Transactions of the Korean Nuclear Society Spring Meeting
Jeju, Korea, May 18-19, 2017

Evaluation of Load Rejection Test of SKN 3 using KISPAC System Analysis

Kim, Eung Seo[a*], Lee, Ju Han[a], Song, In Ho[a], Lee, Gyu Cheon[a], Joo, Bong Soo[b], Bahn, Byeong Hoon[b]
[a]*KEPCO E&C Co., 111, Daedeok-daero 989, Yuseong-gu, Daejeon, 34057*
[b]*KHNP Co., 1655, Bulkuk-ro, Janghang-ri, Yangbuk-myeon, Gyeongju-si, Gyeongsangbuk-do, 38120*
**Corresponding author:eungseo.k@kepco-enc.com*

1. Introduction

The Nuclear Steam Supply System (NSSS) control systems are used for control of plant parameters within control ranges during normal power operation and design operational transients without actuation of reactor trip signals.

The KEPCO E&C Integrated Systems Performance Analysis Code (KISPAC) has been used for simulation of design transients and design of NSSS control systems for APR1400 [1]. The adequacy of the control system design is verified by evaluating the initial startup test results of the as-built plant [2].

In this study, the power ascension test results of Shin-Kori Nuclear Power Plant Unit 3 (SKN 3), the first APR1400 plant, are compared with the output of KISPAC code simulation to evaluate the simulation capability of this computer code as well as to verify the effectiveness of the APR1400 control system design which incorporates improvement compared with design of previous plant.

2. Computer Code and Test Description

2.1 KISPAC Computer Code

The KISPAC computer code is a best-estimate simulation tool developed to evaluate the Performance Related Design Bases Events (PRDBEs) and is used for overpressure protection and natural circulation cooldown analyses for APR1400 [3].

The KISPAC code uses a detailed node and flow path methodology to model the transient behavior of the fluid systems and components of the NSSS. This code performs a mass, energy, and volume balance on each node and a momentum balance on each flow path. The momentum balance includes the effects of inertia, elevation, and frictional and geometric losses. This ensures that all RCS pressures are correctly predicted. The code contains models for all plant control and protection systems. Modelled NSSS control systems include: Reactor Regulating System (RRS), Pressurizer Pressure and Level Control Systems (PPCS/PLCS), Feedwater Control System (FWCS), Steam Bypass System (SBCS) and Reactor Power Cutback System (RPCS). This system design code has shown good simulation ability in analyzing PRDBEs by comparing it results with operational and test data of the previous OPR1000 plants.

2.2 Load Rejection Test

For this evaluation, 'the load rejection to house load from full power' test was selected. This test is a major PRDBE used in ensuring the performance of NSSS control systems because of its dramatic changes in major plant parameters. Therefore, an integrated initial quick response of the NSSS control systems required in order to avoid reactor trips which may occur due to high pressurizer pressure or low steam generator level.

Upon opening of the switchyard breakers to initiate the load rejection, the turbine power decreases immediately to a house load level in response to the Turbine Control System (TCS) control action. The decrease in turbine power causes a dramatic decrease in the steam flowrate to the turbine, and, hence, a sharp increase in the steam generator pressure. In response to the decrease in the steam flowrate and the increase in the steam generator pressure and the pressurizer pressure, the SBCS generates the steam bypass demand and the reactor power cutback demand signals, simultaneously. The SBCS quick-open signal opens all Turbine Bypass Valves (TBVs), and the reactor power cutback signal drops the pre-selected Control Element Assembly (CEA) groups into the core resulting in a rapid reactor power decrease.

Also, the pressurizer pressure increases due to the reduction in the primary to secondary heat transfer, and PPCS actuates the main pressurizer spray to reduce the pressure increase. Initially, the Steam Generator (SG) water level decreases mainly due to the shrink caused by SG pressure increase, and then recovers to the normal water level as the SG pressure stabilizes and the FWCS controls the feedwater flow.

The immediate the control actions described above are followed by slower control system actions such as the modulation steam bypass demand by SBCS to control the steam pressure and the CEA insertion demand by the RRS to match the reactor power to the turbine power. As the reactor power decreases, the SBCS starts to close turbine bypass valves. Based on the decrease in the RCS average temperature (Tavg), the PLCS controls the charging and letdown flow to match the pressurizer water level to the programmed level, and the PPCS controls the pressurizer pressure to its nominal pressure by controlling the pressurizer heater or spray.

3. Evaluation of Test Results

The test data and the KISPAC code prediction for the major plant parameters of the load rejection test performed at SKN 3 are compared in Figures 1 through 6.

Once all the initial conditions were met, a full load rejection was initiated by opening the switchyard breakers. After initiating the event, the plant control systems take automatic control actions to stabilize the plant with house load operation.

