PALO VERDE NUCLEAR GENERATING STATION

UNIT 1 STARTUP REPORT

(Docket No. 50-528)

February 1986

PALO VERDE NUCLEAR GENERATING STATION

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PREFACE

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

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

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

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

The overall objectives of this test program are to:

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

This report will describe the FSAR (i.e. CESSAR) required testing from Fuel Loading though Low Power Physics Testing. Testing is listed and summarized by the applicable section of CESSAR. One or more future supplements will describe the CESSAR required tests of the Power Ascension Testing phase.

1.0 INTRODUCTION AND TEST PROGRAM SUMMARY

1.1 Summary of Test Objectives by Test Phase

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

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

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

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

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

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

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

Testing of the PAT phase will be described in one or more future supplements to this report.

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

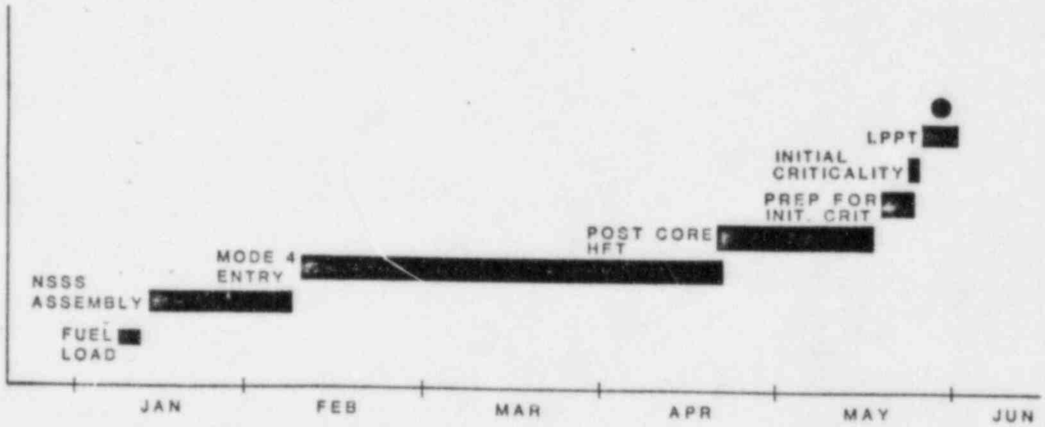
The chronology of startup testing from Fuel Load through Low Power Physics Testing is shown on Figure 1-1. The approximate durations of the activities that took place from the start of fuel load to the completion of LPPT are shown below:

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

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

FIGURE 1-1

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



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

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

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

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

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

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

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

2.0 INITIAL FUEL LOADING
(CESSAR Section 14.2.10.1)

TEST OBJECTIVES AND SUMMARY

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

TEST DESCRIPTION

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

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

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

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

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

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

TEST RESULTS

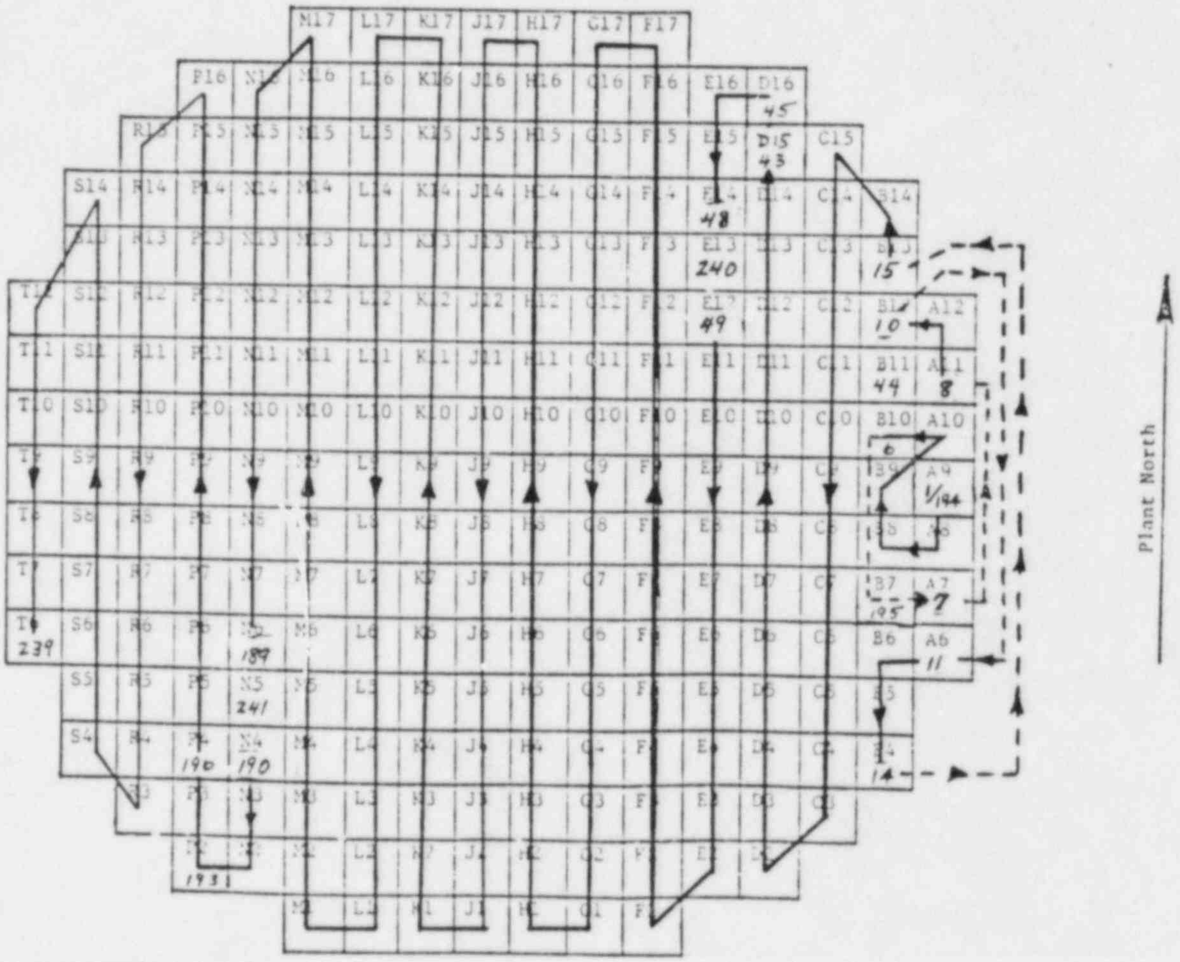
Fuel loading was completed on January 11, 1985. The fuel assemblies were verified to be properly loaded, seated, and oriented, with the exception of one misoriented fuel assembly. This assembly was rotated to the proper orientation. Both startup neutron sources were verified to be properly loaded. Finally,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

