Analysis of a Loss of Condenser Vacuum Event using the SPACE code

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1. Introduction

Since 2010, non-LOCA safety analysis methodology has been developed using SPACE code, being developed to predict the thermal-hydraulic responses of Nuclear Steam Supply System (NSSS) to the anticipated transients and the postulated accidents [1, 2]. Several nodalization schemes for the components such as steam generator and reactor vessel have been tested so far to find out an optimized configuration leading to a preliminary system model for non-LOCA analysis.

In this paper, detailed thermal hydraulic analyses for loss of condenser vacuum (LOCV) with loss of off-site power for the Shin-Kori units 3 and 4 (SKN 3&4) were performed using the SPACE code to evaluate the simulation capability of the SPACE code for the RCS pressurization events. The calculation results were compared with those of CESEC-III code which is the current licensing system code for non-LOCA.

2. Methods and Results

2.1 Input Parameters and Initial Condition

A LOCV may occur due to the failure of the circulating water system to supply cooling water, failure of the main condenser evacuation system to remove noncondensible gases, or excessive in-leakage of air. Immediate cessation of feedwater flow is assumed, and the turbine is assumed to trip immediately coincident with the cause for the loss of condenser vacuum.

The major initial plant condition parameters are pressurizer pressure and level, reactor coolant flow rate, core inlet temperature, and steam generator level. Each initial parameter initialization is preformed as following: The initial steam generator level, pressurizer pressure and pressurizer level are initialized through NSSS control systems and the reactor coolant flow rate is initialized by controlling the geometric K-factor in reactor coolant system. The core inlet temperature is also initialized by controlling the secondary pressure.

Table I: Initial Conditions for LOCV[3]

Parameter	Value
Core Power Level, MWt	4062.66
Core Inlet Coolant Temperature, °C (°F)	287.8 (550)
Core Mass Flow, 10^6 kg/hr (10^6 lbm/hr)	73.3 (161.6)
Pressurizer Pressure, kg/cm^{2}A (psia)	152.92 (2,175)
Pressurizer Water Level, %	21.0
Steam Generator Water Level, %WR	65.0

Table I provides the initial conditions used for LOCV and Figure 1 shows the nodalization for LOCV.

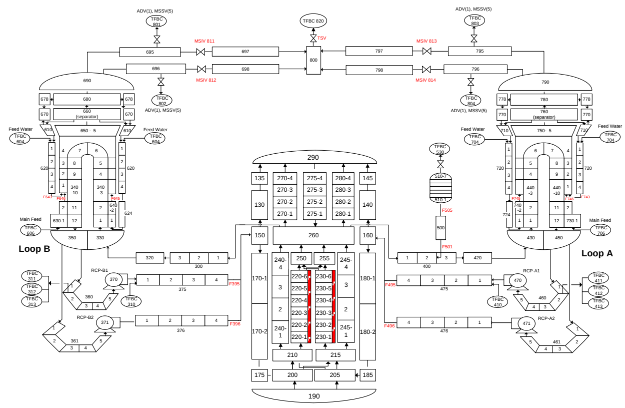

Fig. 1. Nodalization for LOCV

2.2 Results

The results of the initialization test run using control systems, such as pressurizer level control system (PLCS) and feed water control system (FWCS), are presented in Figures 2 and 3.

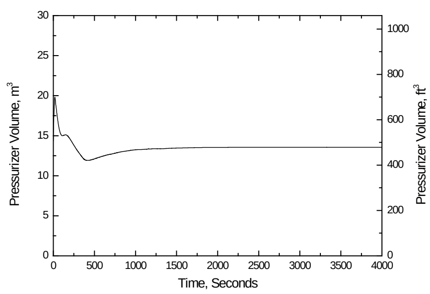

Fig. 2. Pressurizer volume initialization

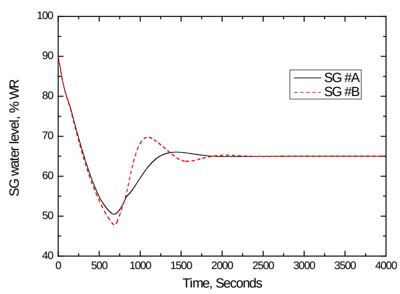

Fig. 3. Steam generation level initialization

The comparison of the dynamic behavior of important NSSS parameters between SPACE and CESEC-III code is presented in Figures 4 through 6.

Sudden reduction of steam flow, caused by the LOCV, leads to a reduction of the primary-to-secondary heat transfer. The moderator reactivity is constant prior to reactor trip due to the assumed zero moderator temperature coefficient (MTC), even though the average core temperature increases from the initial conditions to maximize the reactor coolant system (RCS) pressurization. The pressurizer pilot operated safety and relief valve (POSRV) opens and the maximum RCS pressure is reached. The RCS pressure then decrease rapidly due to the combined effects of reactor trip and opening of primary and secondary safety valves. Auxiliary feedwater automatically begins when the low steam generator level (LSGL) signal is on.

The analysis results agree well with those of SKN 3&4 FSAR with CESEC-III code.

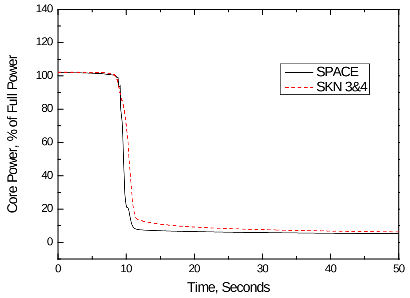

Fig. 4. Core power vs. time

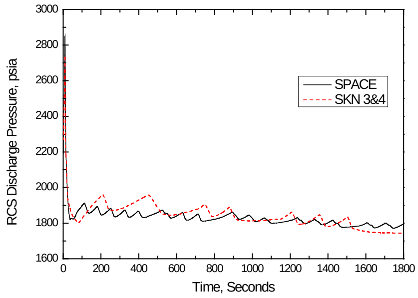

Fig. 5. RCS pressure vs. time

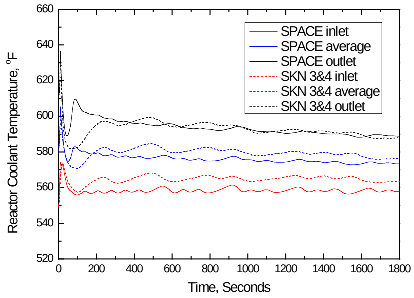

Fig. 6. RCS temperature vs. time

3. Conclusions

LOCV with a loss of offsite power with some delay after turbine trip is calculated using the SPACE code. The analysis results show good agreement to those of SKN 3&4 FSAR with CESEC-III code. This study shows that the SPACE code has sufficient capability to simulate the non-LOCA safety analysis resulting in the RCS pressurization.

Acknowledgment

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REFERENCES

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