PALO VERDE NUCLEAR GENERATING STATION

UNIT 1 STARTUP REPORT

(Docket No. 50-528)

Revision 1

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PREFACE

The Palo Verde Nuclear Generating Station (PVNGS) is a three unit nuclear power station located approximately 50 miles west of downtown Phoenix, Arizona. PVNGS is owned by the Arizona Nuclear Power Project (ANPP), a consortium of southwestern United States utilities. Arizona Public Service Company is the project manager of PVNGS for ANPP.

PVNGS Unit 1 (PVNGS 1) utilizes a System 80 pressurized water reactor nuclear steam supply system (NSSS) manufactured by Combustion Engineering, Inc. (C-E). System 80 is C-E's standardized NSSS design and is described in the Combustion Engineering Standard Safety Analysis Report--Final Safety Analysis Report (CESSAR). PVNGS 1, the first System 80 NSSS to start operation, has a rated core thermal output of 3800 MWt, and a nominal net electric output of 1270 MWe.

The objective of this report is to provide a summary description of the initial startup test program for PVNGS 1. This program consists of a series of tests which satisfy requirements of the Nuclear Regulatory Commission as detailed in the PVNGS Final Safety Analysis Report (FSAR). The FSAR references Chapter 14 of CESSAR, which incorporates the testing requirements of Regulatory Guide 1.68, Revision 0. The test program summarized by this report consists of five phases:

1. Fuel Loading
2. Postcore Hot Functional Tests
3. Initial Criticality
4. Low Power Physics Testing
5. Power Ascension Testing

The overall objectives of this test program are to:

a) Demonstrate that components and systems of the Nuclear Steam Supply System (NSSS) operate in accordance with design requirements.
b) Demonstrate that the NSSS can be safely operated and that performance levels can be maintained in accordance with established safety requirements.
c) Confirm proper transient system operation and thereby verify that the NSSS can be brought to power as well as to shutdown condition in a controlled and safe manner.
d) Provide verification of core physics parameters and baseline performance data for use during normal plant operation.

This report describes the FSAR (i.e. CESSAR) required testing from Fuel Loading though the Power Ascension Testing phase. Testing is listed and summarized by the applicable section of CESSAR.

1.0 INTRODUCTION AND TEST PROGRAM SUMMARY

1.1 Summary of Test Objectives by Test Phase

The initial startup test program described herein begins with Initial Fuel Loading. This phase of the test program provides a systematic process for safely accomplishing fuel load. It also verifies that all fuel assemblies and installed sources are correctly located and oriented. Initial Fuel Loading is described in section 2.

Postcore Hot Functional Tests (HFT) follow Fuel Loading. The objectives of these tests are to provide additional assurance that plant systems necessary for normal plant operation function as expected, and to obtain performance data on core related systems and components. Normal plant operating procedures, in so far as practical, are used to bring the plant from cold shutdown conditions (Operational Mode 5) to hot, zero power conditions (Operational Mode 3). The Postcore Hot Functional Tests provide the first measurements of NSSS and secondary system performance with the core in place. Examples of systems tested under this phase are the control rod drive system, the reactor coolant system (RCS), and the incore neutron monitoring system. Examples of measurements include control rod drop times, reactor coolant system flow rate, flow coastdown following reactor coolant pump trips, and movable incore detector path lengths. The Postcore Hot Functional Test phase is described in section 3.

Initial Criticality follows the Postcore Hot Functional Tests. Because PVNGS 1 is a first-of-a-kind design, initial criticality is performed at RCS conditions of 320 °F/600 psia, instead of the normal hot zero power conditions of 565 °F/2250 psia. This phase of the test program assures a safe and controlled approach to criticality. Section 4 describes Initial Criticality.

Low Power Physics Testing (LPPT) immediately follows Initial Criticality, and is conducted with the reactor critical but producing no measurable heat. Since PVNGS 1 is a first-of-a-kind design, testing is performed at two RCS temperature/pressure plateaus: 320 °F/600 psia, and 565 °F/2250 psia (normal hot zero power conditions). This phase of testing consists of a series of measurements of selected core parameters, such as control rod worth, temperature coefficient of reactivity and soluble boron reactivity worth. These measurements serve to substantiate the safety analyses of the FSAR and the bases of the Technical Specifications on core behavior. The LPPT measurements also demonstrate that core characteristics are within expected limits and provide data for benchmarking the computer algorithms used for predicting core characteristics later in core life. Additionally, the LPPT phase includes the first measurements of radiation shielding by the biological shield. Section 5 describes these tests.

Power Ascension Testing (PAT), the longest phase of testing, follows LPPT. This phase is structured to bring the reactor to full power in stages, with testing performed at intermediate "test plateaus" of approximately 20%, 50%, and 80% of full power, before final testing at full power. PAT demonstrates that the facility operates in accordance with its design during steady power operation and, to the extent that testing is practical, during anticipated transients. Since PVNGS 1 is a first-of-a-kind design, the PAT program is expanded to validate the design methods and to demonstrate new design concepts, notably the Reactor Power Cutback System (RPCS).

Typically, a PAT test plateau begins with confirmation of the reactor power level by secondary heat balance, and calibration of the power instruments as needed. Next, initial plateau testing is performed while equilibrium xenon conditions are allowed to develop, after which time detailed physics testing is performed. Testing of the control systems are performed next, and the test plateaus generally conclude with one or more "transient tests" including:

* Shutdown Outside the Control Room Test
* Loss of Load Tests (with and without RPCS action)
* Loss of Feedwater Pump Tests (with RPCS action)
* Turbine Trip Test
* Generator Trip Test
* Loss of Offsite Power Test
* Natural Circulation Test

Testing of the PAT phase is described in Section 6.

1.2 Chronology of Startup Testing: Fuel Load through Low Power Physics Testing

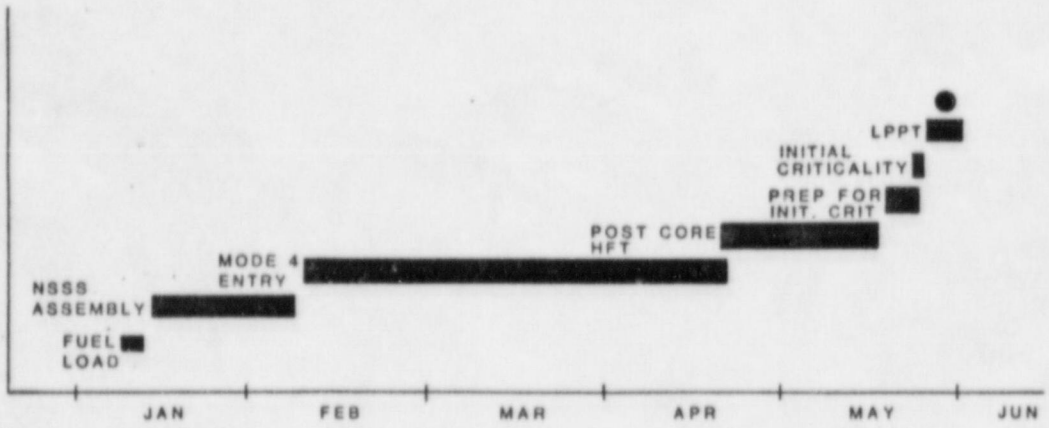
The chronology of startup testing from Fuel Load through Low Power Physics Testing is shown on Figure 1-1. The approximate durations of the activities that took place from the start of fuel load to the completion of LPPT are shown below:

ACTIVITY	DURATION	DATES
Fuel Load	5 days	Jan. 7--Jan. 11, 1985
Post Fuel Load Checks and NSSS Assembly *	27 days	Jan. 11--Feb. 7, 1985
Preparation for Mode 4 Entry (RCS Tavg $\geq$ 210 ^{o}F)	72 days	Feb. 7--Apr. 20, 1985
Postcore HFT	28 days	Apr. 20--May 18, 1985
Preparation for Initial Criticality	5 days	May 18--May 23, 1985
Initial Criticality	2 days	May 23--May 25, 1985
Low Power Physics Testing	7 days	May 25--June 1, 1985
	146 days	Jan. 7--June 1, 1985

* Installation of reactor vessel head, control rod drive power cables, incore detectors, etc.

FIGURE 1-1

**CHRONOLOGY OF STARTUP TESTING:
FUEL LOAD THROUGH LPPT**



●--NRC issues the full power Operating License (6-1-85)

1.3 Summary of Startup Test Results: Fuel Load through Low Power Physics Testing

Fuel Load was completed in 5 days versus a scheduled 8 days with no significant problems. During the post fuel load verification process, one fuel assembly was found to be incorrectly oriented in its core position (all assemblies were placed in the correct core positions). This assembly was re-oriented and procedural changes have been made to reduce the possibility of misorientation during future fuel loads.

Postcore HFT was completed in 27 days versus a scheduled 20 days, with no significant problems. The results of this test phase that are significant to power operations are:

- The performance of the control rod drive system was excellent, and this was further demonstrated throughout LPPT;
- The Reactor Coolant System steady state flow rate was 105.1% of the design volumetric flowrate;
- The reactor coolant pump coastdown flow was measured to be consistent with that assumed in the safety analysis for PVNGS.

Initial Criticality was completed within its scheduled two day duration, with no significant problems. The measured critical soluble boron concentration was 1054 ppm versus a predicted value of 1063 ppm and was within the acceptance criteria.

Low Power Physics Testing was completed in 7 days versus a scheduled 12 days, with no significant problems. The measured core parameters were within their acceptance criteria bands except the worth of control rod Shutdown Group A with the most reactive rod "stuck out" (or, Group "A-1"), which was more worthy than expected. As described in section 5, this condition of higher than expected rod worth was determined to be acceptable. In general, the measured core physics parameters were very well predicted.

1.4 Chronology of Startup Testing: Power Ascension Testing Phase

The approximate durations of the various phases of Power Ascension Testing are tabulated below, and the power level history during this time period is shown on Figure 1-2.

TEST ACTIVITY	DURATION	DATES
Initial increase from zero to 20% power	12 days	June 1--June 12, 1985
First generation of electrical power	-------	June 10, 1985
20% Power Test Plateau	18 days	June 12--June 30, 1985
Initial increase from 20% to 50% power	6 days	June 30--July 6, 1985
50% Power Test Plateau	80 days	July 6--Sept. 24, 1985
Initial increase from 50% to 80% power	2 days	Sept. 24--Sept. 26, 1985
80% Power Test Plateau	74 days	Sept. 26--Dec. 9, 1985
Initial increase from 80% to 100% power	1 day	Dec. 9--Dec. 10, 1985
100% Power Test Plateau	46 days	Dec. 10, 1985--Jan. 25, 1986
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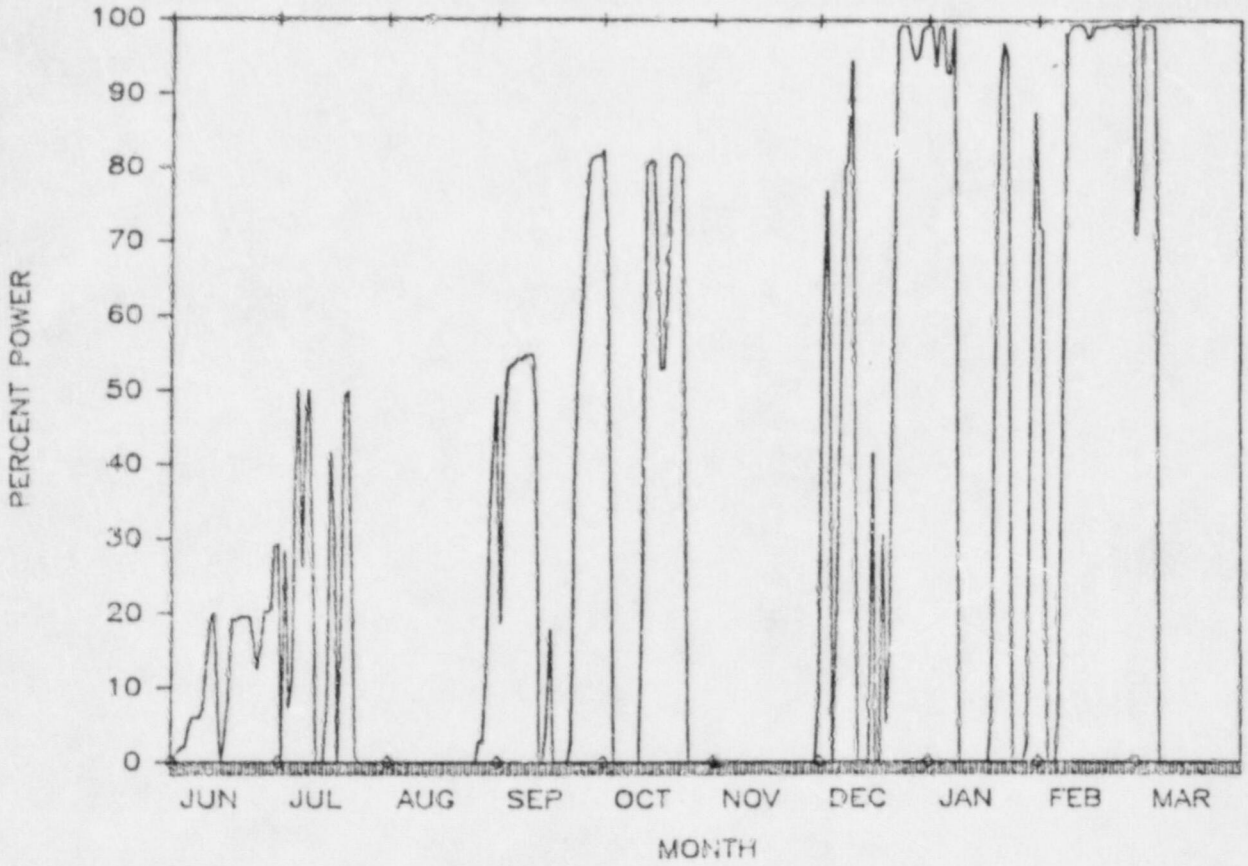
	239 days	June 1, 1985--Jan. 25, 1986

On February 13, 1986, based upon the successful completion of the Power Ascension Test program and of 100 hours of continuous operation above 95% full power, Arizona Public Service Company declared the commencement of commercial operation of PVNGS-1.

FIGURE 1-2

PVNGS UNIT 1 POWER HISTORY:

JUNE 1985 TO MARCH 1986



1.5 Summary of Power Ascension Testing

The Power Ascension Test phase was completed over a period of approximately 34 weeks versus a scheduled 28 weeks. This test phase demonstrated that PVNGS-1 operates in accordance with its design during steady state power operation and during the transient tests performed. The core performance test results compared satisfactorily with the predicted values. The transient tests were satisfactorily completed, although three of the tests had to be reperformed after plant equipment modifications were made or test methods were changed. The Power Ascension Test results have verified the design models used for PVNGS and have confirmed that PVNGS-1 is constructed as designed.

The PVNGS-1 Power Ascension Test program was as extensive as that of any previous C-E plant. Section 6.0 of this report describes the individual Power Ascension Tests that satisfy requirements described in the PVNGS FSAR.

It is noteworthy that PVNGS-1 was the first C-E plant to test the Reactor Power Cutback System, a system which is designed to allow the plant to sustain a loss of a main feedwater pump or a load rejection from full power without suffering a reactor trip. Sections 6.4 and 6.7 of this report describe the successful testing of this system.

There were a total of fourteen reactor trips during Power Ascension Testing. Two of these trips were planned trips as part of Power Ascension Testing (Turbine Trip, Section 6.6, and Shutdown from Outside the Control Room, Section 6.8). Three additional trips occurred inadvertently as a result of scheduled testing (see Sections 6.4 and 6.6). The remaining trips were not related to any testing in progress (although one of these trips, resulting from a loss of offsite power, was subsequently substituted for the planned trip in the Loss of Offsite Power Test--see Section 6.9).

There were two significant outages during Power Ascension Testing. The first was a 35 day outage beginning in late July 1985 to allow for condenser repairs and modifications to the Post Accident Sampling System. The second was a 37 day outage beginning in late October 1985 to modify the Calvert Bus equipment used for electrical distribution, to verify the use of proper bolt material and tensioning of these bolts in the RCS pipe stops, and to modify the response time of the steam generator low level trip circuitry (see Section 6.4).

2.0 INITIAL FUEL LOADING
(CESSAR Section 14.2.10.1)

TEST OBJECTIVES AND SUMMARY

The governing procedure for the loading of the initial core into PVNGS 1 was 72IC-1RX01, "Initial Fuel Loading". The objective of this procedure was to provide a safe, organized plan for accomplishing the fuel loading. Fuel loading was conducted over the period of January 7 through January 11, 1985. During and after fuel load, several checks were performed to assure that the core loading was acceptable. These checks included verification of proper loading pattern, proper fuel assembly seating, and proper fuel assembly alignment. These checks were performed successfully and no problems were determined, with the exception that one fuel assembly was improperly oriented. This assembly was rotated to the proper orientation and the procedure was successfully completed on January 13, 1985.

TEST DESCRIPTION

The initial core loading of PVNGS 1 was performed "dry"; that is, the refueling pool was dry except for the fuel transfer canal area, which was filled with borated water to just above the top of the fuel transfer tube. Before the start of fuel loading, this water was measured to have a boron concentration of 4071 ppm. The water level in the reactor vessel was maintained below the vessel flange, but above the top of the hot legs. This water was measured to have a boron concentration of approximately 2380 ppm at the beginning of fuel load. One shutdown cooling loop was operated almost continuously during the fuel load evolution to ensure a uniform boron concentration throughout the Reactor Coolant System (RCS). Samples of the water were drawn from the reactor vessel and from the fuel transfer canal at least once each day to ensure that the boron concentration remained above the Technical Specification limit of 2150 ppm.

The core loading was initiated by the placement of the first of 241 fuel assemblies on the east side of the core area. This assembly contained a startup neutron source to provide a sufficient population of neutrons for subcritical multiplication monitoring. Succeeding assemblies were loaded in a sequence which assured coupling of the assemblies with the source. In general, the fuel assemblies were loaded in north-south rows proceeding from the east to the west side of the core, as illustrated by Figure 2-1.

Monitoring of the subcritical status of the core was performed using four source range detectors: two temporary detectors, located in the reactor vessel; and the two permanently installed Startup Channel detectors, located outside the reactor vessel. Figure 2-2 shows the relative locations of the four detectors. Each of the temporary detectors was moved once during fuel loading to maintain proper monitoring of the core, and both were removed from the vessel prior to the loading of the final two fuel assemblies. After each fuel assembly was loaded, a series of neutron count rates were recorded from each of these detectors. This data was used to compute the inverse multiplication (1/M) for the fuel assembly for each detector. Engineering personnel reviewed this information to ensure that the next fuel assembly could be loaded safely.

Figure 2-3 shows the inverse multiplication response of the temporary detectors, in their initial locations, for the first 25 assemblies loaded. After the first 7 assemblies, the core subcritical multiplication, as indicated by the 1/M response of the temporary detectors, stabilized and remained essentially the same for the remainder of the fuel loading.

During fuel movement, personnel in the fuel building and in containment independently verified that each fuel assembly was transferred from its storage location to its core location in the prescribed sequence. After each assembly was lowered into the reactor vessel, the elevation of the fuel grapple was checked to ensure that the assembly was seated properly on the core support structure before the assembly was ungrappled.

Following the completion of fuel loading, the underwater television camera on the refueling machine was used to scan the serial numbers of the fuel assemblies to verify that each assembly was in its prescribed Cycle 1 location. Furthermore, this scan verified that each assembly serial number was oriented to the plant north and ensured that both startup neutron sources were properly installed in the core. The performance of this scan was recorded on videotape. A second scan was performed on selected fuel assemblies using the underwater camera to ensure that the center of each assembly was aligned within an acceptable tolerance of the nominal centerline for that core location.

TEST RESULTS

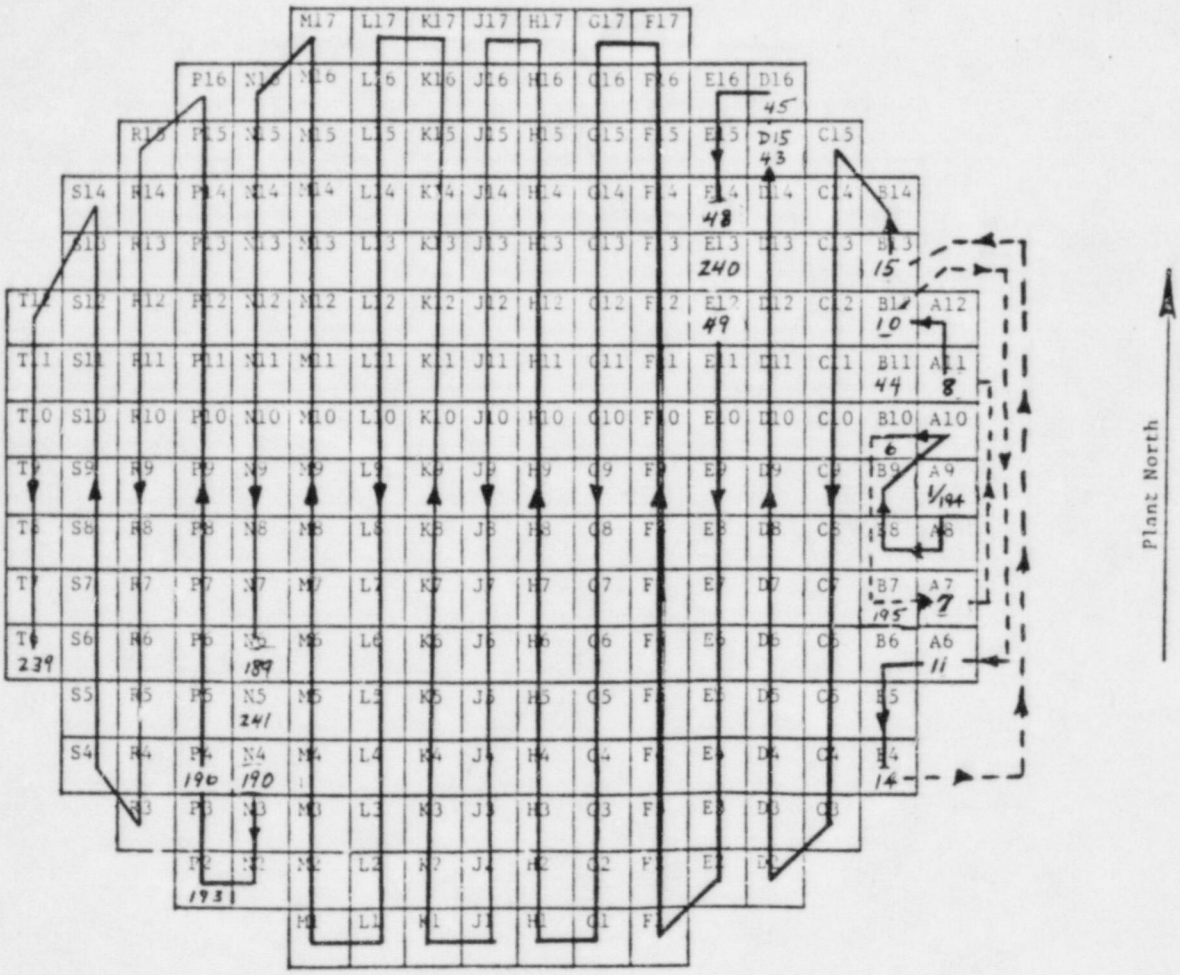
Fuel loading was completed on January 11, 1985. The fuel assemblies were verified to be properly loaded, seated, and oriented, with the exception of one misoriented fuel assembly. This assembly was rotated to the proper orientation. Both startup neutron sources were verified to be properly loaded. Finally, scans of selected assemblies verified that these assemblies were aligned within an acceptable tolerance of the nominal fuel centerlines. 72IC-1RX01 was officially completed, with the results satisfactory, on January 13, 1985.

CONCLUSIONS

The initial fuel loading of PVNGS Unit 1 was successfully accomplished in a safe and controlled manner, in accordance with the objectives and acceptance criteria of 72IC-1RX01.

FIGURE 2-1

CORE LOADING SEQUENCE
PVNGS UNIT 1 INITIAL FUEL LOAD

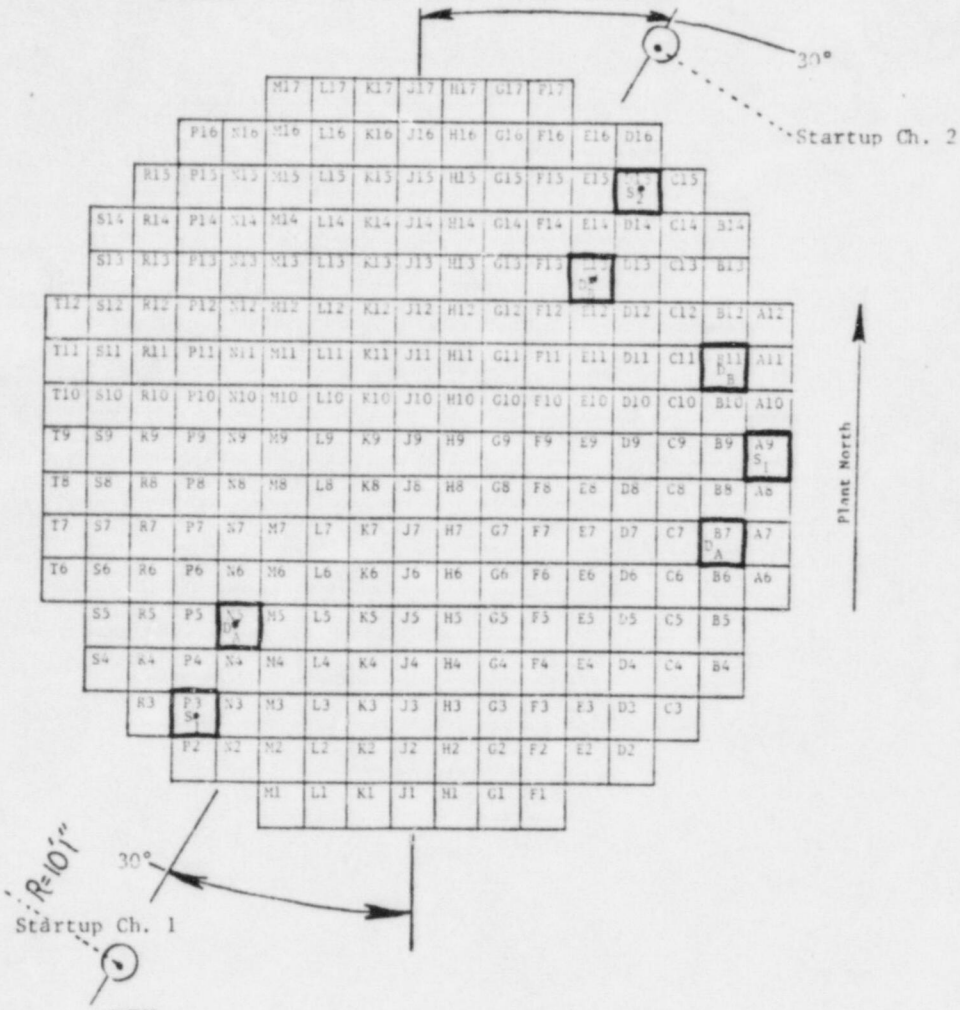


NOTES

The numbers shown above in some core locations correspond to the total number of fuel assemblies in the reactor vessel after that location has been loaded. The first assembly loaded contained a neutron source and was located in position A-9. It was later relocated to position P-3, following the loading of Assembly 193. Assembly 194 was then loaded into the "hole" left in position A-9. Assemblies 44, 195, 240, and 241 were used to fill the "holes" left after the movement or removal of the temporary detectors.

FIGURE 2-2

FUEL LOADING DETECTOR LOCATIONS
PVNGS UNIT 1 INITIAL FUEL LOAD



KEY

S_1---Initial location of neutron source 1 (A-9)

S_1*---Final location of neutron source 1 (P-3)

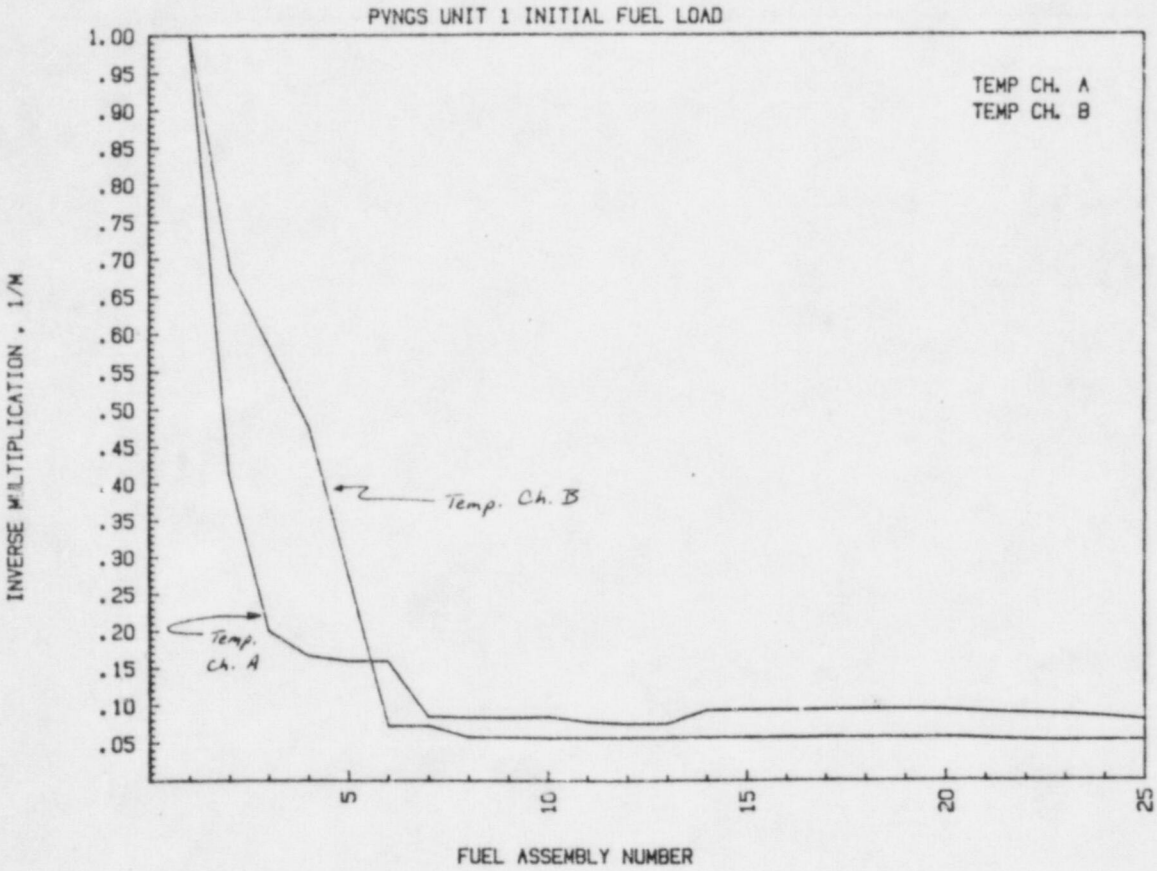
S_2*---Final location of neutron source 2 (D-15)

D_A,D_B---Initial locations of temporary fuel load detectors A (B-7) and B (B-11)

D_A*,D_B*---Final locations of temporary fuel load detectors A (N-5) and B (E-13)

FIGURE 2-3

INVERSE MULTIPLICATION
RESPONSE OF TEMPORARY DETECTORS
PVNGS UNIT 1 INITIAL FUEL LOAD
(First 25 Assemblies)



3.0 POSTCORE HOT FUNCTIONAL TESTS
(CESSAR Section 14.2.12.3)

3.1 Postcore Hot Functional Test Controlling Document
(Section 14.2.12.3.1)

TEST OBJECTIVES AND SUMMARY

The objectives of the Postcore Hot Functional Controlling Document, PVNGS procedure 73HF-1ZZ03, were:

(1) To demonstrate the proper integrated operation of the plant primary, secondary, and auxiliary systems with fuel in the reactor vessel.

