North Anna Power Station
Updated Final Safety Analysis Report

Chapter 14

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CHAPTER 14 INITIAL TESTS AND OPERATION

This chapter describes the scope of tests and operations performed over the time period when construction was sufficiently complete to operate and test individual components and systems through the acceptance run at full power. This time period is divided into two categories:

1. Pre-operational testing: tests performed before the initial core loading.
2. Initial start-up testing: tests and operations from the initial core loading through the acceptance tests.

The preoperational and start-up programs, as outlined in Tables 14.1-1 and 14.1-2, comply with the intent of Regulatory Guide 1.68, *Preoperational and Initial Start-up Test Programs for Water-Cooled Power Reactors*, dated November 1973, in most cases, and use the same wording as much as possible in order to more clearly address the NRC guide requirements. Detailed acceptance criteria were provided in each test procedure that was written to fulfill the testing requirements. The detailed criteria of acceptability were based on various sources, such as equipment technical manuals, system descriptions, plant drawings, manufacturer specifications, and the North Anna Units 1 and 2 FSAR. The tests and their objectives are listed in Tables 14.1-1 and 14.1-2, which also provide a summary of each test. The acceptability of a test is contingent on the successful attainment of the objectives stated in Tables 14.1-1 and 14.1-2.

Because of similarities and differences in the fuel and core characteristics between the two units, certain tests performed for Unit 1 were not repeated for Unit 2, while specific tests were performed for Unit 2 only. A discussion of the start-up physics program differences appears in Section 14.1.3.

14.0.1 Administration of the Preoperational Test Program

The management and direction of the preoperational test program was under the direct control of VEPCO, with the principal responsibility lying with the Supervisor - Engineering Services. In most cases written preoperational test procedures were prepared by the station engineering staff under the direction of the Supervisor - Engineering Services. In those areas where the station engineering staff was not knowledgeable, procedures were provided by the architect-engineer or outside consultants, based on their expertise in the particular areas of concern. Test procedure format generally included the purpose of the test, initial condition requirements, precautions and limitations, instructions, and criteria for acceptability of data. Prior to issuance of test procedures for use in the field, they were reviewed by the Joint Test Group and approved by the Station Nuclear Safety and Operating Committee.

For those procedures provided by the architect-engineer or outside consultants, the preoperational test procedure was used as a cover sheet to their procedure in order to ensure review by the Joint Test Group and approval by the Station Nuclear Safety and Operating

Committee. In some instances the preoperational test procedures were used to review and approve test data from testing performed by equipment vendors off the site (e.g., vendor certifications).

In most instances the conduct and direction of the preoperational tests were the direct responsibility of the VEPCO test engineers designated by the technical supervision at the station. In some instances architect-engineer personnel or outside consultants were responsible for the conduct of tests under the direction of VEPCO by means of written administrative controls. Changes to approved test procedures were documented and became part of the final test results. Administrative controls for making changes to procedures prepared by the station engineering staff were provided in the Nuclear Power Station Quality Assurance Manual. Administrative procedures for making changes to procedures provided by the architect-engineer or outside consultants were formulated by the architect-engineer and approved by VEPCO.

For preoperational testing, the Supervisor - Engineering Services and the Joint Test Group reviewed and analyzed the test results. Assistance from the VEPCO system office, the nuclear steam supply system vendor, and the architect-engineer was solicited as deemed necessary. The test results and evaluations were reviewed by the Station Nuclear Safety and Operating Committee and approved if they were satisfactory. In instances where performance of components or systems deviated from predicted results, further engineering evaluations were made to resolve the discrepancies before the test was considered satisfactory. Systems that had to be modified as a result of the preoperational tests were then retested to verify acceptable performance.

The completed test procedures, along with data and conclusions, were documented and filed as part of the permanent plant records.

Minimum qualifications for the VEPCO test engineers were as follows:

1. A bachelor's degree in engineering or the physical sciences or the equivalent, and at least 1 year of applicable nuclear power plant experience, or:
2. A high school diploma or the equivalent, and at least 3 years of applicable nuclear power plant experience. Credit for up to 2 years of nuclear experience may be given for related technical training on a one-for-one time basis.

Additional information relative to the preoperational test program is provided in the Nuclear Power Station Quality Assurance Manual, in VEPCO station administrative procedures, and in the architect-engineer's administrative procedures.

14.0.2 Administration of the Start-Up Test Program

The management and direction of the start-up test program has been under the direct control of VEPCO, with principal responsibility lying with the Supervisor - Engineering Services. Written start-up test procedures were prepared by the station reactor engineers under the direction

of the Supervisor - Engineering Services. Procedures from the Nuclear Steam Supply System Start-up Manual and assistance from Westinghouse personnel were utilized in many cases. Prior to issuance of test procedures for use in the field, they were approved by the Station Nuclear Safety and Operating Committee.

The conduct and direction of the start-up tests were the responsibility of the reactor engineers designated by the Supervisor - Engineering Services. Changes to approved test procedures were documented and became part of the final results. Administrative procedures for making these changes, including the review and approvals, were formulated and utilized by VEPCO.