The turbine power was decreased from the initial power to the house load in response to the appropriate TCS control action. This decrease in turbine power caused the steam flow to the turbine to decrease drastically and, hence, a sharp increase in the steam generator pressure. (Figures 1 and 3)

After the load rejection, RPCS droped the pre-selected CEA groups, control banks #5 and #4 in this test, to decrease the reactor power to about 42%. After the actuation of RPCS, a small increase in the reactor power was caused by the negative temperature feedback effect of fuel and moderator. (Figure 2)

The RCS reference temperature (Tref), which is directly proportional to the turbine power, was rapidly decreased after the turbine power drops to house load resulting in a large deviation with the measured Tavg. Since the CEA Auto Motion Inhibit (AMI) setpoint was set at 50% power for this test, the SBCS generated AMI signal to block the automatic CEA insertion demand signal. A slight decrease in the reactor power after RPCS actuation is mainly due to a combined reactivity feedback effect of cooldown and the Xenon buildup. (Figure 2)

As the reactor power decreased from the power level of 42% to 24% after the RPCS actuation, the steam generator pressure decreased resulting in a less steam dump demand signal from the SBCS. This caused the TBVs to be subjected to modulation mode. (Figure 3)

Initially, the SBCS sensed the steam flowrate decrease, and generated the steam bypass quick-open demand signal. The quick-open group X and Y signals opened TBVs and RPC signal was generated, simultaneously. All TBVs were opened by the quick-open signal and modulated after the quick-open signal cleared.

The steam generator water level decreased sharply right after the initiation of this test mainly due to shrinking phenomena caused by the rapid increase in the steam generator pressure. After this initial level shrink, the steam generator level was recovered as the steam generator pressure was controlled by SBCS and the feedwater flow was increased by FWCS. The FWCS responded to the steam generator level and automatically controlled the main feedwater flow rate to the steam generator, so that the steam generator level was stabilized at the setpoint of 50%. (Figure 4)

Right after the load rejection, the pressurizer pressure increased due to the reduction of steam flow initially and decreased due to the RPCS actuation and quick opening of TBVs. According to the pressurizer pressure change, the PPCS restored the pressurizer pressure to its nominal pressure of 2250 psia. (Figure 5)

The pressurizer water level changed as the RCS average temperature changed, and the PLCS controlled charging flow to maintain the pressurizer level at the programmed setpoint. (Figure 6)