(2) To act as a sequencing/controlling document for the CESSAR required hot functional tests.

(3) To demonstrate that the plant can be brought from cold shutdown conditions (Mode 5) to hot standby conditions (Mode 3) using station operating procedures.

(4) To sequence/direct the initial performance of certain mode entry technical specification surveillance procedures.

(5) To sequence/control the performance of Precore Hot Functional (Phase 1) carryover tests.

This procedure was performed over the period of April 20 through May 18, 1985. During this time, the plant was brought from Operational Mode 5 to Operational Mode 3, and then returned to Mode 5. Performance of this test successfully demonstrated the integrated operation of the plant primary, secondary, and auxiliary systems during these Mode changes, thereby satisfying the test acceptance criterion. Additionally, the individual tests controlled by 73HF-1ZZ03 were successfully performed and their acceptance criteria satisfied.

TEST DESCRIPTION

Testing commenced with Reactor Coolant System (RCS) at a temperature of approximately 200 °F and a pressure of 365 psia. From this condition, the RCS was heated up and pressurized to 565 °F, 2250 psia using station operating procedures. During the heatup/pressurization, conditions were stabilized at the direction of 73HF-1ZZ03 at five intermediate temperature/pressure plateaus to allow required testing to be performed. The surveillance requirements were verified as being satisfied prior to any changes in operational mode. After completion of all testing at the 565 °F, 2250 psia test plateau, the plant was cooled down and depressurized back to Mode 5 conditions. During the return to Mode 5, conditions were stabilized at the direction of 73HF-1ZZ03 at two intermediate temperature/pressure plateaus to allow the performance of required testing. Table 3-1 lists the various temperature/pressure plateaus at which testing was performed.

TEST RESULTS

The plant was successfully taken from cold shutdown to hot standby and back to cold shutdown under the direction of 73HF-1ZZ03, utilizing the integrated operation of plant systems. The acceptance criterion for this test was thereby satisfied. Furthermore, individual hot functional tests were successfully performed, as described in the following sections, and their accceptance criteria satisfied. Additionally, the carryover testing was satisfactorily completed.

CONCLUSIONS

Proper integrated operation of the PVNGS 1 primary, secondary, and related auxiliary systems was successfully demonstrated during the Postcore Hot Functional Test. Therefore, these systems will functionally support power operation of the plant.

TABLE 3-1

POSTCORE HOT FUNCTIONAL TEST PLATEAUS
(Nominal Conditions)

Date	Time	RCS Temp (°F)	RCS Press (psia)	Mode
4/20/85	1530	197	365	5
4/21/85	1633	280	380	4
4/24/85	1920	340	500	4
4/30/85	0642	450	1100	3
4/30/85	1600	450	1650	3
4/30/85	2300	500	2250	3
5/01/85	0745	565	2250	3
5/16/85	0230	450	1650	3
5/16/85	0545	450	1100	3
5/16/85	1328	<210	366	5

3.2 Postcore Instrument Correlation
(Section 14.2.12.3.2)

TEST OBJECTIVE AND SUMMARY

PVNGS procedure 73HF-1ZZ02, "Postcore Instrument Correlation," was performed over the period of April 21 to May 16, 1985 in Operational Modes 3 and 4. The objective of this test was to verify that the Main Control Room indications of selected plant parameters monitored by the Plant Monitoring System (PMS), Qualified Safety Parameter Display System (QSPDS), Plant Protection System (PPS), Core Protection Calculators (CPC) and Process Instruments were correct and consistent within acceptance criteria that were based on vendor and design accuracies. This objective was satisfactorily met.

TEST DESCRIPTION

Data for this test was gathered at the following nominal test plateaus:

280 °F/380 psia	450 °F/2250 psia
340 °F/500 psia	500 °F/2250 psia
450 °F/1100 psia	565 °F/2250 psia
450 °F/1650 psia	

Specified plant parameters that were displayed by more than one device were observed and the values recorded as simultaneously as possible. These parameters included reactor coolant system (RCS) hot leg temperatures, RCS cold leg temperatures, core exit temperatures, pressurizer pressure, pressurizer level, steam generator pressures, steam generator levels, reactor coolant pump (RCP) differential pressures, reactor vessel differential pressures, steam generator differential pressures, RCP speeds, RCP seal pressures, and RCP seal bleed-off flows. The values recorded for each parameter were then cross-compared to verify that the various indications of that particular parameter were consistent and accurate within the specified acceptable agreement bands.

TEST RESULTS

The selected parameters met the respective acceptance criteria with no outstanding Test Exceptions. Sufficient correlation was established to ensure that the indications observed were correct and consistent within the prescribed criteria.

CONCLUSION

The accuracy and consistency of Control Room indications of selected plant parameters monitored by the PMC, QSPDS, PPS, CPCs, and process instruments were adequate to support plant power operation.

3.3 Post Core Reactor Coolant System Flow Measurement (Section 14.2.12.3.3)

TEST OBJECTIVE AND SUMMARY

PVNGS procedure 73HF-1RC09, "Post-Core Reactor Coolant System Flow Measurements", was conducted over the period of May 4 through May 10, 1985 with the reactor at hot standby conditions (565 °F,2250 psia). The principle objectives of the test were as follows:

(1) To determine the postcore reactor coolant system (RCS) steady state flow rate and flow coastdown characteristics.

(2) To adjust Core Protection Calculator (CPC) and Core Operating Limits Supervisory System (COLSS) flow algorithm constants based on the measured steady state flow rate.

(3) To compare the measured loss of flow coastdown curve (4-pump trip) to that used in the CESSAR Final Safety Analysis.

The measured RCS flow rate (4-pump steady state operation) was 468,430 gpm and was within the acceptance criteria.

TEST DESCRIPTION

For eleven various steady state reactor coolant pump (RCP) configurations and three RCP flow coastdowns, measurements were made of the RCP differential pressures (DPs), RCP speeds, RCS temperature, and RCS pressure. The measured RCP DP values were converted to values of head, and the corresponding pump flow rates were determined from the RCP performance curves relating pump head to flow rate. The total RCS flow rate for each pump configuration and coastdown was determined by summing the four individual RCP flow rates.

In addition, the total 4-pump steady state RCS flow rate was determined using the more accurate ultrasonic flow measurement (UFM) technique. Lithium niobate crystals were mounted on each RCS hot leg to serve as ultrasonic signal transmitters and receivers. The received signals were electronically processed and used to analyze the fluid turbulence patterns as they passed successive crystal pairs. The mean transit time of the fluid between crystal pairs was determined using cross-correlation techniques, and the fluid flow rate was calculated as a function of the mean transit time, crystal spacings, and flow area. The RCS flow rate measured by UFM techniques was then used as the reference, or standard, flow rate for adjusting the CPC and COLSS flow algorithm constants.

TEST RESULTS

The four pump volumetric RCS flow rate determined by UFM techniques was 468,430 gpm, or 105.1% of the design flow rate of 445,600 gpm. This measured flow rate was within the acceptance criteria of greater than or equal to 465,850 gpm and less than or equal to 501,800 gpm. Because the four pump volumetric RCS

flow rate calculated from RCP differential pressure data did not acceptably agree with the ultrasonic flow measurement result, revised RCP performance curves were provided by the NSSS vendor.

In addition to the steady state flow measurements, the measured loss of flow coastdown curve (4-pump trip) was evaluated and found to be consistent with the curve assumed in the safety analysis.*

CONCLUSIONS

The 4-pump steady state RCS flow rate is sufficient to provide proper cooling of the reactor core under power operation conditions. Additionally, the measured 4-pump trip flow coastdown curve was consistent with that assumed in the safety analysis for PVNGS.

* -- The total loss of coolant flow curve used in the safety analysis assumed a complete and sudden interruption of electrical power to the reactor coolant pumps as the most limiting initiating event. However, testing performed during the Power Ascension Test phase identified that under certain circumstances when electrical power is not immediately interrupted to the RCPs, the RCP motors could coastdown faster than the safety analysis case, due to the electrical braking influence of other house loads on the RCP buses. Pending further analysis by the NSSS vendor, the operating margin was decreased to accomodate the observed coastdown effects and to insure that the conclusions of the safety analysis remain valid. This will be discussed further in a future supplement to this report.

3.4 Postcore Control Element Drive Mechanism Performance (Section 14.2.12.3.4)

TEST OBJECTIVES AND SUMMARY

(1) To demonstrate the proper operation of the control rod drive system (Control Element Drive Mechanisms, or CEDMs) including the control rods (Control Element Assemblies, or CEAs), under Hot Shutdown and Hot Zero Power conditions. This objective was met by Tests 73HF-1SF10, "CEDM Coil Testing 260 °F", and 73HF-1SF11, "CEDM Coil Testing 565 °F". 73HF-1SF10 was conducted on April 22, 1985 with the plant in Mode 4, and 73HF-1SF11 was conducted from May 6 through May 12, 1985 with the plant in Mode 3. The acceptance criteria for both tests were based on the successful movement of the CEAs and their respective CEDM coil current traces being normal.
(2) To verify the proper operation of the CEA position indicating system and alarms. This objective was met in Test 73HF-1SF02, "Post-Core CEDM Performance", which was conducted from April 5 through April 15, 1985. The plant was in Mode 5 during the performance of this test. The acceptance criteria were met by verifying that the indicating systems provided the correct CEA position and that the alarms functioned per design.
(3) To measure CEA drop times. This objective was met in Test 73HF-1SF08, "Post-Core CEA Drop Time Test". The test was conducted from May 11 through May 13, 1985 with the RCS at 565 °F and all four RCPs running (Mode 3). The acceptance criteria for CEA drop time (Technical Specification 3.1.3.4) was met by verifying that the CEAs dropped to 90% insertion in less than 4.0 seconds.

TEST DESCRIPTION

73HF-1SF10: Each CEA was withdrawn individually to 20 inches (13% withdrawn) and then inserted to 15 inches (10% withdrawn). The CEA was then dropped by opening the individual CEA breaker. Current traces were taken while the CEA was being withdrawn and inserted. The CEA was verified to drop when the power was removed.
73HF-1SF11: Each CEA was withdrawn individually to 120 inches (80% withdrawn) and then inserted to 7 inches. The CEA was then dropped by opening the individual CEA breaker. Current traces were taken while the CEA was being withdrawn and inserted. The CEA was verified to drop when the power was removed.
73HF-1SF02: Each CEA was withdrawn individually to its upper limit, then inserted to its lower limit, and then dropped by opening the individual CEA breaker. During this evolution, the following were verified:

1) The upper and lower electrical limits were set correctly
2) The upper, lower, and rod drop lamps were operational.
3) The Plant Computer minor and major deviation alarms were set correctly.
4) The CEA Calculator (CEAC) deviation alarm was set correctly.
5) The CEA position indicated correctly on the CPC, CEAC, PMS, and CEA position CRT.
6) The CEA withdrawal and insertion drive speeds were correct (30 in/min).

73HF-1SF08: The CEAs were withdrawn by group to their upper limit and each CEA in that group was verified to be at its upper electrical limit. The CEAs were then dropped, one at a time, by opening the individual CEA breakers. The CEA position was recorded as it dropped, by monitoring the CEDM power and the Reed Switch Position Transmitter (RSPT) output. The recorded data for each CEA was reviewed, and the drop time for 90% insertion was calculated.

TEST RESULTS

The testing was completed for the CEDMs, with no outstanding test exceptions. The alarms, position lamps, and CEA position indicators were verified to respond within their assigned limits. The CEAs moved as required, and their withdrawal and insert CEDM current traces were satisfactory. The drop time of each CEA to 90% insertion was less than 3 seconds, well within the allowed limit of 4.0 seconds.

CONCLUSION

The testing of the CEDMCS proved that the system will operate as designed, and will support plant power operation.

3.5 Postcore Reactor and Secondary Water Chemistry Data (Section 14.2.12.3.5)

TEST OBJECTIVE AND SUMMARY

PVNGS procedure 74HF-1SS01, "Postcore HFT Chemistry Test," was performed (1) to demonstrate that proper water chemistry for the reactor coolant and secondary systems can be maintained from ambient conditions to system operating conditions; and (2) to verify the adequacy of the prescribed sampling frequencies in establishing and maintaining proper chemistry control as well as detecting, and correcting, out-of-specification conditions in a timely manner. Acceptability of the test was based on three criteria:

(1) The procedures for sample collection and analysis were adequate for primary and secondary chemistry control.

(2) The prescribed sampling frequencies were adequate for primary and secondary chemistry control.

(3) The analyses of water samples from the reactor coolant and secondary systems were capable of detecting deviations from the prescribed chemistry specifications in a timely manner.

Monitoring of the chemistry conditions per this procedure was initiated on April 19, 1985 and completed on May 1, 1985. The acceptance criteria were satisfied.

TEST DESCRIPTION

Sampling and chemical analyses of the primary and secondary water systems were performed using the appropriate plant operating procedures, as directed by 74HF-1SS01. Data was taken at the following test plateaus:

Ambient conditions	340 °F/500 psia
200 °F /365 psia	450 °F/1100 psia
280 °F/380 psia	565 °F /2250 psia

At each of these plateaus, samples from the reactor coolant system (RCS), steam generators, feedwater system, condensate system, the reactor makeup water tank (RMWT), and the refueling water tank (RWT) were compared to the operating specifications provided in the test procedure.

TEST RESULTS

Ambient conditions - The steam generators were in "Wet Lay-Up" with Wet Lay-Up chemistry specifications being maintained. The RCS was in Mode 5 with RCS chemistry specifications being maintained. The feedwater system was not operating. The condensate system was in long path recirculation with the condensate polishers in service. The make-up water tanks (RMWT, RWT) were within specification.

200 °F/365 psia plateau - The RCS chemistry was within the Mode 5 specifications per 74HF-1SS01. The steam generators were also within the Mode 5 specifications. The feedwater system was not operating. The condensate system was in long path recirculation with the condensate polishers in service. The Mode 5 specifications for the condensate system were maintained. The RMWT and RWT were within specification.

280 °F/380 psia plateau - The RCS chemistry was within specifications with the exception of a low lithium concentration. The make-up water chemistry was within specification. The steam generator chemistry was within specification with the exceptions of dissolved oxygen and cation conductivity. The feedwater system was not in operation. Auxiliary feedwater was used to feed the steam generators. The auxiliary feedwater was maintained within specification. The condensate system was in long path recirculation with its chemistry in specification, with the exception of dissolved oxygen.

A Test Exception Report was written to document the test exceptions and out-of-specification conditions. Chemistry Control Instructions were initiated to correct the out-of-specification conditions. The chemistry control parameters were within specification prior to proceeding to the next plateau with the exception of the dissolved oxygen in the condensate system and the RCS low lithium concentration. These conditions were evaluated and determined to be acceptable for proceeding to the next test plateau. The bases for these determinations are as follows:

* RCS lithium concentration--The primary concerns of proper system (RCS) chemistry are proper pH level and dissolved oxygen concentration. The specification on the RCS lithium concentration is based on steady state conditions at 565 °F, when lithium is the primary pH control additive and dissolved oxygen is within specification. During the initial heat-up phase, however, hydrazine and ammonia are present in the RCS, and a balance between these two chemicals and lithium is used to establish the proper pH level as well as dissolved oxygen concentration. At low system temperatures, i.e. <250 °F, hydrazine is present to control the dissolved oxygen concentration. During the heat-up phase, the hydrazine decomposes into ammonia, which contributes to the system pH. The ammonia concentration also tends to decrease during the heat-up as the ammonia is lost as an off-gas to the letdown system gas stripper or to the pressurizer steam space. Therefore, during the heat-up phase to 350 °F, the lithium concentration is increased as needed to compensate for the decreasing effects of the hydrazine and ammonia in maintaining the proper system pH. At approximately 350 °F, the hydrazine and ammonia concentrations have decreased to the point where their effect on pH is minimal, and the lithium concentration is adjusted and maintained within the specification of 1 to 2 ppm for the remainder of the heat-up.
* Condensate system dissolved oxygen concentration--High dissolved oxygen concentrations in the condensate system were due to low heat conditions in the condensate system, such that the reaction of the hydrazine with dissolved oxygen proceeded at a slow rate. Because the condensate system was not used to feed the steam generators, it was determined that no additional action was required for the purposes of this test.

340 °F/500 psia plateau - The RCS chemistry was within specification, with the exception of lithium. The RMWT and RWT were within the limits of 74HF-1SS01. The steam generators were within the operating specifications with the exception of dissolved oxygen. The feedwater system was not operating. The auxiliary feedwater system was in service to feed the steam generators. The auxiliary feedwater was maintained within specifications with the exception of dissolved oxygen. The condensate system was in long path recirculation through the condensate polishers. The chemistry of the condensate system was within specifications, with the exception of dissolved oxygen.

A Test Exception Report was written to document the test exceptions and out-of-specification conditions. The chemistry control parameters were within specification prior to proceeding to the next plateau with the exception of the dissolved oxygen in the condensate system and the RCS low Lithium concentration. These conditions were evaluated and, using the same bases detailed above for the 280 °F/380 psia plateau, were determined to be acceptable for proceeding to the next test plateau.

450 °F/1100 psia plateau - The RCS and the steam generators were within specifications. The feedwater system was not in operation. The auxiliary feedwater system was used to feed the steam generators, and its chemistry was maintained within the specifications required by 74HF-1SS01. The condensate system was in long path recirculation with the condensate polishers in service. The specifications, with the exception of dissolved oxygen, were maintained. This condition was evaluated and, using the same basis detailed above for the 280 °F/380 psia plateau, was determined to be acceptable for proceeding to the next test plateau.

565 °F/2250 psia plateau - The RCS, the RMWT, the RWT, and the steam generators all were within the specifications of 74HF-1SS01. Auxiliary feedwater was used to feed the steam generators. The condensate was in long path recirculation through the condensate polishers and was within specification with the exception of dissolved oxygen. The condensate hydrazine concentration was increased to bring the condensate dissolved oxygen within specification.

CONCLUSIONS

The test objectives for 74HF-1SS01 were satisfied in that overall proper chemistry for the RCS and secondary system was maintained at system operating conditions. The prescribed sampling frequency was adequate to ensure proper chemistry control and out-of-specification conditions were detected in a timely manner.

3.6 Postcore Pressurizer Spray Valve and Control Adjustments (Section 14.2.12.3.6)

TEST OBJECTIVES AND SUMMARY

Test 73HF-1RC10, "Pressurizer Spray Valve and Control Adjustment", was performed to establish the proper settings for the continuous (bypass) pressurizer spray valves, to measure the rate at which pressure is reduced by maximum pressurizer spray, and to measure the maximum pressurization rate. The following acceptance criteria applied to the test: (1) the continuous spray flow was to be adjusted such that spray line temperature would be no more than 70 °F lower than the cold leg temperature; and, (2) operation of both pressurizer spray valves together would reduce the pressurizer pressure at a rate equal to or greater than 1.06 psia per second from a nominal pressure of 2250 psia.

This test was performed on May 6, 1985, with the RCS at approximately 565°F, 2250 psia, and full flow conditions. The acceptance criteria were satisfied.

TEST DESCRIPTION

Initial data was gathered to determine the temperature difference between the cold legs and the pressurizer spray line with both main spray valves closed and both continuous spray valves set at 50% open. Permanent plant temperature instrumentation on the cold legs was used to determine the average cold leg temperature. Two strap-on thermocouples located on the spray line near the pressurizer were used to determine the average spray line temperature. The temperature difference was found to be greater than 70 °F, so the continuous spray valves were readjusted (by equal amounts), conditions were allowed to stabilize, and the new temperature difference was determined as before. This procedure was repeated until the acceptance criterion for the temperature difference was satisfied.

The effectiveness of the pressurizer spray was measured by opening the main spray valves, securing all pressurizer heaters, and recording pressurizer pressure as a function of time to determine the depressurization rate. These measurements were performed with both main spray valves open, as well as with each valve opened individually. Following each depressurization, the main spray valves were closed, the pressurizer heaters were energized, and pressurizer pressure was recorded as a function of time to determine the pressurization rate.

TEST RESULTS

The continuous spray valves were set to obtain a temperature difference between the cold legs and the spray line near the pressurizer of less than 70 °F (actual measured temperature difference was approximately 64 °F). Furthermore, the depressurization rate obtained using both main spray valves was measured to be 6.47 psia per second, which is greater than the minimum required rate of 1.06 psia per second. Therefore, the acceptance criteria for this test were satisfactorily met.

It should be noted that the setting of the continuous spray valves which resulted in the 64 °F spray line/cold leg temperature difference also necessitated continuous energization of some of the pressurizer backup heater banks in order to maintain RCS pressure. Although this was acceptable from a test standpoint, it was undesireable from an operational standpoint and the problem was addressed in an Engineering Evaluation Request (EER) to Bechtel and C-E. As an interim disposition of the EER, the continuous spray valves were readjusted slightly to reduce the bypass spray flow while maintaining an acceptable temperature difference. The main source of this problem, however, was later traced to the leakage of spray flow through the main spray valve while in the closed position. Additionally, the method used to measure the spray line temperature was determined to have resulted in excessive bypass flow. The spray line data in this test was measured using strap-on thermocouples. There are inherent inaccuracies in this method which result in an overly conservative (i.e. too low) value for the spray line fluid temperature and, hence, an overly large calculated temperature difference. During the Precore Hot Functional Test on PVNGS Unit 2, spray line temperature data was recorded utilizing spring loaded thermocouples and an auxiliary thermowell in the bonnet of the spray line check valve, which provided a more accurate measurement of the spray line fluid temperature. The data gathered during the Unit 2 test will be evaluated under the aforementioned EER, and a final bypass valve setting for Unit 1 will be determined and implemented.

CONCLUSIONS

The testing required by CESSAR was completed and the acceptance criteria were satisfied. An EER was generated to determine spray bypass valve settings which will minimize pressurizer heater operation while not exceeding the acceptable limit for spray line/cold leg temperature difference. Final disposition of this EER will be based on the correlation of Unit 1 and Unit 2 test data to establish final bypass valve settings.

3.7 Postcore Reactor Coolant System Leak Rate Measurement (Section 14.2.12.3.7)

TEST OBJECTIVE AND SUMMARY

The purpose of this test was to measure the reactor coolant system (RCS) leakage at hot zero power conditions. Testing was performed using the PVNGS surveillance test procedure 41ST-1RC02, "RCS Water Inventory Balance". In this test, "identified" and "unidentified" RCS leakage must be within the limits specified by Technical Specification 3.4.5.2; namely, that identified leakage shall not exceed 10 gpm and that unidentified leakage shall not exceed 1 gpm. Testing was performed at least once every 72 hours during Postcore Hot Functional Testing while the plant was in Modes 3 and 4. The test results were within allowable limits.

TEST DESCRIPTION

The test is performed by measuring the changes in water inventory of the RCS and Chemical Volume Control System (CVCS) over a two hour interval. Changes in the levels of the Pressurizer, Volume Control Tank, Reactor Drain Tank, and Safety Injection Tanks, as well as changes in the RCS temperature and pressure, are recorded and correlated to volume to determine the leakage rates in gallons per minute (gpm).

TEST RESULTS

This surveillance test was performed several times during the course of the Postcore Hot Functional Testing, including six performances with the RCS at hot zero power conditions. Acceptable test results were obtained by each performance. A typical set of test results is illustrated by the May 6, 1985 performance, which measured identified and unidentified leakages of 1.04 gpm and 0.4 gpm, respectively. This particular measurement was conducted with the RCS at 564 ^{o}F and 2235 psia.

CONCLUSION

The RCS leakage determined during postcore testing was well within the limits specified by the Technical Specifications.

3.8 Postcore Incore Instrumentation Test
(Section 14.2.12.3.8)

TEST OBJECTIVES AND SUMMARY

PVNGS procedure 73HF-1RI01, "Post-Core Movable Incore Instrumentation Test", was performed over the period of January 30 to May 13, 1985. The objectives and acceptance criteria of this test were as follows:

1. To determine the Movable Incore Drive System (MICDS) path length measurements by manual means.
2. To determine the MICDS path lengths using the detector drive system encoder, and compare the manual and encoder length measurements. Repetitive measurements of each path were to agree with the average length for the given path within 0.3 inch. The manual and encoder length measurements were to agree within 3.0 inches.
3. To install the permanent plant movable incore detectors (fission chambers).
4. To demonstrate proper computer control of the movable incore detectors.
5. To verify proper operation of the MICDS at Hot Shutdown and Hot Standby conditions by accessing specified core locations in the manual control mode. This would also check changes in the physical fit of the detector in the path and the path growth due to temperature changes.
6. To verify that leakage resistances for the Fixed Incore Detectors are at least 10 megohms at Hot Standby conditions.

TEST DESCRIPTION

With the RCS at ambient conditions, the dummy detector cable for one of the MICDS drive units was removed from the drive unit and manually inserted into each of the movable incore detector paths to measure the path lengths. The dummy cable was then reinstalled into its drive unit and the path lengths were remeasured by driving the dummy cable into each path using the Manual Control Box and recording the encoder readout from the Control Box. The measured path lengths were then compared, manual versus encoder.

Next, the permanent detectors (fission chambers) were installed with their associated cabling. The Manual Control Box was used to operate the system to measure the transfer times and drive rates for use as input to the computer control program. The computer control program was then tested to demonstrate its operability.

Following the heatup of the RCS to Hot Shutdown conditions, three of the path lengths were remeasured to obtain baseline data on tube growth due to temperature. These remeasurements were performed again at Hot Standby conditions. Additionally, while at Hot Standby conditions (565 ^{o}F, 2250 psia nominal), the leakage resistance of each fixed incore detector was measured using a High Resistance Meter, to check for any abnormalities. The automatic test functions of the Fixed Incore Amplifier Bins (zero output; full scale output; insulation resistance) were also initiated and verified.

TEST RESULTS

The manual and encoder path length measurements were performed using the dummy detector and recorded as required. The repetitive measurements were within the 0.3 inch tolerance.

The encoder to manual measurement comparison, based upon data recorded using the dummy detector cable, was within the 3 inch tolerance. One drive unit had an average difference of 0.25 inch, while the average difference for the other drive unit was 2.04 inches. However, when the three incore paths were remeasured at Hot Shutdown and Hot Standby conditions using the actual movable detectors (fission chambers), a difference of over 6 inches was measured between the encoder readings obtained with the fission chambers and the manual readings obtained with the dummy detectors. These differences were initially attributed to excessive tube growth, but the vendor indicated that the tube growth should be less than 0.25 inch. Further investigation determined that the differences were caused by the different helical wraps of the dummy detector cables and the fission chamber cables. This difference in the helical wraps affected the way in which the encoder tracked the length of the detector cable being moved, thus causing a different indication of path length for the two types of cables used when no physical difference in the path length actually existed. Therefore, the 3 inch tolerance was not valid for comparisons between the path lengths determined using the dummy detectors and those determined using the fission chambers. As a final resolution of this problem, a retest was performed to measure the movable incore path lengths using the fission chambers. This set of data will be used for operating the system. It should be noted that during the retest, repetitive measurements of each path agreed with the average length for the path within the 0.3 inch tolerance. It should also be noted that during the final review of the test results by the PVNGS Test Results Review Group, a procedure change was approved to delete the acceptance criterion specifying a 3 inch tolerance between the manual and encoder measurements.

The computer operation of the MICDS was not successful. During testing, the computer would access the correct path, but would not properly control the insertion of the detector into the path. This resulted in the detectors being driven into the end of the path tube at high speed (14.4 in/sec) on at least three occasions (the detectors were visually inspected and electrically checked, but no damage was apparent). Following this, a Test Exception Report was initiated to address this problem. There are two major reasons for the failure: (1) the Plant Computer was overtasked and thus the MICDS program was frequently interrupted; and (2), there are numerous problems in the MICDS program software. Since the performance of this test, the Plant Computer has undergone a hardware upgrade to alleviate the first problem. However, the problem with the Plant Computer prevented the gathering of enough data to identify the parts of the software which are not performing properly, so the second problem still remains to be corrected. Data gathered from the performance of this test on PVNGS Unit 2 will be used to identify the software problem areas, and a course of action will be determined at that time. Until the computer operation mode is successfully implemented, the MICDS may continue to be operated satisfactorily in the manual mode using the Manual Control Box.

The Fixed Incore Detector leakage resistances were well above the minimum acceptable value of 10 megohms, indicating that the detectors and cabling are free from electrical grounds. Additionally, the automatic test functions of the Fixed Incore Amplifier Bins were successfully tested and demonstrated to operate per design.