For start-up testing the reactor engineers and the Supervisor - Engineering Services reviewed and analyzed the test results. The measurements and data analysis for start-up physics tests were performed by the VEPCO Fuel Resources Department. Assistance from the VEPCO system office, the nuclear steam supply system vendor, and the architect-engineer was solicited as deemed necessary. Approval of test results was the responsibility of the Station Nuclear Safety and Operating Committee. The completed test procedures, along with data and conclusions, were documented and filed as part of the permanent plant records.

The minimum qualifications for the reactor engineers, in terms of educational background and experience, are stated in Section 13.1. Minimum qualifications for the test engineers responsible for the preparation and performance of start-up tests were as follows:

1. A bachelor’s degree in engineering or the physical sciences or the equivalent and 2 years of applicable power plant experience, of which at least 1 year shall be applicable nuclear power plant experience, or
2. A high school diploma or the equivalent and 5 years of applicable power plant experience, of which at least 2 years shall be applicable nuclear power plant experience. Credit for up to 2 years of non-nuclear experience may be given for related technical training on a one-for-one time basis.

Additional information relative to the start-up test program is provided in the Nuclear Power Station Quality Assurance Manual and in station administrative procedures.

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14.1 TEST PROGRAM

14.1.1 Pre-Operational Test Program

The pre-operational test program included tests, adjustments, calibrations, and system operations necessary to ensure that initial fuel loading, initial criticality, and subsequent power operation could be safely undertaken.

After installation of individual components and systems was completed, the installed components and systems were tested and evaluated according to approved testing procedures or check-off lists. Analyses of test results were made to verify that systems and components were performing satisfactorily or, if not, to provide a basis for recommended corrective action.

Whenever possible, these tests were performed under the same conditions to be experienced under subsequent station operations. During system tests for which unit parameters were not available, the systems were operationally tested as far as possible without these parameters. The remainder of the tests were performed under plant conditions when the parameters were available. Abnormal unit conditions were simulated during testing as required and when such conditions did not endanger personnel or equipment, or contaminate systems whose cleanliness had been established.

In general, pre-operational testing was completed before core loading. As individual systems were completed, pre-operational tests were performed to verify as nearly as possible the performance of the system under actual operating conditions. Where required, simulated signals or inputs were used to verify the full operating range of the system and to calibrate and align the systems and instruments at these conditions. Later, systems that were used during normal operation were verified under actual operating conditions. Systems that are not used during normal plant operation, but should be in a state of readiness to perform safety functions, were tested before plant start-up. Examples of these systems are the reactor trip system and engineered safety features system logic, operation checks, and setpoint verifications.

Testing performed during the pre-operational test program is outlined in Table 14.1-1. A typical sequence of performance for operational tests is shown in Figure 14.1-1. The actual sequence of tests was formulated before the performance of the tests, considering equipment and system availability. In some cases, it was necessary to complete certain pre-operational tests after core loading. These included such tests as those performed on the complete rod control system, rod position indication, and complete incore movable detector system. These tests have been identified in Table 14.1-1.

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14.1.2 Initial Start-Up Test Program

Fuel loading was begun when all prerequisite system tests and operations were satisfactorily completed. Upon completion of fuel loading, the reactor upper internals and pressure vessel head were installed, and additional mechanical and electrical tests were performed as discussed in pre-operational testing. The purpose of this phase of activities was to prepare the system for nuclear operation and to establish that all design requirements necessary for operation were achieved. The core-loading and postloading tests are described below.

14.1.2.1 **Initial Fuel Loading**

The reactor containment structure was completed and tested before initial fuel loading. Fuel-handling tools and equipment were checked out and dry runs conducted in the use and operation of equipment.

The reactor vessel and associated components were in a state of readiness to receive fuel. Water level was maintained above the bottom of the nozzles.

The overall responsibility and direction for the initial core loading was exercised by the Station Manager assisted by the Superintendent - Station Operation. The overall process of initial core loading was, in general, directed from the operating floor of the containment structure. Procedures for the control of personnel and the maintenance of containment security were in effect during initial fuel loading.

The as-loaded core configuration was specified as part of the core design studies conducted in advance of core loading. In the event mechanical damage to a fuel assembly occurred during core-loading operations, an evaluation would have been performed and a replacement assembly would have been procured if deemed necessary.

The core was assembled in the reactor vessel, containing reactor-grade water with dissolved boric acid to maintain a calculated core effective multiplication factor of 0.95 or lower. The refueling cavity was kept dry during the initial core loading. Core moderator chemistry conditions (particularly, boron concentration) were prescribed in the core-loading procedure document and were verified periodically by chemical analyses of moderator samples taken before and during core-loading operations.

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Core-loading instrumentation consisted of two permanently installed source range (pulse-type) nuclear channels and three temporary incore source range channels. The permanent channels, when responding, were monitored in the main control room by licensed station operators; the temporary channels were monitored by fuel-loading personnel. One permanent channel was equipped with an audible count rate indicator. The neutron flux level from both plant channels was displayed on a strip chart recorder. The temporary channels were indicated on rate meters with one channel recorded on a strip chart recorder. Minimum count rates of two counts per sec, attributable to core neutrons, were required on at least two of the five available nuclear source channels at all times following installation of the initial nucleus of eight fuel assemblies.