CONCLUSIONS

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

FIGURE 2-1

CORE LOADING SEQUENCE
PVNGS UNIT 1 INITIAL FUEL LOAD

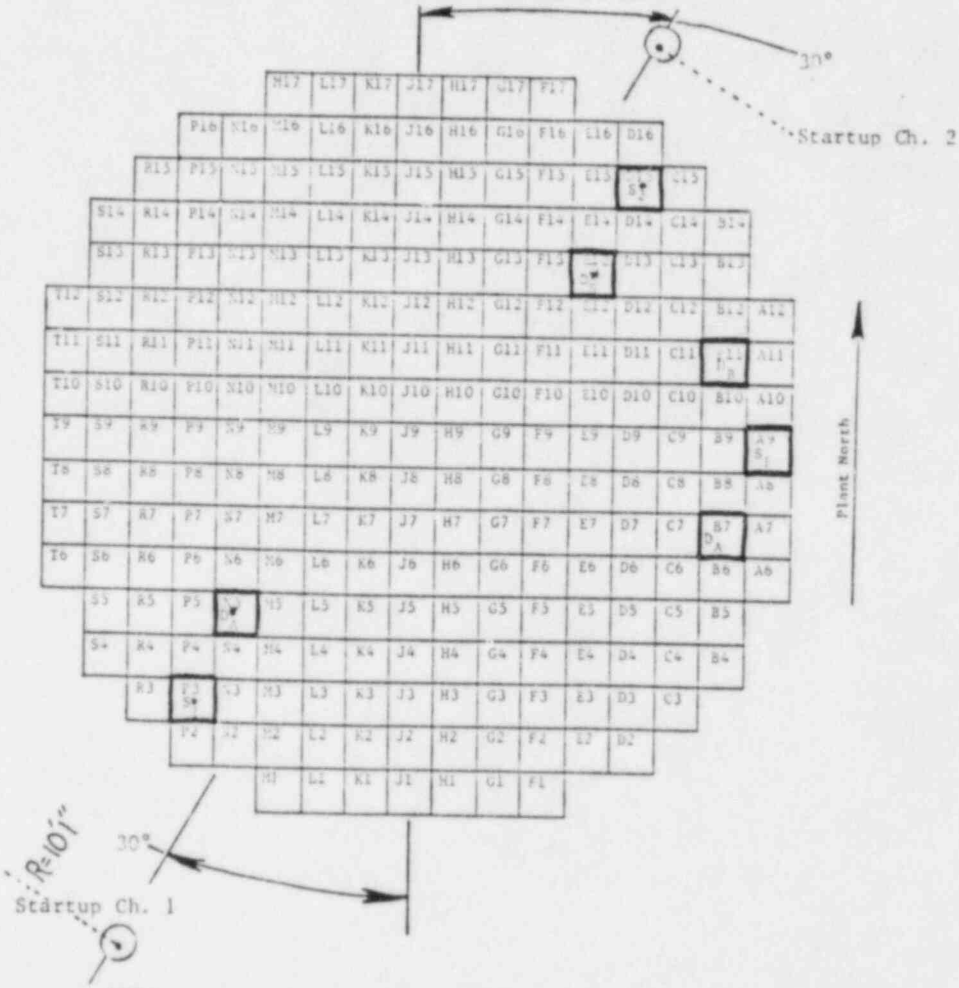


NOTES

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

FIGURE 2-2

FUEL LOADING DETECTOR LOCATIONS
PVNGS UNIT 1 INITIAL FUEL LOAD



KEY

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

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

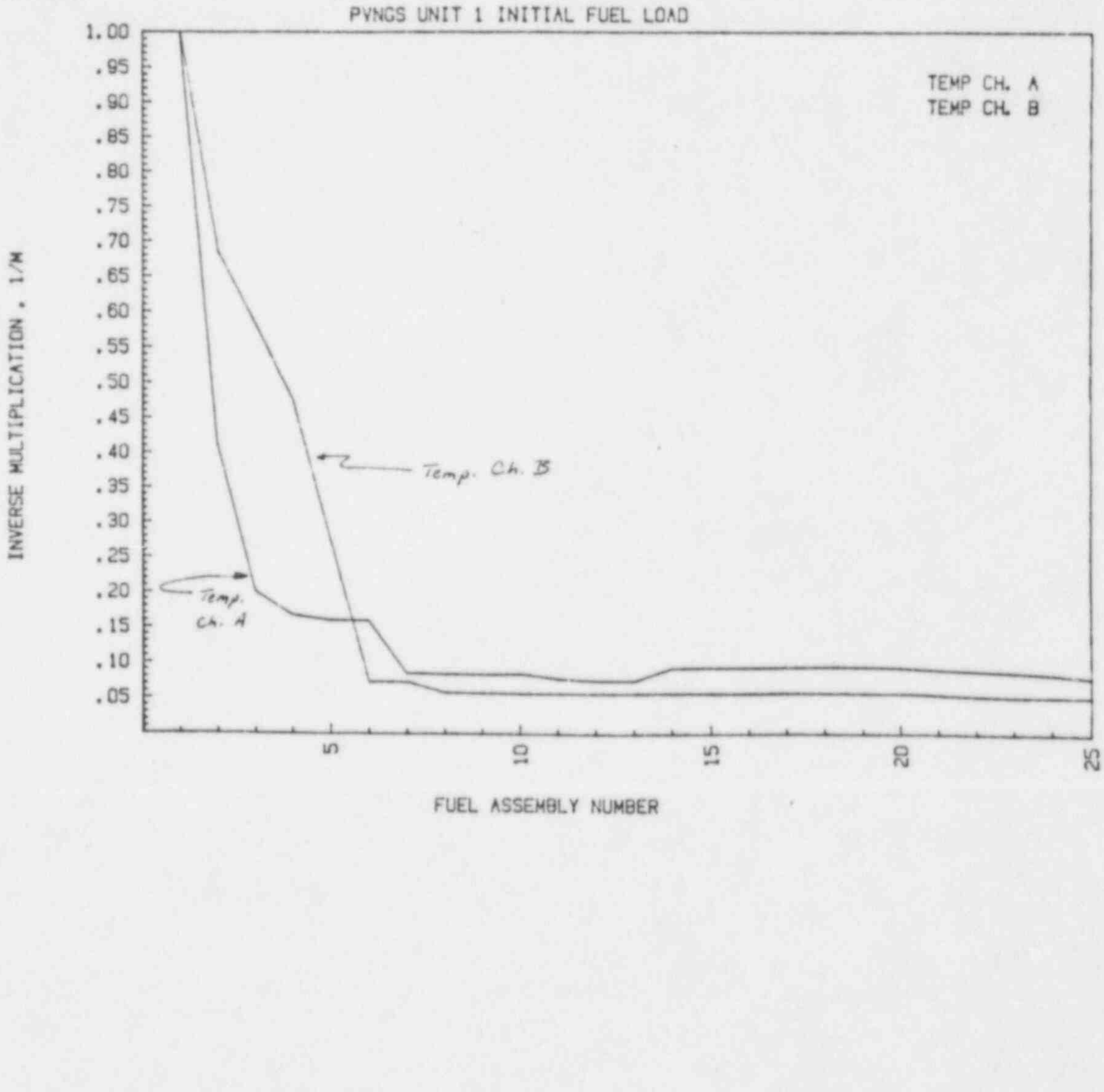
S_2^*---Final location of neutron source 2 (D-15)

