PALO VERDE NUCLEAR GENERATING STATION

UNIT 3 STARTUP REPORT

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PALO VERDE NUCLEAR GENERATING STATION

UNIT 3 STARTUP REPORT

TABLE OF CONTENTS

SECTION	TITLE	PAGE
	Preface	1
1.0	Introduction and Test Program Summary	2
1.1	Summary of Test Objectives by Test Phase	2
1.2	Chronology of Startup Testing: Fuel Load through Low Power Physics Testing	4
1.3	Summary of Startup Test Results: Fuel Load through Low Power Physics Testing	5
1.4	Chronology of Startup Testing: Power Ascension Testing Phase	6
1.5	Summary of Power Ascension Testing	8
2.0	Initial Fuel Loading	9
3.0	Postcore Hot Functional Tests	14
3.1	Postcore Hot Functional Test Controlling Document	14
3.2	Postcore Instrument Correlation	16
3.3	Postcore Reactor Coolant System Flow Measurement	17
3.4	Postcore Control Element Drive Mechanism Performance	19
3.5	Postcore Reactor and Secondary Water Chemistry Data	21
3.6	Postcore Pressurizer Spray Valve and Control Adjustments	23
3.7	Postcore Reactor Coolant System Leak Rate Measurement	25
3.8	Postcore Incore Instrumentation Test	26
4.0	Initial Criticality	28
5.0	Low Power Physics Testing	29
5.1	Low Power Biological Shield Survey Test	30
5.2	Isothermal Temperature Coefficient Test	32
5.3	Shutdown and Regulating CEA Group Worth Test	34
5.4	Differential Boron Worth Test	37
5.5	Critical Boron Concentration Test	38

PALO VERDE NUCLEAR GENERATING STATION

UNIT 3 STARTUP REPORT

TABLE OF CONTENTS
(cont'd)

SECTION			TITLE	PAGE
6.0			Power Ascension Tests	40
	6.1		Variable Tavg (Isothermal Temperature Coefficient and Power Coefficient) Test	40
	6.2		Unit Load Transient Test	43
	6.3		Control Systems Checkout Test	47
	6.4		Reactor Coolant and Secondary Chemistry and Radiochemistry Test	50
	6.5		Unit Load Rejection Test	53
	6.6		Biological Shield Survey Test	57
	6.7		Steady State Core Performance Test	59
	6.8		Intercomparison of PPS, Core Protection Calculator (CPC), and PMS Inputs	64
	6.9		Verification of CPC Power Distribution Related Constants Test	66
		6.9.1	Verification of CEA Shadowing Factors and Radial Peaking Factors	66
		6.9.2	Verification of Temperature Shadowing Factors	69
		6.9.3	Verification of Shape Annealing Matrix and Boundary Point Power Correlation Constants	71
	6.10		Main and Emergency Feedwater Systems Test	75
	6.11		CPC Verification and COLSS Verification	80
	6.12		Steam Bypass Valve Capacity Test	84
	6.13		Incore Detector Test	85
	6.14		Shutdown from Outside the Control Room Test	87

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 3 (PVNGS 3) 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 3, the third System 80 NSSS to start operation, has a rated core thermal output of 3800 MWt, and a nominal net electric output of 1270 MWe. PVNGS 3 is a follow-on unit to PVNGS 1 and 2.

The objective of this report is to provide a summary description of the initial startup test program for PVNGS 3. 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 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 required testing from Fuel Loading through the Power Ascension Testing phase. Testing is listed and summarized by the applicable section of CESSAR. Note that the scope of testing for PVNGS 3 is reduced from that of PVNGS 1 and 2. This scope reduction results from both the elimination of those tests required only for the first-of-a-kind plant (described in CESSAR) and from exceptions taken to the CESSAR tests described in the PVNGS FSAR. Two open items remain. The feedwater piping visual inspection (Section 6.10) and the FM taping portion of the CPC verification (Section 6.11) have not yet been performed. These items will be addressed in a future supplement to this report.

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. Initial criticality is performed at the normal hot zero power RCS conditions of 565 ^{o}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. Testing is performed at 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 relative to 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.

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. If any transient testing is to be performed at the plateau, it is performed at the end of the plateau. The following transient tests were performed for PVNGS 3:

* Shutdown Outside of Control Room Test
* Unit Load Rejection 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 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	Mar. 28—Apr. 2, 1987
Post Fuel Load Checks, NSSS Assembly and Preparation for Mode 4 Entry *	179 days	Apr. 2—Sep. 28, 1987
Postcore HFT	26 days	Sep. 28—Oct. 23, 1987
Mode 4 Entry	————	Oct. 1, 1987
Preparation for Initial Criticality	2 days	Oct. 23—Oct. 25, 1987
Initial Criticality	1 day	Oct. 25, 1987
Low Power Physics Testing	3 days	Oct. 25—Oct. 28, 1987
	215 days	Mar. 28, 1987—Oct. 28, 1987

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

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

Fuel Load was completed in 5 days. There were no significant problems with the fuel loading.

Postcore HFT was completed in 26 days with no significant problems. The results of this test phase that are the most significant to power operations are:

* The performance of the control rod drive system was excellent, and remains excellent throughout the early stages of the fuel cycle.
* The Reactor Coolant System (RCS) steady state flow rate was 105.77% of the design volumetric flow.
* The Reactor Coolant Pump coastdown flow was measured to be slightly faster than that assumed in the safety analysis. Penalty factors were applied to the COLSS and CPC software.

Initial Criticality was completed with no significant problems. The measured critical soluble boron concentration was 1014.1 ppm versus a predicted value of 1015 ppm and was within the acceptance criteria.

Low Power Physics Testing was completed in 3 days with no significant problems, but one minor problem with the testing equipment was encountered (see Section 5.0). All test results fell within their acceptance criteria, and all test results compared well with Unit 1 results. The Unit 2 results are not directly comparable because a different testing methodology was employed on that unit.

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 1-1.

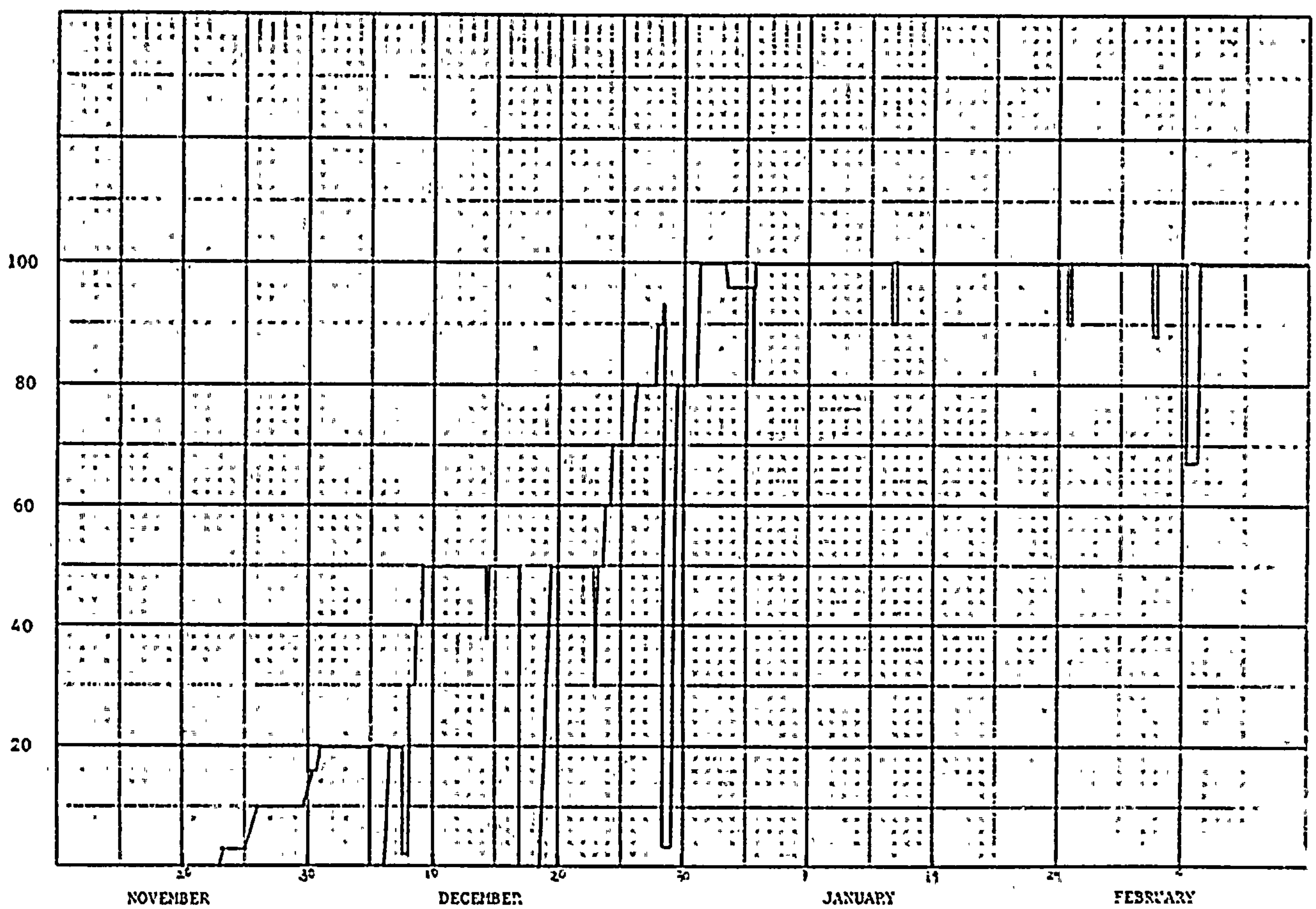
TEST ACTIVITY	DURATION	DATES
Initial increase from zero to 20% power	34 days	Oct. 28—Dec. 1, 1987
Full power operating license issued	———	Nov. 25, 1987
First generation of electrical power	———	Nov. 28, 1987
20% Power Test Plateau	7 days	Dec. 1—Dec. 8, 1987
Initial increase from 20% to 50% power	1 day	Dec. 8—Dec. 9, 1987
50% Power Test Plateau	15 days	Dec. 9—Dec. 24, 1987
Initial increase from 50% to 80% power	2 days	Dec. 24—Dec. 26, 1987
80% Power Test Plateau	2 days	Dec. 26—Dec. 28, 1987
Initial increase from 80% to 100% power	3 days	Dec. 28—Dec. 31, 1987
100% Power Test Plateau	7 days	Dec. 31, 1987—Jan. 7, 1988
	71 days	Oct. 28, 1987—Jan. 7, 1988

On January 7, 1988, based upon the successful completion of the Power Ascension Test program, Arizona Public Service Company declared the commencement of firm power operation of PVNGS 3.

FIGURE 1-1

PVNGS UNIT 3 POWER HISTORY:

NOVEMBER 1987 TO FEBRUARY 1988



1.5 Summary of Power Ascension Testing

The Power Ascension Test phase was completed over a period of approximately 6 weeks. This test phase demonstrated that PVNGS 3 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 the Unit Load Rejection Test was completed by taking credit for an inadvertant turbine trip during power ascension (see Section 6.5). The Power Ascension Test results have verified the design models used for PVNGS and have confirmed that PVNGS 3 is constructed as designed and is similar to PVNGS 1 and 2.

