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Evaluation of SPACE code for simulation of reactor scram due to unplanned loss of RCP power in OPR 1000

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1. Introduction

SPACE (Safety & Performance Analysis Code for Nuclear Power Plants) has been developing by KHNP with the cooperation with KEPCO E&C and KAERI. SPACE code is expected to be applied to the safety analysis for LOCA (Loss of Coolant Accident) and Non-LOCA scenarios. SPACE code solves two-fluid, three-field governing equations and programmed with C++ computer language using object-oriented concepts [1]. To evaluate the analysis capability for the transient phenomena in the actual nuclear power plant, the reactor trip accident due to the loss of RCP power in OPR1000 (Ulchin unit 4) was simulated with SPACE code.

2. Analysis model

2.1 OPR1000 Condition

To evaluate the analysis capability of the SPACE code for natural circulation phenomena in the actual nuclear power plant, the unplanned reactor trip due to the loss of RCP power in Ulchin unit 4 in 11th Sep. 2000 was analyzed. The initiation event of transient was the failure of a lead in current transformer. Sequential RCP trip was followed by the opening of the breaker. The reactor trip signal was occurred through the low DNBR signal [2].

2.2 SPACE Model

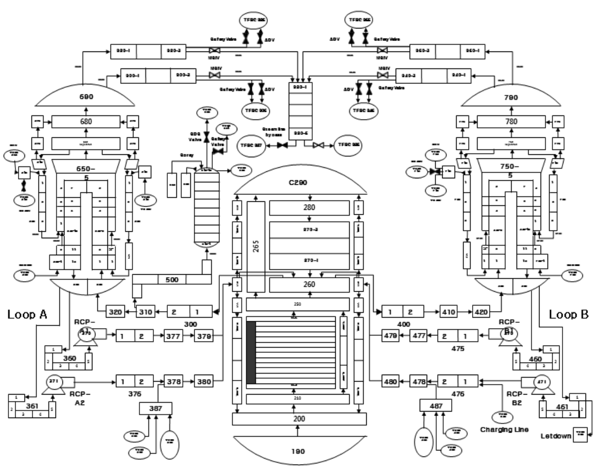

Fig. 1 Nodalization diagram of OPR1000

SPACE model for OPR1000 plant is prepared on the basis of the MARS input model [3] and RETRAN-3D input model [4]. The 1.41 version of SPACE code is used in the analysis [5]. The nodalization diagram of OPR1000 is depicted in Fig. 1. The reactor core is modeled with 12 heat structures and 14 fluid cells. Boundary conditions for the feed water were modeled using control function of SPACE.

3. Analysis results

3.1 Steady-state condition

For steady-state condition, a flow control function for RCP and a proportional-integral control function for feed water are used. The calculation for steady-state condition is performed for 1000 seconds. The calculation results are compared with those of steady-state condition of RETRAN for Ulchin unit 4. The comparisons for the major variables are shown in Table 1. The steady-state condition of SPACE shows good agreement with the RETRAN condition.

Table 1. Comparison of steady-state condition

Plant Parameter	RETRAN	SPACE	Error[%]
Core power [MW_t]	2815	2815	0.0
Core shroud (bypass) flow [kg/s]	36.5	36.5	0.0
Core flow [kg/s]	14945	14945	0.0
Hot leg flow rate [kg/s]	7648.8	7648.7	0.0
Cold leg flow rate [kg/s]	3824.4	3824.4	0.0
Hot leg temperature [K]	600.48	599.94	-0.09
Cold leg temperature [K]	568.98	568.49	-0.09
Pressurizer pressure [bar]	157.6	157.6	0.0
Pressurizer water level [%]	51.5	51.22	-0.53
Downcomer FW flow rate [kg/s]	80.3	80.30	0.0
Economizer FW flow rate [kg/s]	721.02	718.45	-0.4
Steam flow rate [kg/s]	801.3	797.14	-0.52
Steam pressure [bar]	73.7739	73.852	0.11
SG recirculation ratio	3.7	3.6735	-0.7
SG wide range water level [%]	74.1	76.328	3.0

3.2 Transient analysis results

The transient calculation is performed for 575~1200 seconds. The RCP trip transient started at 575 seconds and the log data of Ulchin 4 ended at 1200 seconds.

Major sequence of events for the transient is tabulated in Table 3. The major sequence of events is well predicted in SPACE calculations.

