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Evaluation of Loss of a Main Feedwater Pump Test for UCN 3

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ABSTRACT

The Loss of a Main Feedwater Pump event is one of the major design bases events which characterize the advanced capability of the Korean Standard Nuclear Power Plants. During this event, all NSSS Control Systems including Reactor Power Cutback System are designed to automatically control the plant to prevent reactor trip and continue power operation. The Loss of a Main Feedwater Pump test at 100% power was performed during UCN 3 Power Ascension Test period. All plant control systems worked properly as designed and successfully stabilized the test transients. The test results well agree with computer simulations using the KISPAC code which is a best-estimate plant performance analysis tool used for the design of the UCN 3&4.

1. INTRODUCTION

The Loss of a Main Feedwater Pump (LOMFP) test at 100% power was performed on April 16, 1998 during Power Ascension Test (PAT) period in Ulchin Nuclear Power Plant Unit 3. During this test, one of the two normally operating main feedwater pumps is tripped resulting in a 50% reduction in the feedwater flow. Unless the Nuclear Steam Supply System (NSSS) and Turbine/Generator (T/G) control systems actuate properly, the reactor will be tripped due to a low steam generator water level or a high pressurizer pressure.

In order to prevent a reactor trip and continue power operation during LOMFP event, the UCN 3 is designed with the Reactor Power Cutback System (RPCS) which is an unique design feature for Korean Standard Nuclear Plants [1]. The RPCS is designed to actuate during the LOMFP event, if reactor power is higher than 75%, in order to rapidly reduce the reactor power by dropping the pre-selected control element assemblies 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 Steam Bypass Control System (SBCS), the Feedwater Control System (FWCS), Reactor Regulating System (RRS), and the Pressurizer Pressure and Level Control Systems (PPCS and PLCS) are designed to automatically stabilize the plant conditions at a new steady state.

In this paper, the test results for the LOMFP event are presented. The performance of the

NSSS and T/G control systems is compared to the design capabilities. Also, the measured test data are compared with the results predicted by the plant performance analysis computer code ‘KISPAC’ [2], in order to verify the plant design as well as to validate the computer code.

2. TEST DESCRIPTIONS

2.1 Objectives and Acceptance Criteria

The main objectives of the Loss of a Main Feedwater Pump test are as follows:[3]

1) To demonstrate that the NSSS can accomodate a main feedwater pump trip without initiating a Reactor Protection System (RPS) signal or an Engineered Safety Features Actuation System (ESFAS) signal as well as without opening any primary or secondary safety valves and tripping the turbine.
2) To verify that the feedwater pump which has not tripped will increase feedwater flow to about 65% of total feedwater flow.
3) To verify that the turbine setback and runback control logic responds properly.
4) To assess the performance of the NSSS control systems (SBCS, FWCS, RRS, PPCS, and RPCS) and the Turbine Control System (TCS) following a main feedwater pump trip.

The major acceptance criteria for the test are as follows:[3]

1) The RPS does not initiate a reactor trip.
2) The ESFAS is not actuated.
3) The primary and/or secondary 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/or the turbine setback/runback actuation.

2.2 Expected Plant Performance

Upon tripping of one main feedwater pump, the total feedwater flow to the steam generators decreases rapidly. The SG water level, in turn, decreases mainly due to the decrease in feedwater flow and shrink caused by the SG pressure increase. 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 and opens the economizer feedwater control valves. Upon receiving the loss of feedwater pump signal, the RPCS generates a reactor power cutback signal when reactor power is higher than 75% and a turbine setback signals when reactor power is higher than 60%. The reactor power cutback signal drops a pre-selected CEA groups into the core resulting in a rapid reactor power decrease. On receipt of a setback signal, the TCS decreases the turbine power to 60% at a rate of 10%/sec.