Fuel assemblies together with inserted components (control rod assemblies, burnable poison inserts, source spider, or thimble plugging devices) were placed in the reactor vessel one at a time according to a previously established and approved sequence developed to provide reliable core monitoring with minimum possibility of core mechanical damage. The core-loading procedure documents included detailed tabular check sheets that prescribed and were used to verify the successive movements of each fuel assembly and its specified inserts from its initial position in the storage racks to its final position in the core. Multiple checks were made of component serial numbers and types at successive transfer points to guard against possible inadvertent exchanges or substitutions of components, and fuel assembly status boards were maintained throughout the core-loading operation.

An initial nucleus of eight fuel assemblies, the first of which contained an activated neutron source, is the minimum source-fuel nucleus that permits subsequent meaningful inverse count rate ratio monitoring. This initial nucleus has been determined by calculation and previous experience to be markedly subcritical (k_{eff} less than or equal to 0.90) under the required conditions of loading.

Each subsequent fuel addition was accompanied by detailed neutron count rate monitoring to determine that the just-loaded fuel assembly did not excessively increase the count rate and that the extrapolated inverse count rate ratio was not decreasing for unexplained reasons. The results of each loading step were evaluated before the next prescribed step was started.

Criteria for safe loading require that loading operations stop immediately if:

1. An unanticipated increase in the neutron count rates by a factor of two occurs on all responding nuclear channels during any single loading step after the initial nucleus of eight fuel assemblies is loaded (excluding anticipated changes due to detector and/or source movement).

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2. The neutron count rate on any individual nuclear channel increases by a factor of five during any single loading step after the initial nucleus of eight fuel assemblies is loaded (excluding anticipated changes due to detector and/or source movements).

An alarm in the containment and main control room is coupled to the source range channels with a setpoint equal to or less than five times the baseline count rate. This alarm automatically alerts the loading operation personnel of high count rate and requires an immediate stop of all operations until the situation is evaluated.

Core-loading procedures specified the condition of fluid systems to prevent inadvertent dilution of the reactor coolant, specified the movement of fuel to preclude the possibility of mechanical damage, prescribed the conditions under which loading could proceed, identified responsibility and authority, and provided for continuous and complete fuel and core component accountability.

14.1.2.2 **Initial Postloading Tests**

Upon completion of core loading, the reactor upper internals and the pressure vessel head were installed, and additional mechanical and electrical checks were performed before initial criticality. The final pressure test was conducted after filling and venting were completed to check the integrity of the vessel head installation.

Mechanical and electrical tests were performed on the control rod drive mechanisms. These tests included a complete operational checkout of the mechanisms and calibration of the individual rod position indication.

Tests were performed on the reactor trip circuits to test manual trip operation. The actual control rod assembly drop times were measured for each control rod assembly. The reactor control and protection system was checked with simulated signals to produce a trip signal for the various conditions that require plant trip.

At all times when the control rod drive mechanisms were being tested, the boron concentration in the coolant-moderator was maintained such that the reactor would remain adequately shut down with all control rod assemblies fully withdrawn.

A complete functional electrical and mechanical check was made of the incore nuclear flux mapping system, and reactor coolant system flow measurements were taken to relate reactor coolant pump input power and elbow tap pressure differential to actual reactor coolant loop flow.

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14.1.2.3 **Initial Criticality and Low-Power Physics Tests**

On completion of postloading tests, nuclear operation of the reactor was begun. This final phase of start-up and testing included initial criticality, low-power testing, and power level escalation. The purpose of these tests was to establish the operational characteristics of the unit and core, to acquire data for the proper calibration of setpoints, and to ensure that operation was within license requirements. A brief description of the testing is presented in this section. Table 14.1-2 summarizes the major tests that were performed from initial core loading to rated power. Figure 14.1-2 depicts a typical sequence for these tests; the actual sequence of tests was formulated by station engineering and operating personnel, considering test requirements and equipment availability.

Initial criticality was established by sequentially withdrawing the shutdown and control banks of control rod assemblies from the core, leaving the last withdrawn control bank inserted far enough in the core to provide effective control when criticality would later be achieved, and then diluting the heavily borated reactor coolant until criticality was achieved.

Successive stages of control rod assembly bank withdrawal and of boron concentration dilution were monitored by observing changes in neutron count rate as indicated by the normal plant source range nuclear instrumentation as functions of bank position during rod motion and, subsequently, of reactor coolant boron concentration and primary-water addition to the reactor coolant system during dilution. Throughout this period, samples of the primary coolant were obtained and analyzed for boron concentration.

Inverse count rate ratio monitoring was used as an indication of the proximity and rate of approach to criticality of the core during control rod assembly bank withdrawal and during reactor coolant boron dilution. The rate of dilution was reduced as the reactor approached the boron concentration extrapolated for criticality to ensure that effective control was maintained at all times. Written procedures specified the plant conditions, precautions, and specific instructions for the approach to criticality.