Section 6.0 of this report describes the individual Power Ascension Tests that satisfy requirements described in the PVNGS FSAR.

There were a total of two reactor trips during Power Ascension Testing. One of these trips was a planned trip as part of Power Ascension Testing (Shutdown Outside of Control Room Test, Section 6.14). One additional trip occurred inadvertently as a result of scheduled testing (Verification of CEA Shadowing Factors and Radial Peaking Factors Test, Section 6.9.1).

There was one outage of approximately one day during Power Ascension Testing. The major activity accomplished during this outage was balancing of the main turbine.

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 3 was 72IC-3RX01, "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 March 28 through April 2, 1987. 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 identified.

TEST DESCRIPTION

The initial core loading of PVNGS 3 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 4121 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 4120 ppm at the beginning of fuel load. One shutdown cooling loop was operated 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 all of the 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

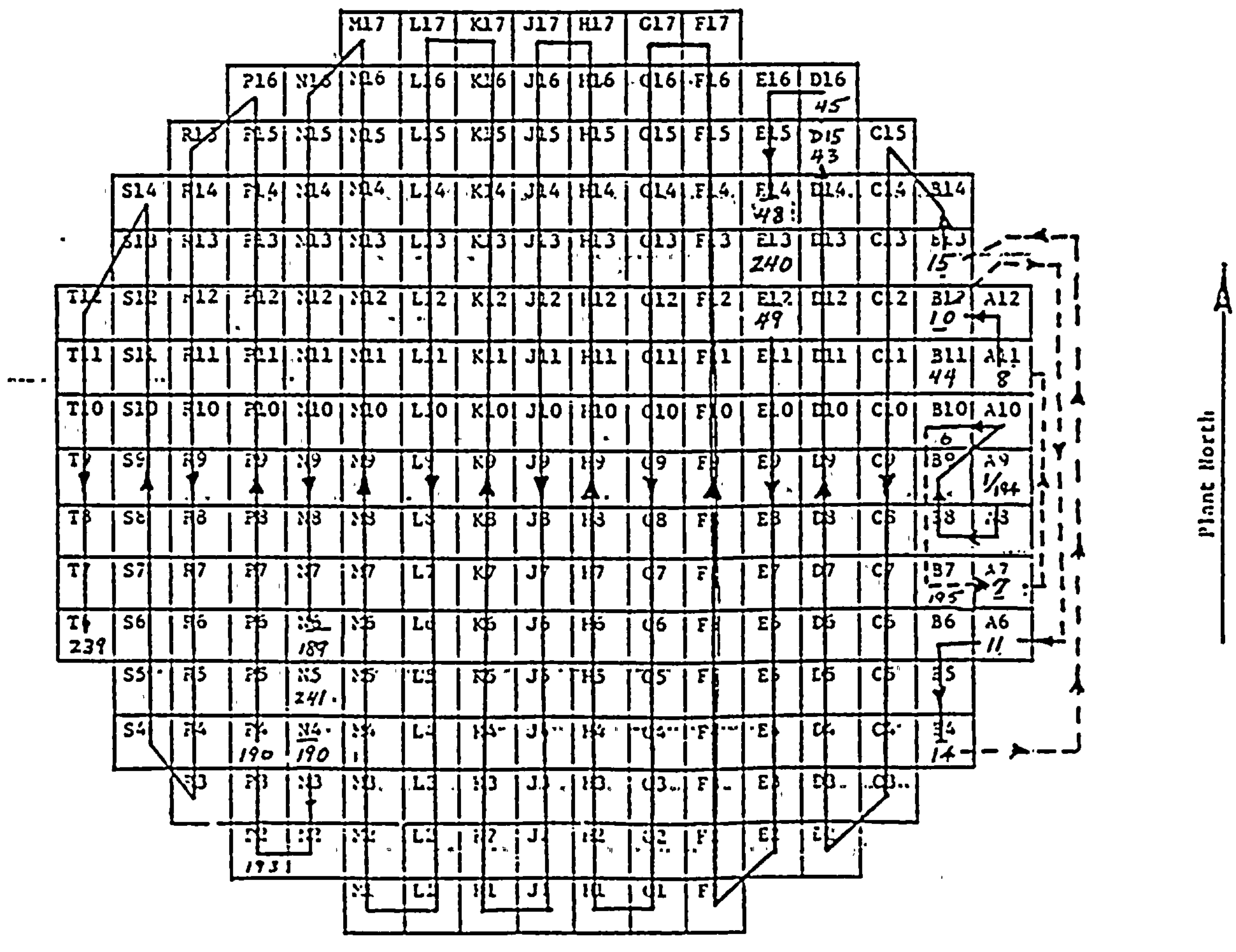
Fuel loading was completed on April 2, 1987. The fuel assemblies were verified to be properly loaded, seated, and oriented. Both startup neutron sources were verified to be properly loaded. Finally, scans of all of the assemblies verified that all of the assemblies were aligned within an acceptable tolerance of the nominal fuel centerlines. 72IC-3RX01 was officially completed with the results satisfactory on April 7, 1987.

CONCLUSIONS

The initial fuel loading of PVNGS Unit 3 was successfully accomplished in a safe and controlled manner, in accordance with the objectives and acceptance criteria of 72IC-3RX01.

FIGURE 2-1

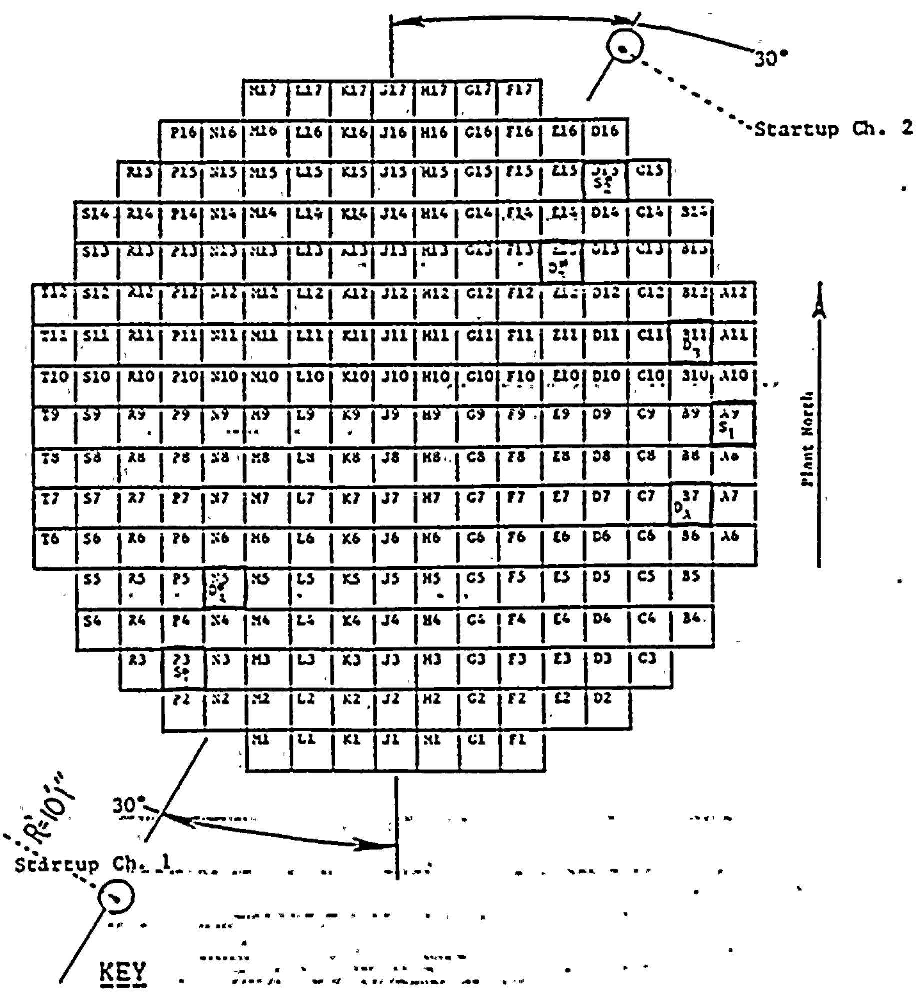
CORE LOADING SEQUENCE
PVNGS UNIT 3 INITIAL FUEL LOAD



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 3 INITIAL FUEL LOAD



KEY

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

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

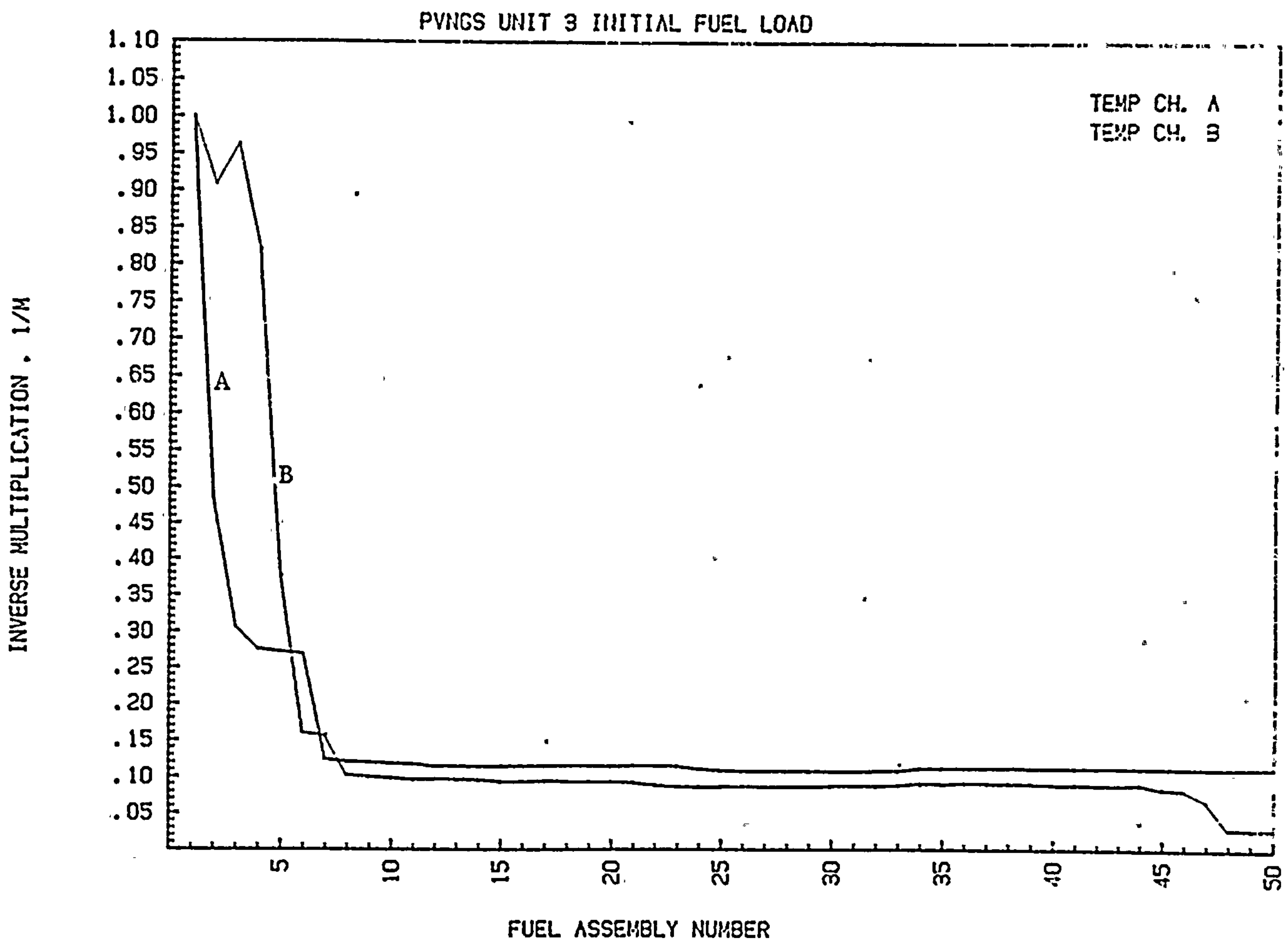
S_2*---Final location of neutron source 2 (D-15)

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 3 INITIAL FUEL LOAD
(First 50 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 72HF-3ZZ03, 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 September 28 through October 23, 1987. During this time, the plant was brought from operational mode 5 to operational mode 3. 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 72HF-3ZZ03 were successfully performed.