CONCLUSIONS

Although several problems were encountered during the performance of this test, the primary objectives and acceptance criteria as specified by CESSAR Chapter 14 were satisfied. That is:

(1) The leakage resistances of the fixed incore detectors were measured to be within design specifications.

(2) The ability of the MICDS to access the various paths was demonstrated using the Manual Control Box. Although the computer control mode was not operable, the system may be operated satisfactorily manually until a fix of the computer mode is implemented.

(3) The path lengths of all movable incore paths were measured by manual and mechanical means using the dummy detectors, and by mechanical means using the actual movable detectors.

Thus, the fixed and movable incore detector systems were determined to be functional to the extent required to support plant power operation. It should be noted that the MICDS is not safety-related and that the Technical Specification on incore detectors (Technical Specification 3.3.3.2) is not impacted by the operability status of the MICDS. In this Technical Specification, the MICDS is considered only as a backup to the fixed incore detectors. The Technical Specification can be satisfied solely through the fixed incore detectors, even if the MICDS is declared inoperable, and plant operation in any of its operational modes will not impacted.

4.0 INITIAL CRITICALITY
(CESSAR Section 14.2.10.2)

TEST OBJECTIVE AND SUMMARY

Initial criticality for PVNGS 1 was achieved under test procedure 72IC-1RX02, "Initial Criticality." The purpose of the procedure was to provide a safe, organized method for attaining the initial criticality of the PVNGS Unit 1 reactor and to verify that at least a one decade overlap existed between the startup excore detector channels and the log range of the safety excore detector channels.

The approach to criticality began on May 23, 1985 and initial criticality was achieved at 0145 on May 25, 1985. The RCS boron concentration at criticality was measured at 1054 ppm. An overlap greater than one decade between each startup channel and each log range of the safety channels was observed.

TEST DESCRIPTION

The approach to criticality began with the reactor coolant system at 320°F, 600 psia, approximately 1766 ppm boron and 2 reactor coolant pumps (RCPs) operating. The control rod banks (or groups) were withdrawn in a specified sequence until the control rods were out of the core, with the exception of a single group of four rods (Regulating Group 5), which was positioned at approximately 75 inches withdrawn (or midcore). The RCS boron concentration was then reduced to achieve criticality, with Group 5 used to control the chain reaction.

Core reponse during the control rod group withdrawal and RCS dilution was monitored in the control room by observing the change in neutron count rate as indicated by the permanent source range nuclear instrumentation (startup channels). Neutron count rate was plotted as a function of control rod group position and RCS boron concentration during the approach to criticality. Primary safety reliance was based on inverse count rate ratio (ICRR or 1/M) monitoring as an indication of the nearness and rate of approach to criticality.

TEST RESULTS

Initial criticality of the Unit 1 reactor was achieved in a safe and controlled manner as described above. The measured RCS boron concentration at criticality, 1054 ppm, fell within the acceptance criteria of 963 ppm to 1163 ppm and differed from the predicted value of 1063 ppm by only 9 ppm. An overlap greater than one decade was verified between each startup channel and the log range of the safety channels.

CONCLUSIONS

Satisfactory completion of this test demonstrated the validity of the core physics predictions for initial criticality, as well as the adequacy and redundacy of plant instrumentation in monitoring the reactor in the source and low power ranges.

5.0 LOW POWER PHYSICS TESTS
(CESSAR Section 14.2.12.4)

With the exception of the Low Power Biological Shield Survey Test (Section 14.2.12.4.1), all Low Power Physics Tests were performed as part of PVNGS procedure 72PY-1RX30, "Low Power Physics Test".

5.1 Low Power Biological Shield Survey Test
(Section 14.2.12.4.1)

TEST OBJECTIVE AND SUMMMARY

PVNGS procedure 75PA-1ZZ01, "Biological Shield Survey", was performed during the phases of Low Power Physics Testing (LPPT) and Power Ascension Testing (PAT) to meet the following objectives:

(1) To measure radiation in accessible locations outside the biological shield.

(2) To obtain baseline radiation levels for comparison with future measurements of level build-up with plant operation.

Acceptance criteria for these measurements are based on predicted radiation levels for 100% power operation and are presented as maximum dose rates for five different access zones. Table 5-1 shows the applicable criteria and defines the access zones.

Baseline background data for this test was gathered on March 14, 1985 with the plant in Mode 5. Low power physics data was gathered on May 27, 1985 with the reactor critical and at 0% full power (FP) and the primary conditions of 320 °F, 600 psia; and on May 30, 1985, with the reactor critical and at 0% FP, and the primary at 565 °F, 2250 psia. The low power data met the acceptance criteria.

TEST DESCRIPTION

With the plant stabilized at the desired conditions, gamma and neutron radiation surveys were performed at over 400 selected locations in accessible areas outside the biological shield. These surveys were performed per the plant radiation survey procedure, and included general area surveys in rooms or areas as well as more detailed surveys around penetrations, shield plugs, and other areas where streaming could be occurring. A scan survey was also performed while the survey team was in transit between designated survey points. Surveys were performed in the Containment Building, Auxiliary Building, Main Steam Support Structure, Turbine Building, Fuel Building, Control Building, Decontamination and Laundry Facility, and at various site locations exterior to the plant.

TEST RESULTS

Baseline data was gathered on March 14, 1985 in the accessible areas. Comparison of this data to the acceptance criteria is not applicable. Low power physics data was gathered on May 27 and May 30, 1985. The data gathered in the accessible plant areas during the low power physics surveys showed no increases above the baseline measurements.

CONCLUSION

Reviews of the low power test results revealed no apparent deficiencies in the plant shielding. Sufficient baseline data was gathered for comparison with future measurements at higher power levels.

Table 5-1
RADIATION ZONE CLASSIFICATION

Zone Designation	Dose Rate (mrem/h)	Allowed Occupancy (Design)
1	Less than 0.5	Uncontrolled, unlimited access (plant personnel)
2	0.5 to 2.5	Controlled, limited access, (40 h/wk to unlimited)
3	2.5 to 15	Controlled, limited access (6 to 40 h/wk)
4	15 to 100	Controlled, limited access (1 to 6 h/wk)
5	Over 100	Normally inaccessible; access only as permitted by radiation protection personnel (1 h/wk)

5.2 CEA Symmetry Test
(Section 14.2.12.4.2)

TEST OBJECTIVE AND SUMMARY

The objective of this test was to demonstrate that no core loading or control rod/fuel fabrication errors existed which would result in measurable control rod (Control Element Assembly, or CEA) worth asymmetries. This testing was performed from May 29 to May 31, 1985 with the reactor at hot zero power conditions (565 °F, 2250 psia). The data recorded during the symmetry test was analyzed to verify that the reactivity worth of each CEA in a symmetric subgroup (of 4 CEAs total), relative to the average worth of a CEA in that same subgroup, was within 1.5 cents. The symmetric CEAs were determined to be within the specified tolerance.

TEST DESCRIPTION

In this test, the worth of each CEA was measured relative to the worth of the other CEAs within its symmetric subgroup. The technique used for the measurement involved the insertion of a "reference CEA" from each subgroup to its Lower Electrical Limit (LEL, or "full in" position) to establish a reference reactivity condition. The next specified CEA was then inserted to its LEL by trading its insertion with withdrawal of the reference CEA to its Upper Electrical Limit (UEL). The deviation of the resulting reactivity condition from the reference condition was recorded, and this CEA was then swapped to its UEL with insertion of the next CEA to its LEL. This process was repeated for all CEAs of the subgroup. The average deviation from the reference condition was then computed for the subgroup, and the deviation of each individual CEA was compared with this average. Differences which were no greater than ± 1.5 cents were acceptable; however, differences greater than this limit may indicate either a misloading of fuel or fabrication errors in the fuel or CEAs.

TEST RESULTS

Each CEA was checked within its symmetric subgroup as described above. Data was recorded and analyzed per procedure and the symmetric CEAs were found to agree to within the acceptance criterion of ± 1.5 cents of the symmetric CEA subgroup average.

CONCLUSIONS

Since the acceptance criterion for this test was satisfactorily met, it can be concluded that the fuel and CEAs were properly fabricated and the core was correctly loaded.

5.3 Isothermal Temperature Coefficient Test
(Section 14.2.12.4.3)

TEST OBJECTIVE AND SUMMARY

The Isothermal Temperature Coefficient (ITC) was measured four times during Low Power Physics Testing at various RCS temperatures, pressures, and control rod (CEA) configurations. The conditions under which the ITCs were measured and the dates of performance are listed below:

1) 320 °F, 600 psia, unrodded----------May 25, 1985
2) 320 °F, 600 psia,
   CEA Gps 5,4,3,2, and 1 inserted------May 26, 1985
3) 565 °F, 2250 psia, unrodded----------May 29, 1985
4) 565 °F, 2250 psia,
   CEA Gps 5,4, and 3 inserted----------May 29, 1985

The measured ITC values were required to be within $\pm$ 0.5 x 10^{-4} delta-K/K/°F of their predicted values, a condition which was met during all the measurements. The moderator temperature coefficient (MTC) was determined from each of the ITCs measured at 565 °F (hot zero power), and verified to be in compliance with the Tech Spec limits (+0.22 x 10^{-4} to -2.80 x 10^{-4} delta-K/K/°F per Technical Specification 3.1.1.3) in each case.

TEST DESCRIPTION

The ITC is defined as the the change in reactivity associated with a uniform change in the moderator and fuel temperature. To measure the ITC, the RCS temperature was changed approximately 5 °F at a rate of about 10-20 °F per hour, using the secondary Steam Bypass Control System. RCS temperature and core reactivity were recorded on a strip chart or, additionally, on an X-Y plotter. Following a short stabilization time at the new temperature, the RCS temperature was then returned to its initial value. Temperature and reactivity were again recorded during the transition.

The change in RCS temperature and the corresponding change in reactivity were obtained from the strip chart for both temperature swings and analyzed to determine the average reactivity change per °F. When an X-Y plotter was used to record the data, the ITC was obtained from the slope of the "best-fit" line drawn through the data points. The MTC was then determined by subtracting the predicted Fuel Temperature Coefficient from the measured ITC.

TEST RESULTS

The measured ITCs agreed with the predicted values within the acceptable band. These results are summarized in Table 5-2. The MTCs derived from the ITCs measured at 565 °F were -0.288 x 10^{-4} and -0.818 x 10^{-4} at the unrodded (ARO) and rodded (Groups 5,4, and 3 inserted) conditions, respectively. Both of these values are within the aforementioned Technical Specification limits.

CONCLUSIONS

The accurracy of the predicted isothermal temperature coefficients, calculated for various conditions of RCS temperature, pressure, and CEA position (i.e. boron concentration), was verified by the measured values, all of which were well within their acceptance criteria. Furthermore, the MTCs determined at hot, zero power conditions were in compliance with the Technical Specification limits.

TABLE 5-2

ISOTHERMAL TEMPERATURE COEFFICIENTS
($x10^{-4}$ delta-K/K/^{o}F)

Conditions	Predicted	Measured	Diff. (M-P)	Accept. Diff.
320 ^{o}F, 600 psia				
unrodded	+0.03	-0.128	-0.158	±0.50
Grps 5 to 1 inserted	-0.25	-0.37	-0.12	±0.50
565 ^{o}F, 2250 psia				
unrodded	-0.19	-0.44	-0.25	±0.50
Grps 5 to 3 inserted	-0.91	-0.97	-0.06	±0.50

5.4 Shutdown and Regulating CEA Group Worth Test (Section 14.2.12.4.4)

TEST OBJECTIVE AND SUMMARY

The purpose of this test was to determine the individual group worths of the regulating and shutdown control rod (Control Element Assembly, or CEA) groups, and to sum those measured group worths to demonstrate the adequacy of the shutdown margin. Figure 5-1 shows the relative locations of the CEA groups in the PVNGS 1 core. The regulating CEA groups (5,4,3,2,1) were measured with the RCS at 320 °F, 600 psia on May 26, 1985; and at hot zero power (565 °F, 2250 psia) on May 28 and 29, 1985. Shutdown groups B and A, less the most worthy CEA ("A-1"), were measured with the RCS at 320°F, 600 psia on May 26 and 27, 1985. The measured individual group worths were required to be within ± 15% of their predicted values, or ± 0.10% delta-K/K (whichever is larger) . The total worth of the CEA groups was required to be within ± 10% of its predicted value.

TEST DESCRIPTION

If the group to be measured was initially withdrawn from the core, a constant dilution of the boron concentration was initiated. Insertion of the desired CEA group was then performed in periodic, discrete steps, to offset the change in core reactivity from the boron dilution, and thus maintain power and reactivity within the desired control bands. Reactivity and power were recorded on a strip chart recorder. Insertion of the group continued until it reached its lower limit, at which time the dilution was secured or insertion of the next CEA group to be measured began. If the group was initially inserted in the core, a constant boration was initiated and the group was withdrawn from the core using the same general technique described above until it reached its upper limit. In either approach, only one group was moved at a time, with no overlap between groups.

The data used to determine the CEA group worths was obtained from the reactivity strip charts. The reactivity change for each discrete group movement was determined and then summed over the length of the entire group to produce an integral worth.

TEST RESULTS

The measured group worths are shown in Table 5-3. With the exception of the A-1 worth, all of the CEA group worth measurements agreed with the predicted values within the allowed tolerances, including the total group worth. A Test Exception Report was generated to evaluate the deviation of the A-1 worth from its acceptance band. The test results were reviewed by Combustion Engineering, and it was concluded that the deviation of the measured worth of A-1 from its prediction had no impact on the Safety Analysis. This conclusion was based on the fact that the measured worth of A-1 was conservative with respect to its prediction, and on the fact that the total group worth was within its acceptance criterion.

CONCLUSIONS

The accuracy of the predicted CEA group worths, with the exception of group A-1, was confirmed by the measured values. The measured worth of group A-1 was more conservative than its prediction and was determined to have no impact on the Safety Analysis. Furthermore, the total measured worth was in acceptable agreement with its prediction.

TABLE 5-3

INDIVIDUAL CEA GROUP WORTHS
(%delta-K/K)

Conditions	Group	Pred.	Meas.	Diff. (M-P)	Accept. Diff.
320 °F, 600 psia					
	5	-0.093	-0.101	-0.008	±0.100
	4	-0.223	-0.241	-0.018	±0.100
	3	-0.795	-0.722	+0.073	±0.119
	2	-0.730	-0.750	-0.020	±0.110
	1	-1.518	-1.418	+0.100	±0.228
	B	-3.353	-3.615	-0.262	±0.503
	A-1	-0.371	-0.547	-0.176	±0.100
	Total (N-1)	-7.083	-7.397	-0.314	±0.708
565 °F, 2250 psia					
	5	-0.260	-0.277	-0.017	±0.100
	4	-0.421	-0.445	-0.024	±0.100
	3	-0.848	-0.790	+0.058	±0.127
	2	-0.994	-1.037	-0.043	±0.149
	1	-1.276	-1.231	+0.045	±0.191

FIGURE 5-1

RELATIVE CORE LOCATIONS
OF CEA GROUPS

					1	2 S	3	4 3	5	6 S	7					
			8	9	10 A	11	12 1	13	14 1	15	16 A	17	18			
		19 4	20	21 2	22	23 P_2	24	25 3	26	27 P_2	28	29 2	30	31 4		
	32	33	34 B	35	36 B	37	38 B	39	40 B	41	42 B	43	44 B	45	46	
	47	48 2	49	50 P_1	51	52	53	54 5	55	56	57	58 P_1	59	60 2	61	
62	63 A	64	65 B	66	67 4	68	69 A	70	71 A	72	73 4	74	75 B	76	77 A	78
79 S	80	81 P_2	82	83	84	85	86	87	88	89	90	91	92	93 P_2	94	95 S
96	97 1	98	99 B	100	101 A	102	103 3	104	105 3	106	107 A	108	109 B	110	111 1	112
113 3	114	115 3	116	117 5	118	119	120	121 P_1	122	123	124	125 5	126	127 3	128	129 3
130	131 1	132	133 B	134	135 A	136	137 3	138	139 3	140	141 A	142	143 B	144	145 1	146
147 S	148	149 P_2	150	151	152	153	154	155	156	157	158	159	160	161 P_2	162	163 S
164	165 A	166	167 B	168	169 4	170	171 A	172	173 A	174	175 4	176	177 B	178	179 A	180
	181	182 2	183	184 P_1	185	186	187	188 5	189	190	191	192 P_1	193	194 2	195	
	196	197	198 B	199	200 B	201	202 B	203	204 B	205	206 B	207	208 B	209	210	
		211 4	212	213 2	214	215 P_2	216	217 3	218	219 P_2	220	221 2	222	223 4		
			224	225	226 A	227	228 1	229	230 1	231	232 A	233	234			
					235	236 S	237	238 3	239	240 S	241					

KEY:

5--Lead Regulating Bank
4--Second Regulating Bank
3--Third Regulating Bank
2--Fourth Regulating Bank
1--Last Regulating Bank

A--Shutdown Bank A
B--Shutdown Bank B
P_1--Part-length Subgroup 1
P_2--Part-length Subgroup 2
S--Spare CEA Locations

5.5 Differential Boron Worth Test
(Section 14.2.12.4.5)

TEST OBJECTIVE AND SUMMARY

The purpose of this test was to determine the differential boron worth at different reactor conditions. The measurement was performed on May 27, 1985 with the RCS at 320 °F, 600 psia; and on May 31, 1985 with the RCS at 565 °F, 2250 psia. The measured differential boron worths were required to be within ± 15 ppm/%delta-K/K of their predicted worths.

TEST DESCRIPTION

The differential boron worth is defined as the change in boron concentration (in ppm) which would cause a 1% delta-K/K change in reactivity. The data required to calculate the differential boron worth was obtained from the measurements of the CEA group worths (see Section 5.4) and of the critical boron concentrations (see Section 5.6). The differential boron worth was determined simply by dividing the change in critical boron concentration in going from one CEA configuration to another, by the total reactivity worth of the CEA groups moved.

TEST RESULTS

Both of the calculated differential boron worths agreed with the predicted values within ± 15 ppm/%delta-K/K. The comparison of the measured and predicted values is summarized in Table 5-4.

CONCLUSIONS

The accuracies of the predicted differential boron worths were confirmed by measurement at various CEA configurations and RCS conditions.

TABLE 5-4

DIFFERENTIAL BORON WORTHS
(ppm/%delta-k/k)

Conditions	Predicted	Measured	Diff. (M-P)	Accept. Diff.
320 °F, 600 psia	-73.8	-72.71	+1.09	± 15.0
565 °F, 2250 psia	-87.0	-87.43	-0.43	± 15.0

5.6 Critical Boron Concentration Test
(Section 14.2.12.4.6)

TEST OBJECTIVE AND SUMMARY

The Critical Boron Concentration (CBC) was measured four times during Low Power Physics testing, at various RCS temperatures, pressures, and control rod (CEA) configurations. The conditions under which the measurements were performed and the dates of performance are listed below:

1) 320 °F, 600 psia, unrodded----------May 25, 1985
2) 320 °F, 600 psia,
   CEA Gps 5,4,3,2 and 1 inserted--------May 26, 1985
3) 565 °F, 2250 psia, unrodded----------May 29, 1985
4) 565 °F, 2250 psia,
   CEA Gps 5,4 and 3 inserted----------May 29, 1985

The measured Critical Boron Concentrations were required to be within ± 100 ppm of their predicted values.

TEST DESCRIPTION

With the reactor critical at the desired CEA configuration and stabilized conditions, boron equilibrium was verified by observing that the reactivity drift was negligible. An RCS water sample was taken and analyzed for boron content. The measured boron concentration was then corrected for the worth of any CEA deviation from the position assumed for the prediction. This was done by inserting or withdrawing the deviating group to the assumed position and measuring the reactivity associated with the move (i.e., the residual worth). This residual worth was converted to ppm, using the differential boron worth, and used to appropriately adjust the measured concentration to provide the CBC for the same conditions assumed for the prediction.

TEST RESULTS

The measured Critical Boron Concentrations agreed with the predicted values within the acceptance criteria of ± 100 ppm. Table 5-5 provides a summary of the measured and predicted values.

CONCLUSIONS

The accuracies of the predicted Critical Boron Concentrations, calculated for various conditions of RCS temperature, pressure, and CEA positions, were confirmed by the measured values, all of which were well within their acceptance critieria.

TABLE 5-5

CRITICAL BORON CONCENTRATIONS
(ppm)

Conditions	Predicted	Measured	Diff. (M-P)	Accept. Diff.
320 °F, 600 psia				
--- unrodded	1067	1057	-10	±100
--- Grps 5 to 1 inserted	819	822	+3	±100
565 °F, 2250 psia				
--- unrodded	1062	1025	-37	±100
--- Grps 5 to 3 inserted	929	893	-36	±100

5.7 Pseudo Dropped and Ejected CEA Worth Test (Section 14.2.12.4.7)

TEST OBJECTIVE AND SUMMARY

This test is performed to measure the worth of the worst "dropped" control rod (CEA) from the all rods out condition, and the worth of the worst "ejected" CEA from the zero power dependent insertion limit (ZPDIL). The test performed measurements of the worths of the following CEAs:

* Worst dropped CEA (CEA 3)
* Next worst dropped CEA (CEA 9)
* Worst dropped part-length CEA (PLCEA) (CEA 31)
* Worst dropped PLCEA subgroup (P1)
* Worst ejected CEA (CEA 87)
* Next worst ejected CEA (CEA 19)

Figure 5-2 shows the relative location of these CEAs. The measurements were performed on May 29, 1985 with the RCS at 565 °F and 2250 psia. Both the measured dropped and ejected rod worths were required to be within ±0.1 %delta-K/K of the predicted worths.

TEST DESCRIPTION

The worths of the dropped CEAs/PLCEAs were measured as follows. The changes in reactivity were recorded by a strip chart recorder.

PLCEA 31-- Due to its small worth, this CEA was simply inserted in one continuous motion to its fully inserted position (Lower Electrical Limit, or LEL), and then withdrawn to its fully withdrawn position (Upper Electrical Limit, or UEL). No changes in RCS boron concentration were made.

CEA 3-- Dilution of the RCS boron concentration was initiated and this CEA was inserted in discrete steps to its LEL.

Subgroup P1-- P1 insertion was traded with withdrawal of CEA 3 until P1 was at its LEL and CEA 3 was at its UEL.

CEA 9-- CEA 9 insertion was traded with P1 withdrawal until CEA 9 was at its LEL and P1 was at its UEL.

With CEA groups 5 and 4 at the LEL, and group 3 partially inserted, the worths of the ejected CEAs were measured as follows. Again, all changes in reactivity were recorded by a strip chart recorder.

CEA 87-- CEA 87 withdrawal was traded with group 3 insertion until group 3 reached its LEL ("near" ZPDIL). At that point, boration of the RCS was initiated and CEA 87 was withdrawn in discrete steps to its UEL.

CEA 19-- CEA 19 withdrawal was exchanged with CEA 87 insertion until CEA 19 reached its UEL and CEA 87 reached its LEL.

To determine the worths of the dropped and ejected CEAs, the reactivity data for the measurements was obtained from the strip chart recorder and analyzed in a manner similar to that used to determine the individual CEA group worths (see Section 5.4).

TEST RESULTS

The measured dropped and ejected rod worths agreed with the predicted values within ±0.1 %delta-K/K. The measured and predicted values are summarized in Table 5-6.

CONCLUSIONS

The accuracies of the predicted dropped and ejected worths were confirmed by the acceptable agreement with the measured values.

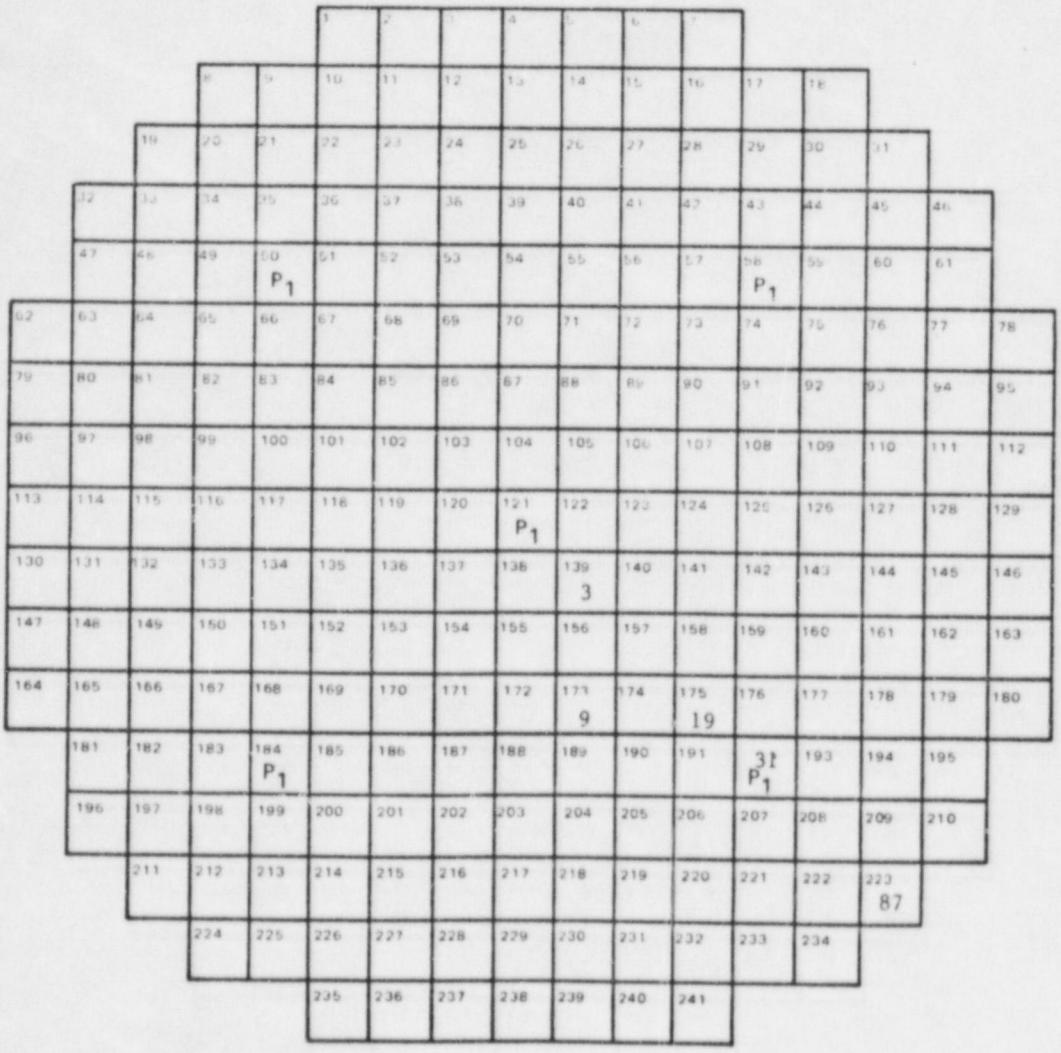
TABLE 5-6

DROPPED AND EJECTED CEA WORTHS
(%delta-K/K)

Case	Predicted	Measured	Diff. (M-P)	Accept. Diff.
Worst Dropped CEA (CEA 3)	-0.101	-0.073	+0.028	±0.100
Next Worst Dropped CEA (CEA 9)	-0.097	-0.069	+0.028	±0.100
Worst Dropped PLCEA (CEA 31)	-0.017	-0.020	-0.003	±0.100
Worst Dropped PLCEA Subgroup (P1)	-0.090	-0.066	+0.024	±0.100
Worst Ejected CEA (CEA 87)	+0.135	+0.147	+0.012	±0.100
Next Worst Ejected CEA (CEA 19)	+0.122	+0.138	+0.016	±0.100

FIGURE 5-2

RELATIVE CORE LOCATIONS
OF DROPPED AND EJECTED CEAS



KEY:

Worst Dropped CEA--CEA 3 (Box 139)
Next Worst Dropped CEA--CEA 9 (Box 173)
Worst Dropped PLCEA--CEA 31 (Box 192)
Worst Dropped PLCEA Subgroup--P_1 (Boxes 50, 58, 121, 184, 192)
Worst Ejected CEA--CEA 87 (Box 223)
Next Worst Ejected CEA--CEA 19 (Box 175)

6.0 POWER ASCENSION TESTS
(CESSAR Section 14.2.12.5)

6.1 Natural Circulation Test
(Section 14.2.12.5.1)

TEST OBJECTIVE AND SUMMARY

PVNGS procedure 72PA-1RX03, "Natural Circulation Test", was conducted on January 24 and 25, 1986 following a reactor trip from 100% full power. The principal objectives of the test were as follows:

(1) To verify natural circulation following the trip of the reactor coolant pumps.

(2) To determine that adequate boron mixing can be achieved under natural circulation conditions.

(3) To demonstrate by performance in combination with analysis the ability to perform a natural circulation cooldown followed by a plant depressurization to shutdown cooling conditions.

(4) To evaluate natural circulation flow conditions and provide data to verify the training simulator models.

(5) To demonstrate the ability to depressurize the RCS using auxiliary spray.

Procedure 72PA-1RX03 brought the unit to Shutdown Cooling initiation conditions (350 ^{o}F, 410 psia), satisfying the testing requirements of CESSAR. Transfer onto Shutdown Cooling under natural circulation conditions and the subsequent cooldown to 250 ^{o}F were performed by a separate procedure.

TEST DESCRIPTION

Natural circulation was established by tripping all four RCPs simultaneously. Once established, adequate flow through the core was verified by calculating a power-to-flow ratio. The power-to-flow ratio under natural circulation conditions should be less than or equal to the full power value.

Adequate boron mixing was demonstrated, while the RCS was maintained in Hot Standby, by increasing the primary boron concentration by a minimum of 100 ppm, and after a mixing period, verifying that 3 consecutive RCS samples were within 20 ppmb. Thereafter, shutdown margin was maintained throughout the test by providing makeup to the RCS from the Refueling Water Tank (RWT).

As required by Branch Technical Position BTP RSB 5-1, the RCS was then cooled and depressurized to Cold Shutdown conditions assuming only on-site (class) power was available, and that the most limiting single failure, an A-train diesel generator start failure, had occurred. Any A-train and non-class equipment used by the operators during the test was recorded.

