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Loss of a Main Feedwater Pump Test Simulation Using KISPAC Computer Code

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Abstract

Among those tests performed during the Yonggwang Nuclear Power Plant Units 3 and 4 (YGN 3&4) Power Ascension Test period, the Loss of a Main Feedwater Pump test at 100% power is one of the major test which characterize the capability of YGN 3&4. In this event, one of the two normally operating main feedwater pumps is tripped resulting in a 50% reduction in the feedwater flow. Unless the NSSS and Turbine/Generator control systems actuate properly, the reactor will be tripped on low SG water level or high pressurizer pressure. The test performed at Unit 3 was successful by meeting all acceptance criteria, and the plant was stabilized at a reduced power level without reactor trip. The measured test data for the major plant parameters are compared with the predictions made by the KISPAC computer code, an updated best-estimate plant performance analysis code, to verify and validate its applicability. The comparison results showed good agreement in the magnitude as well as the trends of the major plant parameters. Therefore, the KISPAC code can be utilized for the best-estimate nuclear power plant design and simulation tool after a further verification using other plant test data.

1. Introduction

An accurate and reliable system simulation computer code for the nuclear power plant design process is an essential element to ensure safety as well as to provide required performance of the plant. The safety analysis uses a conservative analysis methodology in which the postulated accidents are evaluated at the worst combination of the plant conditions with the safety grade fluid and instrument system models only[1]. Therefore, the safety analysis results are intentionally directed to be unrealistic and, in almost all cases, it is impossible to verify the safety analysis code with the measured as-built plant transient data.

The plant performance analysis, on the other hand, utilizes a best-estimate methodology in which all the fluid systems as well as the plant control systems are credited to calculate the nominal plant behavior[2]. The computer code used in the plant performance analysis has the capability of predicting the entire nuclear power plant behavior from the reactor core to the turbine/generator including all necessary plant auxiliaries. Since the real operation of the plant is strongly dependent on the automatic control actions of the plant control systems, the plant performance analysis code should have detailed models for the control systems as well as associated fluid and supporting systems. Also, it is important to verify and validate the performance analysis code through a comparison of the code predictions with the as-built plant test results.

In this paper, the KISPAC (KAERI Integrated Systems Performance Analysis Code), a best-estimate plant performance analysis computer code, is presented with its models and capabilities. Also, the plant test results for the Loss of a Main Feedwater Pump (LOMFP) test performed on February 16, 1995 during Power Ascension Test (PAT) period of the Yonggwang Nuclear Power Plant Unit 3 (YGN 3) are presented [3]. Finally, the KISPAC code predictions of the LOMFP test are compared with the measured plant data to verify the best-estimate simulation capability of the KISPAC computer code.

2. YGN 3&4 Design Features

The YGN 3&4 which generates 2825 MWt power consists of two independent primary loops, each of which has two reactor coolant pumps and a steam generator. A pressurizer is connected to one of the loops and safety injection lines are connected to each of the four cold legs and the two hot legs. The steam generator incorporates integral economizer which is a semicylindrical section of the tube bundle at the cold leg side of the U-tubes. The majority of the feedwater (90% of the total feedwater flow at 100% power) is introduced into the economizer section and the remainder to the downcomer channel [4]. In conjunction with the main fluid systems, the YGN 3&4 incorporates several plant control systems and the plant protection system to provide the desired plant performance as well as the required plant safety. Among the plant control systems, the Feedwater Control System (FWCS) and Reactor Power Cutback System (RPCS) play key roles during the LOMFP event in preventing possible reactor trip and, hence, the functions of those two control systems are briefly described in the following paragraph.

The FWCS of the YGN 3&4 delivers the condensate from the condenser to the steam generators to automatically control the steam generator water level during power operations. Starting at the main feedwater pumps outlets, two identical flow paths proceed to the steam generators. The FWCS receives three input signals of the steam generator level, feedwater flow rate, and steam flow rate and generates the output signal through a series of controllers. This output signal regulates the feedwater flow rate to the steam generators by controlling the main feedwater pump speed and downcomer and economizer feedwater control valves [5].