TEST DESCRIPTION

Testing commenced with reactor coolant system (RCS) at a temperature of approximately 200 ^{o}F and a pressure of 380 psia. From this condition, the RCS was heated up and pressurized to 565 ^{o}F, 2250 psia using station operating procedures. During the heatup/pressurization, conditions were stabilized at the direction of 72HF-3ZZ03 at six intermediate temperature/pressure plateaus to allow required testing to be performed. Table 3-1 lists the various temperature/pressure plateaus at which testing was performed. The surveillance requirements were verified as being satisfied prior to any changes in operational mode. The plant was then maintained at 565 ^{o}F, 2250 psia in accordance with the prerequisites of procedure 72IC-3RX02, "Initial Criticality."

TEST RESULTS

The plant was successfully heated and pressurized from cold shutdown to hot standby using normal plant operating procedures. The integrated operation of the primary, secondary and related auxiliary systems in accordance with CESSAR descriptions was verified by the heatup and pressurization. The implementing test procedures sequenced by this procedure were properly performed and completed. One of the sequenced procedures had acceptance criteria that were not met. In procedure 73HF-3RC09, "RCS Flow Measurements", the 4 pump coastdown did not meet the acceptance criteria. The resolution to this failure is detailed in section 3.3. The remaining individual test procedures acceptance criteria were properly met and documented in each procedure. Additionally, the carryover testing from the Preoperational Test Phase was satisfactorily completed.

CONCLUSIONS

Proper integrated operation of the PVNGS 3 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 (^{o}F)	RCS Press (psia)	Mode
10/1/87	0623	200	380	5
10/1/87	1004	250	380	4
10/1/87	2050	300	380	4
10/2/87	0209	345	380	4
10/4/87	1500	450	1100	3
10/4/87	2147	450	1650	3
10/5/87	0508	500	2250	3
10/5/87	1816	565	2250	3

3.2 Postcore Instrument Correlation (Section 14.2.12.3.2)

TEST OBJECTIVE AND SUMMARY

PVNGS procedure 73HF-3ZZ02, "Postcore Instrument Correlation," was performed over the period of October 1, 1987 to October 12, 1987 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 has been satisfactorily met.

TEST DESCRIPTION

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

300 °F/380 psia	450 °F/1650 psia
345 °F/380 psia	500 °F/2250 psia
450 °F/1100 psia	565 °F/2250 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, steam generator differential pressures, RCP speeds, and auxiliary feedwater 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 data recorded in this test was sufficient and within established criteria. The Test Exception Reports (TERs) generated were successfully retested. The TERs were generated primarily due to failed equipment rather than out-of-tolerance data. One TER, TER 05 (reactor coolant pump speed instruments RCN-ST-184, RCN-ST-185), was generated due to the fact that reactor coolant pump 2B was not running during the early plateaus. Proper operation of these speed instruments was verified during the last plateau.

At the conclusion of the test, 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 PMS, 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-3RC09, "Post-Core Reactor Coolant System Flow Measurements", was primarily conducted over the period of October 6 through October 18, 1987 with the reactor at hot standby conditions (565 ^{O}F,2250 psia). Additional testing (adjustment of CPC and COLSS flow algorithm constants) was conducted on October 28 and 29, 1987. 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.
(4) To update, as needed, the reactor coolant pump performance curves based on the ultrasonic flow measurement results.

The measured RCS flow rate (4-pump steady state operation) was 471,302 gpm and was within the acceptance criteria.

TEST DESCRIPTION

For ten various steady state reactor coolant pump (RCP) configurations and one RCP flow coastdown, 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 471,302 gpm, or 105.77% 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.

The four pump coastdown flow fractions were not conservative with respect to the safety analysis assumptions. The faster coastdown has been postulated to have been attributable to reduced pump efficiencies. Resolution to this problem resulted in the NSSS vendor imposing a penalty factor of approximately 2 percent on the CPC and COLSS to assure that the transients initiated (from 55 percent) up to 100 percent of rated thermal power remain within the safety analysis limits under the condition of faster than anticipated Reactor Coolant Pump coastdown. Specifically, the penalty was imposed to assure that the Loss of Flow (LOF) transient would be bounded by the existing 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 not conservative with respect to the safety analysis. Penalties were installed in the CPCs and COLSS to assure that the limiting event, the Loss of Flow transient, would be bounded by the safety analysis.

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 Zero Power conditions. This objective was met by 73HF-3SF11, "CEDM Coil Testing 565 $^{\circ}$F", conducted from October 7, 1987 through October 10, 1987 with the plant in Mode 3. The acceptance criteria were met by 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-3SF02, "Post-Core CEDM Performance", which was conducted from July 14, 1987 through September 22, 1987. 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-3SF08, "Post-Core CEA Drop Time Test". The test was conducted from October 6, 1987 through October 8, 1987 with the RCS at 565 $^{\circ}$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-3SF11: 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. Two minor problems were noted during the performance of the test: The rod bottom light associated with CEA #18 would not stay lit when CEA #18 was dropped. CEA #18 was subsequently rezeroed and retested successfully. The timing for CEA #87 was incorrect. A new timing board was installed, and CEA #87 was retested successfully.

73HF-3SF02: Each CEA was withdrawn individually to its upper limit, then inserted to its lower limit, then withdrawn six inches, 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-3SF08: 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 insertion 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-3SS01, "Postcore HFT Chemistry Test," was performed (1) to demonstrate that proper water chemistry for the Reactor Coolant System (RCS) and Steam Generators 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 RCS and Steam Generator chemistry control.

(2) The prescribed sampling frequencies were adequate for RCS and Steam Generator chemistry control.

(3) The analyses of water samples from the RCS and Steam Generators 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 September 30, 1987 and completed on October 6, 1987. The acceptance criteria were all satisfied.

TEST DESCRIPTION

Sampling and chemical analyses of the RCS and Steam Generators were performed using the appropriate plant operating procedures, as directed by 74HF-3SS01. Data was taken at the following test conditions:

Ambient conditions (<200 $^{\circ}$F)	450 $^{\circ}$F/1100 psia
250 $^{\circ}$F/380 psia	500 $^{\circ}$F/2250 psia
300 $^{\circ}$F/380 psia	565 $^{\circ}$F/2250 psia
345 $^{\circ}$F/380 psia	

At each of these plateaus, samples from the Reactor Coolant System and Steam Generators were taken, analyzed, and compared to the operating specifications provided in the test procedure.

TEST RESULTS

Ambient condition (<200 $^{\circ}$F) - 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.
250 $^{\circ}$F/380 psia plateau - Both the RCS and steam generators met the Mode 4 specifications at this test plateau.
300°F/380 psia plateau - The RCS and steam generators met the specifications for Mode 4 throughout this plateau.

345 ^{o}F/380 psia plateau - Both the RCS and the steam generators were within specification.
450 ^{o}F/1100 psia plateau - Both the RCS and the steam generators were within specification for this plateau.
500 ^{o}F/2250 psia plateau - Both the RCS and the steam generators were within specification for this plateau.
565 ^{o}F/2250 psia plateau - The RCS and steam generators remained in specification during this plateau.

CONCLUSIONS

The test objectives for 74HF-3SS01 were satisfied in that overall proper chemistry for the RCS and steam generators 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-3RC10, "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 at the pressurizer inlet would be no more than 70 ^{o}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 over the period from October 5 to October 8, 1987, with the RCS at approximately 565^{o}F, 2250 psia, and full flow conditions. The acceptance criteria were satisfied.

TEST DESCRIPTION

Permanent plant temperature instrumentation on the cold legs was used to determine the average cold leg temperature. Four strap-on thermocouples were located at various locations on the spray line check valve near the pressurizer. The spray line temperature differential was then calculated as the difference between the average cold leg temperature and the average of the four thermocouples.

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 in the as found throttled position. The temperature difference was initially determined to be 107.75^{o}F, outside the acceptance criteria. Therefore the continuous spray valves were adjusted. With the continuous spray valves opened to 9 1/4 turns, steady state conditions were re-established and data was retaken. A temperature difference of 67.82^{o}F was measured, within the acceptance criteria of less than 70^{o}F.

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, for information only.

TEST RESULTS

The continuous spray valves were determined to be properly set to obtain a temperature difference between the cold legs and the spray line near the pressurizer of 70^{o}F. The depressurization rate obtained using both main spray valves was measured to be 1.38 psia per second. Therefore, the acceptance criteria of 1.06 psia/sec for this test were satisfactorily met. Furthermore, the pressurization rate was measured to be 0.27 psia/sec using all pressurizer heaters.

CONCLUSIONS

The testing required by CESSAR was completed and the acceptance criteria were satisfied.

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 43ST-3RC02, "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 Mode 3. 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, Equipment 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 ten 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 October 11, 1987 performance, which measured identified and unidentified leakages of 0.120 gpm and 0.119 gpm, respectively. This particular measurement was conducted with the RCS at 565 ^{o}F and 2240 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-3RI01, "Post-Core Movable Incore Instrumentation Test", was performed over the period from September 22 to December 18, 1987. The objectives and acceptance criteria of this test were as follows:

(1) To determine the movable incore detector system (MICDS) path lengths using the detector drive system encoder. Repetitive measurements of each path were to agree with the average length for the given path within 0.3 inch.

(2) To install the permanent plant movable incore detectors (fission chambers).

(3) To demonstrate proper computer control of the movable incore detectors.

(4) 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.

(5) 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 MICDS path lengths were measured by driving the dummy cable into each path using the manual control box and recording the encoder readout from the control box. Three successive path length measurements were taken, then compared to the average length for each path.