After initial criticality, a prescribed program of reactor physics measurements was undertaken to verify that the basic static and kinetic characteristics of the core were as expected and that the values of the kinetic coefficients assumed in the safeguards analysis were indeed conservative.

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The measurements were made at low power and primarily at or near operating temperature and pressure. The measurements included verification of calculated values of control rod assembly bank reactivity worths, of isothermal temperature coefficient under various core conditions, of differential boron concentration reactivity worth, and of critical boron concentrations as functions of control rod assembly bank configuration. In addition, measurements of the relative power distributions were made. Concurrent tests were conducted on the instrumentation, including the source and intermediate range nuclear channels.

Procedures were prepared to specify the sequence of tests and measurements conducted and the conditions under which each was to be performed to ensure both safety of operation and the validity and consistency of the results obtained. Had significant deviations from design predictions existed, or had unacceptable behavior been revealed, or had apparent anomalies developed, then testing would have been suspended and the situation reviewed to determine whether a question of safety was involved before the resumption of testing.

14.1.2.4 **Power Level Escalation**

When the operating characteristics of the reactor and unit were verified by low-power testing, a program of power level escalation in successive stages was used to bring the unit to its full rated thermal power level. Both primary and secondary operational characteristics were examined at each stage of the power escalation program.

Measurements were made to determine the relative power distribution in the core as functions of power level and control assembly bank position.

Secondary system heat balances ensured that the indications of power level were consistent and provided bases for calibration of the power range nuclear channels. The ability of the reactor coolant system to respond effectively to signals from primary and secondary instrumentation under a variety of conditions encountered in normal operations was verified.

At prescribed power levels the dynamic response characteristics of the reactor coolant and steam systems were evaluated. The responses of the systems were measured for design step load changes of 10%, rapid 50% load reduction, and 50% and 100% power plant trips.

Adequacy of radiation shielding was determined by gamma and neutron radiation surveys at selected points inside the containment and the outside area immediately adjacent to the containment at various power levels. Periodic sampling was performed to verify the chemical and radiochemical analyses of the reactor coolant.

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All precritical tests were completed and the results evaluated before initial criticality. Prerequisites for performing a test were specified in the individual test procedure. The sequence of testing was outlined in a start-up test sequence, such that required prerequisite testing was completed before subsequent testing. Any special test instruments required were specified to be installed, calibrated, and checked in the test procedure that specified the test equipment.

14.1.3 Start-Up Physics Test Program Differences Between Unit 1 and Unit 2

After the initial start-up physics program for North Anna Unit 1 was completed, several changes to the program were made before the initial start-up physics program for Unit 2. Table 14.1-3 lists the physics tests that were performed as part of the Unit 2 start-up program. These tests were chosen to:

1. Verify that the core was correctly loaded and that there were no anomalies present that could cause problems later in the cycle.
2. Verify that the calculational model that had been used would correctly predict core behavior during the cycle.
3. Verify the reactivity worth of the control rod banks.
4. Provide data for nuclear instrumentation calibration.
5. Demonstrate the sensitivity of this instrumentation to abnormal core conditions.

In addition, the chosen tests were selected to encompass the physics test goals listed in the NRC Branch Technical Position DOR-1, *Guidance for Reload Submittals, Draft - Spring, 1978*. Table 14.1-4 lists those physics tests that were performed during the Unit 1 start-up, and that were not repeated as part of the Unit 2 start-up program. The deletion of these tests was justified for the following reasons:

1. The successful performance of the abbreviated program was sufficient to achieve the physics testing program goals.
2. The calculational model was verified as a result of the Unit 1 start-up.
3. The fuel and core characteristics of Unit 2 are virtually identical to those of Unit 1, and the results obtained for these tests during the Unit 1 start-up demonstrated that a large margin exists between the measured parameter values and the design values used in the accident analyses. Evidence of this is shown in Table 14.1-5.

When the modified test program for Unit 2 was proposed, several questions were raised by the NRC relating to the modifications. The questions and VEPCO's responses are the subject of Appendix 14A.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

14.1.4 Special Low-Power Tests - Unit 2

This test program consisted of a series of natural circulation tests that demonstrated the plant's cooldown capability in several simulated degraded modes of operation at power levels of up to 3% of rated thermal power.

The objectives of the above tests and the methods used are described below.

1. Natural Circulation Test

   Objective: To demonstrate the capability to remove decay heat by natural circulation.

   Method: The reactor is at approximately 3% power and all reactor coolant pumps are operating. All reactor coolant pumps are tripped simultaneously, with the establishment of natural circulation indicated by the core exit thermocouples and the wide-range resistance temperature detectors.

2. Natural Circulation With Simulated Loss of Offsite Power

   Objective: To demonstrate that following a loss of offsite ac power, natural circulation can be established and maintained while being powered from the emergency diesel generators.

   Method: The reactor is at approximately 3% power and all reactor coolant pumps are operating. All reactor coolant pumps are tripped and a station blackout is simulated. Alternating current power is returned by the diesel generators and natural circulation is verified.