Once the immediate control system actions described above are performed, more slow control actions are followed such as the modulation steam bypass demand by 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. The FWCS increases the feedwater pump speed such that the unaffected pump will deliver the required feedwater flow up to about 65% of total feedwater flow and restore the SG water levels to its normal water level setpoint (44% of narrow range). As the reactor power decreases, the SBCS starts to close turbine bypass valves, if opened. Based on the decrease in primary coolant average temperature (Tavg), the PLCS controls the letdown flow to match the pressurizer water level to the programmed level, and the PPCS controls the pressurizer pressure to its nominal pressure of 2250 psia by controlling the pressurizer heaters or spray.

3. KISPAC CODE DESCRIPTION

The KISPAC code [2] is a best-estimate nuclear power plant simulation tool which is developed on the basis of the LTC computer code [4] 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 including centrifugal charging pump. The 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 1 shows the primary loop model of the reactor coolant system in the KISPAC code. As shown in the figure, the reactor coolant system is divided into 17 nodes plus the pressurizer and reactor vessel 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 pressurizer and the steam generator at which the two phase, i.e., liquid and vapor phase, exists. The Wilson bubble rise correlation [5] 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.

4. TEST RESULTS AND COMPARISON TO EXPECTED RESULTS

The test initial conditions for the major plant parameters are shown in Table 1 with the nominal design values. As shown in the table, all major initial conditions were within the acceptable range for performing the test [3], and all the NSSS and T/G control systems were in automatic mode of operation.

The test data and the KISPAC code predictions for the major plant parameters are plotted in Figures 2 through 9. As the Main Feedwater Pump #1 (MFP 01P) was manually tripped when MFP01P and 02P were running, the total feedwater flow and SG water levels decreased rapidly (Figures 2 and 3). The FWCS increased the feedwater pump speed and opened the economizer feedwater control valves such that the unaffected pump can deliver the required feedwater flow and then restore the SG water levels to their initial values. As compared in Figures 2 through 4, the KISPAC code simulation results agree with the measured data.

The RPCS dropped the selected CEA groups, which was the control bank #5 for this test, on LOSS OF FEEDWATER PUMP STATUS signal. As shown in Figure 5, the reactor power decreased rapidly to about 65%. The RRS further inserted the control 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 (Figure 7). As compared in Figure 5, the KISPAC code predictions follow the trends of measured data exactly.

As shown in Figure 6, the measured turbine power output decreased by the turbine setback to about 53%, which is lower than the target power of 60%. The measured turbine power is decreased in a slower rate than the KISPAC simulation, because there are large steam reservoirs in the Moisture Separator and Reheaters located between the high and low pressure turbines while the turbine power predicted by the KISPAC code is based on the steam flow rate to the turbine.

Based on the decrease in the Tavg, the PLCS controlled the pressurizer water level to match the programmed level by controlling letdown flow (Figure 9), and the PPCS restored the pressurizer pressure to 2250 psia of its nominal pressure (Figure 8). The KISPAC code predicted trends of the pressurizer level and pressure also follows the test data.

5. CONCLUSIONS

The Loss of a Main feedwater Pump tests for UCN 3 was performed successfully. The test acceptance criteria described in Section 2 were satisfied. All the NSSS and T/G control systems responded automatically to prevent the reactor trip and stabilize the test transient. The trends of all major plant parameters were as expected by the design of the plant. The KISPAC computer code used in the performace analysis during the design process of UCN 3&4 predicted the test results successfully.

6. REFERENCES

1) KEPCO, FSAR for UCN 3&4, Chapter 7.
2) KOPEC, "Software Verification and Validation Report for KISPAC-WS," April 7, 1997.
3) KEPCO, "Test Procedure for PAT Reactor Power Cutback System," 3S-I-741-01, Rev.0.
4) ABB-CE, "LTC User's Manual," 1986.
5) J. F. Wilson, et al., "Primary Separation of Steam from Water by Natural Separation," Joint US/EURATOM R&D Program , Allis-Chalmers Topical Report, ACNP-65002, April 15, 1965.