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 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 computer operation of the MICDS was not completely successful. Prior to the test, the system hardware had been modified per lessons learned in PVNGS Units 1 and 2. Extensive troubleshooting in Unit 2 had proved that the hardware modifications and Honeywell plant computer upgrade (45000 model) were both needed. In addition, two more major problems were found in the computer hardware in Unit 2 (I/O cards, interrupt cards). These changes were incorporated into the PVNGS Unit 3 hardware prior to perfomance of this test. Software changes indentified in Unit 2 were also incorporated into Unit 3. Although the A drive could access and scan core locations under computer control, the B drive machine was unable to do so reliably. This prevented the collection of fully automatic scans of the full reactor core, although individual assemblies could be scanned using the A drive and detector operating in the semi-automatic mode. The B drive was affected by occasional path verification failures and discrepancies between detector position indication from the interrupt system and the number of scans collected by the I/O microprocessor during detector withdrawal from the fuel assemblies. The latter problem appears to be a problem of either missed interrupts or spurious actuation of the demand scanner which scans the detector output amplifier during fuel assembly scans. Troubleshooting on the system is on-going. Until the computer operation mode is successfully implemented, the MICDS may continue to be operated satisfactorily using the A drive only in the computer manual or semi-automatic mode or by using the Manual Control Box.

The Fixed Incore Detector leakage resistances were well above the minimum acceptable value of 10 megaohms, 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 fully operable, fuel assembly scans may be obtained under computer control using one drive and the manual control box may continue to be used until a fix of the computer mode is implemented.

(3) The path lengths of all movable incore paths were measured using the manual control box and the dummy 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 be impacted. The manual control boxes can be used in lieu of the computer but with knowledge that accuracy and speed will be degraded.

Although work is continuing to resolve the MICDS computer program problems, the test results met the required CESSAR acceptance criteria.

4.0 INITIAL CRITICALITY
(CESSAR Section 14.2.10.2)

TEST OBJECTIVE AND SUMMARY

Initial criticality for PVNGS 3 was achieved under test procedure 72IC-3RX02, "Initial Criticality." The purpose of the procedure was to provide a safe, organized method for attaining the initial criticality of the PVNGS Unit 3 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 at 0925 on October 25, 1987 and initial criticality was achieved at 1834 on October 25, 1987. The RCS boron concentration at criticality was measured at 1014.1 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 565^{o}F, 2250 psia, approximately 1350 ppm boron and All Rods Out (ARO) with the exception of a single group of four rods (Regulating Group 5), which was positioned at approximately 75 inches withdrawn (at midcore). The RCS boron concentration was then reduced to achieve criticality, with Group 5 used to control the chain reaction.

Core response 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 3 reactor was achieved in a safe and controlled manner as described above. The measured RCS boron concentration at criticality, 1014.1 ppm, fell within the acceptance criteria of 965 ppm to 1065 ppm and differed from the predicted value of 1015 ppm by only 0.9 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 redundancy of plant instrumentation in monitoring the reactor in the source and low power ranges.

5.0 <u>LOW POWER PHYSICS TESTS</u>
(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-3RX30, "Low Power Physics Test".

The Low Power Physics Test for PVNGS 3 utilized a CEA exchange technique. In this method, the critical boron concentration (CBC) and isothermal temperature coefficient (ITC) are first measured with essentially all CEAs withdrawn from the core (ARO). Next the most worthy CEA group, shutdown CEA group B for PVNGS, is inserted into the core while reactor power is held constant by RCS dilution. The reactivity worth of the CEA group is measured concurrently using the reactivity computer. Once this most worthy CEA group is fully inserted, conditions are stabilized and the CBC and ITC are again measured. This most worthy CEA group, also referred to as a reference group, is then exchanged with the next most worthy group, also referred to as the test group, while the RCS boron concentration is maintained constant. When the test group reaches its fully inserted position, conditions are stabilized. The reactivity worth of the test group is then determined from the critical CEA configuration which results. The reference CEA group is then inserted while the test CEA group is withdrawn to maintain the reactor critical. After the test CEA group is completely withdrawn, a new test group is selected and its reactivity worth is measured in the same manner. The process is repeated until all full-length CEA groups have been measured. Due to the widely varying worth of PVNGS CEA groups, two reference groups were used to assure accurate results.

During low power reactor physics testing, one minor problem with some the testing equipment was encountered. One of the picoammeters used as a part of the reactivity computer setup was found to display some non-linearity midway into low power physics testing. This picoammeter was replaced and any testing performed using the suspect picoammeter was redone.

TEST RESULTS

The measured group worths are shown in Table 5-3. All of the CEA group worth measurements agreed with the predicted values within the allowed tolerances, including the total regulating group worth. During the withdrawal of Group 3 and insertion of group 5 (step 9 above) CEA #84 slipped and CEA #82 dropped. Both CEAs were recovered and group 3 was returned to its initial condition. Testing was resumed with test results being unaffected.

CONCLUSIONS

The accuracy of the predicted CEA group worths was confirmed by the measured values. Furthermore, the total measured regulating group 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)	Acceptable Diff.
565 ^{o}F, 2250 psia					
	5	-0.2217	-0.2298	-0.0081	±0.100
	4	-0.445	-0.4577	-0.0127	±0.100
	3	-0.770	-0.731	+0.039	±0.077
	2	-0.539	-0.5586	-0.0196	±0.100
	1	-0.786	-0.7245	+0.0615	±0.121
	B	-2.565	-2.707	-0.142	±0.2565
	A	-2.5328	-2.526	+0.0068	±0.389
	Total	-7.8595	-7.9346	-0.0751	±0.794

5.1 Low Power Biological Shield Survey Test (Section 14.2.12.4.1)

TEST OBJECTIVE AND SUMMARY

PVNGS procedure 75PA-3ZZ01, "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 June 17, 1987 with the plant in Mode 5. Low power physics data was gathered on October 26, 1987 with the reactor critical and at 0% full power (FP) and the primary conditions of 565 ^{o}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 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, Radwaste Building, and at various site locations exterior to the plant.

TEST RESULTS

Baseline data was gathered on June 17, 1987 in the accessible areas. Comparison of this data to the acceptance criteria is not applicable. Low power physics data was gathered on October 26, 1987. 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 Isothermal Temperature Coefficient Test (Section 14.2.12.4.3)

TEST OBJECTIVE AND SUMMARY

The Isothermal Temperature Coefficient (ITC) was measured twice during Low Power Physics Testing at different control rod (CEA) configurations. The conditions under which the ITCs were measured and the dates of performance are listed below:

1) 565 ^{o}F, 2250 psia, unrodded------------October 26, 1987
   retested------------------------------------October 27, 1987
2) 565 ^{o}F, 2250 psia,
   CEA Gp. B inserted--------------------------October 26 and 27, 1987

The measured ITC values were required to be within $\pm$ 0.3 x 10^{-4} delta-K/K/^{o}F of their predicted values, a condition which was met during all the measurements. The moderator temperature coefficient (MTC) was determined from both of the ITCs measured, 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/^{o}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 ^{o}F at a rate of about 10-20 ^{o}F per hour, using the secondary Steam Bypass Control System. RCS temperature and core reactivity were recorded on a strip chart and, 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 X-Y plots for both temperature swings and 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 first ARO ITC measurement was affected by the suspect picoammeter noted in Section 5.0. The ARO ITC measurement was therefore repeated with a replacement picoammeter. 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 were -0.227 x 10^{-4} and -.683 x 10^{-4} at the unrodded (ARO) and rodded (Group B inserted) conditions, respectively. Both of these values are within the aforementioned Technical Specification limits.

CONCLUSIONS

The accuracy of the predicted isothermal temperature coefficients was verified by the measured values, all of which were 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)	Acceptable Diff.
565 ^{o}F, 2250 psia				
--- unrodded	-0.19	-0.381	-0.191	±0.30
--- Grp B inserted	-0.594	-0.837	-0.243	±0.30

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

TEST OBJECTIVE AND SUMMARY

The objectives of this test were as follows:

(1) To determine the individual group worths of all full-length Control Element Assembly (CEA) groups.

(2) 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 3 core. The full-length CEA group worths were measured with the RCS at hot zero power (565 ^{o}F, 2250 psia) over the period of October 26 through October 28, 1987. The measured individual group worths were required to be within ±10% of their predicted values for the reference CEA groups. For the non-reference CEA groups the measured individual group worths were required to be within ±15% or ±0.10% delta-K/K (whichever is larger). The total worth of the regulating CEA groups was required to be within ± 10% of its predicted value.

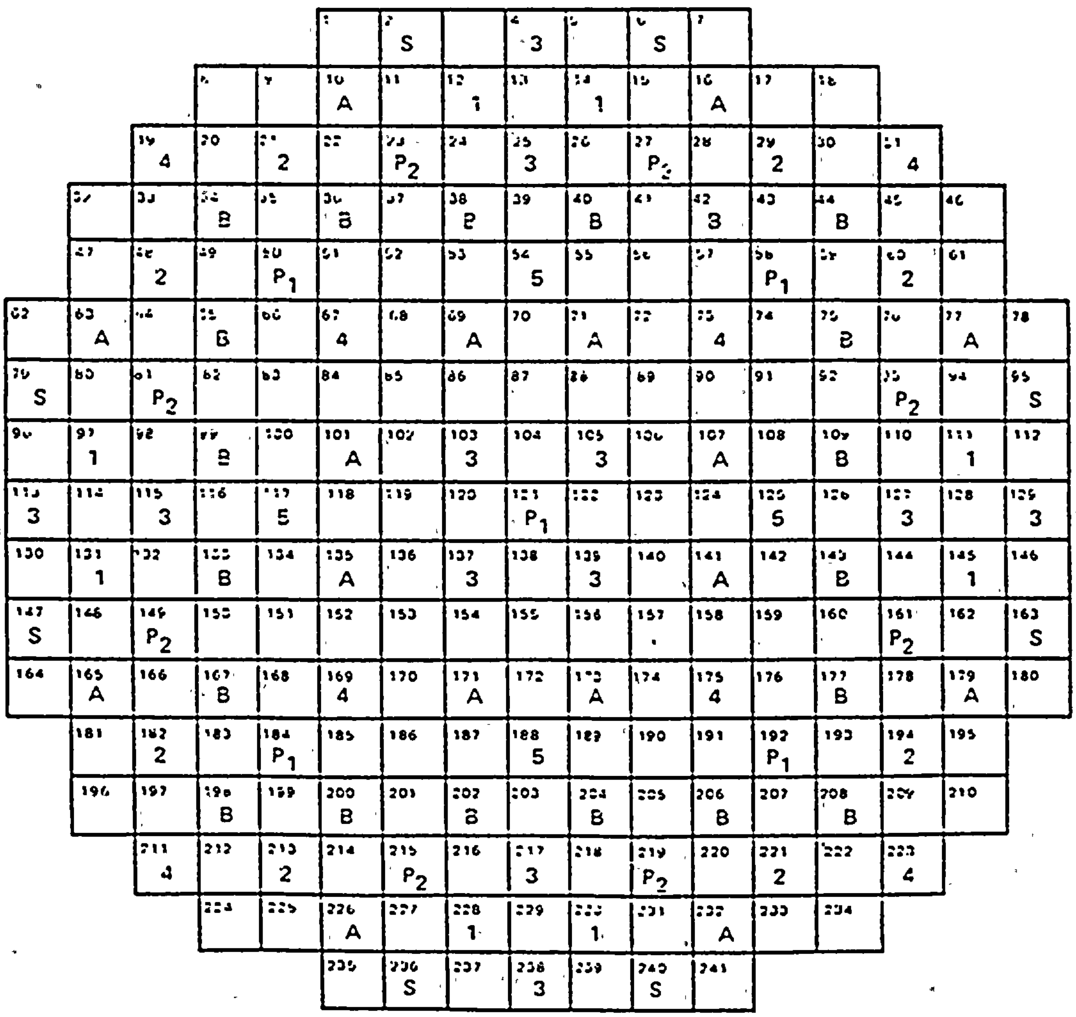
TEST DESCRIPTION

A constant dilution of the boron concentration was initiated. Insertion of CEA group B 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. The relevant system parameters of CEA position, time, reactivity, boron concentration and temperature were recorded. The withdrawal of group B was exchanged with the insertion of Group A until Group A was fully inserted and the reactor was stabilized with the reactivity at approximately zero. The relevant system parameters listed above are again recorded to determine the measured critical position. This process is repeated in essentially the same manner until all full-length CEA groups have been measured. The entire exchange sequence is listed below:

1. Measure shutdown reference bank (group B) by insertion/dilution.
2. Withdraw group B and insert group A until A is fully inserted.
3. Withdraw group A and insert group B until group A is fully withdrawn.
4. Withdraw group B and insert group 3 (regulating reference group) until group 3 is fully inserted.
5. Borate RCS and withdraw group B until group B is fully withdrawn.
6. Withdraw group 3 and insert group 1 until group 1 is fully withdrawn.
7. Withdraw group 1 and insert group 2 until group 2 is fully inserted, after which time insert group 3 until group 1 is fully withdrawn.
8. Withdraw group 2 and insert group 4 until group 4 is fully inserted, after which time insert group 3 until group 2 is fully withdrawn.
9. Withdraw group 3 and insert group 5 until group 5 is fully inserted.
10. Withdraw group 4 and insert group 3 until group 4 is fully withdrawn.
11. Withdraw group 5 and insert group 3 until group 5 is fully withdrawn.
12. Return group 3 to its original position prior to the start of step 6.
13. Measure the worth of group 3 by boration/withdrawal.

FIGURE 5-1

RELATIVE CORE LOCATIONS
OF CEA GROUPS



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 group 1
P_2—Part-length group 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. The measurement was performed on October 28, 1987 with the RCS at 565 ^{o}F, 2250 psia. The measured differential boron worth was required to be within ±10 ppm/%delta-K/K of its predicted worth.

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

The calculated differential boron worth agreed with the predicted value within ± 10 ppm/%delta-K/K. The comparison of the measured and predicted value is summarized in Table 5-4.

CONCLUSIONS

The differential boron worth met the acceptance criteria for the test.

TABLE 5-4

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

Conditions	Predicted	Measured	Diff. (M-P)	Acceptable Diff.
565 ^{o}F, 2250 psia	-83.43	-80.27	+3.16	± 10.0

5.6 Critical Boron Concentration Test (Section 14.2.12.4.6)

TEST OBJECTIVE AND SUMMARY

The critical boron concentration (CBC) was measured twice during Low Power Physics testing, at two different control rod (CEA) configurations. The conditions under which the measurements were performed and the dates of performance are listed below:

1) 565 °F, 2250 psia, unrodded——————October 26, 1987
   Retested——————————————October 27, 1987
2) 565 °F, 2250 psia,
   CEA Gp B inserted——————————October 26, 1987

The measured Critical Boron Concentrations were required to be within ± 50 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 ± 50 ppm. Table 5-5 provides a summary of the measured and predicted values. The unrodded CBC was one of the measurements affected by the suspect picoammeter. The value given in Table 5-5 is the value determined during the retest.

CONCLUSIONS

The accuracies of the predicted critical boron concentrations were confirmed by the measured values, all of which were well within their acceptance criteria.

TABLE 5-5

CRITICAL BORON CONCENTRATIONS
(ppm)

Conditions	Predicted	Measured	Diff. (M-P)	Acceptable Diff.
565 °F, 2250 psia				
---- unrodded	1025	1022	-3	±50
---- Grp B inserted	811	805	-6	±50

6.0 POWER ASCENSION TESTS
(CESSAR Section 14.2.12.5)

6.1 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 50% 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 on December 13, 1987 at 50% power using test procedure 72PA-3RX15 and on January 4 and 5, 1988 using test procedure 72PA-3RX50, "Variable T_{avg} (Isothermal Temperature Coefficient and Power Coefficient) Test" (50% and 100%, respectively).

The measured ITC and MTC were required to be within $\pm 0.3 \times 10^{-4}$ delta-K/K/$^{\circ}$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, with one exception noted below.

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 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 method was performed only at 100% where four cycles were performed.

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 method was performed only at 100% power where four cycles were performed.

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 50% power plateau, the MTC was extrapolated to 100% power to ensure compliance with Technical Specifications (T.S.). The 100% power plateau test was actually performed at approximately 96% power to avoid violating the licensed power limit during the course of this testing. The MTC was then extrapolated to 100% power to ensure T.S. compliance.

TEST RESULTS

The measured ITCs, MTCs and PCs agreed with the predicted values within the acceptable range at both power plateaus with the exception of the ITC and MTC at 100% power. The ITC and MTC at 100% power were determined to be acceptable following calculation of a revised test prediction by the NSSS vendor which took into account the difference in measured and predicted boron concentration, Tavg and core average burnup. These results are summarized in Table 6-1.

CONCLUSIONS

The agreement of the predicted isothermal temperature coefficient, moderator temperature coefficient, and power coefficient with the measured values was verified for the two 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.	(1) Meas.	Diff. (M-P)	Accept. Diff.
50% (12/13/87)	720	ITC	-0.601	-0.761	-0.160	±0.3
		MTC	-0.459	-0.619	-0.160	±0.3
		PC	-1.150	N/A	N/A	±0.2
100% (1/4/88-1/5/88)	620	ITC	-0.814	-1.152(2)	-0.338	±0.3
		MTC	-0.684	-1.021(2)	-0.337	±0.3
		PC	-0.930	-0.925	+0.005	±0.2
		ITC	-1.349(3)	-1.152	+0.197	±0.3
		MTC	-1.218(3)	-1.021	+0.197	±0.3

NOTES — Diff. = Difference = Measured - Predicted
— Accept. Diff. = Acceptable Difference Range
— (1) The measured values have been corrected to predicted boron concentrations of 790 at 50% and 699 at 100%
— (2) Out of tolerance results
— (3) Revised test predictions provided by NSSS vendor for "as tested" condition

6.2 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 both step and ramp load decreases can be performed at the design rates with key plant parameters remaining within design limits.

Testing was accomplished using procedure 73PA-3ZZ05 (50% test) on December 23, 1987 and procedure 73PA-3ZZ07 (100% test) on January 5 and 6, 1988. All control systems responded as designed and key plant parameters remained within their required acceptance band at both power plateaus.

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 (TLI) (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 Steam Bypass Control System (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 Pressurizer Level Control System (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

Both tests, at 50 and 100% power, successfully demonstrated that all control systems responded as designed and all monitored parameters remained within their acceptance band.

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.

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 both tests 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	2201	2280	2150	2335
Pzr. Level	36.3	44.7	30.0	45.0
SG 1 Pressure	1117	1148	1050	1180
SG 2 Pressure	1143	1175	1050	1180
SG 1 Level	70.0	74.0	70.0	88.0
SG 2 Level	70.0	73.3	70.0	88.0
RCS T-avg	574.4	579.4	567.0	587.0
SG 1 T-hot	584.1	593.6	575.0	600.0
SG 2 T-hot	583.9	593.2	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	2210	2300	2150	2335
Pzr Level	32	42	30.0	45.0
SG 1 Pressure	1110	1175	1050	1180
SG 2 Pressure	1110	1175	1050	1180
SG 1 Level	71.0	74.0	70.0	88.0
SG 2 Level	70.0	76.0	70.0	88.0
RCS T-avg	571.0	577.8	567.0	587.0
SG 1 T-hot	576.0	585.0	575.0	600.0
SG 2 T-hot	576.0	585.0	575.0	600.0

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	2228	2280	2150	2335
Pzr Level	51	55	40.0	60.0
SG 1 Pressure	1110	1125	1020	1180
SG 2 Pressure	1112	1125	1020	1180
SG 1 Level	71.5	74.0	70.0	88.0
SG 2 Level	72.0	74.0	70.0	88.0
RCS T-avg	592.0	594.0	580.0	600.0
SG 1 T-hot	613.0	617.0	600.0	625.0
SG 2 T hot	613.0	617.0	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	2220	2290	2150	2335
Pzr Level	47	53	40.0	60.0
SG 1 Pressure	1118	1140	1050	1180
SG 2 Pressure	1113	1138	1050	1180
SG 1 Level	72.0	74.0	70.0	88.0
SG 2 Level	72.0	74.0	70.0	88.0
RCS T-avg	587.0	592.0	580.0	600.0
SG 1 T-hot	608.0	612.0	600.0	625.0
SG 2 T-hot	608.0	612.0	600.0	625.0

6.3 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-3SF01, "Control Systems Checkout at 20% Power" (December 6 & 7, 1987)
73PA-3SF04, "Control Systems Checkout at 50% Power" (December 22 & 23, 1987)
73PA-3SF06, "Control Systems Checkout at 100% Power" (January 5, 1988)

TEST DESCRIPTION

73PA-3SF01, SF04 & 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 ±2°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.

Test Results

73PA-3SF01, 3SF04 & 3SF06 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. All acceptance criteria were met with the exception of two minor items noted below.

At all three plateaus, an offset was noted in S/G 2 level. A total of six test exception reports were written concerning this condition. Because this offset was small, (approximately 2% of the narrow range instrument span), system acceptability was declared.

At the 50% power plateau, the downcomer flow to S/G 2 exceeded 12% of total flow. At the 100% power plateau the downcomer flow to S/G 1 was less than 8%. Resolution to both test exception reports indicated that even though the downcomer flow rates were not as expected, the total flow rate was, and therefore system acceptability was declared.

CONCLUSIONS

These tests demonstrated that the control systems are capable of performing as designed under steady state, minor transient, or major transient conditions.