3. Natural Circulation With Loss of Pressurizer Heaters

   Objective: To demonstrate the ability to maintain natural circulation and saturation margin with the loss of pressurizer heaters.

   Method: Establish natural circulation as in Test 1, and turn off the pressurizer heaters at the main control board. Monitor the system pressures to determine the saturation margin, the depressurization rate, and the effects of charging/letdown flow and steam generator pressure on the saturation margin.

4. Effect of Steam Generator Secondary-Side Isolation on Natural Circulation

   Objective: To determine the effects of steam generator secondary-side isolation on natural circulation.

   Method: Establish natural circulation conditions as in Test 1 but at 1% power. Isolate the feedwater and steam line for one steam generator and establish equilibrium. Return the steam generator to service.

5. Natural Circulation at Reduced Pressure

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Objective: To demonstrate the ability to maintain natural circulation at reduced pressure and saturation margin. The accuracy of the saturation meter will also be verified.

Method: The test method is the same as for Test 3, with the exception that the pressure decrease can be accelerated with the use of auxiliary pressurizer sprays. The saturation margin will be decreased to approximately 20°F.

14.1 REFERENCE DRAWINGS

The list of Station Drawings below is provided for information only. The referenced drawings are not part of the UFSAR. This is not intended to be a complete listing of all Station Drawings referenced from this section of the UFSAR. The contents of Station Drawings are controlled by station procedure.

	Drawing Number	Description
1.	11715-LSK-1-3A	Generator Breaker Closing
2.	11715-LSK-1-3B	Logic Diagram: Power Circuit Breaker Opening
3.	11715-LSK-1-3C	Logic Diagram: “86” Protective Lockout Relays
4.	11715-LSK-1-3D	Logic Diagram: Main Transformer Protection Relays, 86-TL
5.	11715-LSK-1-3E	Logic Diagram: Generator Lockout Relays
6.	11715-LSK-1-3F	Logic Diagram: Main Transformer Differential Lockout Relays, 86-GL & 86-PWIA
7.	11715-LSK-1-3G	Logic Diagram: Main Transformer Coolers
8.	11715-LSK-1-2A	Logic Diagram: External Turbine Trips, Sheet 1
9.	11715-LSK-1-3H	Logic Diagram: Main Transformer Alarms
10.	11715-LSK-1-2B	Logic Diagram: Turbine Trips, Sheet 2
11.	11715-LSK-1-2C	Logic Diagram: Turbine Trips, Sheet 3
12.	11715-LSK-1-2D	Logic Diagram: Turbine Trips, Sheet 4
13.	11715-LSK-1-2E	Logic Diagram: Turbine Trips, Sheet 5
14.	11715-LSK-1-2F	Logic Diagram: Turbine Trips, Sheet 6
15.	11715-LSK-1-2G	Logic Diagram: Turbine Trips, Sheet 7

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Table 14.1-1

LIST OF PREOPERATIONAL TESTS

Test or Measurement	Plant Condition/Prerequisite	Test Objective and Summary of Testing
		I. Plant Instrumentation
1. Nuclear instrumentation (out of core)	Before core loading and initial criticality	Before core loading, nuclear instruments were aligned and source range detector response to a neutron source checked. Just before initial criticality all channels were checked to verify high-level trip functions, alarm setpoints, audible count rates where applicable, and operation of strip chart recorders and any auxiliary equipment.
2. Process instrumentation (temperature, pressure, level, and flow instruments)	Ambient and/or at temperature	Equipment was aligned per manufacturer's instructions and applicable test procedures. Applicable alarm, trip, and control setpoints were checked for conformance with specified values.
3. Seismic instrumentation	Before core loading	This instrumentation was verified for correct installation and operability. A calibration record test was performed to verify operability of the magnetic tape playback units
		II. Reactor Coolant System
1. Vibration and amplitude	Before core loading	Vibration sensors were placed on the main coolant pumps and main cooling piping in order to check for excessive vibration while starting and stopping the pumps
2. Expansion and restraint	Before core loading, during heatup, and during cooldown.	During the heatup to operating temperature, selected points on cooldown components and piping of the reactor coolant system were checked at various temperatures to verify unrestricted expansion. Points of interference detected during the heatup were corrected before increasing the temperature. Following cooldown to ambient temperature, the piping and components were checked to confirm that they returned to their approximate base points.

a. The preoperational test program for the emergency core cooling system (ECCS) meets the requirements set forth in Regulatory Guide 1.79, June 1974, with clarifications noted in Section 3A.40.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-1 (continued)