Reactor vessel head voiding was integrated into the cooldown and depressurization strategy to achieve a cooldown rapid enough to satisfy the limited condensate inventory constraints and availability of nitrogen to the atmospheric dump valves. Following the cooldown to shutdown cooling initiation temperature, the RCS was depressurized to the void formation threshold. Once a void was observed, it was expanded to approximately 1000 ft^3, and then collapsed through charging and opening the reactor head vent. This process flushed cooler water through the hot, stagnant reactor vessel head area.

Prior to the depressurizations using auxiliary spray, letdown was secured and the RCS main spray valves were opened. During the depressurization, the auxiliary spray effectiveness (depressurization rate) was monitored, then the main spray valves were closed, and any changes in the depressurization rate were noted. Changes in the rate would imply that backflow through the main spray valves exists.

TEST RESULTS

The measured power-to-flow ratio was 0.365 and satisfied the acceptance criteria of less than 1.0.

Adequate boration of the RCS was verified under natural circulation conditions by borating the RCS 130 ppm, from 640 to 770 ppm. Three subsequent RCS samples were within 20 ppm of one another (minimum 764, maximum 781).

The RCS was cooled and depressurized from normal operating conditions (565 °F, 2250 psia) to shutdown cooling entry conditions (350 °F, 410 psia) and beyond under natural circulation conditions. Data was collected during the cooldown and is currently being analyzed to verify compliance with the requirements of BTP RSB 5-1. Upon completion of the analysis, the results will be forwarded to the NRC Reactor Safety Branch.

Complete and comprehensive data from the test was transferred to the ANPP Simulator Verification Group to evaluate the training simulator modelling of natural circulation characteristics.

The pressurizer depressurization rate utilizing auxiliary spray did not change when the main spray valves were closed (nor was auxiliary spray flow adjusted to maintain the depressurization rate) thus demonstrating the ability to depressurize using auxiliary spray with the internals of the main spray line check valve removed (implying that no backflow through the main spray lines exists).

CONCLUSIONS

The objectives of the test were met and the test acceptance criteria were satisfied. Natural circulation flow was established after the RCPs were tripped and the measured power-to-flow ratio of 0.365 easily met the acceptance criteria of less than 1.0. A natural circulation cooldown followed by a plant depressurization to shutdown cooling conditions was successfully accomplished. Data was collected and is being analyzed to verify compliance with BTP RSB 5-1.

6.2 Variable Tavg (Isothermal Temperature Coefficient and Power Coefficient) Test (Section 14.2.12.5.2)

TEST OBJECTIVE AND SUMMARY

The objective of the Variable T_{avg} test was to measure the Isothermal Temperature Coefficient (ITC) and the Power Coefficient (PC) at the 20, 50, 80 and 100% power plateaus. Also, the Moderator Temperature Coefficient (MTC) was calculated from the ITC and was extrapolated to 100% to ensure compliance with Technical Specifications upon reaching the higher plateaus. Testing was accomplished using test procedures 72PA-1RX02, 72PA-1RX25, 72PA-1RX35, 72PA-1RX50, "Variable T_{avg} (Isothermal Temperature Coefficient and Power Coefficient)Test" (20, 50, 80 and 100%, respectively).

The measured ITC and MTC were required to be within $\pm 0.5 \times 10^{-4}$ delta-K/K/°F and the measured PC within $\pm 0.2 \times 10^{-4}$ delta-K/K/%PWR of their respective predicted values, a condition which was met for all tests.

TEST DESCRIPTION

This test involved measuring the reactivity changes which accompany changes in temperature and power. The Isothermal Temperature Coefficient (ITC) is a measure of the change in reactivity caused by a change in RCS average temperature, while the Power Coefficient (PC) is the change in reactivity (with RCS temperature constant) associated with a change in reactor power. The following measurement techniques were employed to determine the ITC and PC:

1) ITC Measurement without CEA Movement--The secondary steam loading was adjusted to establish a new core inlet temperature. The reactivity effects of the temperature change resulted in a new power being established. After a brief stabilization period for data collection, the cycle was reversed by adjusting the secondary steam loading in the opposite direction to establish a new temperature and power. After another brief stabilization period for data collection, the secondary steam loading was readjusted and the core inlet temperature returned to its previous value, thus completing one temperature cycle. This cycling of temperature/power was repeated three or four times.

2) ITC Measurement with CEA Movement--The secondary steam loading was adjusted to establish a new core inlet temperature. Core power was held essentially constant by compensating for the reactivity effects of the the temperature change with CEA group 5 movement. After a brief stabilization period for data collection, the secondary steam loading was adjusted in the opposite direction and a new core inlet temperature established. Again, CEA group 5 movement was used to hold reactor power essentially constant. After another brief stabilization period for data collection, the temperature cycle was completed by adjusting the secondary steam loading to return the core inlet temperature to its previous value, while CEA group 5 movement was used to hold reactor power essentially constant. This cycling of temperature with accompanying CEA movement was repeated three or four times.

3) Power Coefficient with CEA Movement--CEA group 5 movement was used to introduce a reactivity change which caused a subsequent power change. The average core coolant temperature (Tavg) was held essentially constant by adjusting the secondary steam load to match reactor power. After a brief stabilization period for data collection, CEA group 5 was moved in the opposite direction (with secondary steam loading adjusted to keep the Tavg constant) and a new power was established. After data was collected, CEA group 5 was moved such that power was returned to its previous value, while the secondary steam loading was adjusted to hold Tavg essentially constant, thereby completing the power cycle. This cycling of power and secondary steam loading was repeated three or four times.

After the data was collected, the ITC and PC were calculated using a reactivity balance which includes the reactivity effects of CEA group 5 movements, the change in average coolant temperature, and the the change in reactor power. The calculation was an iterative one which used the predicted ITC and PC as the starting points and continued until successive iterations yielded agreement of $\pm 0.005 \times 10^{-4}$ for both the ITC and PC.

The MTC was calculated from the ITC by subtracting the predicted fuel temperature coefficient (FTC) as follows:

$$MTC = ITC - FTC$$

At the 20%, 50%, and 80% power plateaus, the MTC was extrapolated to 100% power to ensure compliance with Technical Specifications upon reaching the higher power plateaus.

TEST RESULTS

The measured ITCs, MTCs and PCs agreed with the predicted values within the acceptable range at all four power plateaus. These results are summarized in Table 6-1.

CONCLUSIONS

The agreement of the predicted isothermal temperature, moderator temperature, and power coefficients with the measured values was verified for the four major test plateaus. Furthermore, the MTC determined at each test plateau was verified to be in compliance with the Technical Specification limits.

TABLE 6-1

VARIABLE Tavg RESULTS
($x10^{-4}$ delta-K/K°F) ITC & MTC
($x10^{-4}$ delta-K/K/%PWR) PC

Power (Date)	Boron (ppm)	Coef.	Pred.	Meas.	Diff. (M-P)	Accept. Diff.
20%	820	ITC	-0.527	-0.587	-0.060	±0.5
(6/28/85)		MTC	-0.385	-0.442	-0.057	±0.5
		PC	-1.330	-1.180	-0.150	±0.2
50%	713	ITC	-0.601	-0.837	-0.236	±0.5
(9/5/85)		MTC	-0.459	-0.695	-0.236	±0.5
		PC	-1.150	-1.045	+0.105	±0.5
80%	672	ITC	-0.731	-0.972	-0.241	±0.5
(10/14/85-		MTC	-0.597	-0.838	-0.241	±0.5
10/15/85)		PC	-1.010	-0.986	+0.024	±0.2
100%	636	ITC	-0.814	-1.066	-0.252	±0.5
(12/28/85-		MTC	-0.684	-0.935	-0.251	±0.5
12/29/85)		PC	-0.930	-0.921	+0.009	±0.2

NOTES -- Diff. = Difference = Measured - Predicted
-- Accept. Diff. = Acceptable Difference range

6.3 Unit Load Transient Test
(Section 14.2.12.5.3)

TEST OBJECTIVE AND SUMMARY

The Unit Load Transient Tests were performed at the 50% and 100% power plateaus to demonstrate that the following control systems performed satisfactorily during design NSSS load changes:

Steam Bypass Control System (SBCS)
Feedwater Control System (FWCS)
Reactor Regulating System (RRS)
Pressurizer Pressure Control System (PPCS)
Pressurizer Level Control System (PLCS)
Control Element Drive Mechanism Control System (CEDMCS)

Testing was accomplished using procedure 73PA-1ZZ05 (50% test) on September 12, 1985 and procedure 73PA-1ZZ07 (100% test) on January 3, 1986. At 50% power several parameters did not meet their acceptance criteria due to improper signals being sent to the Reactor Regulating System (RRS). These problems were corrected by adjusting the Turbine Load Index (TLI) signal sent to the RRS and the plant was tested again at the 100% power plateau. At this power level all control systems responded as designed and key plant parameters remained within their required acceptance band.

TEST DESCRIPTION

Unit load ramp decreases of approximately 5% per minute were performed by closing down slightly on the turbine control valves resulting in a mismatch between reactor power and turbine power. As the turbine load demand decreased, the Reactor Regulating System (RRS) detected a decrease in turbine first stage pressure and an associated decrease in the Turbine Load Index (i.e., Turbine Power). The RRS then calculated a power error term (based on the difference between the TLI and reactor power) and a reference temperature (T-ref; based on the TLI). The reference temperature was compared to the actual core average temperature (T-avg) to determine a temperature error. When the summed error (temperature error plus power error) exceeded a specified value (ie, setpoint) the RRS instructed the CEDMCS to insert CEAs. CEA insertion continued until the summed error decreased below the setpoint.

The SBCS, upon sensing the change in turbine load, opened steam bypass valves to relieve excess heat generation. The valve(s) remained open until insertion of the CEAs (by the RRS) reduced heat generation by the primary system. Once the mismatch between the primary and secondary system power was eliminated, the valves reseated.

The decreasing power from the insertion of CEAs caused a reduction in core average temperature. Sensing the lower T-avg, the PLCS lowered the pressurizer water level by increasing the letdown flow and decreasing the charging flow until it matched a programmed level based on the new reactor power.

The decrease in reactor power also caused a decrease in the main steam flow. This decrease resulted in a steam flow/feed flow mismatch which is sensed by the Feedwater Control System (FWCS). To eliminate this mismatch, the FWCS decreased the feedwater pump speed and throttled back on the economizer feedwater control valve until the steam flow and feed flow were approximately equal.

The 10% step decreases were also performed by closing down on the turbine control valves. In this case; however, the valves were closed at a faster rate than they were during the ramp decreases. Control system response to the 10% step changes were similar to those described for the ramp decreases except they occurred over a shorter period of time.

TEST RESULTS

During the performance of this test at the 50% power plateau, several parameters deviated from the maximum and minimum values defined by the Acceptance Criteria (see Tables 6-2 and 6-3). A Test Exception Report was issued to document this problem and to investigate potential solutions. The resulting investigation showed that the Turbine Load Index (TLI) inputs to the Reactor Regulating System were out of calibration. These values were subsequently adjusted prior to the test at the 100% power plateau. The test at the 100% power plateau successfully demonstrated that all control systems responded as designed and all monitored parameters remained within their acceptance band, hence no further testing was required.

Tables 6-2 and 6-3 summarize the key plant parameters during a 5% per minute ramp from approximately 50% to 35% power and a 10% step change from approximately 35% to 25% power. As described earlier, not all parameters remained within their acceptance band at this power level, resulting in adjustments of the TLI inputs to the RRS before this test was performed at the 100% power plateau.

Tables 6-4 and 6-5 summarize the key plant parameters during a 5% per minute ramp from approximately 95% to 80% power and a 10% step change from approximatley 80% to 70% power. Evaluation of the data from this test showed that all parameters remained within their acceptance band and, therefore, that all control systems performed satisfactorily during design NSSS load changes

CONCLUSION

These tests successfully demonstrated that 10% step decreases and 5% per minute ramp decreases can be performed with the plant control systems maintaining key plant parameters within their design limits.

TABLE 6-2

SUMMARY OF TEST RESULTS (50%)
5% PER MINUTE RAMP DECREASE TRANSIENT

Parameter	Test Results Minimum	Test Results Maximum	Acceptance Criteria Minimum	Acceptance Criteria Maximum
Pzr. Pressure	2200	2228	2150	2335
Pzr. Level	33.5	47.0*	30.0	45.0
SG 1 Pressure	1118	1178	1050	1180
SG 2 Pressure	1122	1184*	1050	1180
SG 1 Level	76.0	80.0	70.0	88.0
SG 2 Level	76.0	82.0	70.0	88.0
RCS T-avg	570.5*	582.0	575.0	587.0
SG 1 T-hot	579.5	594.0	575.0	600.0
SG 2 T-hot	578.5	593.5	575.0	600.0

TABLE 6-3

SUMMARY OF TEST RESULTS (50%)
10% STEP DECREASE TRANSIENT

Parameter	Test Results Minimum	Test Results Maximum	Acceptance Criteria Minimum	Acceptance Criteria Maximum
Pzr Pressure	2206	2245	2150	2335
Pzr Level	31.5	47.4*	30.0	45.0
SG 1 Pressure	1120	1195*	1050	1180
SG 2 Pressure	1125	1199*	1050	1180
SG 1 Level	72.0	81.0	70.0	88.0
SG 2 Level	70.0	81.0	70.0	88.0
RCS T-avg	567.5	572.5	567.0	587.0
SG 1 T-hot	573.5*	586.0	575.0	600.0
SG 2 T-hot	572.5*	585.5	575.0	600.0

* Out of tolerance results

TABLE 6-4

SUMMARY OF TEST RESULTS (100%)
5% PER MINUTE RAMP DECREASE

Parameter	Test Results Minimum	Test Results Maximum	Acceptance Criteria Minimum	Acceptance Criteria Maximum
Pzr Pressure	2170	2228	2150	2335
Pzr Level	45.2	55.6	40.0	60.0
SG 1 Pressure	1077.0	1139.3	1020	1180
SG 2 Pressure	1083.0	1145.0	1020	1180
SG 1 Level	72.5	83.0	70.0	88.0
SG 2 Level	72.0	82.0	70.0	88.0
RCS T-avg	587.0	593.8	580.0	600.0
SG 1 T-hot	610.0	618.0	600.0	625.0
SG 2 T hot	609.0	618.0	600.0	625.0

TABLE 6-5

SUMMARY OF TEST RESULTS (100%)
10% STEP DECREASE

Parameter	Test Results Minimum	Test Results Maximum	Acceptance Criteria Minimum	Acceptance Criteria Maximum
Pzr Pressure	2159	2300	2150	2335
Pzr Level	41.0	55.0	40.0	60.0
SG 1 Pressure	1070	1165	1050	1180
SG 2 Pressure	1076	1171	1050	1180
SG 1 Level	75.0	83.0	70.0	88.0
SG 2 Level	75.0	82.0	70.0	88.0
RCS T-avg	582.0	592.0	580.0	600.0
SG 1 T-hot	601.0	614.0	600.0	625.0
SG 2 T-hot	600.2	613.0	600.0	625.0

6.4 Control Systems Checkout Test
(Section 14.2.12.5.4)

Test Objective and Summary

The Control Systems Checkout Tests were performed to demonstrate the satisfactory performance of the NSSS Control Systems during normal operations and under transient conditions. The control systems involved in these tests were:

Feedwater Control System (FWCS)
Steam Bypass Control System (SBCS)
Pressurizer Pressure Control System (PPCS)
Pressurizer Level Control System (PLCS)
Reactor Regulating System (RRS)
Reactor Power Cutback System (RPCS)
Control Element Drive Mechanism Control System (CEDMCS).

The specific objectives of the tests were:

1) To demonstrate that the FWCS, SBCS, PPCS, PLCS, RRS and CEDMCS can control the plant within acceptable tolerances of their programmed values and to also demonstrate that the FWCS and RRS can control steam generator levels and Tavg respectively within acceptable limits following minor adjustments in control setpoints. The procedures which were used to test these control systems during normal operations and their dates of performance are listed below:

73PA-1SF01, "Control Systems Checkout at 20% Power" (June 27-29, 1985)
73PA-1SF04, "Control Systems Checkout at 50% Power" (July 19-20, 1985)
73PA-1SF05, "Control Systems Checkout at 80% Power" (December 8, 1985)
73PA-1SF06, "Control Systems Checkout at 100% Power" (January 2-3, 1986)

2) To demonstrate that the aforementioned control systems enable the plant to sustain a loss of feedpump (tested at 70% and 100% power) or a loss of load (tested at 50% and 80% power): (a) without initiating a Reactor Protection System (RPS) signal or an Engineered Safety Features Actuation System (ESFAS) signal, (b) without opening any primary or secondary safety valves, and (c) without opening any atmospheric dump valves. The procedures used to conduct these tests and their dates of performance were:

73PA-1SF07, "RPCS Test - 50% Loss of Load" (September 16, 1985)
73PA-1SF08, "RPCS Test - 70% Loss of Feedwater" (October 19, 1985)
73PA-1SF09, "RPCS Test - 80% Loss of Load" (December 4, 1985)
73PA-1SF10, "RPCS Test - 100% Loss of Feedwater" (January 6, 1986)

TEST DESCRIPTION

73PA-1SF01 (SF04, SF05 & SF06) Each of these Control System Checkout tests consisted of three parts:

1) A test of the Feedwater Control Systems (FWCS) ability to control steam generator level during steady state and minor transient conditions. The steady state portion of the test simply involved determining whether the FWCS maintained steam generator levels within ±2% of the (steady state) level setpoint, while the transient portion involved both increasing and decreasing the level setpoint in ramp and step functions to determine whether the FWCS controlled steam generator levels to within ±2% of the new level setpoint (after allowing for a brief stabilization period).

2) A test of the Reactor Regulating System's ability to control Tavg with respect to its control signal Tref. A mismatch was created between Tavg (Tavg#1, Tavg#2 and Tavg (average of Tavg#1 and Tavg#2) were all tested) and Tref by either varying Tavg with boration/dilution or varying Tref with Main Turbine Power. The CEDMCS was then placed in the Auto Sequential mode to allow automatic RRS controlled rod movements. After an adequate time for temperature stabilization Tavg and Tref were required to be within $\pm 2^{o}F$.
3) A test of the ability of all the control systems to function in an integrated manner. The control systems were all placed in the automatic mode while the plant was operating at steady state conditions. Data was taken over a thirty minute period to verify that the control systems maintained their respective parameters within the acceptable control band.
73PA-1SF08 (1SF10) These tests were initiated by manually tripping one feedpump while the plant was operating at 70% and 100% power, respectively. The control systems were in the automatic mode and were allowed to perform as designed to counteract and control the effects of the loss of the feedpump. During the 70% test, on the loss of feedpump, the RPCS was expected to initiate a turbine setback, a turbine runback, a main turbine load inhibit increase, and a SBCS valve quick open block, but was not expected to drop any CEAs. The RRS was expected to insert CEA group 5 to match Tavg and Tref, while the other control systems performed as designed. In the 100% test the RPCS was expected to actuate and drop the selected CEAs. The RRS was expected to match Tavg and Tref by inserting CEAs, while the other control systems performed as designed.
73PA-1SF07 (SF09) These tests initiated a loss of load event at 50 and 80% power, repectively, by manually opening a relay to simulate a generator differential trip. All the control systems were in the automatic mode and were allowed to perform as designed. During the 50% test the RPCS was expected to indicate an actuation on the loss of load but was not expected to drop any CEAs. The RRS was expected to insert CEAs to match Tavg and Tref, while the other control systems performed as designed. In the 80% test the RPCS was expected to actuate and drop the selected CEAs. The RRS was expected to match Tavg and Tref by inserting CEAs, while other control systems performed as designed.

Test Results

73PA-1SF01 (1SF04, 1SF05 & 1SF06) The ability of the control systems to perform as designed during steady state operations and minor transients was verified by these tests. The major test results are summarized in Table 6-6. However, some specific acceptance criteria were not satisfied and are explained below:

1) At 50%, 80%, and 100% power the feedwater flow rates did not agree with the expected values. Since one of the objectives of the testing was to verify the ability of the control systems to maintain the steam generator levels within acceptable limits and this was satisfactorily demonstrated (except as noted in item 4 below), the measured feedwater flow rates were considered to acceptable and the associated acceptance criteria inappropriate. Test Exception Reports were generated and accompanying Engineering Evaluation Requests were sent to Combustion Engineering (C-E). C-E reviewed the data and concurred with the acceptability of the measured flow rates.

2) At 20%, 50%, and 80% power, the steam generator pressures exceeded (on the low side) the acceptance criteria. A Test Exception Report was generated which explained that the SBCS does not control steam generator pressure under normal operating conditions. Even if it did, it could not control deviations below the setpoint, since all it could do is open the steam bypass control valves, which would only serve to decrease the steam pressure. The results were consequently accepted as is. This conclusion was also reviewed and concurred with by C-E.

3) At 50% power pressurizer level exceeded its acceptance criteria. A Test Exception Report was generated which stated that the indicator used did not have high enough resolution and that acceptable data recorded from an alternate indicator would be used.

4) Also at 50% power, the Steam Generator # 2 level failed to meet its acceptance criteria. Spikes were found on the tranmitter's signal and a Work Order was generated to increase dampening on the level transmitter. A Test Exception Report detailing the foregoing was generated and no retest was required.

73PA-1SF08 (1SF10) The RPCS performed as designed upon receiving the loss of feedpump--at 70% it did not drop any CEAs, while at 100% it did drop CEA group 5. The RRS and other control systems also performed as designed. Although the overall test was successful, some specific acceptance criteria regarding feedwater flow rates were not satisfied during the performance of 70% loss of feedpump test (73PA-1SF08). As explained in the Test Exception Report this was determined to be of little significance since the overall test objectives of keeping the plant on line with no EFAS signal and no lifting of safety valves was achieved. Also, the steam generator levels recovered after the feedpump trip and were adequately controlled. The acceptance criteria were met during the 100% loss of feedpump test (73PA-1SF10).

73PA-1SF07 (1SF09) The 50% loss of load test (73PA-1SF07) resulted in a loss of offsite power event. The primary cause of this was the failure of the Turbine/Generator to maintain house loads, particularly the Reactor Coolant Pumps (RCPs), after the T/G was separated from the grid. As the reactor coolant pumps began to coast down after the loss of power a trip was initiated by the Core Protection Calculators. Because the acceptance criteria requires that the reactor not trip, a retest was required. A retest was conducted using basically the same test procedure except that the RCPs were being powered from the Startup Transformer and the Generator Differential Relay was used to simulate a loss of load. During this test the RPCS performed as designed i.e. did not drop any CEAs, the RRS and other control systems performed as designed and the test acceptance criteria were satisfied.

The initial performance of the 80% loss of load test also resulted in a reactor trip. The primary cause of this was a pressure pulse which caused a false indication of Low Wide Range Steam Generator level for approximately 192 to 470 milliseconds. The pressure pulse was caused by the sudden closure of the Turbine stop valves and the sensing of that pulse by the Wide Range Steam Generator level reference leg. The duration of pulse was sufficient to cause a PPS Low Wide Range Level Trip on Steam Generator # 2. To correct for this, the PPS time delay after detecting a Low Wide Range Level condition was increased to approximately 550 milliseconds. This adjustment prevents a pressure pulse from causing a PPS trip, yet maintains the overall response time for an actual event within the time delay assumed in the safety analyses. A retest was conducted

following this adjustment. The RPCS dropped CEA group 5 per design, the RRS and the other control systems performed as designed, and the test acceptance criteria were satisfied.

The ability of the RPCS to avoid a reactor trip due to a loss of load from 100% full power was also tested. This test is discussed in Section 6.7, "Unit Load Rejection".

CONCLUSIONS

These tests demonstrated that the control systems are capable of performing as designed under steady state, minor transient, or major transient conditions. The ability of the NSSS to sustain a loss of feedpump or loss of load at various power levels without a reactor trip or a lifting of the primary or secondary safety valves, because of proper control system response (including the RPCS), was also demonstrated.

TABLE 6-6

DEVIATION FROM CONTROL SETPOINTS FOR
CONTROL SYSTEM CHECKOUT TESTS

Acceptance Criteria	Z3PA-1SF01	Z3PA-1SF04	Z3PA-1SF05	Z3PA-1SF06
Feed Water Control:				
Steady state S/G lvls. ±2% from setpoint:	0.5	0.41	1.1	0.49
Transient S/G lvls. ±2% from setpoint:	1.0	1.5	1.0	1.0
Reactor Regulation:				
Tavg ±2°F from Tref.				
Tavg #1:	1.82	1.75	1.75	1.72
Tavg #2:	1.69	1.59	0.92	1.71
Tavg #3:	1.84	1.87	1.58	1.94
Integrated Checkout:				
Tavg ±2 °F of Tref:	1.66	1.8	0.58	0.95
#1 S/G level ±2% from setpoint:	-2 to +1	0 to +2	-.9 to +1	-2 to +1
#2 S/G level ±2% from setpoint:	-1 to +2	0 to +3*	-.3 to +1	-2 to 1
Pzr. pressure ±15 psia from setpoint:	0 to +1	0	0 to 7.5	0 to .49
Pzr. level ±1% from setpoint:	-1 to +1	0 to 1.43*	.02 to .62	0 to .24
#1 S/G pressure ±15 psia from setpoint:	-37.0*	-80.0*	-22.0*	N/A
#2 S/G pressure ±15 psia from setpoint:	-29.0*	-75.0*	-16.0*	N/A
#1 S/G pressure initial value + 15 psia:	N/A	N/A	N/A	1.48
#2 S/G pressure initial value + 15 psia:	N/A	N/A	N/A	0.74

* Denotes out of tolerance value.

6.5 Reactor Coolant and Secondary Chemistry and Radiochemistry Test (Section 14.2.12.5.5)

TEST OBJECTIVE AND SUMMARY

The Reactor Coolant and Secondary Chemistry and Radiochemistry Test 74PA-1SS01 was performed at various power levels throughout the power ascension test program. The principal objectives of the test were as follows.

(1) To conduct chemistry tests at various power levels with the intent of gathering corrosion data and determining activity buildup.

(2) To verify proper operation of the Process Radiation Monitor (PRM) and the Gas Stripper Effluent Monitor (GSEM).

(3) To verify the adequacy of sampling and analysis procedures and ensure proper chemistry control can be established and maintained.

(4) To verify that reactor coolant and secondary activity levels are maintained within the limits of the Technical Specifications.

Monitoring of plant chemistry during power ascension testing per 74PA-1SS01 was initiated on June 1,1985 at 0% power and was completed on December 26,1985 with the plant at 100% rated power.

TEST DESCRIPTION

Sampling and chemical analyses of the reactor coolant and secondary water systems were performed using the applicable plant operating procedures at various power levels during the power ascension. At each power level where chemistry testing was performed, samples from the reactor coolant system (RCS), steam generators (SGs), feedwater system, condensate system, reactor makeup water tank (RMWT), and the refueling water tank (RWT) were analyzed and the results compared to the operating specifications. Out-of-specification conditions were corrected by initiating the applicable Chemistry Control Instruction. Proper operation of the Process Radiation Monitor (PRM) and the Gas Stripper Effluent Monitor (GSEM) were to be verified by comparing PRM and GSEM readings to the laboratory analysis of grab samples which were representative of the fluid monitored by these systems.

TEST RESULTS

The key findings and major activities at each testing plateau are summarized below:

Zero Percent Power The RCS chemistry was within specification with the exception of lithium and hydrogen. A Chemistry Control Instruction was initiated to increase the lithium concentration in the RCS and hydrogen overpressure in the Volume Control Tank (VCT).

6% Rated Power The feedwater system was within specification with the exception of pH and dissolved oxygen. The condensate system was also within specification with the exception of dissolved oxygen. These conditions were corrected by increasing the hydrazine concentration in these systems to the proper value. The steam generators were within specification with the exception of cation conductivity, chlorides, sulfate, and sodium. These exceptions were corrected by initiating normal rate blowdown to the SGs with periodic shifts to abnormal rate blowdown.

13% Rated Power Inline instrumentation was placed in service on the condensate system to monitor sodium, pH, and dissolved oxygen. The accuracy of the inline monitors was verified by comparing readings to the laboratory analysis of grab samples. The SGs were continuously monitored for cation conductivity. Whenever conductivity exceeded the operating range specified for power operation, abnormal rate blowdown was initiated.
20% Rated Power Inline instrumentation was placed in service on the feedwater system to monitor sodium and dissolved oxygen. The accuracy of the inline monitors was verified by comparing readings to the laboratory analysis of grab samples. The SGs were in continuous normal rate blowdown. Blowdown was shifted to abnormal rate whenever cation conductivity, sodium/chloride, or sulfate values exceeded the operating range specified for power operation. The condensate system was monitored with the inline instrumentation whose readings were verified daily by comparing them to the results of a grab sample analysis.
50% Rated Power The feedwater and condensate systems were monitored with the inline instrumentation. The SGs were in continuous blowdown using the same criteria as at 20% power. The PRM and GSEM were inoperable for all test plateaus up through 50% power.
80% Rated Power Since the condensate demineralizers were in service throughout the test plateau, the feedwater pH was maintained in an expanded range of 9.0 to 9.6. This expanded range for feedwater pH is necessary to minimize condensate demineralizer resin exhaustion due to NH_3 removal by the demineralizer. The SGs were in abnormal rate blowdown throughout most of the 80% plateau. Chloride, sodium, sulfate, and cation conductivity were out of specification on several occasions due to testing of the condensate demineralizer bypass valve and to initial lineup of heater drains to the feedwater system. Parameters were returned to normal values by maintaining abnormal rate blowdown.
100% Rated Power Makeup water tanks were maintained within specification with the exception of high RMWT conductivity and boron concentration. This was due to higher than normal concentrations of ammonia and boron in the water returning to the RMWT from radwaste processing. RCS chemistry was maintained within specification with the exception of hydrogen and lithium. These concentrations were low on two occasions due to dilutions performed for a reactor shutdown and startup. The out-of-specification parameters for the RMWT and the RCS were restored to normal ranges. A major condenser tube failure caused the condensate demineralizers to become exhausted. Feedwater and condensate chemistry parameters were out-of-specification for approximately one hour on December 10, 1985 while main feedwater was in service. Reactor power was reduced to approximately 2% and main feedwater was taken out of service. Condensate demineralizers were regenerated, and feedwater and condensate systems were cleaned up using the condensate demineralizers. This same condenser tube leak also caused a loss of control of steam generator chemistry for approximately eight hours. The reactor and steam plant were subsequently shutdown on December 11, 1985 for condenser tube repairs. SG chemistry was recovered using blowdown to the condenser and feeding from the condensate storage tank (CST). Ammonia and hydrazine were added to the non-essential auxiliary feedwater to restore SG pH and to control feedwater oxygen. SG pH was recovered approximately eight hours later. Auxiliary feedwater hydrazine was maintained greater than 1.5 times the oxygen concentration while auxiliary feedwater was in service. Reactor power was restored to greater than 96% on December 22, 1985.