Table 1. Initial Conditions of Major Plant Parameters

Parameters	Test Initial Values	Nominal Design Values
Neutron Flux Power	97.3 %	100 %
Turbine/Generator Power	1037 Mwe	2825 Mwt
Pressurizer Pressure	2252 psia	2250 psia
Pressurizer Level	50.7 %	52.6 %
RCS Average Temperature	589.8 deg.F	592.85 deg.F
RCS Reference Temperature	592.3 deg.F	592.85 deg.F
Steam Generator Pressure	1092 psia	1088 psia
Steam Generator Level	44.3 % of NR	44 % of NR

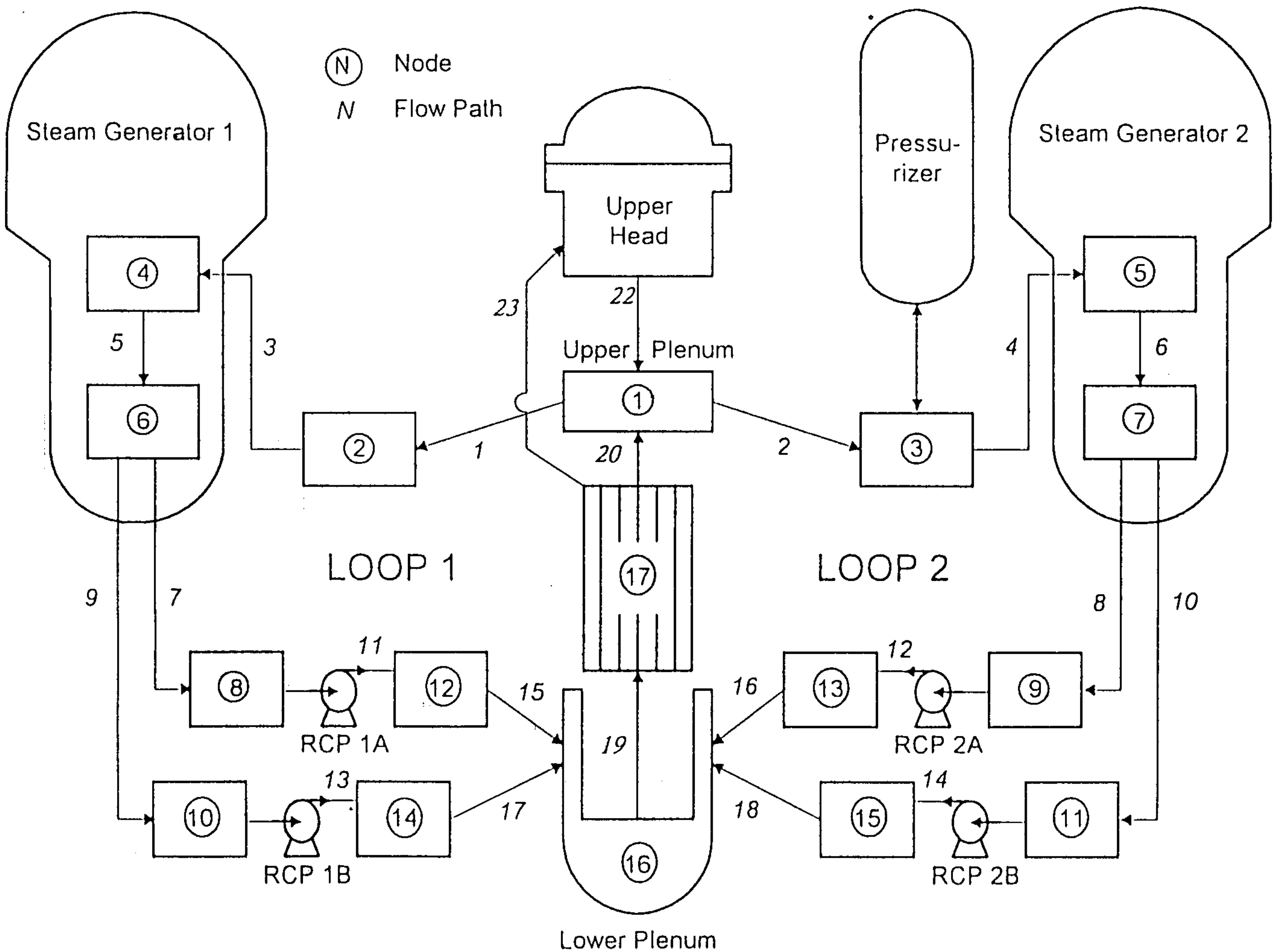

Figure 1. Node and Flow Path Diagram for the KISPAC Code

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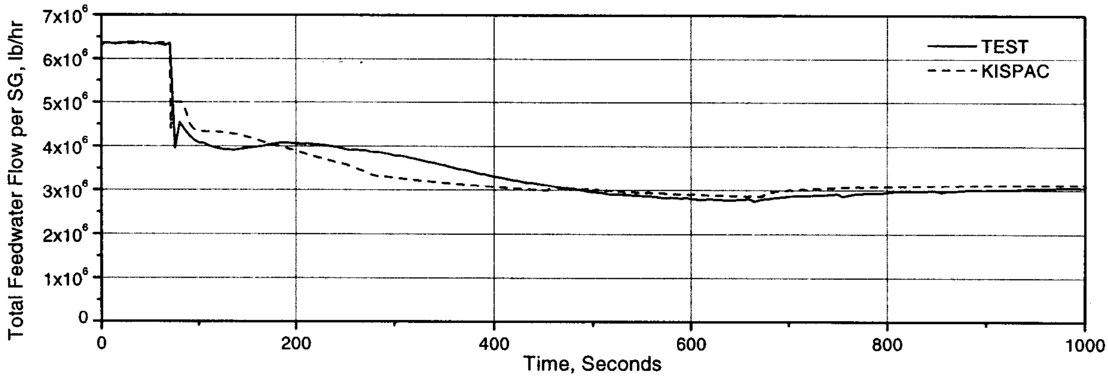