LIST OF PREOPERATIONAL TESTS

Test or Measurement	Plant Condition/Prerequisite	Test Objective and Summary of Testing
	II. Reactor Coolant System (continued)	
3. Integrated hot functional tests	Heatup, at temperature, and during cooldown. (Hydrostatic testing has been satisfactorily completed and reactor coolant system instruments aligned and operational. Associated auxiliary systems were to be operational.)	The reactor coolant system was tested using pump heat to check heatup and cooldown procedures and to demonstrate satisfactorily performance of components and systems exposed to reactor coolant system temperature. Proper operation of instrumentation, controllers, and alarms was checked against operating conditions of auxiliary systems and setpoints verified. Among the demonstrations performed were: a. To verify that water can be charged by the Chemical and Volume Control System at rated flow against normal reactor coolant pressures. b. To check letdown design flow rate for each operating mode. c. To check response of system to change in pressurizer level. d. To check response procedures and components used in boric acid batching and transfer operations. e. To check operation of the excess letdown and seal-water flow paths. f. To check steam generator level instrumentation response to level changes. g. To check thermal expansion of system components and piping. h. To perform isothermal calibration of resistance temperature detectors and incore thermocouples.

a. The preoperational test program for the emergency core cooling system (ECCS) meets the requirements set forth in Regulatory Guide 1.79, June 1974, with clarifications noted in Section 3A.40.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-1 (continued)

LIST OF PREOPERATIONAL TESTS

Test or Measurement	Plant Condition/Prerequisite	Test Objective and Summary of Testing
	II. Reactor Coolant System (continued)	
		i. To operationally check out the residual heat removal system.
		j. Visual and surface examination of the upper and lower core structures after the heatup and cooldown were completed (see item 6 below.)
4. Component tests		
a. Pressurizer	At operating temperature	During the hot functional testing, the pressure-controlling capability of the pressurizer was demonstrated to be within the controlling band. After core loading, with the reactor coolant pumps operating and with full spray, the pressure-reducing capability of the pressurizer was verified. With the spray secured and all heaters energized, the pressure-increasing capability of the pressurizer was verified. Expected rates of pressure decrease/increase with tolerances were specified in the test procedures.
b. Reactor coolant system pumps and motors	At ambient conditions, during heatup, and at temperature	As the pumps and motors were placed in operation they were checked for: a. Direction of rotation. b. Vibration. c. Power requirements. d. Lubrication e. Cooling. f. Recirculation flow. g. Flow and pressure characteristics.

a. The preoperational test program for the emergency core cooling system (ECCS) meets the requirements set forth in Regulatory Guide 1.79, June 1974, with clarifications noted in Section 3A.40.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-1 (continued)

LIST OF PREOPERATIONAL TESTS

Test or Measurement	Plant Condition/Prerequisite	Test Objective and Summary of Testing
	II. Reactor Coolant System (continued)	
		h. Megger and hi-pot tests (as applicable) i. Overload protection. j. Correct power supply voltage During the reactor coolant system cold hydrostatic and hot functional tests, the pumps were operated to verify proper installation. Following core loading, measurements were made to determine flow and input power relationships.
c. Steam generators	At ambient conditions, during heatup, and at temperature. (The secondary system had been satisfactorily hydrostatically tested.)	The proper operation of instrumentation and control systems for steam generators was checked during heatup and at temperature. The heat transfer capability of the steam generators was demonstrated. The functioning of the blowdown system was also checked.
d. Pressurizer relief and safety valves	Pressure conditions	The setpoints of the relief and safety valves were verified from vendor certification data, by bench tests, or by in-plant tests. When verified by in-plant tests, setpoints were checked by using a pressure assist device that adds to the force due to pressure. Once the valve started to lift, this assist device was vented, allowing the valve to reseat immediately. Following lifting and blowdown of any valve, the reseating of the valve was verified.

a. The preoperational test program for the emergency core cooling system (ECCS) meets the requirements set forth in Regulatory Guide 1.79, June 1974, with clarifications noted in Section 3A.40.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-1 (continued)

LIST OF PREOPERATIONAL TESTS

Test or Measurement	Plant Condition/Prerequisite	Test Objective and Summary of Testing
	II. Reactor Coolant System (continued)	
e. Main steam stop valves	At operating temperature (steam flow no required)	At hot conditions and with pressure equalized across the valve, the operation of the main steam stop valves was verified. The operating times were verified to be within expected values as specified by the test procedure.
f. Reactor coolant system (RCS) loop isolation valves	At ambient conditions and at operating temperature and pressure conditions	At ambient and at hot conditions, the operation of the loop isolation valves was checked.
g. Main steam stop valve piping	Before core loading	Main steam stop valve piping was checked for excessive vibration while closing the main steam stop valves with steam available from the heatup of the reactor coolant by the reactor coolant pumps.
h. Pressurizer relief valve discharge piping	Pressure conditions	The discharge piping associated with pressurizer relief valves was checked for excessive vibration during the operation (opening and closing) of the pressurizer relief valves. Following lifting and blowdown of any valve, the re-seating of the valve was verified. (Note that this test may be done in conjunction with item d of this section.)

a. The preoperational test program for the emergency core cooling system (ECCS) meets the requirements set forth in Regulatory Guide 1.79, June 1974, with clarifications noted in Section 3A.40.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-1 (continued)