The RPCS is a control system designed to accommodate large plant load imbalances such as a large turbine load rejection or a turbine trip by providing a step reduction in reactor power. This step reduction in reactor power is accomplished by dropping one or more pre-selected groups of Control Element Assemblies (CEAs) into the core. Also, the RPCS actuation signal is generated whenever the RPCS receives the loss of main feedwater pump signal in order to reduce the reactor power corresponding to the heat removal capability of the remaining one main feedwater pump. In this case, the RPCS also generates the turbine setback signal in order to rapidly reduce the turbine power below 60% [6].

Figure 1 shows the integrated NSSS control systems performance and their major interfaces. As shown in this figure, the FWCS interfaces with other NSSS control systems such as Reactor Regulating System (RRS), Steam Bypass Control System (SBCS), RPCS, and Control Element Drive Mechanism Control System (CEDMCS).

3. LOMFP Test

The LOMFP test at 100% power is one of the major test which shows the characteristics and the capability of YGN 3&4. During this event, one of the two normally operating main feedwater pumps is tripped resulting in a 50% reduction in the feedwater flow. Unless the NSSS and Turbine/Generator (T/G) control systems actuate properly, the reactor will be trip-

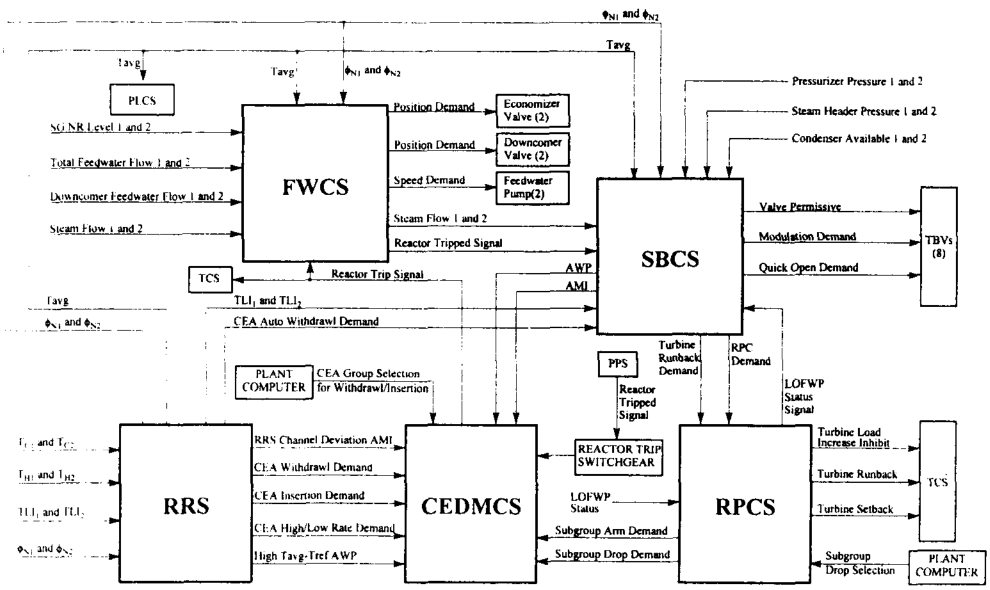

Fig. 1. NSSS Control System Major Interfaces for YGN 3&4

ped due to a low steam generator water level and/or a high pressurizer pressure.

In order to prevent a reactor trip and to continue power operation during LOMFP event, the YGN 3&4 is designed with the RPCS which is an unique design for the ABB-CE type PWR plants. The RPCS is designed to be actuated during the LOMFP event in order to reduce the reactor power rapidly by dropping the pre-selected CEAs into the core so that the plant can be operated at a reduced power level. Along with the RPCS, other control systems such as the SBCS, FWCS, RRS, CEDMCS, and the Pressurizer Pressure and Level Control Systems (PPCS and PLCS) are designed to stabilize the plant conditions automatically at a new steady state.