Figure 2. Total Feedwater Flow per SG at 100% Power

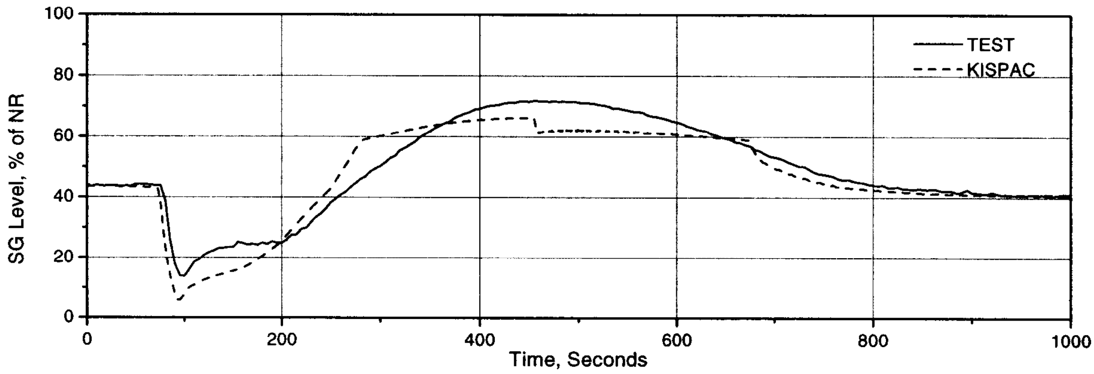

Figure 3. SG Narrow Range Level During LOMFP at 100% Power

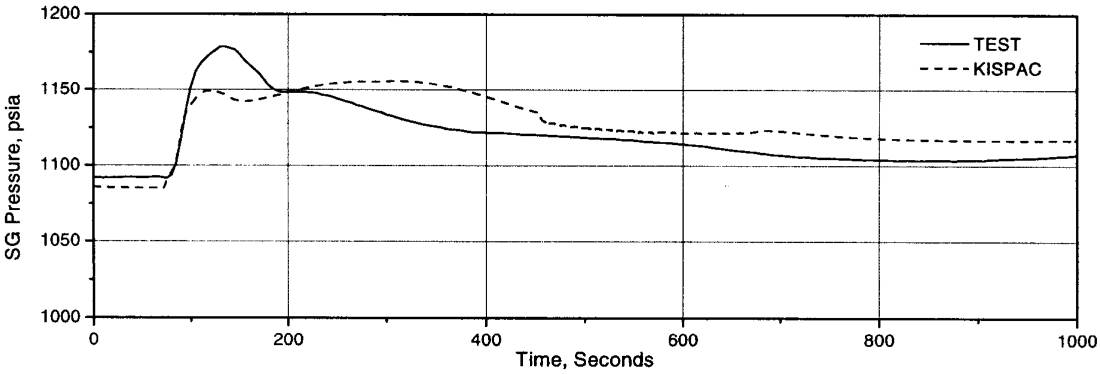

Figure 4. SG Pressure During LOMFP at 100% Power

