OPR1000 CEA Withdrawal at Power Accident Analysis using the SPACE code

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1. Introduction

The Korea nuclear industry has developed a best-estimated two-phase three-filed thermal-hydraulic analysis code, SPACE (Safety and Performance Analysis Code for Nuclear Power Plants), for safety analysis and design of a PWR (Pressurized Water Reactor). As the first phase, the demo version of the SPACE code was released in March 2010. The code has been Verification (V&V) matrix prepared for the SPACE code as the second phase of the development.

In this study, CEA withdrawal at power accident has been simulated using the SPACE code as one aspect of the V&V work. The results from this simulation were compared with results of the RETRAN code which was used to approval of methodology for safety analysis of OPR1000 from regulatory committee.

2. Analysis Method

2.1 CEA withdrawal at power event Description

To begin with, CEA withdrawal at power accident is classified as an ANS condition II event. An uncontrolled sequential withdrawal of CEAs is assumed to occur as a result of a single failure in the Control Element Drive Mechanism Control System (CEDMCS), Reactor Regulating System (RRS), or as a result of operator error.

The withdrawal of CEAs causes a positive reactivity change, resulting in an increase in the core power and heat flux. As a consequence, the reactor coolant temperature and pressurizer pressure increase. After initiation of the transient, a reactor trip on CPC VOPT is actuated. A few minutes later, the trip breakers are opened. Also at this time, the turbine is assumed to trip resulting in an instantaneous loss of offsite power. Subsequently, the CEAs begin dropping into the core and terminate the transient. The minimum DNBR reached during the transient is well above 1. The maximum planar radial peaking factor occurs in the region of the axial power peak and the peak linear heat generation rate during the transient remains below 21 kW/ft (68.9 kW/m)

2.2 Analysis Method

The standard nodalization OPR1000 is as shown in Figure 1. The primary side nodalization includes 12-node reactor core section, 2 steam generators, 2 hotlegs, 4 coldlegs, 4 RCPs (Reactor Coolant Pump) and a pressurizer. The secondary side model includes multimode steam generators, 4 main steam lines, MSSVs (Main Steam Safety Valve), and main/auxiliary feedwater.

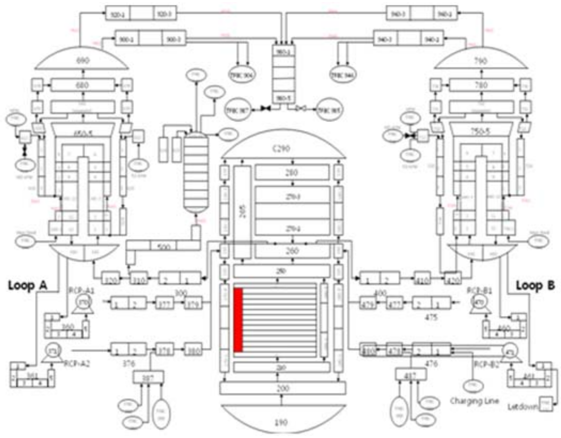

Fig. 1. SPACE Nodal Diagram for OPR1000 CEA Withdrawal At Power Accident.

The CEA withdrawal is modeled as reactivity insertion table in the form of time vs. reactivity. The RETRAN code calculates system parameters, such as core power, heat flux, RCS pressure, temperature, and the time of reactor trip, etc. Core heat flux, core inlet temperature, RCS pressure, core flow rate are calculated by RETRAN code.

3. Analysis Results

3.1 Initial Conditions and Assumptions

At first, the SPACE code deck for calculation of CEA withdrawal at power accident was made using the initial conditions of the RETRAN code deck, which used the OPR1000 safety analysis project for SPACE code capability evaluation. All initial conditions and assumptions used in RETRAN code were equally adapted to CEA withdrawal at power accident SPACE input deck.

Initial conditions for the CEA withdrawal analysis are chosen to minimize initial DNBR. Thus, initial conditions and assumptions are as follows: maximum core power, maximum core inlet temperature, minimum RCS pressure, and minimum core flowrate. The reactivity parameters are chosen to maximize the rate of core power increase. Thus, minimum feedback (moderator feedback and Doppler feedback) coefficients, maximum rod withdrawal speed, maximum rod withdrawal reactivity insertion. The axial power shape for DNBR analysis is ASI=-0.3. The VOPT trip setpoint is assumed to be 115%.