100% Rated Power (cont'd) Feedwater chemistry was maintained within specification during the remainder of the plateau. Feedwater pH was maintained in an expanded pH range due to the condensate demineralizers being in service. SG chemistry was restored to normal operating parameters using abnormal rate blowdown.

Radiation Monitors The PRM and GSEM remained either inoperable or uncalibrated throughout most of the Power Ascension Test program. When the PRM was in service it was useful only as an indicator of RCS activity trends. Readings taken during testing could not be correlated to RCS activity because the monitor was not calibrated to a specific energy. At higher plant power levels the PRM readings went off scale high. This will be corrected by installing a high range tube in accordance with the system technical manual instructions. The purpose of the high range tube is to reduce the size of the fluid sample seen by the PRM detector and thus lower the activity of the sample. This will bring the PRM readings back on scale. An inoperative interface between the PRM, GSEM, and a radmonitor minicomputer will be removed by a design change.

Corrosion Data The gathering of corrosion data took the form of system sampling for various metals such as iron and copper, measuring suspended solids in the SGs, measuring and monitoring RCS and secondary activity levels, and performing RCS sample isotopic analyses. Areas of radioactivity buildup and locations of probable crud traps can be identified by correlating chemistry data with the results of Radiation Protection surveys of RCS piping, letdown lines, and other associated piping. The corrosion data gathered during the power ascension indicated no unusual or unexpected results with the exception of high antimony levels. RCS antimony activities were significantly higher than what has been found in other U.S. PWRs. It was determined that the source of the antimony was the RCP journal bearings.

CONCLUSIONS

The adequacy of the sampling and analysis procedures and the ability to establish and maintain proper chemistry control was demonstrated throughout the power ascension test program. The RCS and secondary activity levels were maintained within the Technical Specification limits and increased as expected with increasing reactor power. The corrosion data gathered during the power ascension indicated no unusual or unexpected results with the exception of the high antimony levels.

The proper operation of the PRM and GSEM was not verified. Data collected during the power ascension testing showed that the monitors need additional work and modification. Additional testing will have to be performed to satisfy the requirements of the CESSAR. This results of this testing will be addressed in future supplement to this report.

6.6 Turbine Trip Test
(Section 14.2.12.5.6)

TEST OBJECTIVES AND SUMMARY

The principal objectives of the turbine trip test 73PA-1MT02 were as follows:

(1) To demonstrate that the Nuclear Steam Supply System (NSSS) responds and is controlled as designed following a turbine trip from 100% rated power, without the Reactor Power Cutback System (RPCS) in service.

(2) To assess the operation of the following control systems following a turbine/reactor trip from 100% load:

- Steam Bypass Control System (SBCS)
- Feedwater Control System (FWCS)
- Pressurizer Pressure Control System (PPCS)
- Pressurizer Level Control System (PLCS)

(3) To collect data for the verification of the CESEC transient analysis code for Palo Verde Unit 1.

This test was performed twice. On January 9, 1986, the first performance of the test resulted in a reactor trip due to a failure of the electrical fast bus transfer from auxiliary to startup transformer power supplies. Although a reactor trip was expected to result from the turbine trip, the cause of the trip was supposed to be high pressurizer pressure and the time to trip was expected to be longer than that observed. Thus, the transient resulting from the turbine trip was not consistent with that assumed in determining the test acceptance criteria. Additionally, demonstration of the electrical fast transfer feature was not one of the objectives of the test. Thus, the procedure was modified to eliminate this feature and the modified test was successfully performed on January 24, 1986 with satisfactory results.

TEST DESCRIPTION

Initially, the reactor was stabilized at 100% rated power with the house electrical loads supplied from the startup transformers. The turbine was tripped by manipulating a generator differential relay. The reactor tripped due to high pressurizer pressure approximately eight seconds after the turbine trip. During the transient, key plant parameters were recorded by several data acquisition systems.

TEST RESULTS

During the 100% turbine trip transient performed on January 24, the steam bypass, feedwater, pressurizer pressure, and pressure level control systems operated automatically to maintain the NSSS within the operating limits. During the sixty seconds following the initiation of the transient, several key parameters were trended and compared to single value acceptance limits derived from simulations of the event using the transient analysis code CESEC. Table 6-7 compares the test results with the single value acceptance criteria and shows that the acceptance criteria were met.

During the first performance of the test on January 9, 1986, the failure of the electrical fast bus transfer from auxiliary to startup transformer power supplies resulted in a four reactor coolant pump coastdown. An analysis of this incident revealed that this coastdown was faster than that assumed in the safety analysis for the total loss of flow incident. The more rapid coastdown was due to the electrical braking influence of other house loads on the RCP motors. The NSSS vendor analyzed the transient and reported that a reactor trip during a loss of flow event will occur more rapidly than assumed in the previous loss of flow analyses. This earlier trip more than offsets the loss of thermal margin that would be experienced due to the more rapid flow coastdown. The more rapid trip ensures acceptable results for this and any subsequent loss of flow events over the entire allowed operating range. Therefore, no penalties needed to be applied to compensate for the more rapid coastdown. PVNGS Unit 1 LER 86-006-00 addressed this event.

CONCLUSIONS

The Nuclear Steam Supply System demonstrated its ability to sustain a turbine trip from 100% rated power. The control systems operated satisfactorily during the resultant transient and maintained plant parameters within design limits.

TABLE 6-7

SINGLE VALUE ACCEPTANCE CRITERIA FOR TURBINE TRIP

Parameter	Test Result	Acceptable Limit
Max. Pressurizer Pressure (psia)	2396.3	2425
Min. Pressurizer Level (%)	25.9	22.9
Min. RCS Hot Leg #1 Temp (^{o}F)	565.5	564
Min. RCS Hot Leg #2 Temp (^{o}F)	566.3	564
Max. SG #1 Pressure (psia)	1177.9	1253
Max. SG #2 Pressure (psia)	1187.2	1253

6.7 Unit Load Rejection
(Section 14.2.12.5.7)

TEST OBJECTIVE AND SUMMARY

The primary objective of this test was to demonstrate that the Nuclear Steam Supply System (NSSS) can accomodate a 100% load rejection (1) without initiating a Reactor Protection System (RPS) signal or an Engineered Safety Features Actuation System (ESFAS) signal, (2) without opening any primary or secondary safety valves, and (3) without causing a turbine trip. Additional objectives of the test were:

1) To collect data for verification of the CESEC transient analysis code for PVNGS Unit 1.
2) To verify that the closing time of the Main Turbine Control Valves was acceptable.
3) To verify that the Power Load Unbalance Circuit (PLU) functions as designed and prevented a turbine trip (caused by development of an overspeed condition) following the 100% load rejection.
4) To assess the operation of the following control systems following a 100% load rejection:
   - Steam Bypass Control System (SBCS)
   - Feedwater Control System (FWCS)
   - Pressurizer Pressure Control System (PPCS)
   - Pressurizer Level Control System (PLCS)
   - Reactor Power Cutback System (RPCS)
   - Control Element Drive Mechanism Control System (CEDMCS).

Testing was accomplished using procedure 73PA-1MA01, "Unit Load Rejection Test" and was successfully completed on January 7, 1986.

TEST DESCRIPTION

This test initiated a unit load rejection by opening the unit generator main output breakers while the plant was operating at essentialy 100% power with a Turbine Generator gross output of 1315 Mwe. The control systems (SBCS, FWCS, RRS, PPCS, PLCS, RPCS and CEDMCS) were all in the automatic mode of operation and were allowed to perform as designed to counteract and control the effects of the load rejection. The RPCS was expected to actuate on the loss of load and drop the selected CEAs. The RRS was expected to match Tavg and Tref by inserting CEAs, while the other control systems performed as designed. To verify that the test was performed successfully, actual plant parameters were compared with the single value acceptance criteria as supplied by the CESEC transient analysis code predictions.

During the performance of this test the house electrical loads were supplied from the startup transformer. This was done to minimize the exposure to a Loss of Offsite Power (LOP) event and to subject the turbine to the maximum credible overspeed condition, i.e. to eliminate any electrical load on the generator from providing any braking effect.

TEST RESULTS

The acceptance criteria were fully satisfied. The Reactor Power Cutback System performed as designed, i.e. the reactor did not trip, no ESFAS signals were initiated and none of the primary or secondary safety valves were opened.

The plant parameters recorded during the sixty seconds following initiation of the transient and their comparison with the single value acceptance criteria supplied by the CESEC transient analysis code are provided in Table 6-8.

CONCLUSIONS

The test demonstrated that the NSSS can sustain a 100% load rejection without a reactor trip, turbine trip, or a lifting of the primary or secondary safety valves. The control systems operated satisfactorily throughout the transient and data was collected to verify the CESEC predictions.

TABLE 6-8

SINGLE VALUE ACCEPTANCE CRITERIA
FOR 100% UNIT LOAD REJECTION

Parameter	Test Results	Acceptance Limit
Max. Pressurizer Pressure (psia)	2339.2	2388
Min. Pressurizer Level (%)	35	29.4
Min. RCS Hot Leg #1 Temp.(^{o}F)	592	574
Min. RCS Hot Leg #2 Temp.(^{o}F)	592	574
Max. SG #1 pressure (psia)	1236.7	1242
Max. SG #2 pressure (psia)	1240.8	1242

6.8 Shutdown from Outside the Control Room Test (Section 14.2.12.5.8)

TEST OBJECTIVE AND SUMMARY

The objective of 73PA-1SF02, "Shutdown Outside the Control Room" was to demonstrate that the plant can be shutdown and maintained in a Hot Standby condition from outside the Control Room. The test was performed on September 16, 1985 while the reactor was operating at 20% of rated thermal power.

The acceptance criterion for this test was to perform a safe shutdown of the plant from outside the control room and maintain selected plant parameters within a specified range for at least 30 minutes using equipment that would normally be available only at the remote shutdown panel. An engineering evaluation of the test data concluded that a stable Hot Standby condition had been established and adequate control of RCS and steam generator parameters was maintained throughout the test.

TEST DESCRIPTION

The test was performed by utilizing a normal operating crew and a standby crew. The standby crew served as Control Room observers and were to take action only if a problem that involved plant safety developed. The operating crew consisted of the minimum shift complement as defined in Table 6.2-1 of the PVNGS Technical Specifications.

The operating crew performed the test by evacuating the Control Room and proceeding to the Remote Shutdown Panel. Once established, they initiated the reactor trip by opening the reactor trip breakers at the local reactor trip switchgear panel. After the trip was verified, they established control of the plant using equipment available at the Remote Shutdown Panel and maintained Hot Standby conditions for approximately one hour and fifteen minutes. Control of the plant was then transferred to the standby crew in the Control Room and the Remote Shutdown Panel was secured.

TEST RESULTS

During the performance of this test, several parameters deviated from the acceptable control range. This was due to a greater than predicted RCS cooldown which resulted from excessive leakage through a steam generator blowdown control valve. Once an auxiliary operator from the operating crew closed the containment isolation valve for the leaking steam generator blowdown valve, the RCS cooldown was mitigated. The operators were then able to demonstrate adequate temperature and pressure control over the Reactor Coolant System and Steam Generators from the Remote Shutdown Panel. Table 6-9 summarizes the key plant parameters monitored during this test.

A Test Exception Report was written to document the deficiencies and an Engineering evaluation of the test data was performed. This evaluation concluded that once Steam Generator blowdown was isolated, the operating crew was able to satisfactorily establish and maintain Hot Standby conditions from the Remote Shutdown Panel within the prescribed tolerances.

CONCLUSION

This test successfully demonstrated the ability of the minimum shift crew to trip the reactor from outside the Control Room and maintain the plant in a stable Hot Standby condition utilizing only the equipment available at the Remote Shutdown Panel.

TABLE 6-9

SHUTDOWN FROM OUTSIDE THE CONTROL ROOM
TEST RESULTS

Parameter	Test Results Minimum	Test Results Maximum	Acceptance Criteria Minimum	Acceptance Criteria Maximum
Pressurizer Pressure(psia)	2204	2254	2200	2275
Hot Leg Temperature(F)	552*	567	559	569
Pressurizer Level (%)	22*	37	greater than 26	
Steam Gen. Pressure(psia)	1056*	1150	1120	1220
Steam Gen. Level (%)	46	69	greater than 35	

* Out-of-tolerance results

6.9 Loss of Offsite Power Test
(Section 14.2.12.5.9)

TEST OBJECTIVE AND SUMMARY

The principal objective of the Loss of Offsite Power Test was to demonstrate that the reactor can be shutdown and maintained in a Hot Standby condition following the loss of all AC power. Testing was scheduled for performance at approximately 50% power in accordance with PVNGS procedure 73PA-1NA01 following completion of testing at the 80% power plateau.

On October 3, 1985, however, before the scheduled performance of 73PA-1NA01, PVNGS Unit 1 experienced an actual loss of all AC power while operating at approximately 52% of rated thermal power. The reactor shut down automatically and the plant was maintained in a stable Hot Standby condition for approximately 25 minutes, at which time offsite power was restored. The responses of the plant equipment and personnel during this unanticipated event were reviewed by the PVNGS Test Results Review Group and the Plant Review Board and were determined to have satisfied the objectives of the power ascension test as well as the intent of the regulatory requirements for the performance of this test.

TEST DESCRIPTION

The initial conditions that were to be used for the performance of 73PA-1NA01, Loss of Offsite Power, are summarized in Table 6-10. In addition, the pretest electrical lineup was to be set up such that when the main generator was tripped (the initiating event), plant electrical loads could not be switched to the startup transformers (ie, no "fast transfer" capability) resulting in a complete loss of all AC (LOAC) power. Sufficient data would then be collected to verify natural circulation through the core, stable steam generator levels (supplied by the motor-driven emergency feedwater pump), secondary system heat removal via the atmospheric dump valves (using a backup nitrogen supply) as well as general plant stability using only emergency power supplies (diesel generators). Stable hot standby conditions were to be maintained for approximately 30 minutes, at which time offsite power was to be restored.

TEST RESULTS

Table 6-10 shows the actual plant conditions as they were just prior to the unanticipated loss of offsite power. This data shows the similarity between the actual plant conditions that existed just prior to the unanticipated LOAC and those that were to be used for the power ascension test 73PA-1NA01. Table 6-11 shows the sequence of events that occurred during this event.

Following the loss of power, the reactor tripped on a Core Protection Calculator (CPC) generated low Departure from Nucleate Boiling Ratio (DNBR) signal due to the reactor coolant pumps (RCPs) coasting down. Due to the particular electrical lineup (in place for other electrical testing) the reactor trip, and subsequent master turbine trip, produced a loss of all AC power. The feedwater control system and the steam bypass control system were temporarily unavailable which resulted in the opening of the Main Steam Safety Valves (MSSVs) for secondary pressure control. Once backup power (diesel generators) was available, the control room operator took manual control of the secondary

pressure with the atmospheric dump valves (ADVs) and the MSSVs reseated. Secondary pressure peaked at 1280 psia then decreased as the operator assumed control. Once natural circulation had been established the secondary pressure stabilized around 820 psia.

The first charging pump was manually started approximately one (1) minute after the trip and suction was transferred to the refueling water tank (which had a boron concentration greater than 4000 ppm) within 15 minutes. Shutdown margin was verified to be greater than 6% delta k/k, per the applicable surveillance procedure, within one hour of the trip.

Verification of natural circulation was accomplished within 15 minutes after the trip. Primary parameters were monitored, and periodically recorded, to verify the following:

(1) Hot leg temperatures were stable or decreasing;
(2) Cold leg temperatures were close to and trending the steam generator saturation temperatures;
(3) Core delta T was less than the full power delta T of 57 °F (i.e., power to flow ratio was less than 1);
(4) Hot leg RTDs and Core Exit Thermocouples were trending consistently.

Reactor vessel head and plenum subcooling were maintained at greater than 28 °F to preclude any loop void formation that could degrade natural circulation. The motor-driven emergency feedwater pump supplied flow to the steam generators which maintained their levels above 38% (wide range), thus providing an adequate heat sink for the primary system. The main steam safety valves and the atmospheric dump valves were utilized to release the transferred heat to the atmosphere.

Pressurizer level was maintained between 39% and 47%, thus remaining above the heater cutout level (26%) throughout the event. RCS pressure peaked at 2290 psia (concurrent with the low secondary heat removal and subsequent pressure increase to the MSSV setpoint) and reached a low of 2120 once natural circulation was in progress. Auxiliary spray was utilized twice (for approximately one minute each time) for primary pressure control. Careful monitoring of the secondary steam and feedwater flows precluded both RCS overcooling and initiation of safety injection (SIAS).

CONCLUSIONS

Following the loss of the main generator and all offsite power, the plant was stabilized and natural circulation was verified. Although plant conditions were only maintained for approximately 25 minutes before offsite power was restored, the trended data indicate that Hot Standby conditions could have been maintained for a greater length of time had offsite power not been available. Thus, the equipment, controls and instrumentation necessary to remove decay heat from the core using only emergency power supplies was demonstrated for a sufficient period of time.

The PVNGS Test Results Review Group and the Plant Review Board subsequently reviewed the responses of the plant equipment and personnel to the unanticipated LOAC and determined that the event satisfied the objective of the power ascension test thereby obviating the need to perform 73PA-1NA01 for PVNGS-1. This position was formally submitted to the Nuclear Regulatory Commission via ANPP letter ANPP-34062-EEVB/BJA, E. E. Van Brunt, Jr. to George W. Knighton, dated November 20, 1985.

TABLE 6-10

LOSS OF OFFSITE POWER TEST
INITIAL CONDITIONS

Parameter	Actual Value	Test Value
Reactor Power (%)	52	50
Cold Leg Temperature (°F)	567	565
Pressurizer Pressure (psia)	2230	2250
Steam Generator Pressure (psia)	1140	stable
Steam Generator Level (%)	54	stable
Boron Concentration (ppm)	715	N/A
CEA Position	ARO*	ARO*
Control System Status		
-Reactor Regulating	Auto	Auto
-Control Element Drive	Standby	Standby
-Feedwater Control	Auto	Auto
-Steam Bypass Control	Auto	Auto
-Pzr. Level Control	Auto	Auto
-Pzr. Pressure Control	Auto	Auto
-Reactor Power Cutback	OOS**	OOS**

* -- ARO = All Rods Out = All control rods fully withdrawn
** -- OOS = Out of Service

TABLE 6-11
SEQUENCE OF EVENTS

Date	Time	Event
10/3/85	16:43:45	Electrical supply breakers open causing loss of offsite power
	16:43:46	Reactor and main turbine trip
	16:44	All CEAs inserted; diesel generators started and loaded
	16:44:35	Charging restored (42 GPM)
	16:45	Primary parameters monitored for verification of natural circulation
	16:48	Shifted RCS heat removal to ADVs
	16:57	Charging suction transferred to refueling water tank; Verified motor driven auxiliary feedwater started and feeding both generators
	17:03	Second charging pump started
	17:09	Offsite power restored
	17:10	Verification of shutdown margin commenced
	17:15	Cooldown of 15 ^{o}F/hr commenced
	17:35	Shutdown margin verified
	17:38	Diesel generator B unloaded and shutdown
	17:55	Charging pump suction transferred back to Volume Control Tank (VCT)
	18:00	RCS boron concentration = 916 ppm
	20:18	Started reactor coolant pump 1A; Forced circulation restored
	20:24	Started reactor coolant pump 2A
	20:26	Started reactor coolant pump 2B
	20:32	Commenced RCS heatup to normal operating temperature (565 ^{o}F)
10/4/85	01:09	Started reactor coolant pump 1B; All 4 RCPs in service

6.10 Biological Shield Survey Test
(Section 14.2.12.5.10)

TEST OBJECTIVES AND SUMMARY

PVNGS procedure 75PA-1ZZ01, "Biological Shield Survey", was performed during Low Power Physics Testing (see Section 5.1) and also during the major test plateaus of the Power Ascension Test program (20%, 50%, 80%, and 100% full power). The principal objectives of this test were:

(1) To measure the radiation levels in accessible locations outside the biological shield;

(2) To obtain baseline radiation levels for comparison with future measurements of level buildup with plant operation;

(3) To determine occupancy times for the measured areas during power operation.

Acceptance criteria for these measurements are based on predicted radiation levels for normal power operation (100% full power) and are presented as ranges of dose rates for five different access zones. Table 6-12 shows the applicable criteria and defines the access zones.

All survey points inside the Containment Building have met the applicable acceptance criteria. Some areas in the Auxiliary Building, however, were determined to be High Radiation areas while the FSAR had listed them as being either Zone 2 or Zone 3. These areas are currently undergoing an engineering evaluation.

TEST DESCRIPTION

With the plant stabilized at the desired conditions, gamma and neutron radiation surveys were performed at over 400 selected locations in accessible areas outside of the biological shield. These surveys were performed per the plant radiation survey procedure, and included general area surveys in rooms or areas as well as more detailed surveys around penetrations, shield plugs, and other areas where streaming could occur. A scan survey was also performed while the survey team was in transit between designated survey points. Surveys were performed in the Containment Building, Auxiliary Building, Main Steam Support Structure, Turbine Building, Fuel Building, Control Building, Decontamination and Laundry Facility, and at various site locations exterior to the plant.

TEST RESULTS

Baseline data was obtained on March 14, 1985 while the plant was in Mode 5 and again on May 27 and May 30 during Low Power Physics Testing. These values were then used for comparison to the data taken during Power Ascension.

On June 5, 1985 a survey was completed while the plant was operating at 3% full power (FP). All values were within the acceptable range for their designated zone.

On June 12, 1985 a survey was completed while the plant was operating at 20% FP. All values were within the acceptable range for their designated zone. Several survey points in the reactor coolant pump bays and near the reactor cavity, however, were found to exceed 500 mrem/hr and, since these areas will not be occupied during power operation, they were deleted from further surveys.

On September 10, 1985 data was obtained while the plant was operating at approximately 50% FP. Although all survey points met their acceptance criteria, the 100 foot and 120 foot elevations on the west side of the Containment Building showed a minor trend toward dose rates slightly higher than the east side.

The survey at 80% FP was performed on October 1, 1985 and showed all points meeting their acceptance criteria. Once again, the trend toward higher dose rates at the west end of the Containment Building was noted.

Data collection for the survey at 100% FP was performed over the period from December 30, 1985 through January 4, 1986. This data showed that several areas in the Auxiliary Building were now posted as High Radiation areas (Zone 5) while the FSAR shows them to be either Zone 2 or Zone 3. An engineering evaluation of this data is currently underway to determine the reasons for this discrepancy. All other survey points were within the acceptance criteria for this test.

CONCLUSIONS

All the data taken inside the Containment Building have met their acceptance criteria indicating the effectiveness of the Biological Shield. Some high radiation areas in the Auxiliary Building are currently being evaluated and will be discussed further in a future supplement to this report.

TABLE 6-12

RADIATION ZONE CLASSIFICATION

Zone Designation	Dose Rate (mrem/h)	Allowed Occupancy (Design)
1	Less than 0.5	Uncontrolled, unlimited access (plant personnel)
2	0.5 to 2.5	Controlled, limited access, (40 h/wk to unlimited)
3	2.5 to 15	Controlled, limited access (6 to 40 h/wk)
4	15 to 100	Controlled, limited access (1 to 6 h/wk)
5	Over 100	Normally inaccessible; access only as permitted by radiation protection personnel (1 h/wk)

6.11 Part Length CEA Xenon Oscillation Control (Section 14.2.12.5.11)

TEST OBJECTIVE AND SUMMARY

The objective of this test was to demonstrate a satisfactory technique for dampening axial xenon oscillations using the part length control element assemblies (PLCEAs). Testing was performed in accordance with PVNGS procedure 72PA-1RX30, "PLCEA Xenon Oscillation Control", over the period of September 5 through 7, 1985 with the reactor operating at approximately 53% full power (FP). The results of the test demonstrated that the PLCEAs can be satisfactorily used to dampen axial xenon oscillations.

TEST DESCRIPTION

This test began with the reactor core unrodded and operating at approximately 53% FP, with equilibrium xenon conditions. An axial xenon oscillation was initiated by inserting control rod groups 5 and P to mid-core, holding this configuration for approximately six hours, and then withdrawing both control rod groups to the full out position. The resulting xenon oscillation was then monitored by observing the shift in the axial power distribution as indicated by the Axial Shape Index (ASI*). The ASI is calculated by the Core Operating Limits Supervisory System (COLSS) using the fixed incore detector signals. As the xenon oscillation caused the ASI to move away from its equilibrium (i.e., steady state) value (called the Equilibrium Shape Index, or ESI), the part-length CEA group , Group P, was moved as needed to shift the axial power distribution and thus maintain the ASI within a specified control band about the ESI. For this test, the control band used was ESI $\pm$ 0.01. By using Group P movement to maintain the axial power distribution near its equilibrium state, the reactor operators would gradually dampen out the xenon oscillation.

TEST RESULTS

Following the withdrawal of groups 5 and P to start the oscillation, the shifting xenon concentration began to force the axial power distribution toward the top half of the core. Thus, Group P insertion was needed to maintain the ASI within its control band. After approximately 8 hours, however, the shifting xenon concentration began to force the axial power distribution back toward the bottom half of the core. Thus, Group P withdrawal was needed to maintain the ASI within its control band. Movement of Group P, and hence this test, continued for approximately 30 hours until the xenon oscillation was dampened.

$$* \text{ ASI} = \frac{(\text{power in lower core half} - \text{power in upper core half})}{\text{total core power}}$$

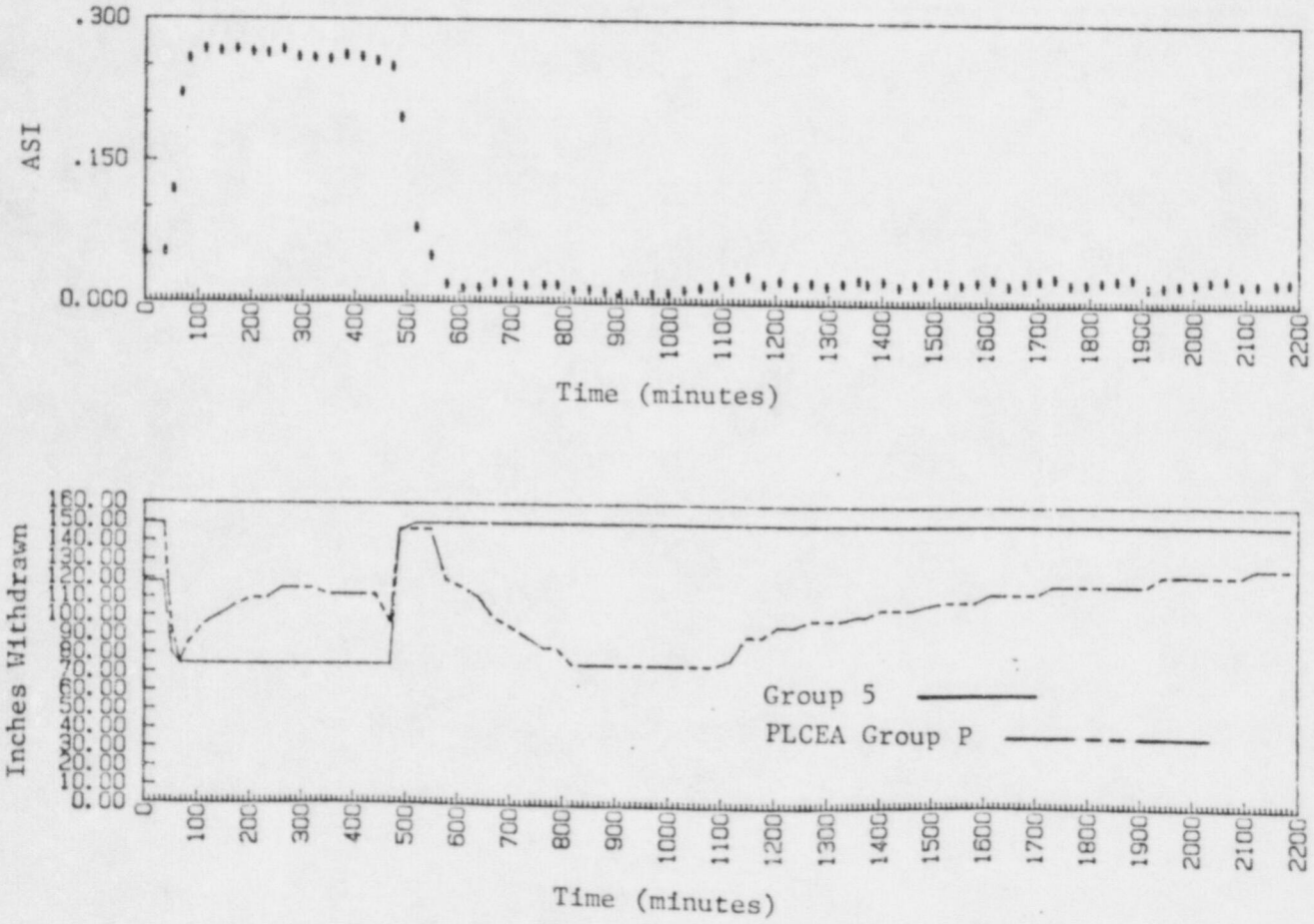
Figure 6-1 illustrates the variation of the COLSS ASI as a function of time as a result of the axial xenon oscillation induced by the insertion and subsequent withdrawal of groups 5 and P. Figure 6-1 also shows the group P motion used by the reactor operators to effectively dampen the oscillation. Also evident is a gradual withdrawal of group P which was required following the initial rod insertion to prevent the CPC ASI from exceeding the test administrative ASI limit of ± 0.4 (a CPC trip is generated when the ASI reaches ± 0.5).

CONCLUSION

The PLCEA xenon oscillation control test successfully demonstrated the technique of using the part length control rods to dampen axial xenon oscillations.