3.1. Objectives and Acceptance Criteria

The main objective of the LOMFP at 100% Power test performed as a part of the PAT for YGN 3 is to demonstrate that the NSSS can accomodate a main feedwater pump trip at greater than 95% power without initiating a Reactor Protection System (RPS) signal or an Engineered Safety Features Actuation System (ESFAS) signal as well as without opening any pressurizer and/or main steam safety valves and tripping the turbine. Other objective of the test includes to assess the performance of the NSSS control systems (SBCS, FWCS, RRS, CEDMCS, RPCS, PPCS, and PLCS) and the Turbine Control System (TCS) following a main feedwater pump trip.

To accomplish the above objectives, the LOMFP test shall meet the following acceptance criteria :

(1) The RPS does not initiate a reactor trip.
(2) The ESFAS is not actuated.
(3) The pressurizer and/or main steam safety valves do not open.
(4) The RPCS drops the selected CEA Groups into the core.
(5) Turbine setback to 60% is initiated.
(6) Turbine runback occurs as necessary to match turbine power to reactor power.
(7) Reactor and turbine are automatically stabilized after the reactor power

cutback and the turbine setback/runback actuation.

3.2. Initial Conditions and Test Method

The test initial conditions are defined as the full power steady state conditions [3], and the measured major plant parameters right before the test are shown in the second column in Table 1. As shown in this table, all major initial conditions were within the acceptable range for performing the test, and all control systems such as FWCS, SBCS, RRS, PPCS, PLCS, and TCS are in the automatic control mode during the LOMFP test period. The test was initiated by manually tripping the #2 Main Feedwater Pump (MFP02) while the #1 and #2 MFP were in service. After tripping MFP #2, the operator did nothing but monitoring the plant status in order to check whether the automatic control actions of the plant control systems are as expected and/or to be prepared to the unexpected transients. The test was successful by meeting all the acceptance criteria, and the test results are discussed later in conjunction with the KISPAC code predictions.

4. Analysis Methodology

4.1. Code Descriptions

KISPAC code [7] is a best-estimate nuclear power plant simulation tool which is developed based on the ABB-CE supplied LTC computer code [8] by updating the control systems as well as associated fluid system models. The major improvements incorporated into the KISPAC code includes the FWCS logic change for downcomer/economizer valve controls, turbine power setback and runback model change in TCS, RCP seal injection model change, and CVCS model and associated PLCS model changes to incorporate various CVCS configurations.

KISPAC code is designed to analyze the thermal-hydraulic responses of the NSSS and major secondary systems during non-LOCA accidents, power range transients, reactor trips, plant heatup and plant cooldown. Major systems modeled in detail include the reactor coolant system, main steam system, main and auxiliary feedwater systems, containment heat transfer and all NSSS control systems. Other systems which influence the response of the major heat transport systems are also modeled. These include the chemical and volume control system, safety injection system and a limited turbine system model. Plant monitoring, control and protection systems, including instrument lag times and instrument decalibration due to environmental effects are also modeled.

Figure 2 shows the primary loop model of the reactor coolant system in the KISPAC code. As shown in this figure, the reactor coolant system is divided into 17 nodes plus the pressurizer and upper head and 23 flow paths. All conservation equations for the reactor coolant system are written on the basis of the single phase incompressible flow excluding the pres-

Table 1. Initial Conditions of Major Plant Parameters

Parameters	Test Values	Nominal Design Values
Neutron Flux Power	99 %	100 %
Turbine/Generator Power	1040 MWe	2825 MWt
Pressurizer Pressure	2230 psia	2250 psia
Pressurizer Level	51 %	52.6 %
RCS Average Temperature	590.5 °F	592.85 °F
RCS Reference Temperature	592.85 °F	592. 85 °F
Steam Generator Pressure	1110 psia	1088 psia
Steam Generator Level	44 % of NR	44 % of NR