TABLE 6-6

DEVIATION FROM CONTROL SETPOINTS FOR
CONTROL SYSTEM CHECKOUT TESTS

Acceptance Criteria	73PA-3SF01	73PA-3SF04	73PA-3SF06
Feed Water Control:			
Transient S/G lvls. ±2% from setpoint: ramp change, S/G 1:	+1.5	+1.2	+1.2
S/G 2:	+3.4(1)	-2.7(1)	-3.2(1)
Transient S/G lvls. ±2% from setpoint, step change, S/G 1:	-0.5	+1.7	+0.9
S/G 2:	2.9(1)	-2.6(1)	-3.7(1)
Reactor Regulation:			
Tavg ±2°F from Tref.	0.72	1.7	+1.51
Integrated Checkout:			
Tavg ±2 °F of Tref:	1.97	-1.7	1.08
#1 S/G level ±2% from setpoint:	+1.7	+1.7	+1.04
#2 S/G level ±2% from setpoint:	-3.5(1)	-2.8(1)	-3.55(1)
Pzr. pressure ±15 psia from setpoint:	10.7	7.6	11.60
Pzr. level ±1% from setpoint:	-0.37	1.0	1.0
Steam bypass valves closed	true	true	true

(1) Out of tolerance results due to S/G level offset.

6.4 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-3SS01 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), also referred to as the letdown radiation monitor.

(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-3SS01 was initiated on October 28, 1987 at 0% power and was completed on January 5, 1988 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) was verified by comparing PRM readings to the laboratory analysis of grab samples which were representative of the fluid monitored by the monitor.

TEST RESULTS

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

Zero Percent Power RCS lithium[7] was low in the specification range at this plateau. The low lithium concentration was corrected by the addition of lithium hydroxide. Steam Generators and makeup water supplies were all within specification for this plant condition.

3% to 5% Power The RCS, steam generators and makeup water supplies were all within specification for this plateau.

<u>10% Rated Power</u> The primary systems had two items out of specification at this plateau. These were RCS hydrogen which was out of specification low and lithium which was out of specification high. These conditions were corrected by increasing the volume control tank hydrogen overpressure and by placing the delithiating deminalizer bed in service. On the secondary side, intermittant out of specification conditions were encountered with steam generator cation conductivity and hotwell and condensate demineralizer influent dissolved oxygen. The steam generator conductivity problem was corrected by increasing blowdown flowrates and the dissolved oxygen problem was corrected by increasing the rate of hydrazine addition to the secondary system..

<u>20% Rated Power</u> The primary system remained in specification throughout this plateau. Intermittant problems in the steam generators and feedwater system with cation conductivity, chlorides, sulfates, and total iron were encountered. Additionally, the hotwells experienced some problems with high sodium concentrations. The steam generator chemistry was partially corrected by increasing blowdown flows, but both the steam generator and hotwell chemistry problems were not completely cleared up until later plateaus. This was in accordance with previous experience during the startup of PVNGS 1 and 2, where several weeks of operation were neccessary to clean up the secondary piping systems. The feedwater out of specification condition was attributed to the leaking by of the condensate demineralizer bypass valve. The valve was reseated and the feedwater problem was corrected. The makeup water sources remained in specification during the entire plateau.

<u>50% Rated Power</u> The primary system had only one item out of specification for a short period of time at this plateau. This was RCS lithium which went out of specification low for a few hours on December 20, 1987. This condition was attributed to the heavy RCS dilution which took place the previous day to support a reactor startup following a reactor trip. A lithium addition corrected the condition promptly. On the secondary side, intermittant problems with high, out of specification cation conductivity, chlorides and sulfates in the steam generators were encountered. The steam generators were blown down continuously with increased flow rates at times to reduce cation conductivity, sodium, chlorides or sulfates as needed. Some problems were encountered with high feedwater cation conductivity and hydrazine and hotwell sodium. The cation conductivity problem cleaned up as secondary system piping cleaned up through operation. The hydrazine problem was corrected by reducing the stroke on the secondary chemical addition pump. The hotwell sodium problems were generally correlated with a reactor trip during the plateau and the performance of the steam bypass valve capacity test (see Section 6.12). Makeup water sources remained within specification for the entire plateau.

<u>80% Rated Power</u> The primary system remained in specification throughout the plateau. The steam generators were in continuous blowdown with increased flow rate to return sodium and sulfates to specification at times. The only significant problem occurred following a turbine trip when hotwell sodium levels rose. The problem corrected itself when the plant was brought back online which increased the flow through the hotwells and condensate demineralizers. Makeup water sources remained within specification for the entire plateau.

100% Rated Power The primary system and steam generators remained in specification throughout the plateau. A high hotwell sodium problem was encountered for a short period of time when steam bypass valves were opened for valve timing checks. The condition corrected itself quickly after the valves were closed.
Radiation Monitors The PRM was operable during this test. The acceptance criteria for agreement between the PRM and gross gamma activity in a grab sample was not met at any plateau. This acceptance is that the PRM shall agree with grab sample activity to within ±16.4 percent. The PRM generally gave readings which were 200-300% above the grab samples, but trended with grab sample results.

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. The high antimony levels experienced in PVNGS Units 1 and 2 were not experienced in Unit 3. This was expected since the antimony containing reactor coolant pump journal bearings were replaced with antimony-free bearings and any remaining antimony deposited during pre-core hot functional testing was cleaned up prior to the start of post-core hot functional testing.

The PRM trended with RCS gross activity determined from grab samples, but did not meet its quantitative acceptance criteria. Subsequent engineering evaluation determined that this acceptance criteria was inappropriate, and that the PRM should be used for activity trending only. The PRM therefore performed as designed and the results were accepted as-is, with no further testing planned.

6.5 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, and (2) without opening any primary or secondary safety valves. Additional objectives of the test were:

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)
- Reactor Regulating System (RRS)

Testing was scheduled to be performed at the end of the 100% power plateau in accordance with procedure 73PA-3MA01, "Unit Load Rejection Test". On December 27, 1987, however, before the scheduled performance of 73PA-3MA01, PVNGS Unit 3 experienced an unplanned turbine trip while operating at approximately 93% of rated thermal power. Reactor power cutback and and SBCS quick open signals were received. No RPS or ESFAS signals were received, no primary or secondary safety valves lifted and reactor power was stablized at approximately 30%. The responses of the plant equipment during this unanticipated event were reviewed by the PVNGS Test Results Review Group 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-3MA01 and the actual conditions at the time of the December 27, 1987 event are listed in table 6-7. Both the planned test and the unplanned turbine trip event allow verification that SBCS, FWCS, RRS, PPCS, PLCS, CEDMCS and RPCS perform their designed functions.

In both the planned test and the unplanned event the load rejection was accomplished by initiating the turbine trip circuitry. This causes all turbine stop and control to rapidly close and opens the main generator output breakers.

As a result of the large reduction in steam flow, the SBCS will signal the RPCS to actuate on "Large Load Reject" (LLR) and open all available steam bypass valves in quick open (Q.O.) mode. The RPCS should drop CEA Group 5 into the reactor core to provide immediate power reduction. Following Q.O. activity the steam bypass valves should transfer to control on steam generator pressure in modulation and the RRS will attempt to match primary to secondary power by inserting regulating groups to lower RCS average temperature until the Automatic Motion Inhibit (AMI) is achieved. The other NSSS control systems will function to maintain the plant within the programmed system operating parameters appropriate for the final power level achieved.

TEST RESULTS

Following the loss of the turbine-generator, the reactor power cutback system in conjunction with the other control systems performed as designed. The reactor did not trip, no ESFAS signals were generated and no primary or secondary safety valves lifted. The single value acceptance criteria of test procedure 73PA-3MA01 were satisfied with the exception of S/G 1 and 2 pressures. These acceptance criteria apply to the first sixty seconds of the transient and are listed in Table 6-8 with the measured results from the turbine trip. The out-of-tolerance results on S/G pressures were determined to be due to misadjustment of the stroke times of the steam bypass valves. The stroke times were subsequently adjusted to provide the proper stroke times.

CONCLUSIONS

The turbine trip event demonstrated that the NSSS can sustain a 100% load rejection without a reactor trip, ESFAS actuation, or lifting of the primary or secondary safety valves. The control systems operated satisfactorily throughout the transient and data was collected to verify the single value acceptance criteria. The incorrectly set steam bypass valve stroke times were corrected.

The PVNGS Test Results Review Group reviewed the responses of plant equipment and personnel and detemined that the event satisfied the objective of the power ascension test, thereby obviating the need to perform 73PA-3MA01. No further testing is planned.

TABLE 6-7

UNIT LOAD REJECTION
INITIAL CONDITIONS

Parameter	Actual Value	Test Value
Reactor Power	93%, increasing	>95%, stable
RCS Average Temperature °F	590	592.5-593.5
Pressurizer Pressure, psia	2250	2234-2265
Steam Generator Pressure, psia	1080	1064-1102
Steam Generator Level, %	54	51.6-53.6
CEA position	ARO*	Above PDIL[1]
Control System Status		
-Reactor Regulating System	Auto	Auto
-Control Element Drive	Standby[2]	Auto
-Feedwater Control System	Auto	Auto
-Steam Bypass Control System	Auto	Auto
-Pressurizer Level Control	Auto	Auto
-Pressurizer Pressure Control	Auto	Auto
-Reactor Power Cutback System	Auto	Auto

* — ARO = All Rods Out = All control rods fully withdrawn

1 — PDIL = Power Dependent Insertion Limit

2 Changed to Auto fifteen seconds after turbine trip by operator

TABLE 6-8

SINGLE VALUE ACCEPTANCE CRITERIA
FOR 100% UNIT LOAD REJECTION

Parameter	Test Results	Acceptance Limit
Max. Pressurizer Pressure (psia)	2375	2388
Min. Pressurizer Level (%)	31	29.4
Min. RCS Hot Leg #1 Temp. (^{o}F)	584	574
Min. RCS Hot Leg #2 Temp. (^{o}F)	584	574
Max. SG #1 pressure (psia)	1247*	1242
Max. SG #2 pressure (psia)	1248*	1242

*Out of tolerance results due to misadjustment of SBCS valve stroke times

6.6 Biological Shield Survey Test (Section 14.2.12.5.10)

TEST OBJECTIVES AND SUMMARY

PVNGS procedure 75PA-3ZZ01, "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 6-8 shows the applicable criteria and defines the access zones.

All survey points met their applicable acceptance criteria.

TEST DESCRIPTION

With the plant stabilized at the desired conditions, gamma and neutron radiation surveys were performed at 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, Radwaste Building, and at various site locations exterior to the plant.

TEST RESULTS

Baseline data was obtained on June 17, 1987 while the plant was in Mode 5 and again on October 26 during Low Power Physics Testing. These values were then used for comparison to the data taken during Power Ascension.

On November 24, 1987 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 December 2, 1987 a survey was completed while the plant was operating at 20% FP. All values were within the acceptable range for their designated zone.

On December 10, 1987 data was obtained while the plant was operating at approximately 50% FP and at 80% on December 28, 1987. Again all survey points met their acceptance criteria.

Data collection for the survey at 100% FP was performed on January 2, 1988. All survey points were within the acceptance criteria. In addition the high radiation areas found in the Auxilliary Building of Unit 1 were not present in Unit 3, however, it is expected that these areas will exist when the shutdown cooling systems are used.

CONCLUSIONS

All the data taken inside the Containment Building have met their acceptance criteria indicating the effectiveness of the Biological Shield. In addition, all other data from survey points in surrounding buildings were within their expected limits.

