Transactions of the Korean Nuclear Society Spring Meeting
Chuncheon, Korea, May 25-26 2006

APR1400 CEA Withdrawal at Power Accident Analysis using KNAP

Dong-Hyuk Lee, Chang-Keun Yang, Yo-Han Kim, Chang-Kyung Sung
Korea Electric Power Research Institute, 103-16, Munji-Dong, Yuseong-Gu, Daejeon, KOREA, 305-380
dhlee1@kepri.re.kr

1. Introduction

KEPRI (Korea Electric Power Research Institute) has been developing safety analysis methodology for non-LOCA (Loss Of Coolant Accident) analysis of OPR1000 (Optimized Power Reactor 1000, formerly KSNP). The new methodology, named KNAP (Korea Non-LOCA Analysis Package), uses RETRAN as the main system analysis code. RETRAN code is a non-LOCA safety analysis code developed by EPRI. The new methodology will replace existing CE (Combustion Engineering) supplied codes and methodologies currently used in non-LOCA analysis of OPR1000. In this paper, we apply KNAP methodology to APR1400 (Advanced Power Reactor 1400).

The CEA (Control Element Assembly) withdrawal at power accident is one of the "reactivity and power distribution anomalies" events and the results are typically described in the chapter 15.4.2 of SAR (Safety Analysis Report).

The APR1400 has been designed to generate 1,400MWe of electricity with advanced features for greatly enhanced safety and economic goals. The CEA withdrawal at power analysis in APR1400 SSAR (Standard Safety Analysis Report) is analyzed with CESEC-III computer code.

In this study, to confirm the applicability of the KNAP methodology and code system to APR1400, CEA withdrawal at power accident is analyzed using RETRAN code and it is compared with results from APR1400 SSAR.

2. Analysis method

2.1 Description of the transient

An uncontrolled sequential withdrawal of CEAs is assumed to occur as a result of a single failure in the CEDMCS (Control Element Drive Mechanism Control System), RRS (Reactor Regulating System), or as a result of operator error. The CEA withdrawal results in positive reactivity insertion to the core. This leads to increase in core power, leading to increased RCS (Reactor Coolant System) pressure and temperature. Increased core power and increase RCS temperature leads to decreased DNBR. The reactor trip occurs as a result of VOPT (Variable OverPower Trip). The transient is terminated as reactor trip occurs and core power and heat flux decreases.

2.2 Analysis method

The KNAP methodology is used to analyze CEA withdrawal at power transient. The main analysis code is RETRAN-3D. For DNBR calculation, CETOP-D code is used.

The standard nodalization of APR1400 is as shown in Figure 1. The primary side nodalization includes 6-node reactor core section, 2 steam generators, 2 hotlegs, 4 coldlegs, 4 RCPs (Reactor Coolant Pump) and a pressurizer. The secondary side model includes multi-node steam generators, 4 main steam lines, MSSVs (Main Steam Safety Valve), and main/auxiliary feedwater.

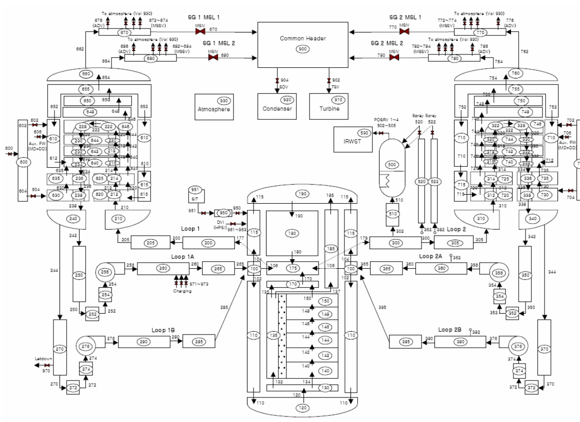

Figure 1. RETRAN Nodalization for APR1400

The CEA withdrawal is modeled as reactivity insertion table in the form of time vs. reactivity. The RETRAN-3D 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-3D and transferred to CETOP-D for DNBR calculation.

3. Analysis results

3.1 Initial conditions and assumptions

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 (30 inches/min), maximum rod withdrawal reactivity insertion ($3.95\times10^{-5}\Delta\rho$/sec). The axial power shape for DNBR analysis is ASI=-0.3. ROPM (Required Over Power Margin) of 118% is assumed.