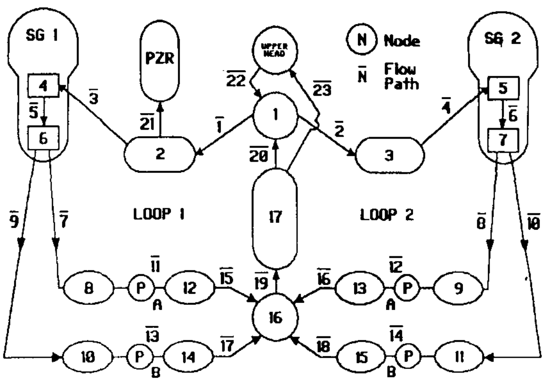

Fig. 2. Node and Flow Path Diagram for the KISPAC code

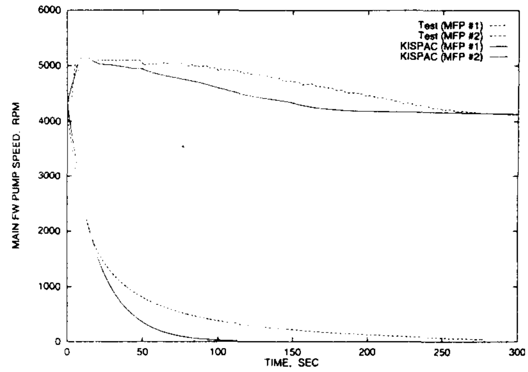

Fig. 3. Main Feedwater Pump Speed during LOMFP at 100% Power

surizer and the steam generator at which the two phase, i.e., liquid and vapor phase, exists. The Wilson bubble rise correlation[9] is used for the modeling of the two phase heat transfer in the pressurizer and steam generator. The conservation equations are solved by implicit finite difference method. In addition to the fluid system modeling, the KISPAC code includes detailed models for all NSSS control systems such as FWCS, SBCS, RRS, PLCS, PPCS, RPCS, and TCS.

4.2. Simulation Method

As listed in the third column of Table 1, the initial conditions used for the code simulation of LOMFP test are defined as the full power steady state conditions[3]. Though the initial conditions used in the simulation are slightly different from the test initial conditions, the difference can be neglected in verifying the capability of the KISPAC code. All NSSS and T/G control systems are set to be in automatic control mode with the as-built setpoints. The KISPAC code simulation is also performed using the as-built plant data which deviate from the warranted plant conditions expected at the end of 40 year plant life.

5. Test Results and Comparison to Expected Results

The test data and the KISPAC code predictions for the major plant parameters are plotted in Figures 3 through 12. As shown in Figure 3, the Main Feedwater Pump #2 (MFP 02P) was manually tripped at 0.1 second when MFP 01P and 02P were running. Upon tripping of one main feedwater pump, the total feedwater flow to the steam generators decreases instantaneously to about 60% of the full power flow rate (see Figure 4). As a result, the SG water level (see Figure 5) decreases rapidly to about 12% narrow range mainly due to the decrease in feedwater flow and level shrink caused by the SG pressure increase (see Figure 9). This decrease in the steam generator water level causes the FWCS to respond with an increased demand signal which increases the main feedwater pump speed (see Figure 3) and opens the economizer feedwater control valves. The FWCS increases the feedwater pump speed such that the remaining one pump can deliver the required feedwater flow, and then restores the SG water levels to normal water level (44% of narrow range). Figure 4 shows that the feedwater flow initially decreases rapidly, and then increases to about 70% of the total feedwater flow owing to the flow delivery increase by the unaffected pump and stabilizes at approximately 50% where the steam generator water level stabilizes to its setpoints, as shown in Figure 5.