TABLE 6-8

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)

6.7 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 four times during power ascension testing at various power levels. These measurements were compared to predictions to confirm assumptions in the safety analysis and to verify expected core behavior. Measurements were performed at the 20%, 50%, 80%, and 100% full power (FP) levels. The conditions of the power distribution measurements and the dates of performance are listed in Table 6-9. Additionally, the 20% test was used to verify proper fuel loading.

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 3% and if the measured peaking factors were within ±7.5% of their predicted values with the exception at the 20% power plateau where the values were 5% and 10% respectively.

At the 20% power plateau the following additional acceptance criteria were applied to verify proper fuel loading:

1. The difference between the measured relative power in each assembly and the predicted power is less than ±10%.
2. The difference between the measured power in each assembly of an instrumented symmetric group and the average power in the same symmetric assemblies is less than ±10%.
3. The difference between the measured power in each quadrant and the average of the quadrant powers is less than ±10%.

The acceptance criteria were 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 6-10. 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 was loaded correctly and is behaving as expected.

TABLE 6-9

STEADY STATE CORE PERFORMANCE
TEST CONDITIONS

	20% FP	50% FP	80% FP	100% FP
Performance Dates	12-3-87	12-10-87	12-27-87	1- 2-87
Actual Reactor Power	18.6%	51.1%	79.6%	99.2%
RCS Tcold ($^{\circ}F$)	562.9	563.7	562.5	564.6
Primary Pressure	2248 psia	2253 psia	2252 psia	2261 psia
Boron Concentration	850 ppm	739 ppm	660 ppm	630 ppm
CEA Position	Unrodded	Unrodded	Unrodded	Unrodded
Core Average Burnup	53 MWD/T (1 EFPD)	106 MWD/T (3 EFPD)	422 MWD/T (11 EFPD)	555 MWD/T (14 EFPD)
Axial Shape Index (ASI)*	-0.005	+0.0215	+0.0597	+0.0701

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

TABLE 6-10
(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	1.38	1.46	-5.5	±10%
F_r Core Intgrtd Radial Peaking Factor	1.36	1.43	-4.9	±10%
F_z Core Axial Peaking Factor	1.28	1.27	-0.1	±10%
F_q Core 3-D Peaking Factor	1.76	1.83	-3.8	±10%

50% FP TEST

Peaking Factor	Measured Value	Predicted Value	% Diff	Accept Criteria
F_{xy} Core Planar Radial Peaking Factor	1.34	1.38	-2.9	±7.5%
F_r Core Intgrtd Radial Peaking Factor	1.33	1.35	-1.5	±7.5%
F_z Core Axial Peaking Factor	1.26	1.25	+0.1	±7.5%
F_q Core 3-D Peaking Factor	1.71	1.68	+1.8	±7.5%

80% FP TEST

Peaking Factor	Measured Value	Predicted Value	% Diff	Accept Criteria
F_{xy} Core Planar Radial Peaking Factor	1.35	1.36	-0.7	±7.5%
F_r Core Intgrtd Radial Peaking Factor	1.31	1.34	-2.2	±7.5%
F_z Core Axial Peaking Factor	1.29	1.28	-0.8	±7.5%
F_q Core 3-D Peaking Factor	1.72	1.71	-0.6	±7.5%

TABLE 6-10
(Part 2 of 2)

STEADY STATE CORE PERFORMANCE
TEST RESULTS

100% FP TEST

Peaking Factor	Measured Value	Predicted Value	% Diff	Accept Criteria
F_{xy} Core Planar Radial Peaking Factor	1.35	1.35	0.0	±7.5%
F_r Core Intgrtd Radial Peaking Factor	1.31	1.34	-2.2	±7.5%
F_z Core Axial Peaking Factor	1.29	1.28	+0.8	±7.5%
F_q Core 3-D Peaking Factor	1.72	1.72	0.0	±7.5%

RMS DIFFERENCES

	(1) 20%	50%	80%	100%	Acceptance Criteria
Radial Dist.	2.87%	1.92%	1.53%	1.68%	<3%
Axial Dist.	3.40%	2.83%	2.95%	2.69%	<3%

(1) — Acceptance criteria at 20% was ≤ 5%.

6.8 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 procedure 72PA-3SB01 at all testing plateaus. Testing was performed on December 3, 1987 (20% power); December 10, 1987 (50% power); December 27 (80% power) and on January 1, 1988 (100% power). All instruments were within their required acceptance band at all plateaus with the exception of RCS cold leg temeperature instrument RCD TT-112CD at the 20% plateau. A work order was written to have this instrument repaired, and the instrument was subsequently determined to be within its respective tolerance band at the 50, 80 and 100% power plateaus.

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

All of the instruments at each of the four major power plateaus were found to be within the specified tolerance bands with the exception of RCS temperature instrument RCD TT-112CD at the 20% power plateau. A Test Exception Report was written to document this instrument which did not meet its acceptance criteria. The instrument was then repaired and successfully retested at the next power level.

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.

6.9 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.

6.9.1 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 initially performed on December 17, and a retest was performed on December 21, 1987. In both cases the reactor was at approximately 50% full power, and test procedure 72PA-3RX18, "CEA Shadowing Factor/Radial Peaking Factor Test" was used.

TEST DESCRIPTION

The test was initiated with the reactor 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
* Group P 75% inserted with group 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 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 (with part lengths inserted, use only middle detector response, all other cases, sum top, middle and bottom

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 and new shadowing factor multipliers (ASM_i) were calculated for input to the CPCs using the following equation.

$$ASM_i = \frac{F_x(\text{calculated})}{F_x(\text{CPC Data Base})}$$

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 four measured configurations (one unrodded case plus three 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 to increase the values used in the CPC power distribution calculation. For CEA configurations not measured, a determination was made if adjustment was needed based on the test results from the unrodded configuration.

TEST RESULTS

This test was initially attempted on December 17, 1987. During the insertion of the part-length CEA group a malfunction in the control element drive mechanism control system resulted in an inadvertant reactor trip. The test was was repeated in its entirety on December 21, 1987.

The CEA Shadowing Factors for all rod configurations were determined and are summarized in Table 6-11. New shadowing factor multipliers (ASM) were calculated for each condition and input to the CPCs.

The measured planar radial peaking factors were less than the predicted planar peaking factors for all CEA configurations. No adjustments were required and the test results are summarized in Table 6-12.

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 6-11

MEASURED CEA SHADOWING FACTORS (CSF)						
			CSFs Computed from Test Data			
CEA Group/Position	Predicted CSF	Expected Range	CPC A	CPC B	CPC C	CPC D
5 / Full in	1.104	1.071 to 1.137	1.1042	1.1045	1.1032	1.1027
5 / Full in P / 75% in	1.125	1.091 to 1.159	1.0916	1.0873	1.0870	1.0905
P / 75% in	1.011	0.981 to 1.041	0.9807	0.9782	0.9791	0.9817

TABLE 6-12

MEASURED PLANAR RADIAL PEAKING FACTORS (ALL CPC CHANNELS)		
CEA Group/Position	Predicted Peaking Factor	Measured Peaking Factor
(Unrodded)	1.39	1.3550
5 / Full in	1.54	1.4938
5 / Full in P / 75% in	1.54	1.5138
P / 75% in	1.39	1.3681

6.9.2 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 safety channel detector responses associated with variation of the cold leg (reactor inlet) temperature. This test was performed at approximately 50% power on December 14, 1987 using test procedure 72PA-3RX22, "Temperature Decalibration Test (50%)".

TEST DESCRIPTION

The temperature shadowing factors are used by the CPCs to compensate the calculated neutron flux power for changes in the reactor inlet temperature (Tcold), since 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^{o}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, with boration/dilution used as neccessary to maintain reactor power constant. 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 were 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,

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

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 ^{o}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 and is listed below.

CPC Channel ==>	A	B	C	D
Measured Temp. Shadowing Factor ==>	0.0061	0.0063	0.0063	0.0064

Acceptance Criteria is measured temeperature shadowing factor is between 0.0048 and 0.0068.

CONCLUSIONS

The temperature shadowing factor measured for each CPC channel was in acceptable agreement with the predicted value.

6.9.3 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 December 4, 1987 using test procedure 72PA-3RX04, "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 set of SAM and BPPCCs for each CPC channel. Testing was accomplished on December 21 through 23, 1987 using test procedure 72PA-3RX19, "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 Departure from Nucleate Boiling (DNBR) and Linear Power Density (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 except channel D. A test exception report was written and subsequent investigation revealed that one of the channel D excore subchannels was reading differently than the other subchannels. The subchannel amplifier was recalibrated and the channel D results were determined to be acceptable and no further testing was performed. Therefore, no change to the CPC data base was required. The results are summarized in Table 6-13.

50% FP Test--The measured SAM and BPPCC values differed from the previously installed values by an amount that exceeded the acceptance criteria of 2.0% and 3.0% respectively, thereby requiring installation of the measured values. 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 6-14 and 6-15.

CONCLUSIONS

The SAM/BPPCC constants initially installed in the CPCs were satisfactory for operation up to the 50% power test plateau. New constants were determined and installed at 50% power.

TABLE 6-13

RMS VALUES 20% SAM TEST
(%)

CPC Channel	RMS Value	Acceptance Criteria
A	3.018	5.0
B	2.630	5.0
C	4.141	5.0
D	5.025(1)	5.0

TABLE 6-14

BOUNDARY POINT POWER CORRELATION CONSTANTS
(ALL CPC CHANNELS)

Parameter	Original Installed Value	Measured Value	%Diff.*
BPPCC1	0.01389	0.01399	+0.720
BPPCC2	0.08113	0.07986	-1.570
BPPCC3	0.01433	0.01423	-0.698
BPPCC4	0.08141	0.07123	-12.500

(1) Failed due to out of calibration excore subchannel amplifier.

* Where,

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

for i = 1 to 4

TABLE 6-15

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	3.4995	3.6673	4.79	3.4944	3.5931	2.82
S12	-0.3993	-0.6678	-69.79	-0.4052	-0.6235	53.87
S13	-0.2116	-0.0296	-86.01	-0.2029	-0.0182	-91.03
S21	-0.7857	-0.7101	-9.62	-0.6600	-0.6014	-8.88
S22	4.5093	4.4113	-2.17	4.2663	4.2602	-0.15
S23	-0.8257	-0.7742	-6.24	-0.6252	-0.6770	8.29
S31	0.2861	0.0428	-85.04	0.1656	-0.0083	-94.99
S32	-1.1100	-0.7435	-33.02	-0.8612	-0.6367	-26.07
S33	4.0371	3.8038	-5.78	3.8281	3.6953	-3.47
Test Matrix Value	3.7366			3.6536		

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	3.5430	3.6166	2.08	3.6220	3.5975	-0.68
S12	-0.3814	-0.6130	60.72	-0.5224	-0.5963	14.15
S13	-0.2287	-0.0461	-79.84	-0.1571	-0.0520	-66.90
S21	-0.7587	-0.6093	-19.69	-0.8612	-0.7458	-13.40
S22	4.4281	4.2274	-4.53	4.6055	4.4865	-2.58
S23	-0.8265	-0.6258	-24.28	-0.8896	-0.8222	-7.58
S31	0.2157	-0.0073	-103.40	0.2392	0.1483	-38.00
S32	-1.0467	-0.6144	-41.30	-1.0832	-0.8903	-17.81
S33	4.0552	3.6719	-9.45	4.0467	3.8743	-4.26
Test Matrix Value	3.6438			3.8055		

Where,

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

for i and j = 1 to 3

6.10 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-3FW01, "FWCS Test at 10% Power" evaluated the performance of the FWCS at a power level of 10% FP and was performed on November 28 through December 1, 1987.
2) 73PA-3FW02, "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 December 1, 1987.
3) 73PA-3FW03, "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 December 1, 1986.
4) 73PA-3FW04, 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-3ZZ03, "BOP Piping Dynamic Transient Test ", was performed. It consisted of two sections concerning the feedwater system:

1) Section 8.5 monitored the feedwater transfer from the downcomer to economizer during power increases and was performed on December 1, 1987.
2) Section 8.6 monitored the feedwater transfer from the economizer to downcomer during power decreases and was performed on December 7, 1987.