Table 2. Major sequence of events

Event	Time [sec]		
	RETRAN	SPACE	Error
RCP trip (Initiation of event)	575.0	575.0	0.0
RCS low flow signal	578.99	578.98	0.01
Reactor trip due to lo RCS flow	579.70	579.68	0.02
Feed Water (ECO) valve close	579.70	579.69	0.02
Turbine Stop Valve close	588.80	588.69	0.11

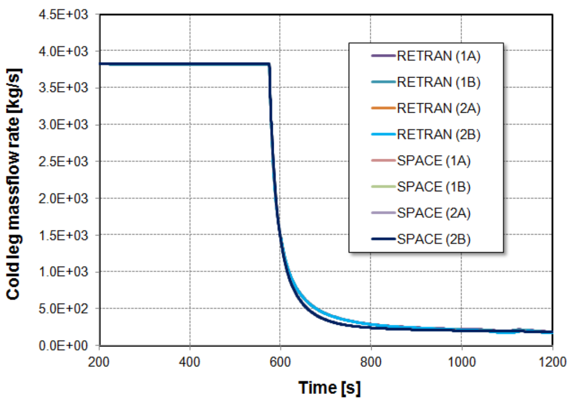

Fig. 2 Mass flow rate of cold leg

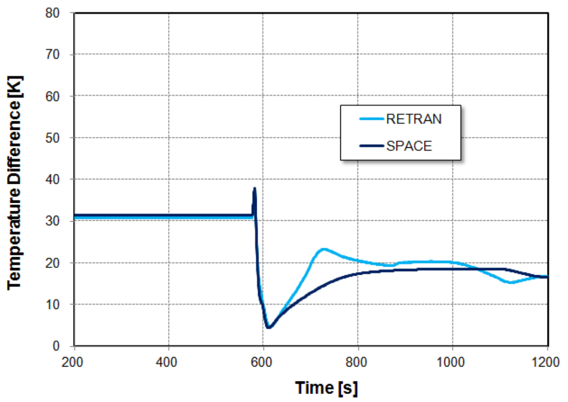

Fig. 3 Temperature difference between hot leg and cold leg

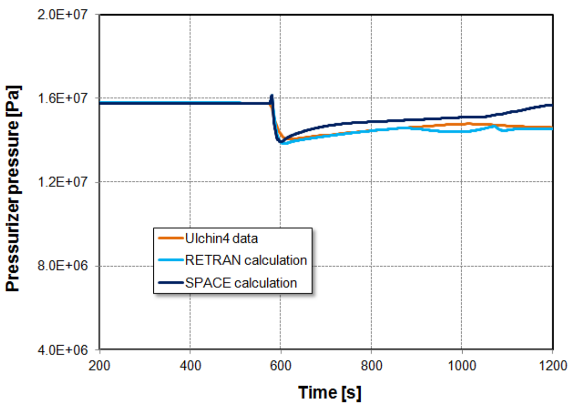

Fig. 4 Transient pressure behavior of pressurizer

The coastdown of RCP is well predicted when compared to RETRAN results as depicted in Fig. 2. The slight difference is caused by the temperature difference between hot leg and cold leg as shown in Fig. 3. The abrupt depressurization of the pressurizer due to the reactor scram and the gradual increase of pressure due to decrease of heat removal of secondary side are well estimated as presented in Fig. 4 when compared to the measured data and RETRAN calculation.

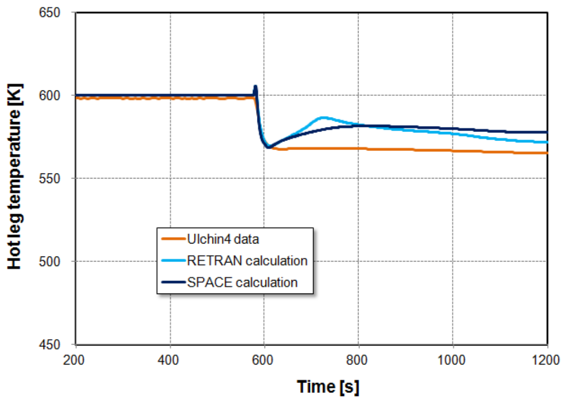

Fig. 5 Transient temperature behavior of hot leg

The under-estimation of primary-secondary heat transfer rate is considered as the main cause of the over-estimation of hot leg temperature as shown in Fig. 5.

4. Conclusions

To evaluate the analysis capability of SPACE code in the actual nuclear power plant, the reactor trip accident due to the loss of RCP power was simulated with SPACE code. The steady-state condition of SPACE input model shows good agreement with the reference code (RETRAN) condition. The transient calculation is performed for 575~1200 seconds. The major sequence of events is well predicted in SPACE calculations. The primary mass flow rate shows good agreement with the reference value. When compared to the measured data, the SPACE calculation shows physically valid results. A sensitivity study on the heat transfer rate and the initial water level of steam generators could be performed as a further work.

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