As compared in Figures 3 through 5, the KISPAC

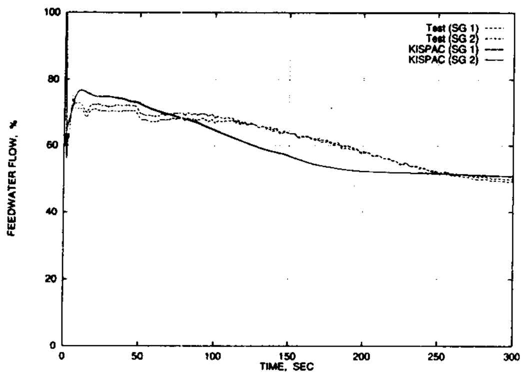

Fig. 4. Feedwater Flow Rate during LOMFP at 100% Power

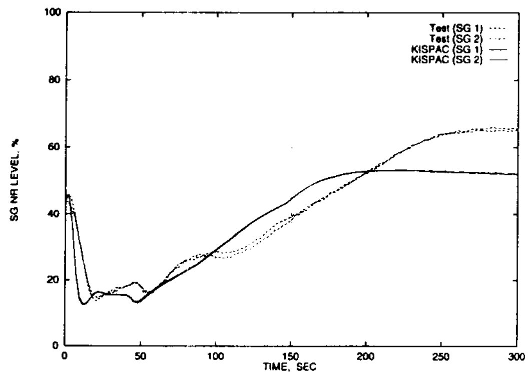

Fig. 5. SG Water Level (Narrow Range) during LOMFP at 100% Power

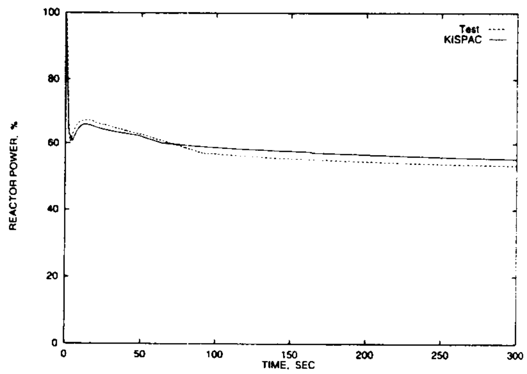

Fig. 6. Reactor Power during LOMFP at 100% Power

code simulation results agree well with the measured data. Especially, the code predicted amount of SG water level shrink right after the test initiation is exactly same with the measured data (see Figure 5). However, the response time of the KISPAC code results is a little shorter than that of the test data. Minor differences in the maximum pump speed (see Figure 3), the feedwater flow trend between 100 seconds and 250 seconds (see Figure 4), and the SG level after 100 seconds (see Figure 5) are considered to be resulted from the differences in the code input data and the as-built field data (e.g., pump characteristic curve, feedwater control valve characteristic curve including non-ideal behavior, feedtrain delay time, etc.).

Upon receiving the loss of feedwater pump signal, the RPCS generates the reactor power cutback and the turbine setback signals simultaneously. The reactor power cutback signal drops a pre-selected CEA groups into the reactor core resulting in a rapid reactor power decrease. As shown in Figure 6, the RPCS dropped the selected CEA groups, which is the control rod bank #5 for this test, on loss of feedwater pump signal, and the RRS further inserted the control rod bank #4 to match the reactor power to the turbine power. The CEA insertion results in a corresponding reactor power decrease and, in turn, the RCS Tavg decrease (see Figure 7). As compared in Figures 6 and 7, the KISPAC code predictions follow the trends of the measured data satisfactorily.