FIGURE 6-1

COLSS ASI AND CEA POSITION
VS TIME



6.12 Ejected CEA Test and Dropped CEA Test
(Sections 14.2.12.5.12 and 14.2.12.5.13)

TEST DESCRIPTION AND SUMMARY

The objective of this testing was to measure the radial power distributions resulting from the drop of a full length control rod (Control Element Assembly, or CEA), the drop of a part-length CEA, and the simulated ejection of a full length CEA. Testing was performed in accordance with PVNGS procedure 72PA-1RX47 on October 18, 1985 from an initial core power level of approximately 53% FP. The full length and part-length CEA drop tests were initiated from an unrodded core configuration whereas the full length CEA ejection test was initiated with regulating CEA group 5 positioned at 90 inches (60%) withdrawn.

The test results were consistent with the predictions provided by the NSSS vendor (Combustion Engineering) and all appropriate test acceptance criteria were satisfied.

TEST DESCRIPTION

After the appropriate initial plant operating conditions were established (i.e., 53% full power, all rods out configuration, equilibrium xenon conditions, and axial power distribution stability), the fixed incore detector signals were recorded for determination of the pre-drop radial power distribution for the full length CEA drop. CEA #37, a 12-finger full length CEA (refer to Figure 6-2 for relative core location), was then dropped by opening its circuit disconnect breaker. Fixed incore detector signals were subsequently recorded to calculate the post-drop radial power distribution. Following the collection of data, CEA #37 was returned to its full out position.

Plant conditions were allowed to stabilize before the fixed incore detector signals were recorded for determination of the pre-drop radial power distribution for the part-length CEA drop. Part-length CEA #52 (refer to Figure 6-2) was then dropped by opening its circuit disconnect breaker. Fixed incore detector signals were then recorded to calculate the post-drop radial power distribution. CEA #52 was subsequently withdrawn to its full out position.

Following the part-length CEA withdrawal, Regulating CEA Group 5 was inserted to 90 inches (60%) withdrawn in preparation for the ejected CEA test. The RCS boron concentration was adjusted during the insertion to maintain the reactor power at approximately 53%. Once plant conditions were stable, fixed incore detector signals were recorded to determine the pre-ejection radial power distribution. CEA #16 (refer to Figure 6-2) was then withdrawn individually to its full out position to simulate an ejected CEA condition. Fixed incore detector data was then recorded for determination of the post-ejected CEA radial power distribution. CEA #16 was then reinserted to 90 inches withdrawn while CEA #14 (symmetric to CEA #16) was simultaneously withdrawn to the full out position. Additional fixed incore detector data was recorded for this configuration after which CEA #14 was realigned with Regulating Group 5.

The fixed incore detector data was then processed using the Combustion Engineering incore detector analysis code CECOR to determine the measured radial power distributions for the dropped and ejected CEA cases. For each fuel assembly, the measured relative power density was compared to the predicted value using Equations 1 and 2 for the CEA drop and CEA ejection data, respectively.

Equation 1:

$$\left[\frac{(RPD\ (i)\ \text{post-drop})}{(RPD\ (i)\ \text{pre-drop})}\right]_{pred} - \left[\frac{(RPD\ (i)\ \text{post-drop})}{(RPD\ (i)\ \text{pre-drop})}\right]_{meas} = A(i)$$

Equation 2:

$$\left[\frac{(RPD\ (i)\ \text{post-eject})}{(RPD\ (i)\ \text{pre-eject})}\right]_{pred} - \left[\frac{(RPD\ (i)\ \text{post-eject})}{(RPD\ (i)\ \text{pre-eject})}\right]_{meas} \Bigg/ \left[\frac{(RPD\ (i)\ \text{post-eject})}{(RPD\ (i)\ \text{pre-eject})}\right]_{meas} * 100\% = B(i)$$

where,

i = 1 to 241 fuel assemblies
RPD (i) = the relative power density for fuel assembly "i"
A(i) = the difference between the measured and predicted ratios of the post-dropped CEA to pre-dropped CEA relative power densities for fuel assembly "i"
B(i) = the percent difference between the measured and predicted ratios of the post-ejected CEA to pre-ejected CEA relative power densities for fuel assembly "i"

The acceptance criterion for the full and part-length CEA drop tests was that each A(i) had to be within the range of ± 0.2. For the ejected CEA test, the acceptance criterion was that each B(i) had to fall within the range of ± 20%.

TEST RESULTS

The measured and predicted ratios of relative power densities for the full length CEA drop, the part-length CEA drop, and the ejected CEA are presented in Figures 6-3, 6-4, and 6-5, respectively. The comparison of measured and predicted ratios shows that all acceptance criteria were satisfied.

Only one significant event occurred during the tests. Following the drop of the high worth 12-finger CEA, the azimuthal tilt increased to 30% within 3 minutes. Since the magnitude of the tilt increase was larger than expected, CEA

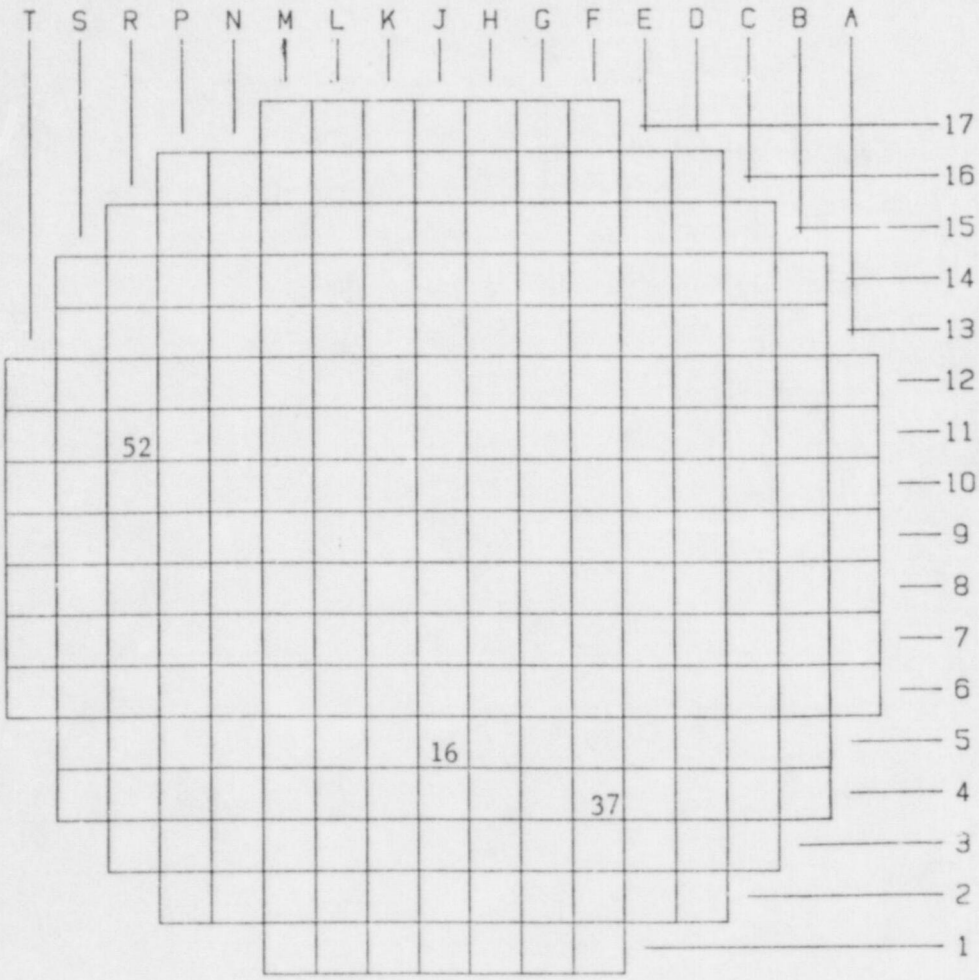
realignment actions were promptly initiated to decrease the tilt to an acceptable magnitude. Sufficient fixed incore detector data was recorded following the full length CEA drop, however, to determine the post-drop radial power distribution.

CONCLUSION

Measurement of the relative change in radial power distributions following dropped and ejected CEA events provided results which satisfactorily confirmed agreed with the test predictions, thereby confirming the design analyses for the System 80 NSSS.

FIGURE 6-2

RELATIVE CORE LOCATIONS
OF DROPPED AND EJECTED CEAS



KEY:

Dropped CEA--CEA 37 (F-4)
Dropped PLCEA--CEA 52 (R-11)
Ejected CEA--CEA 16 (J-5)

FIGURE 6-3

DROPPED FULL LENGTH CEA TEST RESULTS

	A	B	C	D	E	F	G	H	J	K	L	M	N	P	R	S	T
1						.548 .815 .067	.811 .871 .060	.697 .754 .057	.798 .835 .049	.884 .908 .042	.920 .953 .033	.958 .974 .018					
2				.490 .585 .075	.485 .540 .075	.491 .588 .077	.570 .629 .059	.678 .725 .047	.782 .817 .035	.888 .898 .028	.931 .948 .017	.978 .982 .006	1.015 1.013 .002	1.039 1.031 .008			
3			.578 .637 .061	.510 .584 .074	.418 .497 .079	.354 .438 .084	.494 .551 .057	.656 .708 .052	.783 .814 .031	.879 .898 .017	.945 .960 .015	.993 .995 .002	1.028 1.031 .003	1.051 1.047 .004	1.069 1.070 .001		
4		.683 .729 .046	.634 .688 .054	.552 .619 .067	.380 .461 .081	.281 .306 .045	.435 .474 .039	.667 .706 .039	.802 .832 .030	.897 .908 .011	.952 .956 .004	1.009 1.004 .005	1.040 1.036 .004	1.064 1.053 .011	1.080 1.069 .011	1.092 1.077 .015	
5		.745 .782 .037	.704 .748 .044	.638 .686 .048	.535 .597 .062	.449 .521 .072	.580 .624 .044	.729 .766 .037	.842 .860 .018	.923 .928 .005	.982 .988 .006	1.028 1.020 .008	1.058 1.054 .002	1.078 1.066 .012	1.092 1.079 .013	1.101 1.088 .013	
6	.857 .879 .021	.822 .850 .028	.784 .813 .029	.742 .778 .036	.699 .736 .037	.698 .729 .041	.734 .763 .029	.818 .836 .018	.894 .903 .009	.957 .958 .001	1.006 1.002 .004	1.045 1.033 .012	1.072 1.061 .011	1.082 1.079 .013	1.104 1.088 .006	1.113 1.100 .013	1.122 1.108 .014
7	.899 .908 .009	.882 .892 .010	.859 .878 .019	.837 .854 .017	.822 .837 .015	.824 .848 .024	.853 .867 .014	.901 .909 .008	.951 .959 .008	.995 .991 .004	1.034 1.032 .002	1.066 1.054 .012	1.089 1.074 .015	1.105 1.094 .011	1.117 1.103 .014	1.123 1.111 .012	1.127 1.116 .011
8	.945 .948 .001	.938 .938 .002	.924 .927 .003	.916 .924 .008	.912 .914 .002	.930 .923 .003	.940 .949 .009	.970 .969 .001	1.001 .997 .004	1.033 1.023 .010	1.052 1.051 .011	1.087 1.075 .012	1.105 1.090 .015	1.118 1.100 .018	1.128 1.110 .018	1.132 1.118 .014	1.134 1.119 .015
9	.988 .991 .003	.985 .991 .006	.980 .976 .004	.979 .972 .007	.980 .971 .009	.986 .982 .004	1.001 .997 .004	1.021 1.019 .002	1.044 1.037 .007	1.068 1.053 .013	1.088 1.070 .018	1.106 1.089 .017	1.120 1.105 .015	1.130 1.113 .017	1.137 1.119 .018	1.141 1.125 .016	1.142 1.134 .008
10	1.025 1.031 .006	1.028 1.022 .006	1.027 1.016 .011	1.029 1.016 .013	1.030 1.018 .012	1.036 1.032 .004	1.048 1.034 .012	1.060 1.047 .013	1.077 1.061 .016	1.093 1.079 .014	1.108 1.089 .019	1.122 1.103 .019	1.133 1.114 .019	1.142 1.125 .017	1.147 1.129 .018	1.149 1.131 .018	1.150 1.132 .018
11	1.059 1.043 .016	1.057 1.046 .011	1.052 1.050 .012	1.064 1.057 .007	1.069 1.053 .016	1.073 1.058 .015	1.081 1.063 .018	1.091 1.072 .019	1.103 1.083 .020	1.115 1.094 .021	1.128 1.104 .022	1.137 1.119 .018	1.148 1.124 .022	1.151 1.131 .020	1.158 1.138 .018	1.155 1.137 .018	1.155 1.132 .023
12	1.071 1.058 .015	1.079 1.068 .011	1.086 1.078 .008	1.091 1.077 .014	1.095 1.077 .018	1.102 1.084 .018	1.108 1.088 .020	1.116 1.094 .022	1.124 1.107 .017	1.133 1.111 .022	1.142 1.117 .025	1.149 1.126 .023	1.155 1.132 .023	1.159 1.138 .021	1.161 1.140 .021	1.160 1.143 .017	1.159 1.140 .019
13		1.098 1.089 .009	1.104 1.090 .014	1.109 1.092 .017	1.115 1.094 .021	1.120 1.099 .021	1.128 1.107 .019	1.132 1.109 .023	1.139 1.116 .023	1.146 1.125 .021	1.152 1.128 .024	1.157 1.135 .022	1.162 1.139 .023	1.163 1.141 .022	1.165 1.144 .021	1.165 1.143 .022	
14		1.112 1.094 .018	1.116 1.099 .017	1.122 1.108 .014	1.128 1.107 .021	1.134 1.111 .023	1.139 1.116 .023	1.145 1.119 .026	1.149 1.124 .025	1.155 1.130 .025	1.160 1.136 .024	1.163 1.140 .023	1.165 1.145 .020	1.166 1.145 .021	1.168 1.149 .017	1.168 1.147 .021	
15			1.125 1.108 .019	1.130 1.113 .017	1.136 1.121 .015	1.142 1.120 .022	1.148 1.126 .022	1.151 1.129 .022	1.155 1.131 .024	1.160 1.139 .021	1.165 1.141 .024	1.166 1.146 .020	1.168 1.145 .023	1.167 1.147 .020	1.169 1.149 .020		
16				1.137 1.116 .021	1.141 1.120 .021	1.146 1.126 .020	1.151 1.130 .021	1.155 1.137 .018	1.158 1.136 .022	1.162 1.140 .022	1.165 1.145 .020	1.167 1.146 .021	1.170 1.145 .025	1.171 1.148 .023			
17						1.152 1.132 .020	1.154 1.132 .022	1.158 1.138 .020	1.159 1.137 .022	1.163 1.140 .023	1.165 1.143 .022	1.169 1.145 .024					

KEY:

X
Y
Z

Where X = Ratio of predicted relative power densities (post-drop to pre-drop)

Y = Ratio of measured (CECOR) relative power densities (post-drop to pre-drop)

Z = X - Y

FIGURE 6-4

DROPPED PART-LENGTH CEA TEST RESULTS

	A	B	C	D	E	F	G	H	J	K	L	M	N	P	R	S	T	
						1.030 1.020 .010	1.029 1.019 .010	1.028 1.019 .009	1.026 1.019 .007	1.024 1.017 .007	1.023 1.017 .006	1.021 1.019 .002						1
				1.030 1.022 .008	1.029 1.022 .007	1.028 1.021 .007	1.027 1.018 .009	1.026 1.019 .007	1.024 1.019 .005	1.023 1.019 .004	1.021 1.016 .005	1.018 1.013 .005	1.016 1.010 .006	1.014 1.008 .006				2
			1.031 1.028 .005	1.029 1.024 .005	1.028 1.022 .006	1.027 1.024 .003	1.025 1.022 .003	1.023 1.021 .002	1.022 1.018 .004	1.020 1.016 .004	1.017 1.015 .002	1.015 1.011 .004	1.013 1.009 .004	1.011 1.005 .006	1.009 1.003 .006			3
		1.031 1.028 .005	1.030 1.028 .004	1.028 1.024 .004	1.027 1.024 .003	1.025 1.023 .002	1.022 1.022 .000	1.021 1.020 .001	1.018 1.018 .000	1.016 1.015 .001	1.014 1.013 .001	1.012 1.009 .003	1.010 1.008 .004	1.007 1.001 .006	1.005 .997 .008	1.003 .994 .009		4
		1.030 1.028 .004	1.029 1.025 .004	1.027 1.028 .001	1.026 1.025 .001	1.023 1.023 .000	1.021 1.021 .000	1.018 1.020 .002	1.015 1.017 .002	1.012 1.014 .002	1.009 1.011 .002	1.007 1.007 .000	1.004 1.004 .000	1.001 .997 .004	.999 .994 .005	.997 .989 .008		5
1.031 1.028 .005	1.030 1.028 .004	1.029 1.028 .003	1.028 1.027 .001	1.024 1.024 .000	1.022 1.021 .001	1.018 1.020 .002	1.015 1.018 .003	1.012 1.015 .003	1.009 1.012 .004	1.004 1.008 .004	1.000 1.002 .002	.997 .997 .000	.994 .991 .003	.991 .987 .004	.988 .981 .007	.983 .968 .015		6
1.031 1.025 .006	1.030 1.025 .005	1.028 1.025 .003	1.025 1.025 .000	1.029 1.023 .006	1.030 1.021 .001	1.016 1.019 .003	1.013 1.017 .004	1.008 1.014 .006	1.004 1.009 .005	.999 1.005 .006	.993 .998 .003	.988 .988 .000	.983 .985 .002	.979 .977 .002	.977 .972 .005	.975 .978 .001		7
1.031 1.025 .006	1.030 1.025 .005	1.027 1.025 .002	1.024 1.025 .001	1.021 1.023 .002	1.019 1.020 .002	1.015 1.017 .002	1.010 1.014 .004	1.005 1.010 .005	1.000 1.004 .004	.993 .998 .005	.985 .991 .006	.977 .979 .002	.969 .968 .001	.963 .960 .003	.959 .957 .002	.959 .957 .002		8
1.031 1.028 .005	1.029 1.025 .004	1.027 1.024 .003	1.024 1.024 .000	1.021 1.022 .001	1.017 1.021 .004	1.013 1.018 .005	1.008 1.015 .007	1.003 1.009 .006	.998 1.002 .006	.988 .991 .003	.977 .980 .003	.963 .969 .003	.948 .950 .002	.935 .941 .006	.933 .940 .007	.937 .934 .003		9
1.031 1.028 .005	1.029 1.028 .003	1.028 1.025 .001	1.024 1.024 .000	1.020 1.022 .002	1.017 1.021 .004	1.013 1.017 .004	1.007 1.012 .005	1.001 1.007 .006	.994 1.001 .007	.983 .987 .004	.969 .971 .002	.948 .951 .003	.917 .924 .007	.886 .914 .029	.898 .930 .032	.912 .933 .021		10
1.030 1.028 .004	1.029 1.024 .005	1.028 1.027 .041	1.024 1.023 .001	1.021 1.022 .001	1.017 1.019 .002	1.012 1.018 .004	1.007 1.011 .004	1.000 1.005 .005	.993 .997 .004	.982 .984 .002	.965 .988 .003	.941 .944 .003	.993 .913 .020	.787 .795 .008	.888 .905 .037	.894 .917 .023		11
1.030 1.025 .005	1.029 1.024 .005	1.027 1.023 .004	1.024 1.022 .002	1.021 1.020 .001	1.018 1.019 .001	1.013 1.018 .003	1.007 1.011 .004	1.002 1.005 .003	.994 .997 .003	.982 .984 .002	.987 .989 .002	.948 .948 .002	.914 .923 .009	.881 .912 .031	.888 .929 .043	.891 .933 .042		12
	1.027 1.025 .001	1.028 1.023 .003	1.024 1.021 .003	1.021 1.020 .001	1.019 1.018 .000	1.014 1.017 .003	1.009 1.010 .001	1.004 1.005 .001	.998 .999 .003	.988 .987 .001	.973 .975 .002	.957 .958 .001	.938 .942 .004	.923 .933 .010	.914 .934 .020			13
	1.028 1.024 .004	1.028 1.022 .004	1.025 1.020 .005	1.023 1.019 .004	1.019 1.018 .001	1.015 1.015 .000	1.011 1.010 .001	1.005 1.008 .001	.998 .998 .000	.990 .989 .001	.981 .978 .003	.959 .957 .002	.957 .954 .003	.945 .948 .001	.934 .941 .007			14
		1.028 1.021 .005	1.025 1.020 .005	1.023 1.019 .004	1.020 1.018 .004	1.017 1.014 .003	1.014 1.010 .004	1.008 1.005 .003	1.003 1.001 .002	.995 .992 .003	.987 .984 .003	.978 .973 .005	.968 .963 .005	.959 .954 .005				15
			1.025 1.019 .006	1.024 1.018 .006	1.021 1.016 .005	1.019 1.014 .005	1.018 1.010 .008	1.011 1.008 .005	1.008 1.002 .004	1.000 .998 .002	.993 .987 .006	.984 .979 .005	.977 .971 .006					16
					1.022 1.017 .005	1.020 1.014 .006	1.018 1.010 .008	1.013 1.007 .006	1.009 1.002 .007	1.003 .999 .004	.999 .992 .007							17

KEY:

X
Y
Z

Where X = Ratio of predicted relative power densities (post-drop to pre-drop)

Y = Ratio of measured (CECOR) relative power densities (post-drop to pre-drop)

Z = X - Y

FIGURE 6-5

EJECTED CEA TEST RESULTS

A	B	C	D	E	F	G	H	J	K	L	M	N	P	R	S	T	
					1.089 1.067 1.528	1.089 1.056 1.231	1.077 1.087 .937	1.080 1.072 .748	1.077 1.084 1.272	1.069 1.053 1.519	1.063 1.039 2.904						1
			1.036 1.013 2.270	1.045 1.027 1.759	1.059 1.041 1.729	1.062 1.050 1.143	1.060 1.074 1.117	1.091 1.085 .953	1.088 1.055 1.572	1.062 1.058 .598	1.059 1.043 1.534	1.045 1.029 1.955	1.036 1.021 1.469				2
		1.024 1.008 1.299	1.030 1.014 1.578	1.040 1.028 1.365	1.057 1.038 1.820	1.071 1.071 0.000	1.113 1.094 1.737	1.130 1.129 .089	1.113 1.111 .180	1.071 1.088 .281	1.057 1.047 .955	1.040 1.022 1.781	1.030 1.015 1.478	1.024 1.003 2.094			3
	1.011 .994 1.710	1.018 .997 1.908	1.028 1.010 1.787	1.024 1.021 1.273	1.055 1.050 .478	1.097 1.101 -.382	1.174 1.175 -.085	1.262 1.238 2.104	1.174 1.184 -.845	1.098 1.097 -.091	1.054 1.058 -.189	1.034 1.029 .488	1.023 1.014 .888	1.018 1.005 1.095	1.011 1.000 1.100		4
	1.005 .990 1.515	1.007 .995 1.238	1.014 1.008 .735	1.028 1.021 .490	1.050 1.044 .375	1.098 1.112 -1.183	1.242 1.223 1.054	1.715 1.775 -3.980	1.242 1.247 -.401	1.099 1.094 .497	1.050 1.054 -.380	1.028 1.022 .391	1.014 1.010 .398	1.007 .999 .801	1.005 .995 1.005		5
.995 .979 1.034	.997 .981 1.831	.998 .989 .910	1.002 .997 .502	1.014 1.018 -.197	1.039 1.043 -.959	1.075 1.095 -1.828	1.151 1.174 -1.075	1.231 1.244 -1.845	1.231 1.179 -2.375	1.075 1.091 -1.487	1.039 1.048 -1.243	1.014 1.020 -.588	1.002 1.002 0.000	.998 .987 1.114	.997 .986 1.118	.995 .985 1.015	6
.989 .975 1.438	.999 .979 1.021	.999 .993 .510	.991 .992 -.101	.997 1.008 -.905	1.013 1.018 -.431	1.036 1.034 -1.708	1.054 1.048 -2.208	1.078 1.092 -1.292	1.064 1.087 -2.118	1.036 1.045 -.881	1.012 1.028 -1.365	.997 1.008 -1.091	.991 .991 0.000	.988 .984 .407	.989 .981 .815	.989 .977 1.228	7
.983 .971 1.238	.982 .973 .925	.980 .978 .324	.979 .981 -.306	.982 .984 -1.207	.989 1.004 -1.434	1.002 1.013 -1.086	1.012 1.037 -2.411	1.018 1.045 -2.584	1.012 1.041 -2.798	1.002 1.022 -1.57	.989 1.002 -1.297	.982 .995 -1.307	.979 .985 -.711	.980 .979 .102	.982 .978 .815	.983 .988 1.530	8
.978 .984 1.452	.978 .988 1.035	.971 .970 .103	.989 .979 -.717	.988 .984 -1.829	.970 .990 -2.022	.977 .997 -2.006	.989 1.003 -1.894	.984 1.008 -2.187	.989 1.009 -2.577	.977 1.002 -2.428	.970 .991 -2.118	.988 .975 -.929	.989 .973 -.411	.971 .972 -.103	.975 .970 .515	.978 .988 1.033	9
.973 .957 1.872	.970 .982 .832	.988 .984 .207	.992 .995 -.725	.998 .971 -1.545	.957 .978 -1.847	.982 .982 -2.007	.984 .988 -2.429	.985 .990 -2.525	.984 .987 -2.230	.982 .985 -2.205	.957 .977 -2.047	.956 .989 -1.342	.959 .983 -.415	.965 .985 0.000	.970 .984 .522	.979 .988 .517	10
.971 .958 1.357	.988 .959 .938	.982 .950 .313	.958 .959 -.104	.952 .983 -1.038	.952 .988 -1.853	.959 .972 -1.955	.952 .978 -2.439	.951 .978 -2.581	.952 .978 -2.439	.959 .974 -2.156	.952 .988 -1.449	.953 .984 -1.141	.958 .980 -.208	.981 .958 .313	.987 .980 .729	.971 .955 1.875	11
.989 .958 1.380	.988 .958 1.048	.982 .957 .522	.957 .958 -.104	.954 .980 -.625	.950 .981 -1.145	.948 .984 -1.088	.945 .984 -1.971	.943 .981 -1.973	.945 .988 -2.174	.948 .985 -1.792	.950 .982 -1.247	.954 .959 -.521	.957 .958 .105	.981 .958 .418	.986 .957 .835	.989 .970 -.103	12
		.985 .953 1.259	.981 .955 .628	.958 .958 .209	.954 .958 -.209	.950 .957 -.731	.948 .959 -1.253	.942 .959 -1.773	.941 .957 -1.672	.942 .960 -1.875	.948 .959 -1.358	.950 .959 -.836	.954 .958 -.209	.958 .958 .209	.981 .956 .523	.985 .957 .836	13
		.984 .953 1.154	.984 .954 1.048	.980 .954 .829	.957 .954 .314	.952 .954 -.210	.948 .954 -.629	.945 .955 -1.047	.943 .954 -1.153	.945 .954 -.943	.948 .954 -.629	.952 .955 -.314	.956 .954 .210	.960 .954 .629	.983 .953 1.049	.984 .955 .942	14
			.984 .954 1.048	.982 .953 .944	.958 .953 .525	.955 .951 .421	.951 .950 .105	.950 .950 0.000	.949 .950 -.105	.950 .949 .105	.951 .952 -.105	.958 .951 .421	.957 .952 .525	.962 .951 1.157	.984 .952 1.261		15
				.981 .952 .945	.980 .951 .946	.959 .949 1.054	.958 .948 .844	.958 .948 1.057	.958 .950 .528	.958 .950 .632	.958 .951 .528	.959 .951 .841	.960 .950 1.053	.981 .948 1.371			16
					.960 .947 1.373	.960 .948 1.298	.958 .947 1.182	.958 .948 1.055	.958 .950 .842	.960 .950 1.053	.960 .949 1.159						17

KEY:

X
Y
Z

Where X = Ratio of predicted relative power densities (post-ejection to pre-ejection)

Y = Ratio of measured (CECOR) relative power densities (post-ejection to pre-ejection)

Z = ((X - Y)/Y) x 100%

6.13 Steady State Core Performance
(Section 14.2.12.5.14)

TEST OBJECTIVE AND SUMMARY

The reactor core power distributions and core peaking factors were measured five times during power ascension testing at various power levels and control element assembly (CEA) configurations. These measurements were compared to predictions to confirm assumptions in the safety analysis and to verify expected core behavior. Measurements were performed once at the 20%, 50%, and 80% full power (FP) levels and twice at the 100% FP level. The conditions of the power distribution measurements and the dates of performance are listed in Table 6-13.

The test acceptance criteria was satisfied if the root mean square (RMS) differences between measured and predicted power distributions were less than or equal to 5% and if the measured peaking factors were within ±10% of their predicted values. The acceptance criteria was satisfied for all measurements.

TEST DESCRIPTION

Core power distributions and peaking factors were measured at steady state equilibrium conditions using fixed incore detector signals. The detector signals were recorded on magnetic tape using a plant computer snapshot function and then transferred to a main frame computer for further analysis. The incore detector analysis code CECOR was used to synthesize radial and axial power distributions from the fixed incore detector signals and to calculate core peaking factors from the synthesized power distributions. The measured power distributions derived from the incore detector signals were compared to predicted distributions by calculating the root mean square difference between nodes. Core peaking factors were compared to predicted values on an individual basis.

TEST RESULTS

The measured and predicted core peaking factors and the RMS differences between measured and predicted power distributions are presented in Table 6-14. Additionally, Figures 6-6 and 6-7 show the axial and radial power distribution results from the rodded 100% power test. The test acceptance criteria were satisfactorily met for all measurements.

CONCLUSIONS

Since the acceptance criteria for this test were satisfactorily met, it can be concluded that the safety analysis assumptions concerning core peaking factors are valid and that the core is behaving as expected.