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

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

FIGURE 2-3

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



3.0 POSTCORE HOT FUNCTIONAL TESTS
(CESSAR Section 14.2.12.3)

3.1 Postcore Hot Functional Test Controlling Document
(Section 14.2.12.3.1)

TEST OBJECTIVES AND SUMMARY

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

(1) To demonstrate the proper integrated operation of the plant primary, secondary, and auxiliary systems with fuel in the reactor vessel.
(2) To act as a sequencing/controlling document for the CESSAR required hot functional tests.
(3) To demonstrate that the plant can be brought from cold shutdown conditions (Mode 5) to hot standby conditions (Mode 3) using station operating procedures.
(4) To sequence/direct the initial performance of certain mode entry technical specification surveillance procedures.
(5) To sequence/control the performance of Precore Hot Functional (Phase 1) carryover tests.

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

TEST DESCRIPTION

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

TEST RESULTS

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

CONCLUSIONS

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

TABLE 3-1

POSTCORE HOT FUNCTIONAL TEST PLATEAUS
(Nominal Conditions)

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

3.2 Postcore Instrument Correlation
(Section 14.2.12.3.2)

TEST OBJECTIVE AND SUMMARY

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

TEST DESCRIPTION

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

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

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

TEST RESULTS

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

CONCLUSION

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

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

TEST OBJECTIVE AND SUMMARY

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

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

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

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

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

TEST DESCRIPTION

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

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

TEST RESULTS

The four pump volumetric RCS flow rate determined by UFM techniques was 468,430 gpm, or 105.1% of the design flow rate of 445,600 gpm. This measured flow rate was within the acceptance criteria of greater than or equal to 465,850 gpm and less than or equal to 501,800 gpm. Because the four pump volumetric RCS flow rate calculated from RCP differential pressure data did not acceptably agree with the ultrasonic flow measurement result, revised RCP performance curves were provided by the NSSS vendor.

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

CONCLUSIONS

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

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

3.4 Postcore Control Element Drive Mechanism Performance (Section 14.2.12.3.4)

TEST OBJECTIVES AND SUMMARY

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

TEST DESCRIPTION

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

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

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

TEST RESULTS

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

CONCLUSION

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

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

TEST OBJECTIVE AND SUMMARY

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

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

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

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

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

TEST DESCRIPTION

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

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

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

TEST RESULTS

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

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

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

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

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

340 °F/500 psia plateau - The RCS chemistry was within specification, with the exception of lithium. The RMWT and RWT were within the limits of 74HF-15501. The steam generators were within the operating specifications with the exception of dissolved oxygen. The feedwater system was not operating. The auxiliary feedwater system was in service to feed the steam generators. The auxiliary feedwater was maintained within specifications with the exception of dissolved oxygen. The condensate system was in long path recirculation through the condensate polishers. The chemistry of the condensate system was within specifications, with the exception of dissolved oxygen.