The VOPT trip setpoint is assumed to be 115%. After reactor trip, LOOP (Loss of Offsite Power) is assumed with 3 sec. delay from time of turbine trip.

3.2 Analysis Results

The CEA withdrawal 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 (Figure 3). 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 about 5 seconds later than SSAR. And peak pressure occurs later and the value is smaller for RETRAN.

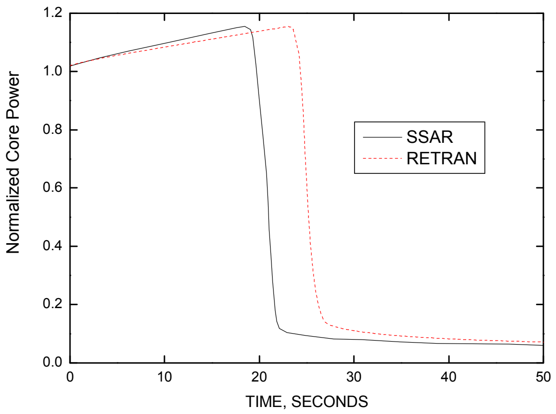

Figure 2. Core power vs time

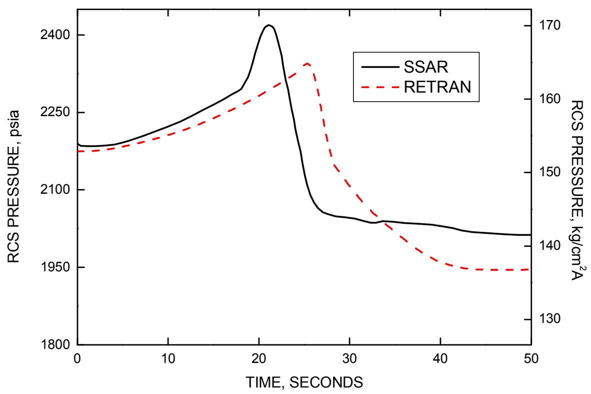

Figure 3. Pressurizer pressure vs time

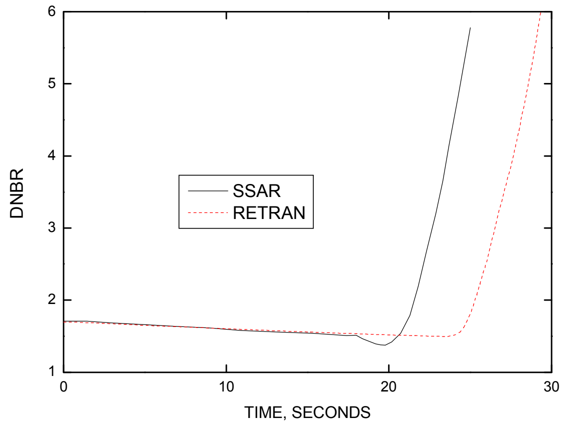

Figure 4. DNBR vs time

The DNBR is calculated by CETOP-D. The minimum DNBR also occurs later for RETRAN. The SSAR DNBR curve show a dip in DNBR around t=20sec. This is due to assumption of LOOP with 0-second delay for RCP trip. For RETRAN analysis, 3-second delay for LOOP is applied and this gives smooth DNBR curve. The minimum DNBR remains above 1.30 and safety criterion is met.

4. Conclusion

The KNAP methodology is applied to APR1400 CEA withdrawal at power analysis and the results are compared with those mentioned in APR1400 SSAR. Although there is some difference in reactor trip time, the results from RETRAN calculation and SSAR (using CESEC-III) show similar trends.

The maximum RCS and secondary pressures do not exceed 110% of design pressure and the minimum DNBR remains above 1.30. So the safety criteria for CEA withdrawal at power is met.

This analysis supports the applicability of KNAP methodology for APR1400.

Acknowledgements

This study is funded by the Ministry of Commerce, Industry & Energy (MOCIE).

REFERENCES

[1] M.P.Paulsen et al, “RETRAN-3D – a Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems,” EPRI, July 2001.

[2] CESEC-III manual, “CESEC Digital Simulation of a Combustion Engineering Nuclear Steam Supply System,” LD-82-001, Enclosure 1-p, Combustion Engineering Inc., 1981.

[3] RETRAN Basedeck, “Generation of the RETRAN-3D Basedeck for APR1400,” FNC-TR-04-070 Rev.00, FNC Technology Co., 2004.

[4] APR1400 Standard Safety Analysis Report, Chapter 15, June, 2002.