Comparative Analysis of SLB for APR1400 by using the SPACE code and CESEC-III

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ACKNOWLEDGMENTS

This work was supported by the Nuclear Research & Development of the Korea Institute of Energy Technology Evaluation and Planning (KETEP) grant funded by the Ministry of Knowledge Economy. (No. R-2006-1-243)

1. Introduction

Non-LOCA safety analysis methodology has been developed using SPACE code, designed to predict the thermal-hydraulic response of Nuclear Steam Supply System (NSSS) to the anticipated transients and the postulated accidents[1,2]. Since 2010, several nodalization schemes for the components such as steam generator and reactor vessel have been tested so far to find out optimized configuration for preliminary system model for Non-LOCA analysis. And the steady state initialization and a test to evaluate the code initialization capability were also performed [3].

A lot of effort is now being made to evaluate the applicability of the SPACE code to Non-LOCA analysis, by comparing the analysis results with those from the current licensing Non-LOCA system code, CESEC-III. The comparative simulations of Pressurizer Level Control System(PLCS) Malfunction and Loss of Condenser Vacuum(LOCV), which have been accomplished by U. S. Kim[4] and E. J. Lee[5], respectively, already showed that the SPACE code is applicable to the analysis of Non-LOCA events.

In this paper, detailed thermal hydraulic analyses for Steam Line Break(SLB) without loss of off-site power were performed using the SPACE code. The calculation results were also compared with those of the CESEC-III which is used for Advanced Power Reactor 1400(APR1400).

2. Event Analysis

2.1 Initial conditions and assumption

Double ended guillotine break is assumed to occur in the steam line upstream of Main Steam Isolation Valve (MSIV). The same initial conditions and assumptions are applied to the analysis using both the SPACE and CESEC-III codes. Initial core power is 3,983 Mwt. Initial reactor coolant flow rate, pressurizer and steam generator level, pressurizer pressure, feed water enthalpy are assumed to be at full power steady state condition. Conservative Doppler and moderator temperature coefficient are applied in terms of core power increase before reactor trip and return to power after reactor trip.

Reactor trip setpoints of low steam generator and variable over power are assumed to be 898 psia and 4,122.4 Mwt, respectively. Turbine admission valve closure and feedwater flow decrease are assumed to occur simultaneously at the time of reactor trip. Loss of offsite power and single failure are not assumed.

2.2 Description on SLB event

The pressure in the steam genernator adjacent to the break decreases due to steam line break, resulting in an increase in heat transfer from primary system to secondary system. Reactor can be tripped by low steam generator pressure, low primary system pressure, low steam generator level, variable over power, or low DNBR. MSIVs are closed due to low steam generator pressure. Eventually, the affected steam generator is isolated by interrupting main feed water supplied to the steam generators and closing all the MSIVs. The pressurizer pressure decrease to the Safety Injection Actuation Signal (SIAS) setpoint. Also, the affected steam generator level decreases and auxiliary feedwater (AFW) is supplied for decay heat removal.

2.3 Break locations

Fig. 1 shows the break locations for the SLB event in the SPACE code. The break locations are assumed for ignoring the friction and flow interference effect at the main steam line and common header like CESEC-III.

The break flow is discharged from intact and affected steam generators after event initiation. And then, the break flow is discharged from only one steam line of affected steam generator after Main Steam Isolation Signal (MSIS) by the low steam generator pressure.

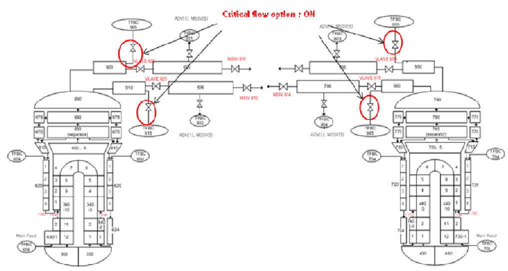

Fig. 1. Break locations

3. Analysis Results

Fig. 2 shows the comparison of core power variation between the CESEC-III and SPACE code analyses.

After break in steam line, excessive vapor discharged from the affected steam generator causes the decrease in reactor coolant temperature and steam generator level. Consequently, cold water reaches active core within about 3 or 4 second time delay, and then core power increases rapidly due to negative feedback effect of fuel and moderator. Both in the CESEC-Ш and SPACE code analyses, reactor trip occurs at about 10 seconds into the transient due to variable over power. Including the reactor trip time, the overall behavior of core power shows a good agreement between the two code analyses.

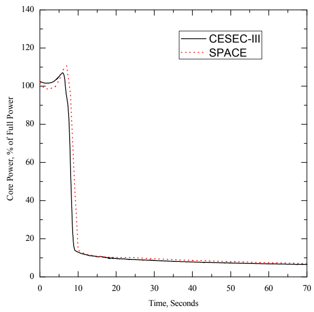

Fig. 2. Normalized core power

Fig. 3 shows the comparison of break discharge flow rate. Break flow is discharged from both the intact and affected steam generators through the steam pipes and the common header until MSIVs are closed at about 20 seconds by low steam generator pressure. During this period, the discharge flow is limited by the area of the flow restrictor installed a top of the steam generator dome. After MSIV closure, break flow is discharged only from the affected steam generator. The critical flows are calculated by Henry-Fauske/moody model and CRITCO correlation in SPACE code and CESEC-III, respectively. Even though the Henry-Fauske/moody model in the SPACE code shows a higher break flow than the CRITCO correlation, the overall tends of break flow agree well with each other.

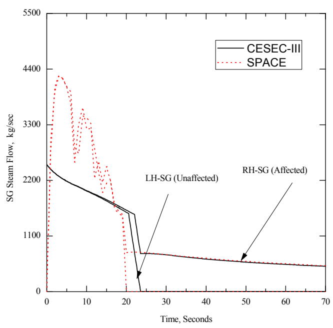

Fig. 3. Break discharge flow rate

Fig. 4 shows variations of pressurizer pressure. The break flow increases heat transfer from primary to secondary system, resulting in a decrease in primary system pressure. The reactor trip at about 10 seconds also contributes to decrease primary system pressure and temperature, but it is compensated by the immediate main steam isolation. The pressurizer pressure gradually decreases after 10 seconds of the accident both in the SPACE and CESEC-III analyses. The change of the depressurization rate seems due to the void formation in the upper head region. Apparently, the upper head behaves like an additional pressurizer, and contributes to mitigate the primary pressure decrease once void is formed in it. Upper head void starts to form at about 10 seconds in the CESEC-III, and void is also formed at the nearly same time in the SPACE analysis.

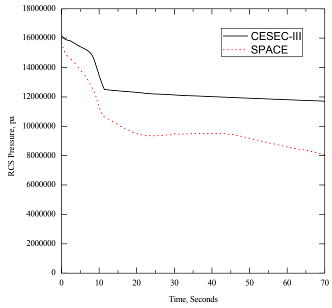

Fig. 4. Pressurizer pressure variation

4. Conclusions

In order to evaluate the SPACE code applicability for non-LOCA analysis, a comparative SLB simulation is performed for APR1400, using SPACE and CESEC-III codes. The accident analysis results from the SPACE code are reasonable not only in the qualitative but also in the quantitative aspect. The break discharge flow difference between the analyses is assumed to be due to the calculated correlation.

Based on the comparative analysis, it is concluded that SPACE code is applicable to the analysis of thermal hydraulic response to SLB accident. Moreover, the SPACE code is expected to be useful to find additional safety margin, with more realistic simulation of two phase flow and relevant phenomena.

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