TABLE 6-13

STEADY STATE CORE PERFORMANCE
TEST CONDITIONS

	20% FP	50% FP	80% FP	100% FP	100% FP
Performance Dates	6-19-85	7- 8-85	10- 1-85	12-26-85	12-31-85
Actual Reactor Power	19.4%	49.9%	82.9%	99.4%	99.4%
RCS Tavg (°F)	569.2	578.7	588.4	592.1	592.0
Primary Pressure	2227 psia	2225 psia	2230 psia	2248 psia	2251 psia
Boron Concentration	829 ppm	719 ppm	661 ppm	628 ppm	632 ppm
CEA Position	Unrodded	Unrodded	* Group 5 @ 120 in wd	* Group 5 @ 120 in wd	Unrodded
Core Average Burnup	53 MWD/T (1 EFPD)	199 MWD/T (5 EFPD)	802 MWD/T (21 EFPD)	1685 MWD/T (44 EFPD)	1915 MWD/T (50 EFPD)
Axial Shape Index (ASI)**	-0.012	+0.021	+0.076	+0.103	-0.071

* -- Position given in "inches withdrawn"; 120 in wd = 80% withdrawn

$$** \text{ -- } ASI = \frac{(\text{power in lower core half} - \text{power in upper core half})}{\text{total core power}}$$

TABLE 6-14
(Part 1 of 2)

STEADY STATE CORE PERFORMANCE
TEST RESULTS

20% FP TEST

Peaking Factor	Measured Value	Predicted Value	% Diff	Accept Criteria
F_{xy} Core Planar Radial Peaking Factor	1.50	1.46	+2.7	<10%
F_r Core Intgrtd Radial Peaking Factor	1.45	1.43	+1.4	<10%
F_z Core Axial Peaking Factor	1.26	1.27	-0.8	<10%
F_q Core 3-D Peaking Factor	1.88	1.83	+2.7	<10%

50% FP TEST

Peaking Factor	Measured Value	Predicted Value	% Diff	Accept Criteria
F_{xy} Core Planar Radial Peaking Factor	1.43	1.38	+3.6	<10%
F_r Core Intgrtd Radial Peaking Factor	1.41	1.35	+4.4	<10%
F_z Core Axial Peaking Factor	1.25	1.25	0.0	<10%
F_q Core 3-D Peaking Factor	1.80	1.68	+7.1	<10%

80% FP TEST

Peaking Factor	Measured Value	Predicted Value	% Diff	Accept Criteria
F_{xy} Core Planar Radial Peaking Factor	1.41	1.38	+2.2	<10%
F_r Core Intgrtd Radial Peaking Factor	1.40	1.35	+3.7	<10%
F_z Core Axial Peaking Factor	1.30	1.30	0.0	<10%
F_q Core 3-D Peaking Factor	1.83	1.74	+5.2	<10%

TABLE 6-14
(Part 2 of 2)

STEADY STATE CORE PERFORMANCE
TEST RESULTS

100% FP TEST (1)

Peaking Factor	Measured Value	Predicted Value	% Diff	Accept Criteria
F_{xy} Core Planar Radial Peaking Factor	1.41	1.38	+2.2	<10%
F_r Core Intgrtd Radial Peaking Factor	1.40	1.35	+3.7	<10%
F_z Core Axial Peaking Factor	1.32	1.29	+2.3	<10%
F_q Core 3-D Peaking Factor	1.86	1.74	+6.9	<10%

100% FP TEST (2)

Peaking Factor	Measured Value	Predicted Value	% Diff	Accept Criteria
F_{xy} Core Planar Radial Peaking Factor	1.41	1.37	+2.9	<10%
F_r Core Intgrtd Radial Peaking Factor	1.39	1.34	+3.7	<10%
F_z Core Axial Peaking Factor	1.28	1.26	+1.6	<10%
F_q Core 3-D Peaking Factor	1.81	1.71	+5.8	<10%

RMS DIFFERENCES

	20%	50%	80%	100%(1)	100%(2)	Acceptance Criteria
Radial Dist.	3.39%	2.84%	2.68%	2.56%	2.57%	<5%
Axial Dist.	3.71%	2.74%	2.03%	3.06%	3.88%	<5%

(1) -- 100% power with CEA group 5 at 120 inches withdrawn
(2) -- 100% power unrodded

FIGURE 6-6

RELATIVE AXIAL POWER DISTRIBUTION
100% FP TEST (RODDED)

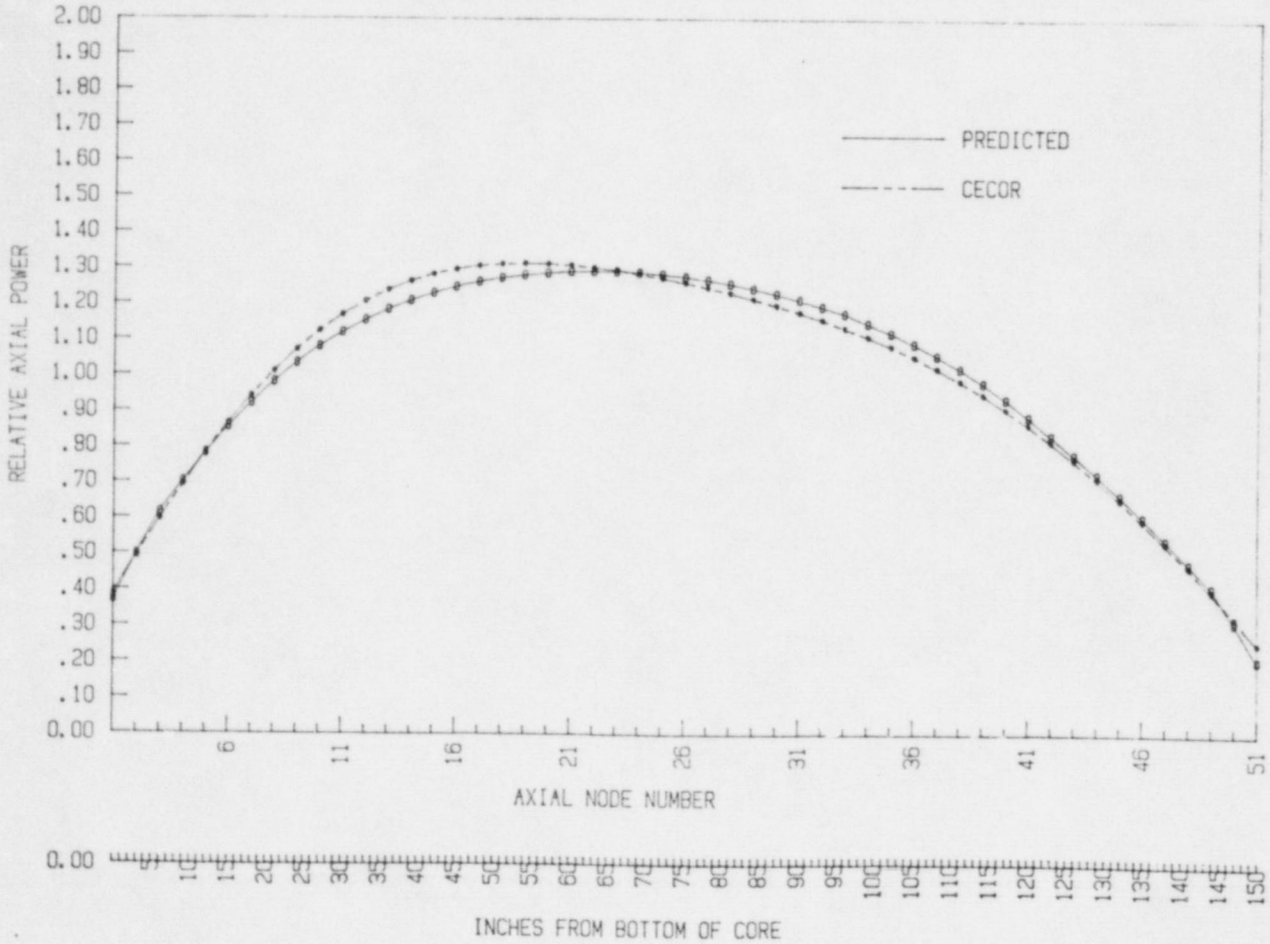


FIGURE 6-7

RELATIVE RADIAL POWER DISTRIBUTION
100% FP TEST (RODDED)

A	B	C	D	E	F	G	H	J	K	L	M	N	P	R	S	T	
					.61	.80	.84	.84	.84	.80	.61						
					.80	.79	.83	.84	.84	.82	.63						1
					2.06	1.75	.95	.13	-.34	-1.88	-2.48						
			.62	.83	1.01	.96	.97	.94	.97	.96	1.01	.83	.62				
			.83	.89	1.00	.99	.95	.93	.95	.99	1.06	.88	.64				2
			-1.40	.11	.11	3.05	1.57	1.43	2.19	-2.35	-4.92	-3.96	-2.94				
		.71	1.04	.97	1.10	1.00	1.05	1.01	1.05	1.00	1.10	.97	1.07	.71			
		.70	1.05	.95	1.09	.98	1.04	.99	1.06	1.08	1.15	1.00	1.10	.73			3
		1.32	-1.55	2.29	.81	3.73	1.54	1.82	-.57	-2.75	-4.50	-2.73	-5.41	-0.59			
	.22	1.04	1.07	1.15	1.06	1.14	1.09	1.06	1.03	1.14	1.08	1.15	1.07	1.04	.62		
	.61	1.04	1.06	1.19	1.02	1.12	.98	1.06	1.01	1.18	1.09	1.20	1.11	1.03	.80		4
	1.87	-.13	.95	2.05	4.53	1.88	4.17	2.13	1.29	-3.81	-1.94	-4.84	-3.34	-4.45	-1.56		
	.89	.97	1.15	1.08	1.17	1.07	1.10	1.00	1.13	1.07	1.17	1.09	1.15	.97	.89		
	.89	.97	1.16	1.04	1.03	1.02	1.12	.99	1.16	1.09	1.22	1.11	1.16	.98	.84		5
	-.13	.49	-1.20	3.94	1.85	5.02	1.13	1.48	-2.10	-2.58	-4.04	-2.86	-3.28	-.44	-1.79		
.81	1.01	1.10	1.06	1.17	1.07	1.19	1.08	1.14	1.06	1.19	1.07	1.17	1.06	1.10	1.01	.81	
.81	1.01	1.14	1.09	1.17	1.02	1.11	1.02	1.16	1.04	1.14	1.08	1.19	1.05	1.10	1.03	.82	6
.07	-.88	-3.35	-2.45	-.23	1.02	2.41	3.21	-1.25	1.42	-.91	1.14	-1.91	1.46	.54	-2.23	-.90	
.80	.96	1.00	1.14	1.07	1.13	1.11	1.14	1.07	1.14	1.11	1.13	1.07	1.14	1.00	.98	.80	
.82	.99	1.04	1.18	1.05	1.11	1.07	1.13	1.09	1.14	1.08	1.14	1.05	1.12	.98	.98	.82	7
-1.62	-2.35	-3.79	-3.73	1.91	2.59	3.69	.83	-1.77	-.34	4.48	-.11	1.91	1.43	2.24	-1.25	-2.10	
.84	.97	1.05	1.03	1.13	1.08	1.14	1.08	1.18	1.08	1.14	1.08	1.13	1.03	1.05	.97	.84	
.85	.97	1.08	1.04	1.15	1.02	1.09	1.03	1.19	1.06	1.14	1.08	1.08	1.01	1.05	.98	.86	8
-1.40	-.65	-2.59	-1.04	-1.73	3.51	4.03	4.48	-1.15	1.91	.01	-.48	7.16	1.78	.10	-.95	-2.54	
.84	.94	1.01	1.09	1.00	1.14	1.07	1.16	1.09	1.18	1.07	1.14	1.00	1.08	1.01	.94	.84	
.85	.91	1.01	1.09	1.00	1.16	1.03	1.14	1.08	1.20	1.08	1.17	1.02	1.10	.99	.95	.87	9
-.59	3.44	.37	-1.51	-.04	-1.17	3.84	3.36	1.22	-1.81	1.29	-2.01	-1.92	-1.90	1.89	-1.13	-3.20	
.84	.97	1.05	1.03	1.13	1.08	1.14	1.08	1.18	1.08	1.14	1.08	1.13	1.03	1.05	.97	.84	
.87	.97	1.08	1.01	1.16	1.08	1.14	1.04	1.19	1.09	1.15	1.04	1.18	1.05	1.08	.99	.87	10
-2.78	-.03	-.28	1.59	-2.44	-2.60	-.52	3.38	-.98	-1.35	-.80	2.01	-2.60	-2.66	-2.50	-1.96	-2.78	
.80	.96	1.00	1.14	1.07	1.13	1.11	1.14	1.07	1.14	1.11	1.13	1.07	1.14	1.00	.98	.80	
.82	.98	.98	1.13	1.05	1.18	1.10	1.13	1.04	1.15	1.06	1.10	1.05	1.19	1.02	.98	.82	11
-1.88	-1.25	1.62	.98	1.23	-2.35	.41	.72	3.34	.54	2.93	3.52	1.62	-4.13	-2.18	-2.15	-1.98	
.81	1.01	1.10	1.06	1.17	1.07	1.13	1.06	1.14	1.06	1.13	1.07	1.17	1.06	1.10	1.01	.81	
.62	1.03	1.11	1.05	1.19	1.08	1.16	1.03	1.11	1.01	1.10	1.02	1.18	1.11	1.14	1.01	.61	12
-1.06	-2.70	-1.18	1.58	-1.86	-1.06	-1.78	2.90	3.97	4.74	2.78	5.32	-.85	-9.98	-3.09	-.29	.23	
	.83	.97	1.15	1.08	1.17	1.07	1.13	1.00	1.13	1.07	1.17	1.08	1.15	.97	.83		
	.84	.97	1.16	1.07	1.20	1.09	1.15	.98	1.11	1.01	1.12	1.04	1.16	.88	.82		13
	-1.56	.38	-.94	.89	-2.97	-1.75	-1.18	2.93	1.77	5.56	4.32	4.04	-1.37	1.01	.47		
	.82	1.04	1.07	1.15	1.06	1.14	1.03	1.08	1.03	1.14	1.06	1.15	1.07	1.04	.82		
	.82	1.06	1.05	1.18	1.08	1.17	1.01	1.07	.98	1.11	1.01	1.12	1.05	1.03	.80		14
	-.13	-2.29	1.34	-2.62	-1.01	-3.07	1.49	1.08	4.59	2.52	5.06	2.80	1.34	1.04	2.98		
		.71	1.04	.97	1.10	1.00	1.05	1.01	1.05	1.00	1.10	.97	1.04	.71			
		.72	1.08	1.01	1.15	1.02	1.06	.99	1.02	.95	1.09	.95	1.06	.70			15
		-.25	-3.92	-3.08	-4.25	-1.49	-.19	2.30	3.13	5.03	1.42	2.49	-1.78	1.90			
			.62	.83	1.01	.96	.97	.94	.97	.96	1.01	.83	.62				
			.84	.86	1.08	.98	.95	.92	.94	.92	1.00	.83	.64				16
			-2.39	-4.35	-4.73	-2.05	1.97	1.98	3.58	5.11	.91	-.01	-3.55				
					.81	.80	.84	.84	.84	.80	.81						
					.83	.81	.84	.84	.89	.78	.80						17
					-2.17	-1.50	.01	.73	1.83	3.19	3.25						

KEY:

X
Y
Z

Where X = Predicted relative power density
Y = Measured (CECOR) relative power density
Z = ((X - Y)/X) x 100%

6.14 Intercomparison of PPS, Core Protection Calculator (CPC), and PMS Inputs (Section 14.2.12.5.15)

TEST OBJECTIVES AND SUMMARY

The principal objective of the "Intercomparison" test was to verify that all Main Control Room indications of selected plant parameters monitored by the Plant Protection System (PPS), Core Protection Calculators (CPC), Plant Monitoring System (PMS) and the Main Control Board (MCB) process instruments were correct and consistent with acceptance criteria that were based on vendor and design accuracies.

Testing was accomplished using PVNGS procedures 72PA-1SB02 (20% power) on June 20, 1985, 72PA-1SB06 (50% power) on July 7, 1985, 72PA-1SB10 (80% power) on September 27, 1985 and 72PA-1SB15 (100% power) on December 23, 1985. All instruments were either within their required acceptance band or sufficient data was sent back to the vendor to justify a broadening of the acceptance criteria.

TEST DESCRIPTION

PPS, CPC, PMS and MCB data were collected (as simultaneously as possible from all available indications) for the following process intrumentation at the four major power ascension test plateaus:

(1) Reactor Coolant System (RCS) hot and cold leg temperatures
(2) Reactor Coolant Pump (RCP) speeds and differential pressures
(3) Pressurizer pressure and level
(4) Steam Generator pressure and level
(5) Reactor Core differential pressure
(6) Steam Generator differential pressure

At all four power plateaus the data from each instrument was cross-compared to verify that the various indications of that particular parameter were consistent and accurate to within the specified acceptable tolerance bands.

TEST RESULTS

Most of the instruments at each of the four major power plateaus were found to be within the specified tolerance bands. Test Exception Reports were written to document those instruments that did not meet their acceptance criteria. These instruments were then recalibrated and either successfully retested at that particular power level or were accepted on the condition that they met their required acceptance criteria at the next higher power plateau.

At the 100% power plateau, all instruments either cross-correlated well (including those that were not within their required tolerance bands at the lower power plateaus) or Test Exception Reports were written and the data sent back to the vendor for further evaluation. The vendor performed some additional analyses to justify a broader acceptance band. Thus, upon completion of this test at the 100% power plateau, all process instrumentation were within their required acceptance band or were evaluated by the vendor and found to be acceptable.

CONCLUSIONS

The accuracy and consistency of Control Room indications of selected plant parameters monitored by the PPS, CPC, PMS, MCB, and process instruments were acceptable for power operation.

6.15 Verification of CPC Power Distribution Related Constants Test (Section 14.2.12.5.16)

The testing performed in accordance with this section of CESSAR verified the agreement of the various constants used by the Core Protection Calculators (CPCs) in the power distribution calculation with those values determined by measurement. These constants were the rod (CEA) shadowing factors, the planar radial peaking factors, the temperature shadowing factors, the shape annealing matrix, and the boundary point power correlation constants. The verification of these constants was performed in three major tests, as described in the following sections.

6.15.1 Verification of CEA Shadowing Factors and Radial Peaking Factors

TEST OBJECTIVES AND SUMMARY

This test verified that the CEA shadowing factors and the radial peaking factors used in the CPC power distribution calculation are valid by comparing the measured values with the predicted values which are part of the CPC data base. If the measured and predicted values did not agree within the acceptance criteria, correction multipliers were adjusted in the CPCs to correct the predicted values. This test was performed on September 8 through 9, 1985, with the reactor at approximately 50% full power, using test procedure 72PA-1RX18, "CEA Shadowing Factor/Radial Peaking Factor".

TEST DESCRIPTION

The test was initiated with the reactor fully unrodded (All Rods Out, or ARO) and at a stable power level and equilibrium xenon conditions. Baseline data was taken, including measurements of the excore detector responses, incore detector responses, and the secondary calorimetric power level. Several rodded configurations were then established while core power was held essentially constant using boron dilution or addition. These rodded configurations were:

* Regulating group 5 fully inserted
* Regulating groups 4 and 5 fully inserted
* Partlength group P 75% inserted with 4 and 5 fully inserted
* Group P 75% inserted with only 5 fully inserted
* Group P 75% inserted with no other rods inserted.

Data was recorded for each rodded configuration, including the excore detector responses, the incore detector responses, and the secondary calorimetric power.

CEA Shadowing Factors--The CEA shadowing factors are used to account for the alteration in the neutron flux seen by the excore detectors when the control rods (Control Element Assemblies, or CEAs) are inserted (assuming no change in gross power level). A CEA shadowing factor (F_x) for each rodded condition was determined from the measured data using the general relation:

$$F_x = \frac{D_R}{D_{ARO}} \times \frac{P_{sec(ARO)}}{P_{sec(R)}}$$

Where: D_R = Excore detector response with CEAs inserted
D_{ARO} = Excore detector response for the unrodded condition
$P_{sec(ARO)}$ = Secondary calorimetric power for the unrodded condition
$P_{sec(R)}$ = Secondary calorimetric power with CEAs inserted

The CEA shadowing factors determined from the measured data were then compared with the predicted values. For each rodded configuration in which the measured CEA shadowing factor did not agree within 3% of the predicted value, the CPCs were adjusted such that the measured factor would be used in the CPC power distribution calculation.

Radial Peaking Factors--The radial peaking factors account for the change in overall radial power distribution caused by CEA insertion by ensuring that the most limiting radial peak for the existing CEA configuration is used in the CPC calculation. "Measured" radial peaking factors were determined from an analysis of the fixed incore detector data taken with the various CEA configurations using the code CECOR. The CECOR calculated radial peaking factor value (CECOR F_{xy}) for each of the six measured configurations (one unrodded case plus five rodded cases) was then compared with the predicted value used internally by the CPCs for the particular configuration. For each case in which the measured peaking factor was larger than the predicted peaking factor, the CPCs were adjusted such that the measured factor would be used in the CPC power distribution calculation.

TEST RESULTS

The only measured CEA shadowing factor within the acceptable range was for the configuration of Group 5 fully inserted. For the other CEA configurations, the CPCs were adjusted to the measured values. Table 6-15 summarizes these results.

The measured radial peaking factors were less than the predicted peaking factors for all CEA configurations except the unrodded case. Consequently, the CPCs were adjusted to the measured unrodded peaking factor. The test results are summarized in Table 6-16.

CONCLUSIONS

All CEA shadowing factors and radial peaking factors used by the CPCs were verified to be accurate, or the appropriate adjustments were made to correct the CPC values to the measurement.

TABLE 6-15

MEASURED CEA SHADOWING FACTORS

CEA Group/Position	Predicted CSF	Acceptable Range	Factors Computed from Test Data CPC A	CPC B	CPC C	CPC D
5 / Full in	1.104	1.071 to 1.137	1.0942	1.0951	1.0948	1.0994
5 & 4 / Full in	1.038	1.007 to 1.069	0.9771	0.9853	0.9783	0.9715
5 & 4 / Full in P / 75% in	1.054	1.023 to 1.085	0.9370	0.9458	0.9390	0.9306
5 / Full in P / 75% in	1.125	1.091 to 1.159	1.0718	1.0719	1.0813	1.0838
P / 75% in	1.011	0.981 to 1.041	0.9669	0.9684	0.9734	0.9765

TABLE 6-16

MEASURED RADIAL PEAKING FACTORS (ALL CPC CHANNELS)

CEA Group/Position	Predicted Peaking Factor	Measured Peaking Factor
(Unrodded)	1.37	1.4210
5 / Full in	1.58	1.5549
5 & 4 / Full in	1.68	1.6515
5 & 4 / Full in P / 75% in	1.70	1.6793
5 / Full in P / 75% in	1.59	1.5703
P / 75% in	1.56	1.4167

6.15.2 Verification of Temperature Shadowing Factors

TEST OBJECTIVE AND SUMMARY

The purpose of this test was to verify the adequacy of the temperature shadowing factors used in the CPC power distribution calculation by measuring the decalibration of the excore detector responses associated with variation of the cold leg (reactor inlet) temperature. If the factors determined from the measured data differed from the predicted factors by more than 10%, the measured values were installed in the CPCs. This test was performed at approximately 50% power on September 8, 1985 using test procedure 72PA-1RX22, "Temperature Decalibration Test (50%)".

TEST DESCRIPTION

The temperature shadowing factors are used by the CPCs to adjust the calculated neutron flux power for changes in the reactor inlet temperature (Tcold), the theory being that temperature variations in Tcold will be accompanied by density changes in the water which will affect neutron attenuation across the downcomer of the reactor vessel and, hence, the response of the excore detectors.

To obtain the data needed to determine the actual temperature shadowing factor, Tcold was increased about 3 degrees above the initial temperature (~563°F) in 1 degree increments, followed by a decrease of 12 degrees (generally in 1.5 degree steps), followed by a return to the initial temperature in 1.5 degree steps. The temperature changes were made via turbine loading changes and boration/dilution. After a short stabilization period at each new value of Tcold, the following test data were recorded: excore detector responses, cold leg temperature, and secondary calorimetric power.

From each set of data, the ratio of the excore detector responses to the secondary calorimetric power was determined. Because the excore detector responses are affected by variations in cold leg temperature, but secondary calorimetric power is not, changes in this ratio are a direct indication of the impact on detector response caused by cold leg temperature variations. Each of these ratios was then normalized to the ratio determined from the data taken at the initial Tcold, to correct for any actual variations in reactor power over the course of the testing. That is, at each different Tcold (t),

$$\text{RATIO}_t = \frac{(\text{Excore detector response})_t}{(\text{Secondary calorimetric power})_t} = \frac{D_t}{P_{sec(t)}}$$

and,

$$(\text{NORMALIZED RATIO})_t = \text{RATIO}_t \times \frac{P_{sec(563^\circ F)}}{D_{563^\circ F}}$$

A least squares fit of the normalized ratios versus Tcold was then performed to produce the best estimate of the excore detector response variation as a function of Tcold. The slope of the line resulting from this fit (i.e. the change in excore response per °F change in Tcold) was the measured temperature shadowing factor.

TEST RESULTS

The temperature shadowing factor measured for each CPC channel was in acceptable agreement with the predicted value. Table 6-17 summarizes the test results.

CONCLUSIONS

The temperature shadowing factor measured for each CPC channel was in acceptable agreement with the predicted value. Therefore, the installed value used in the CPC calculation was satisfactory and no adjustments were necessary.

TABLE 6-17

MEASURED TEMPERATURE SHADOWING FACTORS (/°F)

CPC channel	Meas.	Pred.	%Diff*	Accept. % Diff
A	.0055	.0058	-5.2	± 10
B	.0054	.0058	-6.9	± 10
C	.0054	.0058	-6.9	± 10
D	.0055	.0058	-5.2	± 10

$$* \ \%Diff = \frac{(Meas - Pred)}{Pred} \times 100\%$$

6.15.3 Verification of Shape Annealing Matrix and Boundary Point Power Correlation Constants

TEST OBJECTIVE AND SUMMARY

The Shape Annealing Matrix and Boundary Point Power Correlation Test was performed at the 20% and 50% full power (FP) plateaus. The objective of the test performed at the 20% FP plateau was to verify that the installed CPC Shape Annealing Matrix (SAM) and Boundary Point Power Correlation Constants (BPPCCs) were suitable for power ascension to 50% FP. This was accomplished on June 23, 1985 using test procedure 72PA-1RX04, "Shape Annealing Matrix", which performed a comparison of the measured average axial power distribution and the axial power distribution calculated by each CPC channel to verify acceptable agreement between the two.

At the 50% FP plateau, the objective was to actually measure and install (if necessary) a new SAM and BPPCCs for each CPC channel. Testing was accomplished on September 10 through 12, 1985 using test procedure 72PA-1RX19, "Shape Annealing Matrix (50%)". The measured SAM and BPPCCs were compared to the installed CPC values for each CPC channel to determine the adequacy of the latter prior to power ascension above 50% power. This comparison of measured and installed values did not meet the acceptance criteria for any of the CPC channels, necessitating the installation of the measured SAM and BPPCCs into the CPC data base.

TEST DESCRIPTION

20% FP Test--In this test, the reactor core average axial power distribution was measured with all rods withdrawn from the core and equilibrium xenon conditions established. A "snapshot" was recorded of the fixed incore detector responses concurrently with the recording of the CPC calculated axial power distributions. The "measured" axial power distribution was determined from an analysis of the fixed incore detector responses using the code CECOR. The axial power distribution determined by CECOR was then compared to that calculated by the CPCs to verify that the root-mean-square (RMS) error between the two was no greater than 5%. If the RMS error exceeded 5%, the error between the measured and calculated axial peaks and axial shape indices would be further examined to determine whether the measured and CPC calculated values were in acceptable agreement. If this agreement was not verified, a measurement of the actual SAM/BPPCC values would be performed and these values installed in the CPCs before the reactor power was increased above the 20% power level.

50% FP Test--The SAM/BPPCC constants are used by the CPCs to calculate an accurate axial power profile from the excore detectors. In this test, the data used to determine the SAM and BPPCCs were measured over a range of various axial power shapes to ensure that this data would be representative of the range of axial power distributions expected throughout Cycle 1. To accomplish this, a free running axial xenon oscillation was established. For the next thirty hours (approximately the length of one free xenon oscillation cycle) the excore detector responses for each CPC channel were recorded simultaneously with fixed incore detector responses at approximately fifteen minute intervals.

Each set of incore detector responses was processed using the code CECOR to provide a set of "measured" peripheral axial power distribution information. A least squares analysis of the measured power distribution data from CECOR versus the corresponding excore detector data was then performed to determine the best set of SAM/BPPCC constants for relating measured excore detector responses to the true peripheral axial power distribution.

The results of the least squares analysis are subsequently used to compute a SAM "Test Matrix" value for each CPC channel which gives an indication of the acceptability of the SAM. Test matrix values in the range of 3.0 to 6.0 ensure that the design CPC power synthesis uncertainty factors are adequate and will result in conservative CPC DNBR and LPD calculations.

To determine whether the SAM values measured during the test needed to be installed into the CPC data base, the following criteria were used:

1) For each CPC channel, if the difference between the measured and predicted SAM is less than or equal to 2.0% for all elements, no adjusments are required.

2) For each CPC channel, if the difference is greater than 2.0%, the SAM test matrix value shall be calculated. If the test matrix value is in the range of 3.0 to 6.0, all of the measured SAM elements shall be installed in the CPC data base.

To determine if the BPPCCs measured during the test needed to be installed into the CPC data base, the following criteria were used:

1) For each CPC channel if the difference between the measured and predicted BPPCC is less than or equal to 3.0% for each constant, no adjustment is required.

2) If the difference between the measured and predicted value is greater than 3.0%, the measured BPPCC shall be installed in the CPC data base.

TEST RESULTS

20% FP Test--The RMS error between the CECOR measured and the CPC calculated axial power distribution was less than 5% for all CPC channels. Therefore, no further action nor any change to the CPC data base was required. The results are summarized in Table 6-18.

50% FP Test--The measured SAM and BPPCC values differed form the previously installed values by an amount that exceed the acceptance criteria of 2.0% and 3.0% respectively. Consequently, the measured SAM and BPPCC values were installed in the CPC data base. As indicated by the results, the test matrix values for all CPC channels were within the 3.0 to 6.0 test acceptance range. The test results are summarized in Tables 6-19 and 6-20.