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

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

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

CONCLUSIONS

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

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

TEST OBJECTIVES AND SUMMARY

Test 73HF-1RC10, "Pressurizer Spray Valve and Control Adjustment", was performed to establish the proper settings for the continuous (bypass) pressurizer spray valves, to measure the rate at which pressure is reduced by maximum pressurizer spray, and to measure the maximum pressurization rate. The following acceptance criteria applied to the test: (1) the continuous spray flow was to be adjusted such that spray line temperature would be no more than 70 ^{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 on May 6, 1985, with the RCS at approximately 565^{o}F, 2250 psia, and full flow conditions. The acceptance criteria were satisfied.

TEST DESCRIPTION

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

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

TEST RESULTS

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

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

CONCLUSIONS

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

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

TEST OBJECTIVE AND SUMMARY

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

TEST DESCRIPTION

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

TEST RESULTS

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

CONCLUSION

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

3.8 Postcore Incore Instrumentation Test
(Section 14.2.12.3.8)

TEST OBJECTIVES AND SUMMARY

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

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

TEST DESCRIPTION

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

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

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

TEST RESULTS

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

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

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

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

CONCLUSIONS

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

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

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

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

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

4.0 INITIAL CRITICALITY
(CESSAR Section 14.2.10.2)

TEST OBJECTIVE AND SUMMARY

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

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

TEST DESCRIPTION

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

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

TEST RESULTS

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

CONCLUSIONS

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

5.0 LOW POWER PHYSICS TESTS
(CESSAR Section 14.2.12.4)

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

5.1 Low Power Biological Shield Survey Test
(Section 14.2.12.4.1)

TEST OBJECTIVE AND SUMMARY

PVNGS procedure 75PA-12Z01, "Biological Shield Survey", was performed during the phases of Low Power Physics Testing (LPPT) and Power Ascension Testing (PAT) to meet the following objectives:

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

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

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

Baseline background data for this test was gathered on March 14, 1985 with the plant in Mode 5. Low power physics data was gathered on May 27, 1985 with the reactor critical and at 0% full power (FP) and the primary conditions of 320 ^{o}F, 600 psia; and on May 30, 1985, with the reactor critical and at 0% FP, and the primary at 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 over 400 selected locations in accessible areas outside the biological shield. These surveys were performed per the plant radiation survey procedure, and included general area surveys in rooms or areas as well as more detailed surveys around penetrations, shield plugs, and other areas where streaming could be occurring. A scan survey was also performed while the survey team was in transit between designated survey points. Surveys were performed in the Containment Building, Auxiliary Building, Main Steam Support Structure, Turbine Building, Fuel Building, Control Building, Decontamination and Laundry Facility, and at various site locations exterior to the plant.

TEST RESULTS

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

CONCLUSION

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

Table 5-1
RADIATION ZONE CLASSIFICATION

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

5.2 CEA Symmetry Test
(Section 14.2.12.4.2)

TEST OBJECTIVE AND SUMMARY

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

TEST DESCRIPTION

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

TEST RESULTS

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

CONCLUSIONS

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

5.3 Isothermal Temperature Coefficient Test
(Section 14.2.12.4.3)

TEST OBJECTIVE AND SUMMARY

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

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

The measured ITC values were required to be within $\pm$ 0.5 x 10^{-4} delta-K/K/^{o}F of their predicted values, a condition which was met during all the measurements. The moderator temperature coefficient (MTC) was determined from each of the ITCs measured at 565 ^{o}F (hot zero power), and verified to be in compliance with the Tech Spec limits (+0.22 x 10^{-4} to -2.80 x 10^{-4} delta-K/K/^{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 or, additionally, on an X-Y plotter. Following a short stabilization time at the new temperature, the RCS temperature was then returned to its initial value. Temperature and reactivity were again recorded during the transition.