LIST OF PREOPERATIONAL TESTS

Test or Measurement	Plant Condition/Prerequisite	Test Objective and Summary of Testing
II. Reactor Coolant System (continued)		
5. Pressure boundary integrity tests		
a. Hydrostatic tests	Below 200°F (after verification of cleanliness, and fill of system)	Cold hydrostatic testing of the reactor coolant system pressure boundary was performed at test pressures as specified by ASME standards for the system. Prior to pressurization, the system was heated above the minimum temperature for pressurization. The pressure was then increased in increments and at each increment inspections were made for leakage. Leaky valves or mechanical joints were not a basis for rejecting the test. Relief valves were provided to prevent inadvertent over-pressurization of the system.
b. Baseline data for inservice inspection	During preoperational testing	Systems and components that require inspections in accordance with Section XI of the ASME Code were examined for baseline data either following the cold hydrostatic test or following hot functional testing, depending on the system and component and its availability and accessibility. Data from these inspections provided baseline data for inservice inspections
c. Nondestructive testing of stainless steel safe ends and critical components	Before hydrostatic test	All reactor coolant system weld joints were nondestructively tested using liquid penetrant and/or radiographic tests as required by Section III of the ASME Code.

a. The preoperational test program for the emergency core cooling system (ECCS) meets the requirements set forth in Regulatory Guide 1.79, June 1974, with clarifications noted in Section 3A.40.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-1 (continued)

LIST OF PREOPERATIONAL TESTS

Test or Measurement	Plant Condition/Prerequisite	Test Objective and Summary of Testing
	II. Reactor Coolant System (continued)	
6. Vibration monitoring on reactor internals	During and after hot functional testing	Comprehensive vibration measurements had been made during hot functional testing before core loading for Carolina Power & Light Company's H. B. Robinson Unit 2. The results of these tests had been documented and submitted to the Directorate of Reactor Licensing. These data were the basis for acceptance of following plants, such as North Anna Units 1 and 2, without repeating these tests. During hot functional testing, the plant was operated with full flow for a minimum of 240 hours in order to achieve approximately 20 million cycles on the internal components. Following hot functional testing, the internals were removed and inspected for vibration effects before core loading.
	III. Reactivity Control System	
1. Chemical and volume control system	At ambient and/or at operating conditions. System components were operationally checked out before fuel loading.	Makeup and letdown operations were conducted with the Chemical and Volume Control System to check out the difference modes of dilution and boration and verify flows in the different modes. The adequacy of heat tracing to maintain the highest concentration in solution was verified. The ability to adequately sample and the sampling techniques were demonstrated.
2. Emergency boron shutdown system	During hot functional testing	The pressure/flow characteristics of the emergency boration system were verified by pumping into the reactor coolant system.

a. The preoperational test program for the emergency core cooling system (ECCS) meets the requirements set forth in Regulatory Guide 1.79, June 1974, with clarifications noted in Section 3A.40.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-1 (continued)

LIST OF PREOPERATIONAL TESTS

Test or Measurement	Plant Condition/Prerequisite	Test Objective and Summary of Testing
	III. Reactivity Control System (continued)	
3. Automatic reactor power control system	Preoperational conditions (installation checks had been made)	The system alignment was verified at preoperational conditions to demonstrate the response of the system to simulated inputs. These tests were performed to verify that the systems would operate satisfactorily at power. The alignment of the system was verified at power by programmed step changes and under actual test transient conditions to verify that controlled parameters were within tolerances specified by test procedures.
4. Incore monitor system		
a. Incore thermocouples	During heatup and at temperature	During heatup and at temperature, the incore thermocouples were calibrated to the average of the reactor coolant system resistance temperature detectors. All readout and temperature-compensating equipment was checked during the calibration, and isothermal corrections for the operative thermocouples were determined.
b. Movable detector system	At ambient conditions, before core loading, and after core loading and critical testing	Before core loading, the installation checkout of the movable detector system was completed. The response of each channel was verified using simulated detector inputs. After core loading and insertion of the thimbles in the core, a dummy cable was used to check indexing and to ensure free passage to all positions and set the limit switches based on data obtained during critical testing. During flux mapping at power, the detector responses to neutron flux were verified.

a. The preoperational test program for the emergency core cooling system (ECCS) meets the requirements set forth in Regulatory Guide 1.79, June 1974, with clarifications noted in Section 3A.40.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-1 (continued)

LIST OF PREOPERATIONAL TESTS

Test or Measurement	Plant Condition/Prerequisite	Test Objective and Summary of Testing
	III. Reactivity Control System (continued)	
5. Control rod system		
a. Rod control system	Ambient conditions, before core loading, and hot conditions after core loading	During the installation check of this system, it was energized and operationally checked out with mechanisms connected to each power supply. The ability of the system to step the mechanism was verified, the alarm and inhibit functions checked out, and correct values of system parameters adjusted to specified values. After core loading, the operation of each rod over its full range of travel was demonstrated.
b. Rod drop tests	Cold and hot plant conditions after core loading	At cold and hot plant conditions after core loading, the drop times of the full-length rods were measured. The drop time was measured from the release of the rod until the rod enters the top of the dashpot. This time was verified to be less than the maximum value specified in the Technical Specifications.
c. Rod position indication	At ambient conditions and at temperature after core loading	During rod control system tests, the rod position indication system was aligned to provide rod movement indication. Rod setpoints were also adjusted during these tests. After plant heatup, individual rod positions were calibrated to within tolerances specified by the test procedure.