CONCLUSIONS

The SAM/BPPCC constants initially installed in the CPCs were satisfactory for operation up to the 50% power test plateau. Measurement of the constants at 50% power, however, indicated unacceptable agreement with the installed values, thereby necessitating installation of the measured values in the CPCs.

TABLE 6-18

RMS VALUES 20% SAM TEST
(%)

CPC Channel	RMS Value	Acc. Criteria
A	4.801	5.0
B	4.477	5.0
C	4.853	5.0
D	4.641	5.0

TABLE 6-19

BOUNDARY POINT POWER CORRELATION CONSTANTS
(ALL CPC CHANNELS)

Parameter	Original Installed Value	Measured Value	%Diff.*
BPPCC1	0.01183	0.01389	17.41
BPPCC2	0.07056	0.08113	14.98
BPPCC3	0.01039	0.01433	37.92
BPPCC4	0.04640	0.08141	75.45

* Where,

$$\%Diff = \frac{(BPPCi(measured) - BPPCCi(original\ installed))}{BPPCCi(original\ installed)} * 100\%$$

for i = 1 to 4

TABLE 6-20

Shape Annealing Matrix Elements

SAM Element	Channel A Original Installed Value	Channel A Meas. Value	Channel A %Diff	Channel B Original Installed Value	Channel B Meas. Value	Channel B %Diff
S11	6.0514	3.4995	-42.17	6.0514	3.4944	-42.25
S12	-2.7624	-0.3993	-85.55	-2.7624	-0.4052	-85.33
S13	0.2375	-0.2116	-189.1	0.2375	-0.2029	-185.4
S21	-3.1013	-0.7857	74.67	-3.1013	-0.6600	-78.72
S22	7.9216	4.5093	-43.08	7.9216	4.2664	-46.14
S23	-2.7876	-0.8257	-70.38	-2.7876	-0.6252	-77.57
S31	0.0594	0.2861	381.65	0.0594	0.1656	178.79
S32	-2.1656	-1.1100	-48.74	-2.1656	-0.8612	-60.23
S33	5.5588	4.0372	-27.37	5.5588	3.8281	-31.13
Test Matrix Value	3.8979			3.7494		

SAM Element	Channel C Original Installed Value	Channel C Meas. Value	Channel C %Diff	Channel D Original Installed Value	Channel D Meas. Value	Channel D %Diff
S11	6.0514	3.5430	-41.45	6.0514	3.6220	-40.15
S12	-2.7624	-0.3814	-86.19	-2.7624	-0.5224	-81.09
S13	0.2375	-0.2287	-196.3	0.2375	-0.1571	-166.2
S21	-3.1013	-0.7587	-75.74	-3.1013	-0.8612	-72.23
S22	7.9216	4.4281	-44.10	7.9216	4.6055	-41.86
S23	-2.7876	-0.8265	-70.35	-2.7876	-0.8896	-68.09
S31	0.0594	0.2157	263.13	0.0594	0.2392	302.69
S32	-2.1656	-1.0467	-51.67	-2.1656	-1.0832	-49.98
S33	5.5588	4.0552	-27.05	5.5588	4.0467	-27.20
Test Matrix Value	3.8578			3.9111		

Where,

$$\%Diff = \frac{(Sij(measured) - Sij(original\ installed))}{Sij(original\ installed)} * 100\%$$

for i and j = 1 to 3

6.16 Main and Emergency Feedwater Systems Test (Section 14.2.12.5.17)

TEST OBJECTIVE AND SUMMARY

The primary objectives of this test were to verify the satisfactory operation of the Main and Emergency Feedwater Systems and also to verify the adequacy of the associated piping systems and supports.

Four test procedures were performed to evaluate the low power operation of the Feedwater Control System (FWCS), the downcomer-economizer valve transfer which occurs at approximately 15% full power (FP) and the performance of the main feedwater pumps:

1) 73PA-1FW01, "FWCS Test at 10% Power" evaluated the performance of the FWCS at a power level of 10% FP and was performed on June 25 and 26, 1985.
2) 73PA-1FW02, "FWCS Valve Transfer Checkout Test with Power Decreasing", evaluated the transfer of the main feedwater flow from the economizer to the downcomer during a decrease from 20% to 10% FP and was performed on June 26 and 27, 1985.
3) 73PA-1FW03, "FWCS Valve Transfer Checkout Test with Power Increasing", evaluated the transfer of the main feedwater flow from the downcomer to the economizer during an increase from 10% to 20% FP and was conducted on June 11 and 12, 1985.
4) 73PA-1FW04, Feedwater System Operability" evaluated the performance of the main feedwater pumps by collecting data at each 10% power increment (10% to 100% FP). This test also includes removal of one high pressure feedwater heater train from service while operating at 100% FP to determine if there is any plant capacity degradation.

To verify the adequacy of the piping systems and supports, test procedure 73PA-1SG04, "Dynamic Transient Test (Main Feedwater)", was performed. It consisted of four sections:

1) Section 8.1 monitored the feedwater transfer from the downcomer to economizer during power increases and was performed on June 26, 1985.
2) Section 8.2 monitored the feedwater transfer from the economizer to downcomer during power decreases and was performed on June 27, 1985.
3) Section 8.3 monitored the system piping during a feedwater pump trip and was performed on January 24, 1986.
4) Section 8.4 monitored the system response during the restart of the second main feedpump at approximately 65% FP and was performed on October 19, 1985.

TEST DESCRIPTION

73PA-1FW01--This was a test of the FWCS's ability to maintain steam generator level within ±5% of the control setpoint during steady state and minor transient conditions. The steady state portion simply involved placing the FWCS in automatic and observing control of steam generator level vis-'a-vis the setpoint. The transient portion of the test involved both increasing and decreasing the level setpoint (approximately 5%) in both ramp and step functions to determine whether the FWCS controlled steam generator levels to within 5% of the setpoint (after allowing for a brief stabilization period).

73PA-1FW02--The first portion of this test involved a manual (operator controlled) transfer of feedwater flow from the economizer to the downcomer while reactor power was decreasing through the 15% FP switchover region. The second portion was performed with the FWCS in the automatic mode to evaluate its ability to execute the transfer automatically. Data was taken taken during and after the transfer and was evaluated against specific acceptance criteria.
73PA-1FW03--This test was performed in essentially the same manner as 73PA-1FW02 described above, except that the transfer was from the downcomer to the economizer with reactor power increasing. Again, a manual and automatic transfer were made and similar data were collected for comparison with acceptance criteria.
73PA-1SG04--To measure any loads that may have been imposed on the piping systems and restraints, thirteen load-sensing pins were installed at various hanger locations. Data was collected during the various evolutions previously mentioned and was evaluated against the acceptable loads calculated for each load pin. Also, a visual inspection of the piping, the supports and adjoining structures was performed after the transient portions of the test.
73PA-1FW04--A data snapshot was taken on the plant computer at the start of a feedpump, after feedwater had been transferred to the feedpump from the Auxiliary Feedwater System, and at 10% power increments from 10% to 100% FP. Single pump data was collected for each pump from 10% to 50% FP and dual pump data was collected above 50% FP. Additionally, at 100% FP, one high pressure feedwater train was to be removed from service and data collected to determine if there is any resulting degradation of plant power.

TEST RESULTS

73PA-1FW01--The acceptance criteria for the test were satisfied. The actual test results are compared with the acceptance criteria in Table 6-21.
73PA-1FW02--As can be seen in the test results listed in Table 6-22, steam generator pressure exceeded its acceptable limits during both the automatic and manual valve transfers. A Test Exception Report was generated and the results were declared acceptable based on the fact that the data was recorded almost 10 minutes after the valve transfer, whereas the acceptance criteria values supplied by Combustion Engineering were assumed to be those which would prevail almost immediately after the valve transfer. Combustion Engineering also reviewed the test data via an Engineering Evaluation Request and concurred that the results were acceptable.
73PA-1FW03--The acceptance criteria were met during performance of the automatic transfer portion of the test. However, this was not the case during the manual transfer--steam generator pressure and three RCS cold leg temperatures fell outside the acceptable limits. A Test Exception Report was generated which determined that the results were acceptable since the primary intent of the test was to evaluate the automatic operation of the FWCS, which indeed proved superior to manual operation. Also, since manual operation will vary somewhat with each operator, it was deemed inappropriate to place too much emphasis on failure to meet the aforementioned criteria. The test results are listed in Table 6-23.
73PA-1FW04--Data has been collected at the specified power levels, but because of feedwater drain tank level control valve problems, the removal of the high pressure feedwater heater trains has not been accomplished. The test results recorded to date have been satisfactory; however, since the test has not yet been completed, the final results are pending.

73PA-1SG04--The observed loads were well below the maximum acceptable loads (which varied from 1,700 to 55,000 lbf) calculated for the specified load pins. The loads observed were, in fact, at or below 100 lbf. The visual inspection revealed that no damage was sustained by the piping, the supports, or the adjoining structures. Thus, the acceptance criteria were satisfied.

CONCLUSIONS

The ability to perform the downcomer-economizer valve transfer both automatically and manually with no adverse impact on overall plant control was demonstrated by these tests. Also, the ability of the FWCS to perform as designed under a number of different circumstances was demonstrated. In addition, it was confirmed that the design and construction of the main and auxiliary feedwater systems and associated hangers are adequate to support any normally encountered operating modes without sustaining damage or apparent degradation.

TABLE 6-21

FWCS CHECKOUT AT 10% POWER (73PA-1FW01)
TEST RESULTS

Acceptance Criteria	Maximum Deviation
S/G levels do not deviate more than ± 5% from set pt. for steady state conditions	0.87
S/G levels do not deviate more than ± 5% from set pt. for ramp set pt. changes	2.50
S/G levels do not deviate more than ± 5% from set pt. for step set point changes	2.00

TABLE 6-22

FWCS VALVE TRANSFER--POWER DECREASE (73PA-1FW02)
TEST RESULTS

ACCEPTANCE CRITERIA (deviation from initial value)	AUTOMATIC Actual Value	MANUAL Actual Value
#1 Nuc power decreased no more than 4%	4.0	2.25
#2 Nuc power decreased no more than 4%	4.0	3.5
S/G #1 level increased less than 30%	12.0	15.0
S/G #2 level increased less than 30%	8.5	8.0
S/G #1 pressure decreased less than 30 psia	35.0*	33.0*
S/G #2 pressure decreased less than 30 psia	35.0*	36.0*
RC cold leg temps. increased no more than 6 degrees		
Tc 1A:	3.0	4.5
Tc 1B:	3.8	6.0
Tc 2A:	1.5	4.5
Tc 2B:	1.4	6.0

* Denotes out of tolerance value.

TABLE 6-23

FWCS VALVE TRANSFER--POWER INCREASE (73PA-1FW03)
TEST RESULTS

ACCEPTANCE CRITERIA (deviation from initial value)	AUTOMATIC Actual Value	MANUAL Actual Value
#1 Nuc power increased less than 5%	3.75	2.8
#2 Nuc power increased less than 5%	3.75	2.9
S/G #1 level decreased less than 40%	18.0	16.95
S/G #2 level decreased less than 40%	16.0	21.5
S/G #1 pressure increased no more than 50 psia	38.0	52.0*
S/G #2 pressure increased no more than 50 psia	38.0	50.0
RC cold leg temps decreased less than 8 degrees		
Tc 1A:	3.7	9*
Tc 1B:	6.0	11.5*
Tc 2A:	5.9	9*
Tc 2B:	6.5	7.4

* Denotes out of tolerance value.

6.17 CPC Verification and COLSS Verification (Sections 14.2.12.5.18 and 14.2.12.5.20)

TEST OBJECTIVE AND SUMMARY

The objectives of these tests were to verify the calculations of Departure from Nucleate Boiling Ratio (DNBR) and Local Power Density (LPD) performed by the Core Protection Calculators (CPCs) and the Core Operating Limit Supervisory System (COLSS), in addition to evaluating the effect of process instrument noise on the CPC system.

Testing was performed in accordance with the "COLSS/CPC Verification" test which was directed by PVNGS procedures 72PA-1SB02, at 20% full power (FP) on June 19, 1985; 72PA-1SB06, at 50% FP on July 7, 1985; 72PA-1SB11, at 80% FP on September 28, 1985; and 72PA-1SB16, at 100% FP on January 9, 1986. The results from each of these tests were satisfactory.

TEST DESCRIPTION

The calculations performed by each CPC channel are verified by comparing the values of Local Power Density (LPD) and Departure from Nucleate Boiling Ratio (DNBR) recorded from each channel with the values calculated by the Combustion Engineering CPC FORTRAN simulator code CEDIPS. When provided with a known variation of input data recorded from each of the CPC channels, the CEDIPS code calculates a range of values for LPD and DNBR which would be expected to bound the actual values observed on each of the CPC operator display devices. The CPC data used as input to CEDIPS is manually gathered from the CPC display devices as maximum and minimum values observed over a specified period of time and consists of: pressurizer pressure, RCP speeds, control rod positions, RCS cold and hot leg temperatures, and excore detector responses. The CEDIPS DNBR and LPD values are compared to those observed and recorded during the test. If the observed DNBR and LPD values are within the range of expected values, the functioning of each CPC channel is considered to be verified and process instrument noise has not affected the CPC operation.

As a further step in evaluating the effect of process instrument noise, input signals to the "noisiest" CPC channel and its analog outputs were recorded on FM tape over a period of approximately two hours. The determination of the "noisiest" CPC channel was accomplished by monitoring the maximum variation in the DNBR value calculated by each CPC channel at 1 minute intervals over a 10 minute period. This data was gathered for evaluation by the NSSS vendor and had no specific test acceptance criteria.

For the COLSS calculations, a statistical analysis of different sensor inputs measuring the same parameters was performed to ensure that the instruments were consistent and functioning properly. The statistical analysis is performed automatically on demand by the COLSS Sensor Deviation Statistical Routine which is executed on the Plant Monitoring System (PMS). Additionally, COLSS input and output values were collected via PMS data snapshots.

Following completion of testing at each test plateau, the COLSS statistical data, the COLSS input and output data snapshots, and the FM recorded CPC data were transmitted to the NSSS vendor for evaluation.

TEST RESULTS

The CPC calculated minimum/maximum DNBR and LPD values recorded during the test for the 20, 50, 80 and 100% power plateaus are provided in Table 6-24 along with the CEDIPS calculated range of expected values. All of the CPC calculated DNBR and LPD values were bounded by the corresponding CEDIPS range of values with the exception of the minimum LPD value calculated by Channel B at the 20% test plateau, which was lower than the CEDIPS lower LPD limit. A transcription error and/or error in reading the CPC operator display module was thought to be the reason for Channel B's failure to meet the test acceptance criteria. A Test Exception Report was written to perform a retest for Channel B. The retest results, given in Table 6-25 show that the Channel B CPC values for DNBR and LPD were satisfactorily bounded by the CEDIPS calculated range of values.

The COLSS data recorded during the testing have been evaluated by the NSSS vendor and the COLSS calculations have been determined to be acceptable.

CONCLUSION

The CPC and COLSS calculations of DNBR and LPD were satisfactorily confirmed at each of the major test plateaus.

TABLE 6-24
(Part 1 of 2)

CPC/CEDIPS COMPARISON VS POWER LEVEL

PARAMETER		CHANNEL A		CHANNEL B	
20% FP TEST		CEDIPS	CPC	CEDIPS	CPC
LPD	MAX	6.27808	6.0911	5.89966	5.3367
	MIN	6.01026	6.0594	5.34798	5.3139
DNBR	MAX	6.35754	6.2730	7.44165	7.0398
	MIN	6.04197	6.2537	6.35023	7.0174
50% FP TEST					
LPD	MAX	12.6953	12.658	12.9985	12.546
	MIN	12.2061	12.347	12.1998	12.476
DNBR	MAX	2.3136	2.2389	2.2976	2.1682
	MIN	2.1805	2.1983	2.0855	2.1410
80% FP TEST *					
LPD	MAX	8.22113	8.0561	8.39350	8.0565
	MIN	7.86225	7.9484	7.87477	7.9516
DNBR	MAX	3.96545	3.8703	3.89445	3.8004
	MIN	3.72580	3.7616	3.57324	3.7398
100% FP TEST					
LPD	MAX	14.789	14.512	14.722	14.494
	MIN	14.291	14.491	14.305	14.491
DNBR	MAX	1.757	1.7334	1.7630	1.6884
	MIN	1.672	1.7076	1.6590	1.6849

*-- After the performance of the 50% FP test, subsequent testing resulted in several adjustments to the CPCs which affected the power distribution calculation (and thus, the calculated values of DNBR and LPD). Foremost among these adjustments was the installation of a new Shape Annealing Matrix in each channel (see Section 6.15.3). These adjustments account for the decrease in LPD and the increase in DNBR observed between the 50% and 80% FP test results.

TABLE 6-24
(Part 2 of 2)

CPC/CEDIPS COMPARISON VS POWER LEVEL

PARAMETER		CHANNEL C		CHANNEL D	
20% FP TEST		CEDIPS	CPC	CEDIPS	CPC
LPD	MAX	6.33491	6.0175	5.86722	5.6620
	MIN	5.8880	5.9595	5.62132	6.6514
DNBR	MAX	6.72688	6.3478	6.82789	6.6321
	MIN	5.93464	6.2944	6.36672	6.5936
50% FP TEST					
LPD	MAX	12.6339	12.206	12.998	12.382
	MIN	11.9504	12.142	12.198	12.354
DNBR	MAX	2.3498	2.2407	2.2870	2.1804
	MIN	2.1468	2.2090	2.0330	2.1220
80% FP TEST *					
LPD	MAX	8.28922	7.9497	8.19140	7.9795
	MIN	7.78874	7.8801	7.84869	7.9116
DNBR	MAX	3.98046	3.7663	3.90966	3.7802
	MIN	3.59845	3.6800	3.59382	3.6304
100% FP TEST					
LPD	MAX	14.695	14.171	15.050	14.411
	MIN	13.975	14.157	14.156	14.401
DNBR	MAX	1.7760	1.6723	1.7740	1.7084
	MIN	1.6200	1.6693	1.5770	1.7084

*-- After the performance of the 50% FP test, subsequent testing resulted in several adjustments to the CPCs which affected the power distribution calculation (and thus, the calculated values of DNBR and LPD). Foremost among these adjustments was the installation of a new Shape Annealing Matrix in each channel (see Section 6.15.3). These adjustments account for the decrease in LPD and the increase in DNBR observed between the 50% and 80% FP test results.

TABLE 6-25
CPC/CEDIPS COMPARISON FOR RETEST AT 20% FP
(Performed on 6/22/85)

PARAMETER		CHANNEL B	
20% FP RETEST *		CEDIPS	CPC
LPD	MAX	4.1825	3.8351
	MIN	3.80220	3.8278
DNBR	MAX	10.0014	9.5610
	MIN	8.63536	9.5412

*-- In the interim between the 20% FP test and the retest on Channel B, an adjustment of the excore detector linear subchannel amplifiers was performed. This adjustment affected the relative distribution of the excore signals (i.e. upper vs. lower signals) and thus accounts for the large change in recorded DNBR and LPD values between the two tests.

6.18 Steam Bypass Valve Capacity Test
(Section 14.2.12.5.19)

TEST OBJECTIVE AND SUMMARY

The atmospheric steam dump and steam bypass control system valve capacity test 73PA-1SG01 was conducted on August 31 and September 23, 1985 with the reactor initially stabilized at 30% power. The principal objectives of the test were as follows:

(1) To verify that the capacity of each atmospheric dump valve (ADV) and steam bypass control system (SBCS) valve is less than 11% of the total full power steam flow rate.

(2) To verify that each ADV capacity is greater than 6% of the total full power steam flow rate.

(3) To verify that the total capacity of the 8 SBCS valves is greater than 55% of the total full power steam flow rate.

These percentages are based upon a full power steam flow rate of 17,118,144 lbm/hr at a steam generator pressure of 1070 psia. The measured capacity of each SBCS valve and ADV met the design criteria.

TEST DESCRIPTION

Stable plant conditions were established with the 4 ADVs and 8 SBCS valves closed and reactor power equal to turbine load. A baseline feedwater flow rate was determined. The capacity of each valve was measured individually by cycling the valve full open and then closed. As the valve position was cycled, reactor power and feedwater flow were adjusted to maintain the turbine load as steady as possible. The valve capacity was derived from the difference in feedwater flow rate with the one valve fully open and the baseline condition of all valves closed. This difference in feedwater flow rate was corrected for the difference between full power steam pressure and the test condition steam pressures.

TEST RESULTS

The capacity of each SBCS valve and each ADV was measured to be within the design criteria of greater than or equal to 6% and less than 11% of total full power steam flow rate. The minimum measured valve capacity was 9.46% and the maximum measured valve capacity was 10.89%. The total capacity of the 8 SBCS valves was 80.36% and satisfied the minimum criteria of 55%.

CONCLUSIONS

The capacity of each SBCS valve and each ADV was measured to be within the design criteria and satisfied the safety analysis assumptions concerning the maximum capacity of a single valve.

6.19 Incore Detector Test
(Section 14.2.12.5.20)

TEST OBJECTIVE AND SUMMARY

Testing of the fixed incore detector system (FICDS) was performed in accordance with the "Incore Detector (Fixed) Test" which was directed by PVNGS procedures 72PA-1RI01, at 20% full power (FP) on June 14 through 20, 1985; 72PA-1RI05, at 50% FP on July 7, 1985; 72PA-1RI10, at 80% FP on September 26, 1985; and 72PA-1RI15, at 100% FP on December 31, 1985. The objectives of this testing were:

1) To verify the operability of the system via execution of automatic test functions on the Plant Monitoring System (PMS);
2) To record and review the fixed incore detector voltages to identify potential detector/amplifier failures;
3) To verify that the detector signals received at the input of the PMS were consistent with those measured at the output of the amplifiers;
4) To measure the background voltages.

Testing of the movable incore detector system (MICDS) was performed in accordance with the "Movable Incore Detector Check", which was directed by PVNGS procedures 72PA-1RI02, at 20% FP on June 12 through 22, 1985; 72PA-1RI06, at 50% FP on July 10, 1985; 72PA-1RI11, at 80% FP on September 30, 1985; and 72PA-1RI16, at 100% FP on December 30, 1985. The objectives of this testing were:

1) To establish the initial operating parameters for the system (i.e., amplifier gain settings, detector bias voltage levels, etc.);
2) To gather baseline detector signal data and performance information (i.e., detector positioning, signal repeatability, etc.);
3) To demonstrate the use of the MICDS in obtaining an axial flux profile during core traverses.

The operability of the FICDS was verified at each test plateau and all test objectives were satisfied. The MICDS was not fully functional during the testing due to an inoperable moveable incore detector (one of two in the system). However, sufficient data was recorded with the operable detector to satisfy the test objectives.

TEST DESCRIPTION

Fixed Incore Detector Testing--The operability of the FICDS was verified by executing automatic test functions programmed into the PMS. The three test functions are:

1) Conversion of a zero current input to each amplifier to a zero voltage output to within ± 0.025 vdc (ZERO OFFSET).
2) Conversion of a full scale input signal (10 microamps) to each amplifier to a full scale ouput (10 volts) to within ± 0.136 vdc (AMPLIFIER GAIN).
3) Measurement of the cable leakage resistance in each detector and evaluation of the measured resistance values to a minimum acceptable value of 1000 k ohms.

The fixed incore detector test functions were executed via test pushbuttons located in each Fixed Incore Amplifier Bin. The PMS is programmed to evaluate the data and summarize the comparison of the measured values with the incorporated tolerances.

A set of fixed incore detector voltage signals and uncompensated flux signals were also obtained via the PMS and reviewed to verify that the signal levels were within an expected range for the appropriate power level and core location. This evaluation was performed by comparing signals from symmetric detectors and/or from detectors located in surrounding assemblies.

At 100% FP, raw detector voltage signal levels were measured at the amplifier assembly card test points (amplifier output) and compared to the signal read by the PMS to demonstrate that the voltages agree within acceptable levels (i.e., ± 1%). A measurement of the detector background signal contribution was also performed via the amplifier assembly card test points to verify that the actual background was equal to or lower than the background correction terms incorporated in the PMS data base.

Moveable Incore Detector Testing--The MICDS testing was comprised of four phases. Phase 1 determined the bias voltage and amplifier gain settings which produce an amplifier output voltage of approximately -5 vdc when the moveable detector is positioned at the core mid-plane in an assembly located near the center of the core. With a fixed gain setting, the applied detector voltage was varied in increments of 10 vdc over the range of 30 to 150 vdc and the corresponding amplifier output voltage was recorded. Based on the data collected, a detector bias voltage and amplifier gain setting was selected for use during the other test phases.

Phase 2 of the MICDS testing collected background signal data for a fuel assembly near the core center and an assembly near the core periphery. Using the amplifier gain setting previously determined, the amplifier output voltages were recorded as the detector was positioned at 5 inches above and 5 inches below the active fuel region in the selected assemblies. In addition, data was collected as the detector traversed through the active fuel region. Two additional sets of data were also obtained for the measured fuel assemblies, using different amplifier gain settings.

Phase 3 of the testing was performed to ascertain the repeatability of the moveable incore detector output signal. Utilizing the amplifier gain setting and applied detector bias voltage determined in Phase 1, the detector was inserted to approximately mid-core and then to 15 inches below the top of the active fuel region. Amplifier output voltage readings were recorded for both axial locations. The detector was then withdrawn to a position below the active core, allowed to come to thermal equilibrium, and the process was repeated. A total of six data sets were obtained in this manner.

Phase 4 of the testing performed an axial flux mapping of the core. The detector was positioned below the active core region for each core location and allowed to come to thermal equilibrium. From this position, the detector was inserted to a position above the active fuel region and then withdrawn. During the withdrawal, the amplifier output was connected to a strip chart recorder which produced a trace of the signal, showing the axial flux profile for the particular core location.

TEST RESULTS

Fixed Incore Detector Test--The zero and full scale amplifier signal checks were satisfactorily completed at each test plateau. However, PMS calculated cable leakage resistance values of less than the minimum acceptable limit of 1000 k ohms were obtained for several detectors at each test plateau. A review of the corresponding detector voltages and uncompensated flux signal indicated that the detector signals were valid. The NSSS vendor performed independent cable leakage measurements using a high resistance meter and determined that all

cable leakage resistances were greater than the 1000 k ohm acceptance limit. A check of the test methodology and associated test electronics also indicated everything to be functioning properly. The problem has been identified as a possible capacitance problem in the mineral insulated detector connector cable. The vendor is confident that the fixed incore detectors are not being degraded during normal power operation and is continuing its investigation into the cause of the capacitance problem.

Measurements performed during 100% FP testing verified that the difference between the measured signal at the amplifier output and the signal read by the PMS were within the acceptance criteria limit of ± 1%. In addition, measurement of the detector background signal levels showed the background to be less than the acceptance criteria limit of 5% of the flux signal from the fixed incore detector located at the core mid-plane for the particular core location.

Moveable Incore Detector Check--The test was successfully completed at each major test power plateau. Axial flux maps for thirty-one instrumented fuel assemblies were obtained using moveable incore detector B during testing at the 20%, 50%, and 80% FP plateaus. A full core flux map (refer to Figure 6-8 for the locations of the 61 instrumented fuel assembly locations) was performed using detector B at 100% FP.

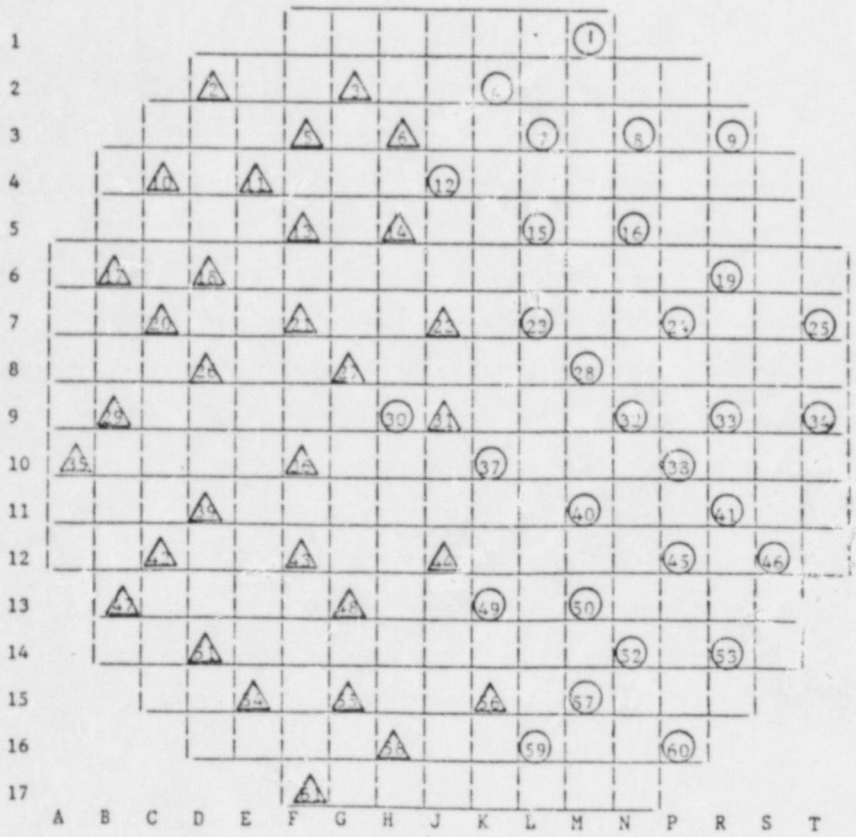
Moveable detector A was inoperable during the entire PAT program and consequently baseline data, performance characteristics information, and axial flux traces were not obtained. The test objectives, however, were satisfied since performance data was obtained for detector B which can be used to obtain a full core flux map (as demonstrated at 100% full power). The operability of both moveable incore detectors, while not required to support plant power operation, provides for a measure of redundancy and increases the speed of data acquisition.

CONCLUSION

The operability of the fixed incore detector system has been verified at each of the test plateaus. Initial operating parameters, performance data, and axial flux traces were obtained for moveable incore detector B at the 20%, 50%, 80%, and 100% FP test plateaus, thereby satisfying the test objectives.

FIGURE 6-8

INCORE DETECTOR LOCATIONS
(FIXED AND MOVABLE)



NORMAL Mode Operation

○ instrument strings accessed by Detector A

△ instrument strings accessed by Detector B

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