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

TEST RESULTS

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

CONCLUSIONS

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

TABLE 5-2

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

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

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

TEST OBJECTIVE AND SUMMARY

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

TEST DESCRIPTION

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

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

TEST RESULTS

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

CONCLUSIONS

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

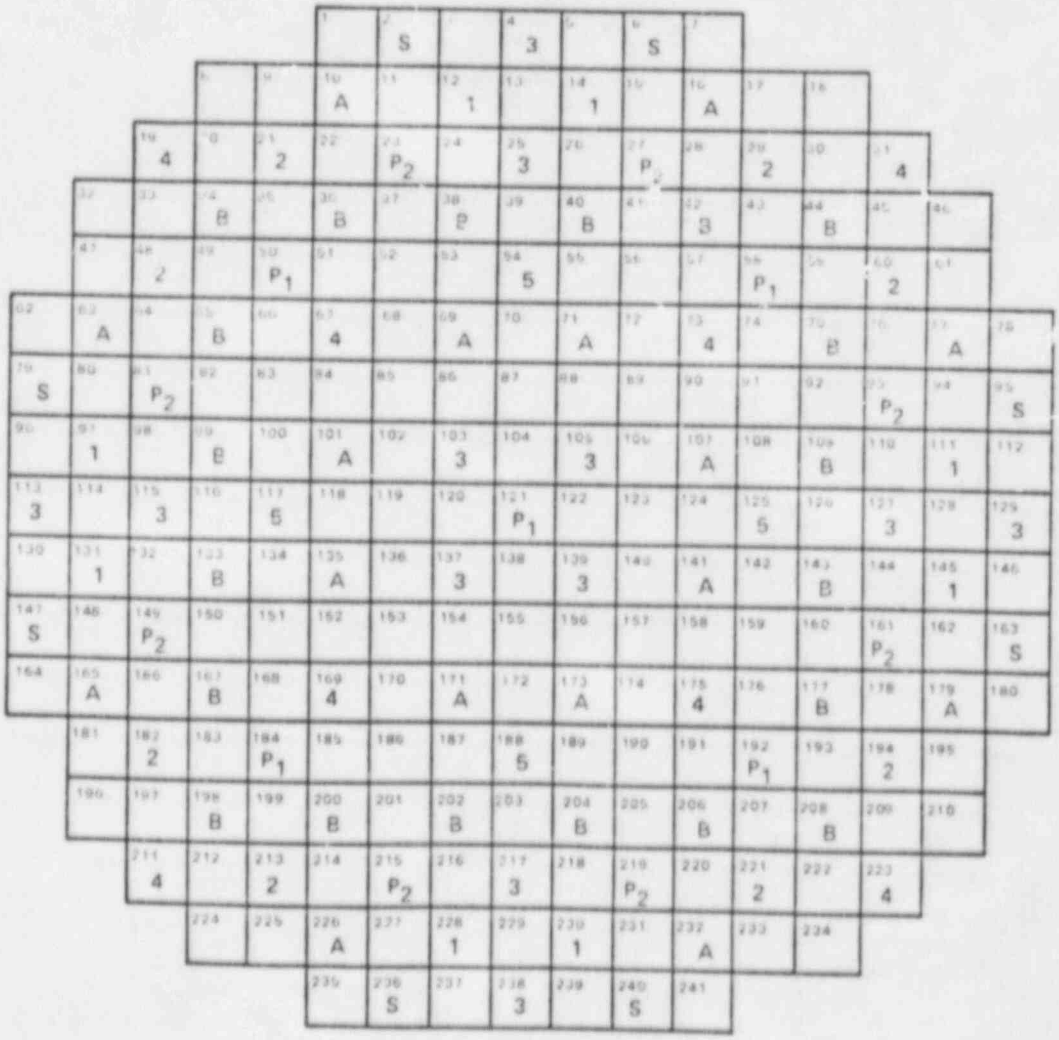
TABLE 5-3

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

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

FIGURE 5-1

RELATIVE CORE LOCATIONS
OF CEA GROUPS



KEY:

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

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

5.5 Differential Boron Worth Test
(Section 14.2.12.4.5)

TEST OBJECTIVE AND SUMMARY

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

TEST DESCRIPTION

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

TEST RESULTS

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

CONCLUSIONS

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

TABLE 5-4

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

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

5.6 Critical Boron Concentration Test
(Section 14.2.12.4.6)

TEST OBJECTIVE AND SUMMARY

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

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

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

TEST DESCRIPTION

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

TEST RESULTS

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

CONCLUSIONS

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

TABLE 5-5

CRITICAL BORON CONCENTRATIONS
(ppm)

