Comparative Analysis of SLB for OPR1000 by using MEDUSA and CESEC-Ш Codes

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1. Introduction

The MEDUSA is a system thermal hydraulics code developed by Korea Power Engineering Company (KOPEC) for Non-LOCA and LOCA analysis, using two-fluid, three-field governing equations for two phase flow. The detailed descriptions for the MEDUSA code are given in Reference[1].

A lot of effort is now being made to investigate the applicability of the MEDUSA code especially to Non-LOCA analysis, by comparing the analysis results with those from the current licensing code, CESEC-III: The comparative simulations of Pressurizer Level Control System(PLCS) Malfunction and Feedwater Line Break(FLB), which have been accomplished by C.E.Park[2] and M.T.Oh[3], respectively, already showed that the MEDUSA 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 MEDUSA code. The calculation results were also compared with the CESEC-Ш**,** which is used for Optimized Power Reactor 1000(OPR1000), for the purpose of the code verification.

2. Event Analysis

2.1 Initinal conditions and assumptions.

Double ended guillotine break is assumed to occur in the upstream of MSIV.

The same the initial conditions and assumptions are applied to both the MEDUSA and CESEC-Ш codes. Initial core power is 2815 Mwt. Initial reactor coolant flow rate, pressurizer level, steam generator level, pressurizer pressure, feed water enthalpy are assumed to be at full power steady state condition. The most negative values are applied to the doppler reactivity coefficient and moderator temperature coefficient.

The setpoints of steam generator low pressure reactor trip and over power reactor trip are assumed to be 888.5 psia and 3104 Mwt, respectively. Turbine stop 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 SLB accident decreases pressure in the steam genernator adjacent to the break, resulting in an increase in heat transfer from primary system to secondary system. Reactor can be tripped by steam generator low pressure, primary system low pressure, steam generator low level, core over power, or low DNBR. The affected steam generator experiences a decrease in pressure. Later on, Main Steam Isolation Valves (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. The event decreases pressurizer pressure to Safety Injection Actuation Signal (SIAS) setpoint. Also, the affected steam generator level decreases and auxiliary feedwater (AFW) is supplied for decay the heat removal.

3. Analysis Results

3.1 Results

Fig. 1 shows the comparison of normalized core power variation between the CESEC-Ш and MEDUSA analyses. After break in steam line, excessive vapor discharge from the affected steam generator causes the reduction of reactor coolant temperature and steam generator level. Consequently cold water reaches active core with about 3 or 4 second time delay, and then core power increases rapidly. Both in the CESEC-Ш and MEDUSA analyses, reactor trip occurred at around 10 seconds into the transient due to over power. Including the reactor trip time, the overall behavior of core power shows a good agreement between the two code analyses.

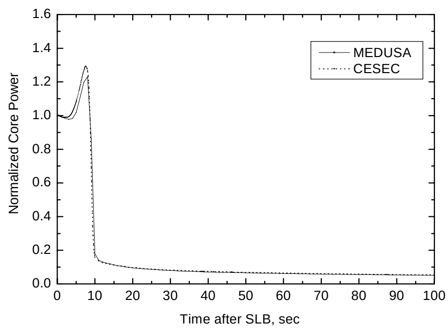

Figure 1. Normalized core power

Fig 2. 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 around 10 seconds by low steam generator pressure. During this period, the discharge flow is limited by throat area, 0.94 ft^2, of the flow restrictor installed at each steam pipe. After MSIV closure, break flow is discharged only from the affected steam generator. Even though critical flows are calculated by isentropic chocked flow model and CRICO correlation in MEDUSA and CESEC-III, respectively, the over all tends of break flow agree well with each other.

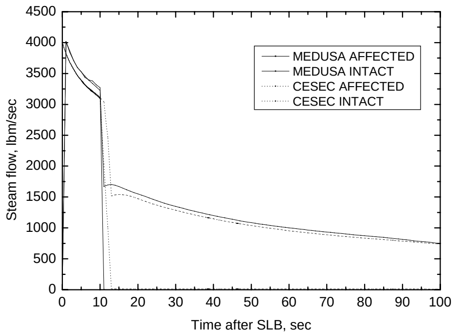

Figure 2. Break discharge flow rate

Fig. 3 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 around 10 seconds also contributes to decrease primary system pressure and temperature, but it is mostly compensated by the immediate main steam isolation. Thus, pressurizer pressure gradually decreases before 50 seconds of the accident both in the MEDUSA and CESEC-III analyses. However after 50 seconds, pressurizer pressure decreases more rapidly in the MEDUSA analysis. The reason of the pressure difference seems due to the over conservatism in the model of the reactor vessel upper head in CESEC-III. That is, the bypass flow from the upper downcomer to the upper head is not taken into account in CESEC-III. In addition, the initial temperature in the upper head volume is assumed to be the same as in the upper plenum. These conservative models result in early formation of void in the upper head: Upper head void starts to form at around 40 seconds in the CESEC-III, while void is formed at around 70 seconds in the MEDUSA analysis. Apparently, the upper head behaves like an additional pressurizer, and contributes to mitigate the primary pressure decrease once void is formed in it.

The variation of other major thermal hydraulic parameters such as hot leg temperature, cold leg temperature, steam generator pressure show a good agreement between the CESEC-III and MEDUSA analyses.

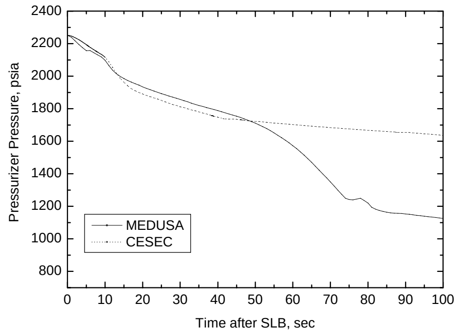

Figure 3. Pressurizer pressure

4. Conclusion

As a part of the MEDUSA code verification, a comparative SLB simulation is performed for OPR1000(UCN 3 and 4), using MEDUSA and CESEC-III codes. The accident analysis results from the MEDUSA are reasonable not only in the qualitative but also in the quantitative aspect. The pressure difference between the comparative analyses is found to be due to the over conservatism in the model of the reactor vessel upper head in CESEC-III. As a result, it is concluded that MEDUSA is applicable to the analysis of thermal hydraulic response to SLB accident. Moreover, the MEDUSA code is expected to be useful to find additional safety margin, with more realistic simulation of two phase flow and relevant phenomena.

REFERENCES

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