TEST DESCRIPTION

73PA-3FW01—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-3FW02--This test evaluated the response to an automatic valve transfer from the economizer to downcomer feed systems at approximately 15% power for adherence to acceptance criteria. The decreasing power transient was accomplished by leading the steam demand of the secondary plant and allowing the NSSS to follow. Data was taken during and after the transfer and was evaluated against specific acceptance criteria.
73PA-3FW03--This test was performed in essentially the same manner as 73PA-2FW02 described above, except that the transfer was from the downcomer to the economizer with reactor power increasing. Again, an automatic transfer was made and similar data were collected for comparison with acceptance criteria.
73PA-3ZZ03--To measure any loads that may have been imposed on the piping systems and restraints, twelve 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 is performed after the transient portions of the test.
73PA-3FW04--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.

TEST RESULTS

73PA-3FW01--The acceptance criteria for the test were satisfied. The actual test results are compared with the acceptance criteria in Table 6-16.
73PA-3FW02--As can be seen in the test results listed in Table 6-17, steam generator pressure exceeded its acceptable limits following the automatic valve transfer. A Test Exception Report was generated and the results were declared acceptable based on information provided by the NSSS vendor and the minimal consequences of steam generator pressure oscillations during a feedwater control valve transfer.
73PA-3FW03--Again, as can be seen in the test results listed in Table 6-18, the steam generator pressures exceeded their respective acceptance criteria. Again a Test Exception Report was generated and the results were declared acceptable based on information supplied by the NSSS vendor.
73PA-3FW04--The feed pumps delivered water to the steam generators at the required flows and temperatures through all power levels. A review of all data indicated the feedwater system to be performing as required.
73PA-3ZZ03--The observed loads were well below the maximum acceptable loads (which varied from 4,300 to 55,000 lbf) calculated for the specified load pins. The loads observed during the control valve transfer with power increasing varied from 7332 lbs to less than 100 lbs. During the control valve transfer with power decreasing all observed loads were at or below 100 lbf. Thus, the acceptance criteria were satisfied. Due to the low loadings observed, the visual inspection has not yet been performed. The visual inpection results will be addressed in a future supplement to this report.

CONCLUSIONS

The ability to perform the downcomer-economizer valve transfer automatically 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 6-16

FWCS CHECKOUT AT 10% POWER (73PA-3FW01)
TEST RESULTS

Acceptance Criteria	Maximum Deviation
S/G levels do not deviate more than ± .5% from set pt. for steady state conditions	3.10
S/G levels do not deviate more than ± 5% from set pt. for ramp set pt. changes	3.30
S/G levels do not deviate more than ± 5% from set pt. for step set point changes	2.80

TABLE 6-17

FWCS VALVE TRANSFER--POWER DECREASE
(73PA-3FW02) TEST RESULTS

ACCEPTANCE CRITERIA (deviation from initial value)	Actual Value
#1 Nuc power decreased no more than 4%	-3.0
#2 Nuc power decreased no more than 4%	-2.4
S/G #1 level increased less than 30%	9.64
S/G #2 level increased less than 30%	12.51
S/G #1 pressure decreased less than 50 psia	-70.4*
S/G #2 pressure decreased less than 50 psia	-65.8*
RC cold leg temps. increased no more than 6 degrees	
Tc 1A:	1.35
Tc 1B:	1.3
Tc 2A:	2.7
Tc 2B:	1.3

* Out of tolerance results

TABLE 6-18

FWCS VALVE TRANSFER—POWER INCREASE (73PA-3FW03) TEST RESULTS	
ACCEPTANCE CRITERIA (deviation from initial value)	Actual Value
#1 Nuc power increased less than 5%	2.08
#2 Nuc power increased less than 5%	2.08
S/G #1 level decreased less than 40%	-25.20
S/G #2 level decreased less than 40%	-27.08
S/G #1 pressure increased no more than 50 psia	50.3*
S/G #2 pressure increased no more than 50 psia	57.30*
RC cold leg temps do not decrease more than 8 degrees	
Tc 1A:	-6.51
Tc 1B:	-6.25
Tc 2A:	-5.00
Tc 2B:	-5.86

*Out of tolerance results

6.11 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 procedure 72PA-3SB02 at all power plateaus. Testing was performed on October 29, 1987 at 0% power; December 11, 1987 at 50% power and on January 1, 1988 at 100% power. 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 (at 100% power only) 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 (100% power only) is transmitted to the NSSS vendor for evaluation.

TEST RESULTS

The CPC calculated minimum/maximum DNBR and LPD values recorded during the test for the 0%, 50% and 100% power plateaus are provided in Table 6-19 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, thus all acceptance criteria were met. The FM taping portion of this test has not yet been performed. This will be addressed in a future supplement to this report.

CONCLUSION

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

TABLE 6-19
(Part 1 of 2)

CPC/CEDIPS COMPARISON VS POWER LEVEL

PARAMETER		CHANNEL A		CHANNEL B	
0% FP TEST		CEDIPS	CPC	CEDIPS	CPC
LPD	MAX	5.27758	5.2637	5.3727	5.2637
	MIN	5.2519	5.2637	5.2495	5.2631
DNBR	MAX	6.43415	6.4048	6.6790	6.4012
	MIN	6.27629	6.4025	6.0034	6.3327
50% FP TEST					
LPD	MAX	8.54162	8.3590	8.46524	8.4400
	MIN	8.25375	8.3399	8.35239	8.4199
DNBR	MAX	3.95888	3.9011	3.88906	3.8459
	MIN	3.73922	3.7899	3.76921	3.7999
100% FP TEST					
LPD	MAX	14.3209	13.923	14.0567	13.755
	MIN	13.6324	13.744	13.6813	13.713
DNBR	MAX	1.9563	1.9270	1.9245	1.8980
	MIN	1.7394	1.8953	1.7785	1.8726

TABLE 6-19
(Part 2 of 2)

CPC/CEDIPS COMPARISON VS POWER LEVEL

PARAMETER		CHANNEL C		CHANNEL D	
0% FP TEST		CEDIPS	CPC	CEDIPS	CPC
LPD	MAX	5.2751	5.2646	6.58879	5.7498
	MIN	5.2535	5.2643	5.25223	5.7300
DNBR	MAX	6.4203	6.4000	6.43652	6.4142
	MIN	6.2666	6.3352	5.75013	6.3435
50% FP TEST					
LPD	MAX	8.73318	8.2179	8.51163	8.3069
	MIN	8.09486	8.1824	8.23536	8.2871
DNBR	MAX	3.97976	3.9064	3.92132	3.8773
	MIN	3.80292	3.8559	3.71704	3.7954
100% FP TEST					
LPD	MAX	14.0569	13.833	14.833	14.059
	MIN	13.6795	13.753	13.626	13.728
DNBR	MAX	1.9142	1.8961	1.9307	1.8715
	MIN	1.7780	1.8631	1.7521	1.7988

6.12 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-3SG01 was conducted on December 23, 1987 with the reactor initially stabilized at 35% power. The principal objectives of the test were as follows:

(1) To verify that the capacity of each of the Atmospheric Dump Valves (ADVs) upstream of the Main Steam Isolation Valves (MSIVs) is greater than 6% and less than 11% of the total main steam flow to be encountered at full power conditions.

(2) To verify that the total capacity of the Steam Bypass Control System (SBCS) controlled turbine bypass 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 10.00% and the maximum measured valve capacity was 10.90%. The total capacity of the 8 SBCS valves was 85.38% 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.

6.13 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 procedure 72PA-3RI01 at all plateaus. Testing was performed on December 2, 1987 at 20% power; December 9, 1987 at 50% power; December 26, 1987 at 80% power and January 1, 1988 at 100% power. 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.

The operability of the FICDS was verified at each test plateau and all test objectives were satisfied.

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 for information only.

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.

TEST RESULTS

Fixed Incore Detector Test--The amplifier zero output test was satisfactorily completed at each test plateau. The amplifier full output test was satisfactorily completed at each plateau with the exception of the 20% power plateau, where eight detectors spread over three detector strings failed. A Test Exception Report was written to document this and a work request was written to troubleshoot. This troubleshooting traced the problem to a bad input card in the plant monitoring system. The bad card was replaced and testing at subsequent plateaus was completed successfully.

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.

CONCLUSION

The operability of the fixed incore detector system has been verified at each of the test plateaus and all test objectives satisfied.

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

TEST OBJECTIVE AND SUMMARY

The objective of 73PA-3SF02, "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 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. All applicable acceptance criteria were met, showing that a safe shutdown could be conducted and the plant stablized in hot standby conditions from outside the control room.

TEST DESCRIPTION

The test is performed by utilizing a normal operating crew and a standby crew. The standby crew serves as control room observers and are to take action only if a problem that involves plant safety develops. The operating crew consists of the minimum shift complement as defined in Table 6.2-1 of the PVNGS Technical Specifications. The test was performed on December 5, 1987 while the plant was operating at 20% of rated power.

The operating crew performs the test by evacuating the control room and proceeding to the remote shutdown panel. Once established, they initiate the switchgear panel. After the trip is verified, they establish control of the plant using equipment available at the remote shutdown panel and maintain hot standby conditions for approximately one hour and fifteen minutes. Control of the plant is then transferred to the standby crew in the control room and the remote shutdown panel is secured.

TEST RESULTS

The single value acceptance criteria and test results are summarized in Table 6-20. As can be seen, all acceptance criteria were satisifed.

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 6-20

SHUTDOWN FROM OUTSIDE THE CONTROL ROOM
TEST RESULTS

Parameter	Test Results Minimum	Test Results Maximum	Acceptance Criteria Minimum	Acceptance Criteria Maximum
Pressurizer Pressure (psia)	2200	2250	2150	2275
Hot Leg Temperature (^{o}F)	563	564	550	569
Pressurizer Level (%)	32	33	greater than 26	
Steam Generator Press. (psia)	1160	1170	1020	1220
Steam Gen. Level	60	73	greater than 35	

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