Table I : Initial Conditions

Parameter	RETRAN	SPACE
Power level (MWt)	2871.3	2871.3
Core inlet temperature (K)	572.03	573.4
Mass flow rate(10^6 lbm/hr)	112	112
Pressurizer Pressure (psia)	2000	2000
Steam Generator Pressure (psia)	1132.7	1137.0
Axial shape index	-0.3	-0.3
Max. reactivity insertion rate($10^{-5}\Delta\rho$ /sec)	5.45	5.45
Moderator temperature coefficient($10^{-5}\Delta\rho$ /sec)	0.0	0.0

For the conservatism, the SPACE input deck for CEA withdrawal at power accident was not made at 100% power condition but at 102% power condition. The SPACE input deck for the CEA withdrawal at power accident was run from 0 seconds to 1000 seconds for steady-state confirmation.

3.2 Analysis Results

The CEA withdrawal at power accident transient is initiated by uncontrolled withdrawal of CEA bank. The resulting reactivity insertion leads to increase in core power as shown in Figure 2. As the core generates more heat than is removed by steam generator, the primary coolant temperature rises and coolant expands in volume, leading to increase in primary pressure. As core power reaches 115% (VOPT trip setpoint), reactor trip occurs. As control rods drop to core and shutdown reactivity is inserted, core power, RCS temperature and pressure begin to decrease, terminating the transient. The RETRAN results show reactor trip occurs similar time with SPACE code result.

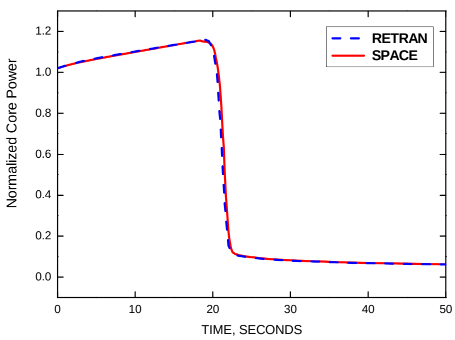

Fig. 2. Core power VS. Time

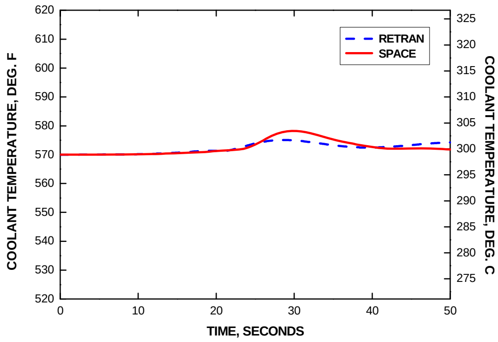

Fig. 3. Core inlet temperature

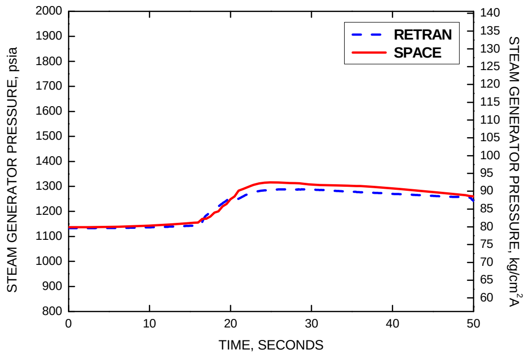

Fig. 4. SG pressure

3. Conclusions

The KNAP methodology is applied to OPR1000 CEA withdrawal at power accident analysis and the results are compared with those mentioned in OPR1000 results of RETRAN code. Although there is some difference in peak temperature and SG pressure, the results from RETRAN calculation show similar trends.

Through this evaluation of a OPR1000 CEA withdrawal at power accident analysis using the SPACE code, it is concluded that the SPACE code has the capability to predict the system response caused by a CEA withdrawal at power accident.

REFERENCES

[1] TR-KHNP-0009, "Topical Report on Non-LOCA Analysis Methodology, KNAP." Korea Hydro and Nuclear Power Co., 2004.
[2] TM.01NE19.M2005.712, "Development of the Unified Safety Analysis Code System-2005 Annual Report," C. K. Sung, et al., Korea Electric Power Research Institute, 2005.
[3] NP-7450, Vol. 1 & 3, "RETRAN-3D User's Mannual," M. P. Paulsen, et al., Electric Power Research Inst., 1996.
[4] NP-4498, Vol. 3, "The Reactor Analysis Support Package(RASP)," L. J. Agee, et al., Electric Power Research Inst., 1986.