Comparison of the PLCS Malfunction Event Analyses by using the CESEC III and MEDUSA Codes

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1. Introduction

The MEDUSA is a code developed by Korea Power Engineering Company (KOPEC) for non-LOCA and LOCA analysis, providing two-fluid, three-field representation of two-phase flow [1]. In this paper, as an effort to verify the MEDUSA code, comparative simulation for Pressurizer Level Control System (PLCS) malfunction with loss of off-site power is performed for UCN 3 & 4 plants. The results by the MEDUSA are compared with those calculated by the CESEC-III, a licensing analysis code used for Korean Standard Nuclear Power Plant (KSNP).

2. Analysis Methodology and Results

2.1 Difference between MEDUSA and CESEC III codes

The CESEC-III code predicts the plant response for non-LOCA initiating events at a wide range of operating conditions. The code, which numerically integrates the one-dimensional conservation equations, assumes a homogeneous equilibrium node/flow-path network to model the Nuclear Steam Supply System (NSSS).

The MEDUSA computer code solves the compressible three dimensional, two-fluid, three-field equations for two-phase flow. The three fields are the vapor field, the continuous liquid field, and the liquid drop field. The conservation equations for each of the three fields are solved using a semi-implicit finite-difference numerical technique. The MEDUSA permits the user to nodalize the wide variety of geometries encountered in nuclear reactor system, using the concept of section, channel and gap.

The detailed descriptions for the MEDUSA code are given in Reference [1].

2.2 Description on PLCS Malfunction with Loss of Off-site Power

When in the automatic mode, the PLCS responds to changes in pressurizer level by changing charging and letdown flows to maintain the programmed level. If the pressurizer level controller fails low or the level setpoint generated by the reactor regulating system fails high, a low-level error signal can be transmitted to the controller. In response, the controller will start all the available charging pumps and close the letdown control valve to its minimum opening, resulting in the maximum mass addition to the Reactor Coolant System (RCS). The PLCS malfunction causes a reactor trip, on high pressurizer pressure, resulting in the maximum RCS pressure in the first 2 to 5 seconds following reactor trip. The loss of off-site power is basically assumed in this event analysis. It results in loss of power to the reactor coolant pumps, the pressurizer pressure control system (PPCS) and PLCS, the reactor regulating system, the feedwater control system, and the steam bypass control system. The unavailability of pumps and control systems by loss of power aggravates further the RCS pressure increase.

2.3 Results

Failure of the PLCS causes an increase in reactor coolant system inventory initiated by the startup of the third charging pump coupled with the decrease in letdown flow to its minimum.

An increase in RCS inventory results in a pressurizer pressure increase to the reactor trip analysis setpoint. Since the steam bypass control system is unavailable, and the rate of closure of the turbine stop valves is faster than the rate of control rod insertion, pressurizer pressure increases rapidly. After a short time, pressurizer pressure reaches the pressurizer safety valves (PSVs) opening setpoint, and pressure decreases by PSVs opening. As shown in Figure 1, the time variation of pressure is in good agreement with CESEC-III prediction.

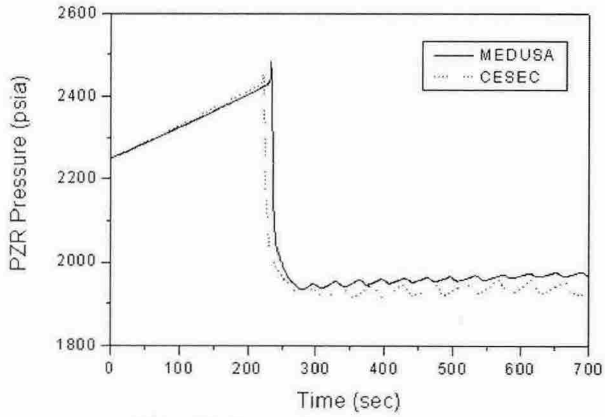

Figure 1. Pressurizer pressure

The slight delay of reactor trip is mainly due to the difference of pressurizer model between the MEDUSA and CESEC-III codes. It seems that the difference of the condensation model of each code affects the increasing rate of pressurizer pressure, but the difference is small enough to neglect.

Figure 2 shows steam generator pressure. The unavailability of the steam bypass valves causes the steam generator pressure to increase, causing the main steam safety valves to open. The decreasing core power and the safety valves function to limit the steam generator pressure. The steam generator level decrease due to the turbine trip and the interruption of the feedwater flow is shown in Figure 3.

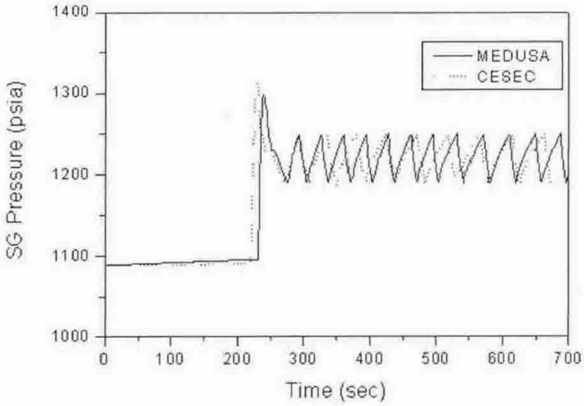

Figure 2. Steam generator pressure

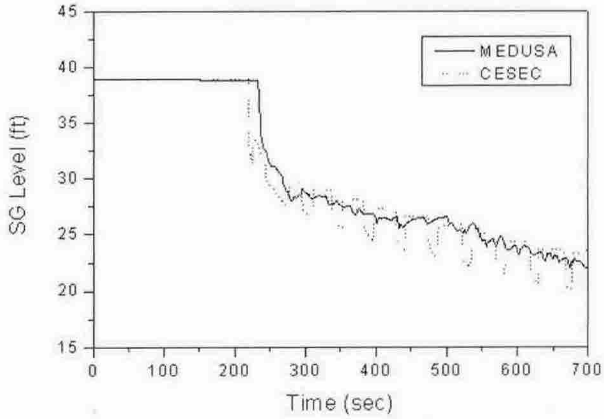

Figure 3. Steam generator level

3. Conclusion

A comparative simulation by the CESEC-III and MEDUSA computer codes for PLCS malfunction with loss of off-site power is performed for UCN 3 and 4. As shown at the above results, the predictions of the transient behavior by the MEDUSA show a good agreement with those by the CESEC-III simulation. Based on this, it can be concluded that the MEDUSA code is applicable to the analysis of thermal hydraulic response to PLCS malfunction accident.

However, in order that the MEDUSA is fully verified as a system analysis code, more intensive study including comparison of its predictions on wide variety of LOCA and non-LOCA events with the results from licensing codes or experiments are still necessary.

REFERENCES

[1] C. E. Park, MEDUSA and its Application to Non-LOCA Analysis, 2nd Safety Analysis Symposium, 2004.

[2] M. J. Thurgood, et. al., "COBRA/TRAC - A Thermal-Hydraulics Code for Transient Analysis of Nuclear Reactor Vessels and Primary Coolant Systems", NUREG/CR-3046, 1982.