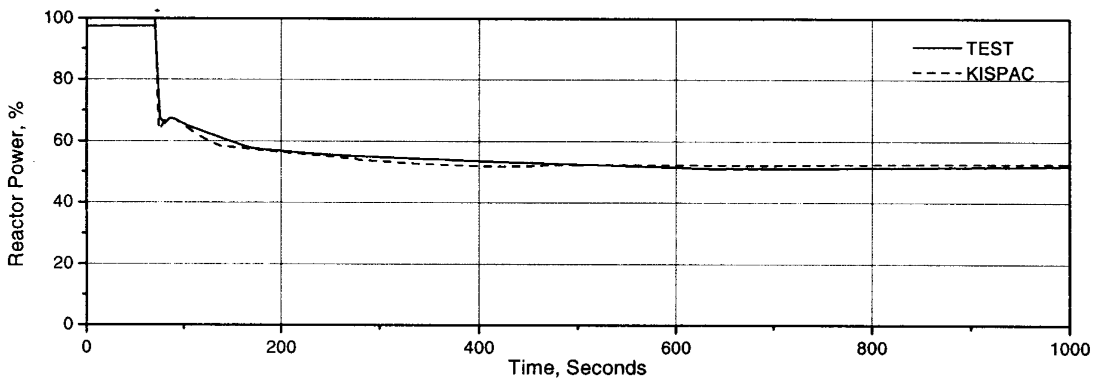

Figure 5. Reactor Power During LOMFP at 100% Power

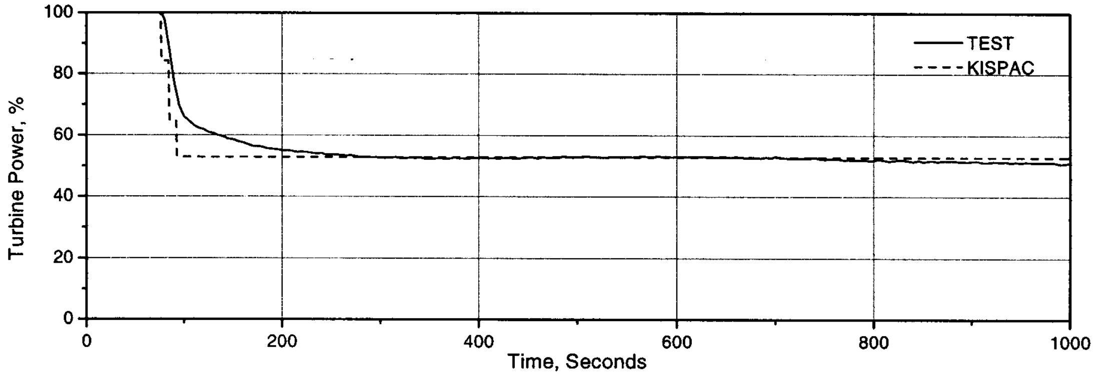

Figure 6. Turbine Power During LOMFP at 100% Power

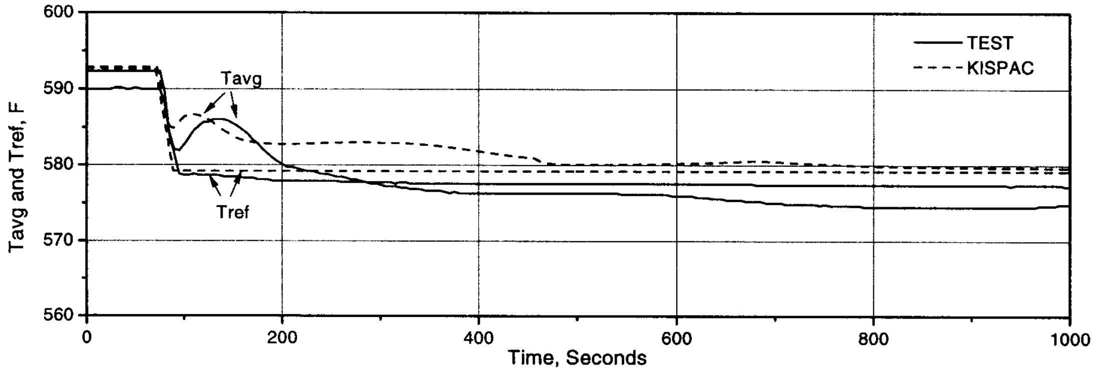