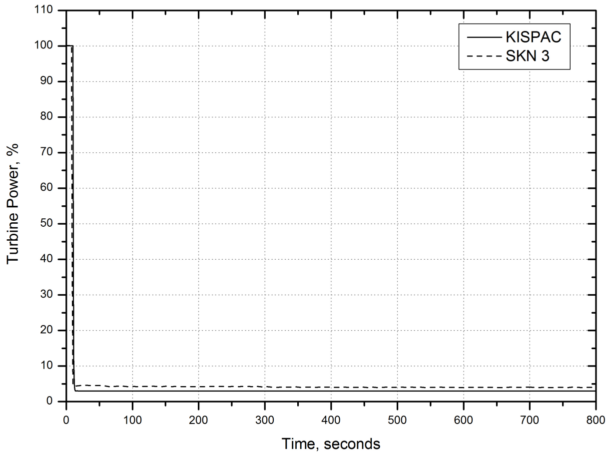

Figure 1. Turbine Power

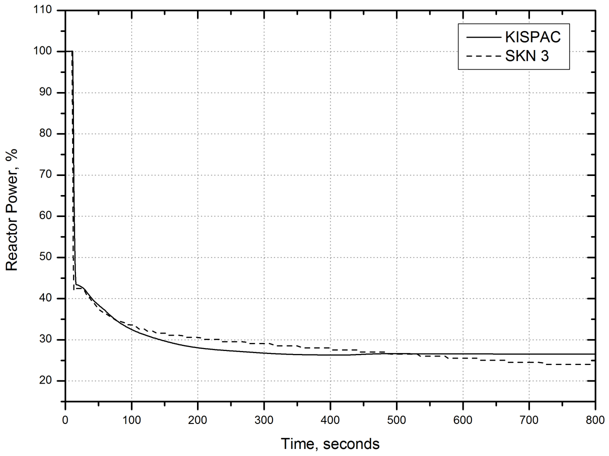

Figure 2. Reactor Power

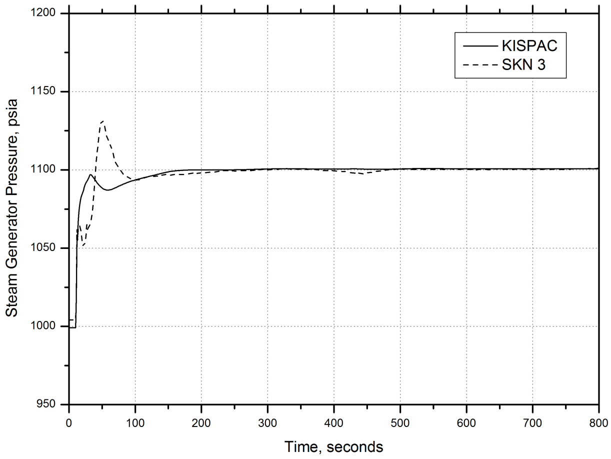

Figure 3. Steam Generator Pressure

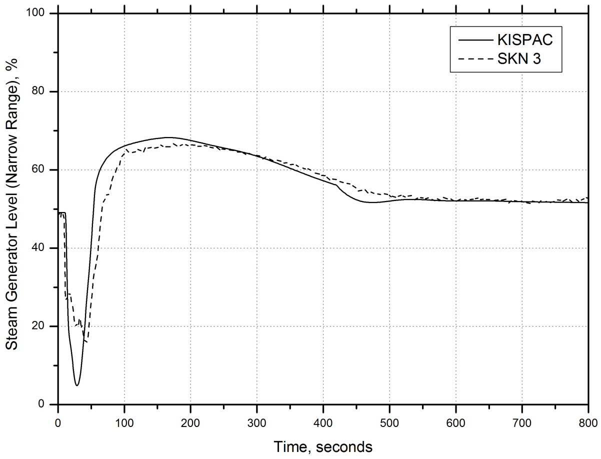

Figure 4. Steam Generator Level

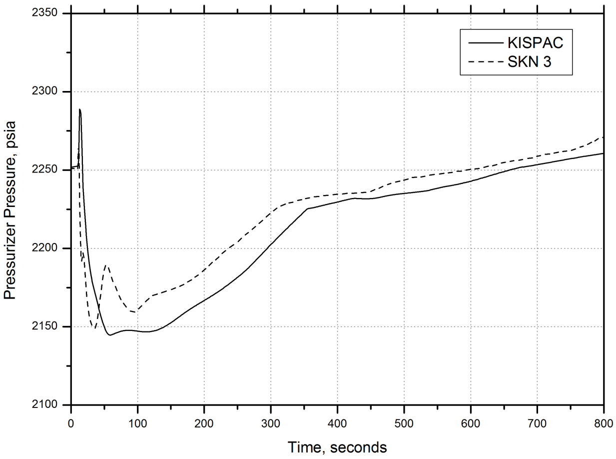

Figure 5. Pressurizer Pressure

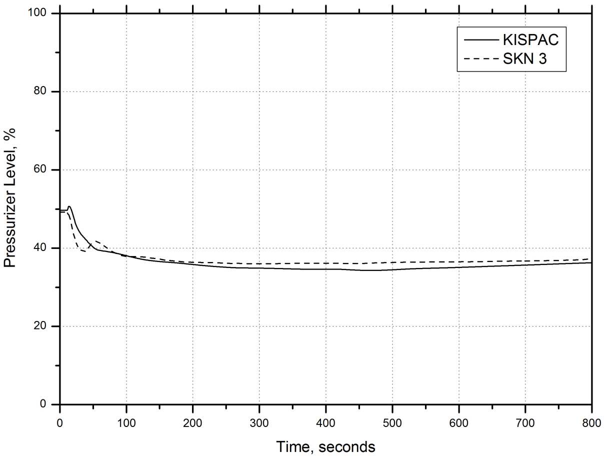

Figure 6. Pressurizer Level

In general, good agreement was observed between the KISPAC code predictions and the measured plant test data. Of the parameters compared, the steam generator peak pressure in the KISPAC analysis was lower than the test results (Figure 3). This deviation turned out to be caused by the difference in the initial design data and the as-built condition (capacity and actuation time of the TBVs). By incorporating the as-built TBVs characteristics, the KISPAC code simulation shows almost identical steam generator pressure response as test data (Figure 7).

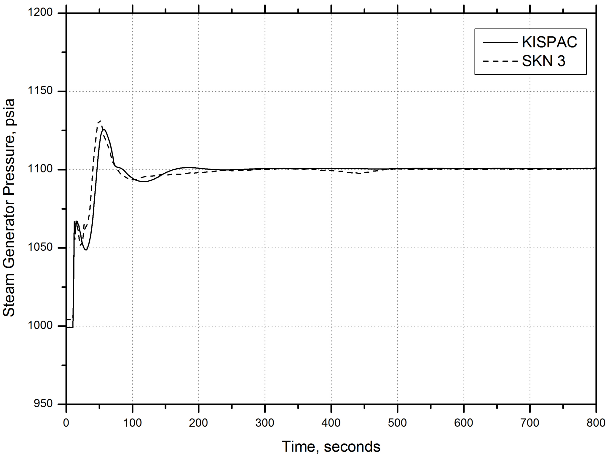

Figure 7. Steam Generator Pressure with As-Built Condition

4. Conclusions

To evaluate the effectiveness of the NSSS control systems design for the first APR1400 plant, comparison has been performed between the system code simulation outputs and SKN 3 test results.

The evaluation has shown that the test results agree well the control systems functioned as designed to mitigate the event consequence and the NSSS system design code adequately simulates the transient behavior.

Therefore, it is concluded that APR1400 NSSS control systems show good performance to maintain continued operation during design operational transients, and the KISPAC code is an useful simulation tool for design of APR1400 plants.

REFERENCES

[1] SKN 3&4, FSAR Chapter 7, Instrumentation and Control.

[2] SKN 3&4, FSAR Chapter 14, Initial Test Program.

[3] SKN 3&4, FSAR Chapter 5, Reactor Coolant System and Connected Systems.