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

System Response Analysis of LOCV for APR1400 Using KNAP

Yo-Han Kim, Chang-Kyung Sung, Hwang-Yong Jun
Korea Electric Power Research Institute, 103-16 Munji-dong Yuseong-gu, Daejeon, 305-380
johnkim@kepri.re.kr

1. Introduction

Korea Electric Power Research Institute (KEPRI) of Korea Electric Power Corporation (KEPCO) has been developed the non-loss-of-coolant accident (non-LOCA) analysis methodology, called as the Korea Non-LOCA Analysis Package (KNAP), for the typical Optimized Power Reactor 1000 (OPR1000) plants. Considering current licensing methodology conducted by ABB-CE, however, the KNAP could be applied to Advanced Power Reactor 1400 (APR1400). In spite of some difference in components of two plant types, there is a close resemblance between their nuclear steam supply systems (NSSS). So, in this study, the loss of condenser vacuum (LOCV) event were analyzed using KNAP for APR1400 to estimate the feasibility of the application and the results were compared with those given in APR1400 Standard Safety Analysis Report (SSAR), which were calculated using the CESEC-III code of ABB-CE. Through the study, it was concluded that the KNAP could be applicable to APR1400 on the view point of LOCV event.

2. Plant Modeling

Prior to analysis, the reactor coolant system (RCS) of object plants, APR1400, was modeled with several control volumes and junctions to simulate the event. The core was partitioned into 6 vertical control volumes and related hydraulic channels, respectively. And the point kinetics assumption was used to represent the reactor core. In the case of steam generators, tubes and secondary sides were modeled with 14 volumes, respectively, to represent the U-tube bundles and two feedwater-paths, economizer, downcomer, *etc*. Entire loops were modeled separately to ensure the capability of analyzing the loop asymmetry events. In fact, the RCS model for APR1400 was developed according to the standard model used in the KNAP.

3. LOCV Analysis

3.1 Descriptions

In chapter 15 of SSAR, 5 categories of accidents have been analyzed such as increase/decrease in heat removal by secondary system, decrease in reactor coolant flow rate, reactivity & power distribution anomalies, increase/ decrease in RCS inventory, and radioactive material release from a subsystem. The LOCV event is marked as the most severe accident in the second category, *i.e.*, decrease in heat removal by the secondary system. In SSAR, this event was prepared with CESEC-III code of ABB-CE.

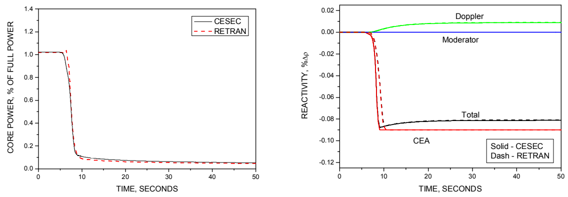
The LOCV is one of the events that can cause a turbine trip. For typical ABB-CE plants including OPR1000 and APR1400, the event is usually the limiting turbine trip event. That is, any other turbine trip events are bounded by the LOCV. The event may occur due to the failure of the circulating water system to supply cooling water, the failure of the main condenser evacuation system to remove non-condensable gases, or the excess in-leakage of air. In APR1400, the event will cause an automatic reactor trip and/or the activation of steam bypass system (SBS) or reactor power cutback system (RPCS) to reduce the excessive heat from primary system. In the licensing analyses, however, those systems are typically excluded for more conservative assumption. So, following the LOCV, the reactor will be tripped by the high pressurizer pressure signal.

The event is analyzed to assure that the RCS pressure does not exceed 110% of design under the transient ambiance and the minimum departure from nucleate boiling ratio (MDNBR) exceeds 1.35 in 95/95 confidence level. The peak linear heat rate should be below to 21 kW/ft.

3.2 Accident Analysis

To analyze the event, the immediate cessation of feedwater flow was assumed, and the turbine was assumed to trip immediately coincident with the beginning of the event. The primary side transient is caused by a decrease in heat removal from the primary to the secondary system, due to the swift termination of steam flow to the turbine, accompanied by a reduction of feedwater flow. The temperature and pressure (Fig. 2a) of the primary system were increased following the feedwater and the reactor trip signal was activated due to the pressurizer high pressure (Fig. 1). All the while, the main steam safety valves (MSSVs) repeated the open and close motion to reduce the steam pressure (Fig. 2b).

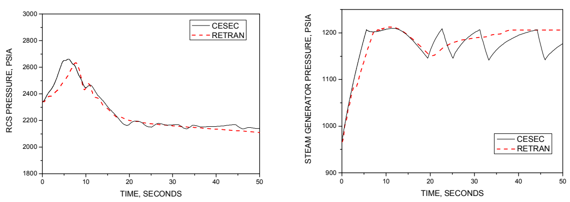
Figure 1 shows the power and reactivity trends during the transient following LOCV. As depicted in Fig. 1a, the high pressure reactor trip signal of RETRAN was delayed about 0.8 seconds compared to that mentioned in SSAR. It could be caused by the difference in pressure trend as depicted in Fig. 2a. The difference of pressure could be resulted from the pressurizer models of the codes, RETRAN and CESEC-III. The RETRAN adopt the comprehensive non-equilibrium model as the special component model for pressurizer, which is different from the general control volume model for that in the case of CESEC-III. And as the other reason of the difference, the multi-node approach of RETRAN model would be raised. The multi-node method tends to calculate more realistic values than those estimated by single node approach during the transient. And the propensity usually shows slower trends. In spite of the difference, the power or reactivity show the similar trends and the delay in trip could be resulted to more conservative state due to longer power period.



(a) Power (b) Reactivity

Figure 1. Power & Reactivity comparison

Figure 2 shows the pressurizer and steam generator secondary side pressure, respectively. The results of RETRAN show somewhat lower pressure trends than those mentioned in SSAR. As depicted in Fig. 2b, the secondary side pressure showed the saw-like trend due to the repeated open-and-close movement of MSSVs in the case of SSAR. In the case of RETRAN, however, the frequency is longer due to the migrated pressure propagation of multi-node secondary side model. On a standpoint of variation, however, they show the similar trends each other.



(a) RCS (b) SG Secondary Side

Figure 2. Pressure comparison

4. Conclusion

The LOCV event was analyzed to estimated the feasibility of the KNAP application to APR1400. The results of the analysis were compared with those mentioned in SSAR, which are calculated by CESEC-III code of ABB-CE. Through the feasibility study, it was concluded that the KNAP application showed the acceptable results and could be used further works.

Acknowledgements

This study has been going on under the funding of Korea Electric Power Corporation and Ministry of Commerce, Industry & Energy (MOCIE).

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] TM.S05.R1999.302, “LOCV Accident Analysis of KNGR with RETRAN,” Y. H. Kim and H. Y. Jun, Korea Electric Power Research Institute, 1999.

[4] NP-7450, Vol. 1 & 3, “RETRAN-3D User's Manual,” M. P. Paulsen, *et al*., Electric Power Research Inst., 1996.

[5] NP-4498, Vol. 3, “The Reactor Analysis Support Package (RASP),” L. J. Agee, *et al*., Electric Power Research Inst., 1986.