Figure 7. RCS Average and Reference Temperature During LOMFP at 100% Power

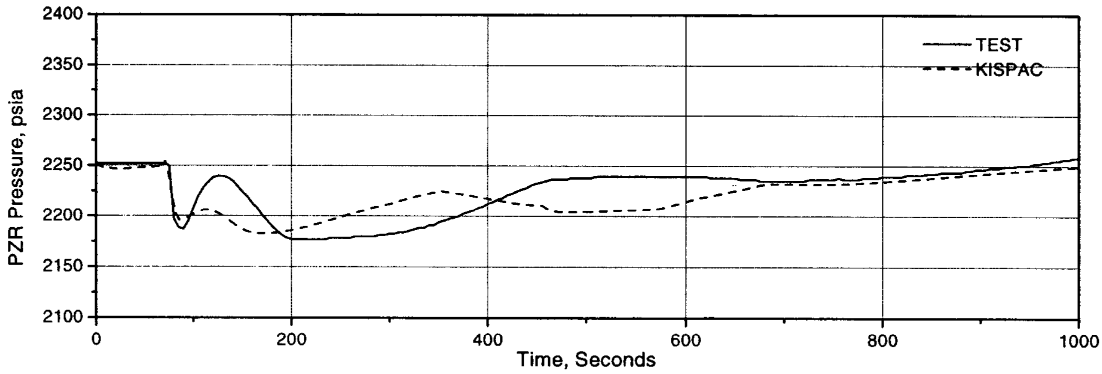

Figure 8. PZR Pressure During LOMFP at 100% Power

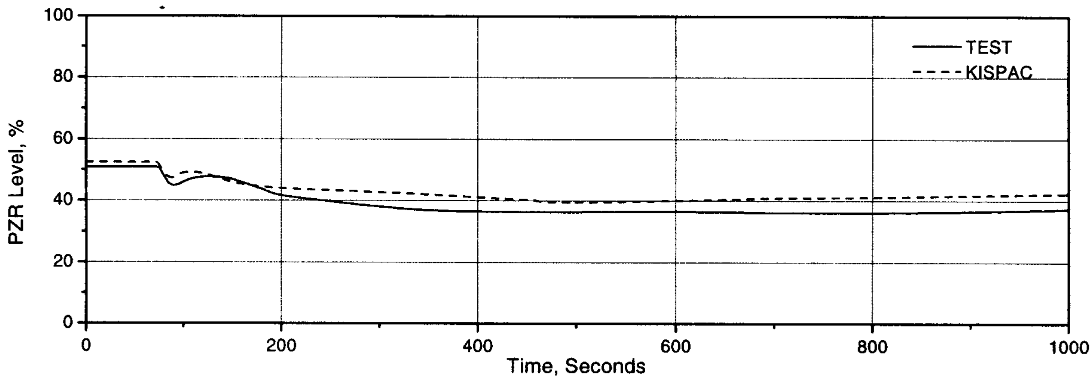

Figure 9. PZR Level During LOMFP at 100% Power