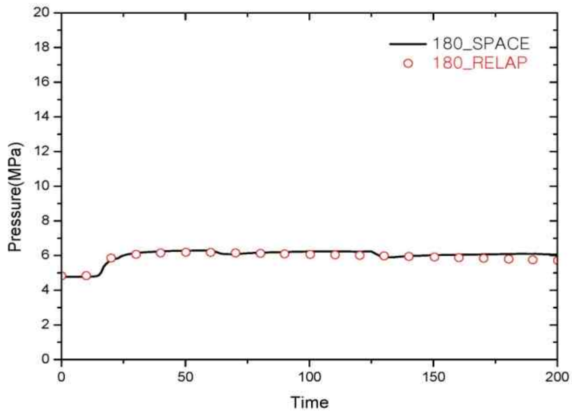

Fig. 4. Steam Generator System Pressure of Intact Loop

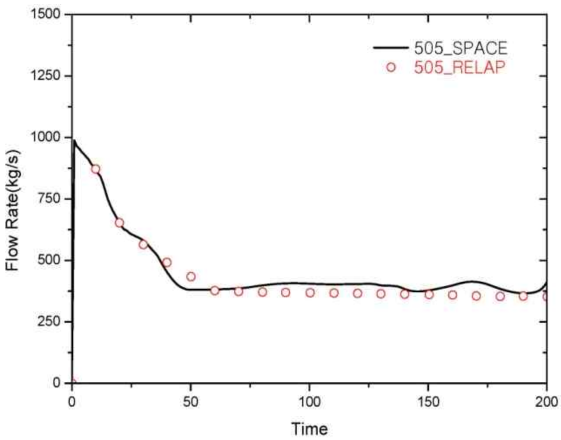

Fig. 5. Mass Flow Rate at Small Break

The rapid decrease of the pressure is presented from 0 to 50 seconds due to the generated break. Energy removed at the break and by the thermal release of steam generators, the core decay heat and stored energy in the vessel heat structures and the system pressure decreased. The system pressure was driven by the break mass flow rate and choking at the break occurred in both calculations as show in Fig. 5. Two code calculations used the H-F critical flow model (Cd 1.0) was used to model critical flow. At about 50 seconds a steam generator to reach a quasi-equilibrium conditions and the discharge flow rate is decreased (Fig. 4.).

Fig. 6. shows the speed of RCP (reactor coolant pump). By reactor shutdown and loss of coolant, primary system pressure was reached for trip setpoint as 12.82 MPa and RCPs began coasting down.

Instant vaporization is occurred at the upper plenum and hot leg and steam generator, while the primary system pressure is reduced. The void fraction at the core outlet is shown in Fig. 7. This result indicates similar trend for the remaining liquid in the upper core volume.

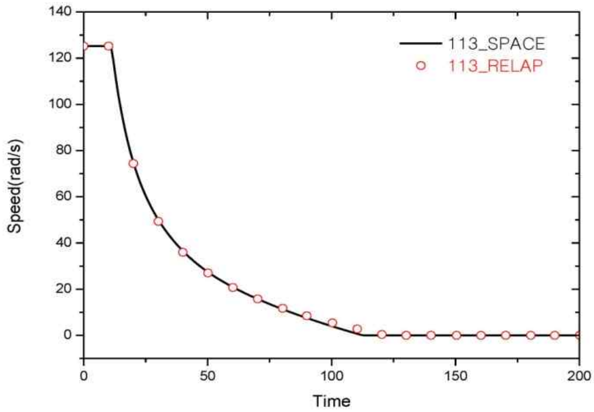

Fig. 6. Reactor Coolant Pump Speed of Intact Loop

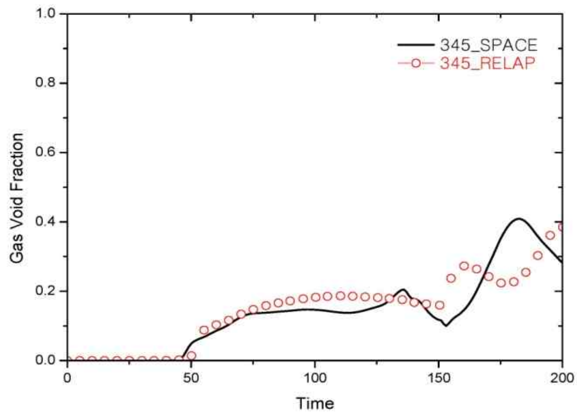

Fig. 7. Void Fraction of Core Outlet

4. Conclusions

The cold leg break SB-LOCA, was simulated using the SPACE code and the results were compared those of RELAP5 code. Through the evaluation, it was concluded that the SPACE code could effectively simulate SB-LOCA accidents.

REFERENCES

[1] SPACE 2.14 Users Manual, KHNP, 2013

[2] NRC, RELAP5/MOD3.3 CODE MANUAL VOLUME Ⅲ : DEVELOPMENTAL ASSESSMENT PROBLEMS, 2001