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

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

TEST OBJECTIVE AND SUMMARY

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

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

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

TEST DESCRIPTION

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

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

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

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

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

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

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

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

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

TEST RESULTS

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

CONCLUSIONS

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

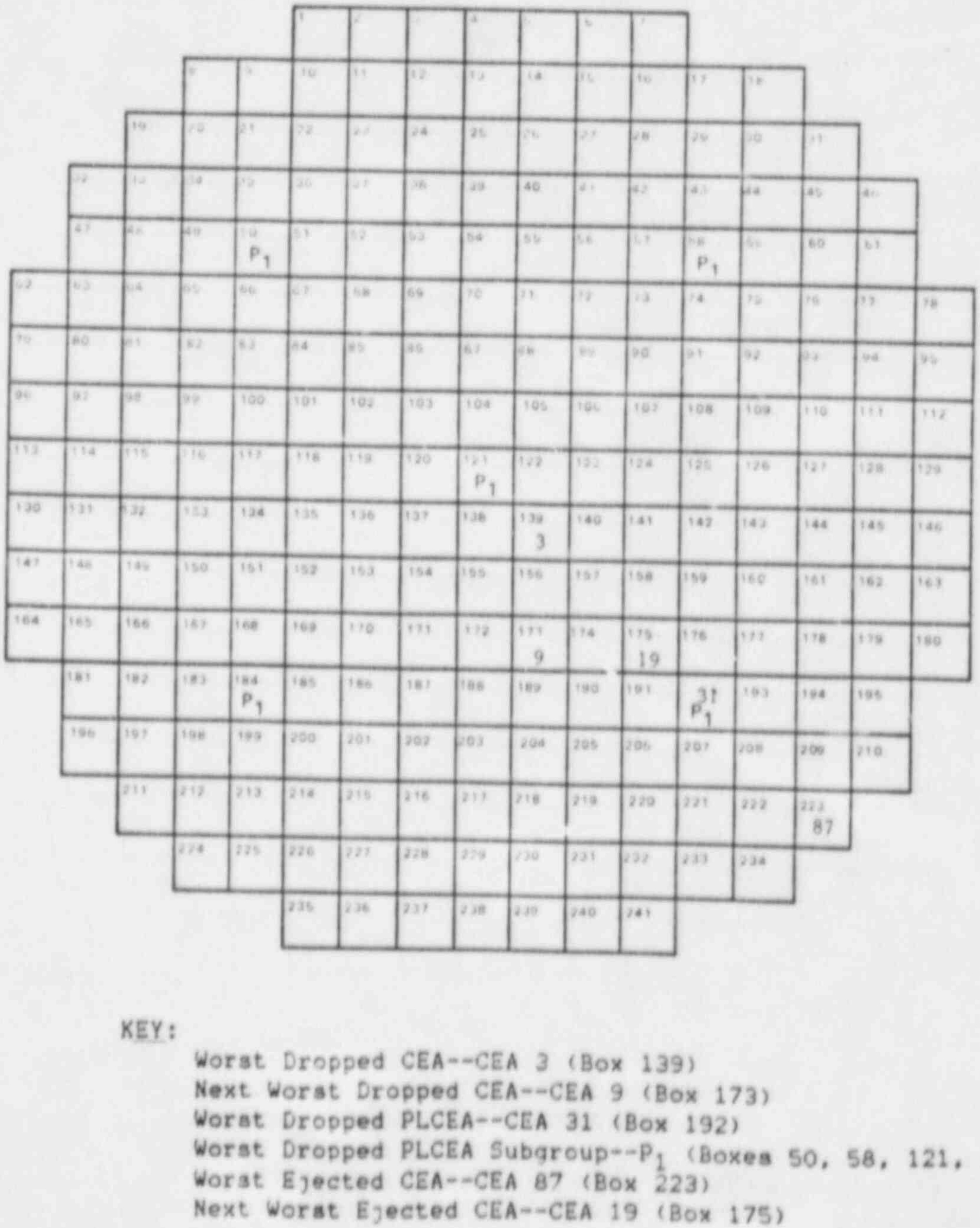
TABLE 5-6

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

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

FIGURE 5-2

RELATIVE CORE LOCATIONS
OF DROPPED AND EJECTED CEAS



KEY:

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