On receipt of the setback signal, the TCS decreases the turbine power to 60% at a rate of −600%/min. As shown in Figure 8, the measured generator electrical power output initially decreased to about 70% by the turbine setback and followed by a further decrease to the final steady state value of 52% by the subsequent turbine runback signal. Although the initial decrease in the generator power output was less than the turbine setback target value of 60%, the steam flowrate to the turbine decreased to the target value. This difference in the generator output power and the turbine steam flowrate is de-

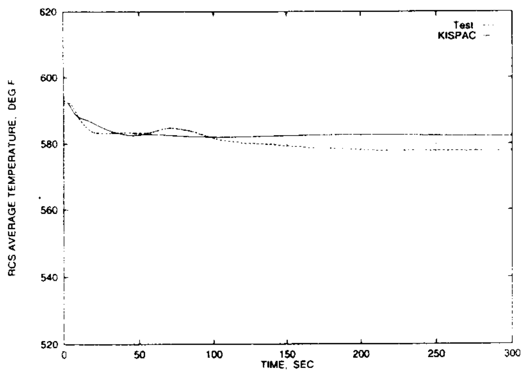

Fig. 7. RCS Average Temperature during LOMFP at 100% Power

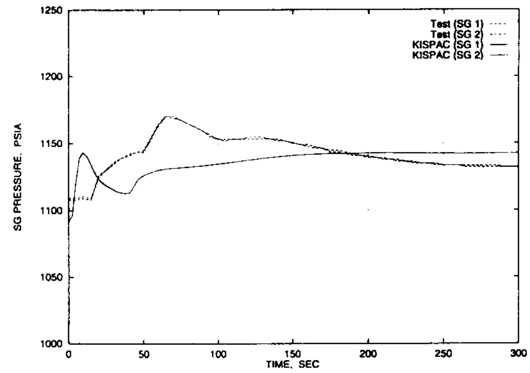

Fig. 9. SG Pressure during LOMFP at 100% Power

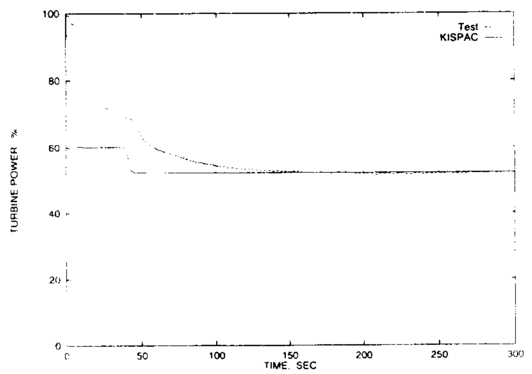

Fig. 8. Turbine Power during LOMFP at 100% Power

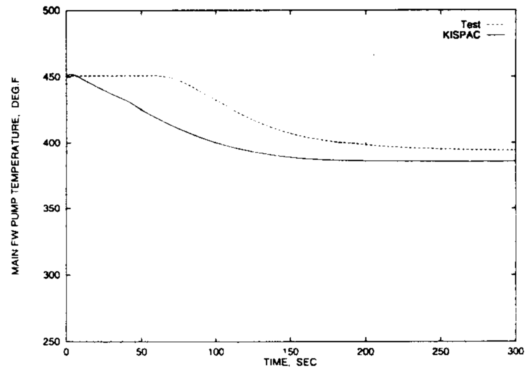

Fig. 10. Main Feedwater Temperature during LOMFP at 100% Power

termined to be caused by the existence of large steam reservoirs in the Moisture Separator and Reheaters (MSRs) which are located between the high and low pressure turbines. Therefore, the turbine power predicted by the KISPAC code, which is based on the steam flow rate to the turbine, agrees well with the test data.

Once the immediate control system actions described above are performed, more slow control actions are followed such as the modulation steam bypass demand by the SBCS to control the steam pressure, the CEA insertion demand by the RRS to match the reactor power with the turbine power, and the turbine runback demand by the TCS after the initial setback demand. As the reactor power decreases, the SBCS starts to close turbine bypass valves, if opened.

Since the secondary heat removal is rapidly reduced by the turbine setback, the SG pressure rapidly increased and then actuate SBCS to open turbine bypass valves in modulation mode. The quick open mode of the SBCS was blocked by the LOMFP signal. Figure 9 shows that the SG pressure increased and then was stabilized at the pressure corresponding to the final steady state turbine power of 52% (refer to Figure 8). As shown in this figure, the rapid decrease of the SG pressures between 10 and 40 seconds in the KISPAC results and between 65 and 100 seconds in the test data is mainly due to the

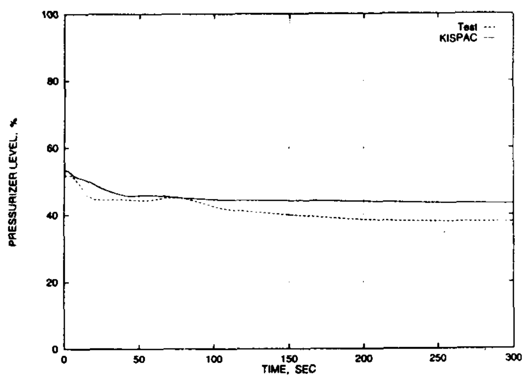

Fig. 11. Pressurizer Level during LOFMP at 100% Power

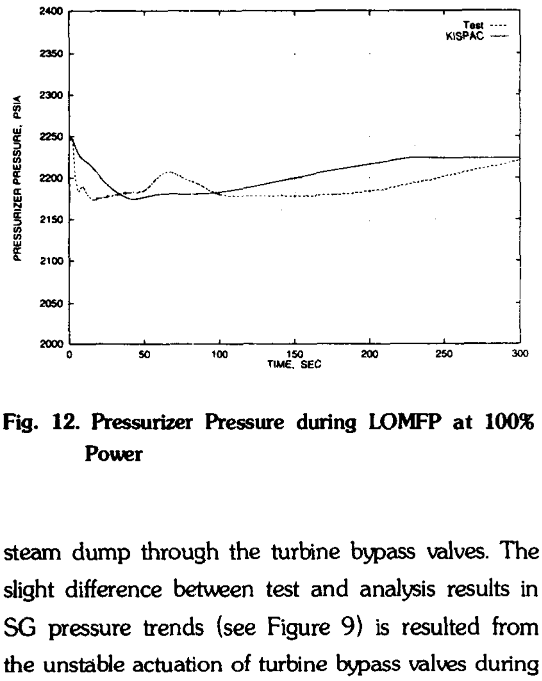

Fig. 12. Pressurizer Pressure during LOMFP at 100% Power

steam dump through the turbine bypass valves. The slight difference between test and analysis results in SG pressure trends (see Figure 9) is resulted from the unstable actuation of turbine bypass valves during the LOMFP test.

In the meantime, the feedwater temperature is gradually decreased due to the steam heating reduction as shown in Figure 10. While the actual feedwater temperature stays at 450 °F during 75 seconds and then decreases, the calculated result shows an immediate decrease right after LOMFP. The current KISPAC computer code does not model the deaerator[10], whose functions are to remove the unwanted air in feedwater flow and to supply the Net Positive Suction Head on the feedwater booster pumps, and therefore the response of the KISPAC computer code is faster than that of the actual data. However, this difference results in a minor effect on feedwater flow response and does not significantly deteriorate the performance of the KISPAC code.

Based on the decrease in the RCS Tavg, the PLCS controls the letdown flow to match the pressurizer water level to the programmed level (see Figure 11), and the PPCS controls the pressurizer pressure to its nominal pressure of 2250 psia by controlling the pressurizer heaters or spray (see Figure 12). The KISPAC code predicted pressurizer level and pressure follow the test data satisfactorily.

6. Conclusions

The Loss of a Main feedwater Pump at 100% Power test for YGN unit 3 was successfully simulated using the KISPAC computer code. The comparison of the LOMFP test data with the KISPAC computer code predictions showed good agreement in the magnitude as well as the trends of the major plant parameters. Especially, the code successfully predicted the swell and shrink phenomena of the steam generator water level and other primary and secondary thermal-hydraulic parameters. Additionally, the code well simulated the control system performance and associated fluid system responses. Therefore, the KISPAC code can be utilized for the best-estimate nuclear power plant design and simulation tool after a further verification using other plant test data.

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