a. The preoperational test program for the emergency core cooling system (ECCS) meets the requirements set forth in Regulatory Guide 1.79, June 1974, with clarifications noted in Section 3A.40.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-1 (continued)

LIST OF PREOPERATIONAL TESTS

Test or Measurement	Plant Condition/Prerequisite	Test Objective and Summary of Testing
	III. Reactivity Control System (continued)	
6. Auxiliary start-up instrumentation test	Before core loading	Three separate temporary source range instruments were installed in the core during core-loading operations. One of these channels served as a spare to the other two channels. During the core loading operations, these detectors were relocated at specific loading steps to provide the most meaningful neutron count rate within minimum acceptable levels, as specified by the core-loading procedures. The response of each channel to a neutron source was verified before core loading.
	IV. Protection System	
1. Reactor protection system	Before core loading (installation checks had been performed)	Before core loading, the reactor trip system was tested to demonstrate operability, proper logic, redundancy, coincidence, independence, and safe failure on power loss. The protection channels were verified through to tripping of the reactor trip breakers. The trip time of each reactor protection signal was also measured from the output of the sensor to tripping of the reactor trip breaker. These times were verified to be less than the values identified in the safety analysis report.

a. The preoperational test program for the emergency core cooling system (ECCS) meets the requirements set forth in Regulatory Guide 1.79, June 1974, with clarifications noted in Section 3A.40.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-1 (continued)

LIST OF PREOPERATIONAL TESTS

Test or Measurement	Plant Condition/Prerequisite	Test Objective and Summary of Testing
	IV. Protection System (continued)	
2. Engineered safety features	Before core loading (installation checks had been performed)	Before core loading, the engineered safety features logic systems were tested to demonstrate operability, proper logic, redundancy, coincidence, and independence. The protection channels were verified through to actuation of the output relays. The response time of each protection signal was also measured from the output of the sensor to actuation of the output relay. Their times were verified to be less than the values identified in the safety analysis report. Operation of the engineered safety features components (i.e., motors, valves, diesel generators) was checked in other tests.
	V. Power Conversion System	
1. System tests		
a. Vibration frequency and amplitude	Hot functional testing and or plant heatup after initial criticality	When the main turbine was rolled, vibration readings were monitored. (Turbine vibrations were also monitored throughout the power escalation program.)
		Major equipment (e.g., feedwater pumps and condensate pumps) was operated as it became available and was observed for indications of excessive vibration.
b. Expansion and restraint	During heatup, at temperature, and cooldown before core loading	During heatup to operating temperature, selected points on the components and piping of the systems were checked at various temperatures to verify that they could expand unrestricted. After cooldown, these components were verified to have returned to their approximate cold position.

a. The preoperational test program for the emergency core cooling system (ECCS) meets the requirements set forth in Regulatory Guide 1.79, June 1974, with clarifications noted in Section 3A.40.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-1 (continued)

LIST OF PREOPERATIONAL TESTS

Test or Measurement	Plant Condition/Prerequisite	Test Objective and Summary of Testing
V. Power Conversion System (continued)		
2. Components and individual systems		
a. Steam generator pressure relief and safety valves	Pressure conditions	The setpoints of safety valves were verified from vendor certification data, by bench tests, or by in-plant tests, setpoints were checked by using a pressure assist device that adds to the force due to pressure. Once the valve left the seated position, the assist device was vented, allowing the valve to re-seat immediately. Steam relief valve setpoints were made during instrument alignment and verified by plant transient tests.
b. Emergency feedwater (auxiliary) system	Before initial criticality	During hot functional testing before initial criticality, the emergency feedwater system was checked out to verify its ability to feed the steam generators. Automatic starting was verified during testing of the safeguards logic system tests. The auxiliary feedwater piping was checked for excessive vibration while starting and stopping the auxiliary feedwater pumps with normal operation of the associated motor-operated and hand-control discharge valves in the auxiliary feedwater system.
c. Turbine control and bypass valves	Hot functional testing and/or power operation after initial criticality	During hot functional testing, the turbine control system was demonstrated in turbine operation up to and including a period of operation at synchronous speed. The turbine bypass valves to the condenser and their associated control systems were operationally checked out before and during hot functional testing. Other testing on the turbine bypass valves was completed after initial criticality.

a. The preoperational test program for the emergency core cooling system (ECCS) meets the requirements set forth in Regulatory Guide 1.79, June 1974, with clarifications noted in Section 3A.40.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-1 (continued)

LIST OF PREOPERATIONAL TESTS

Test or Measurement	Plant Condition/Prerequisite	Test Objective and Summary of Testing
		V. Power Conversion System (continued)
d. Feedwater and feedwater control system	Before hot functional testing and at power	The feedwater and condensate pumps were operationally checked out before hot functional testing. During power escalation, the power was slowly increased and the ability of the feedwater pumps and control system to maintain level in the steam generators was verified. Steam generator level indicators were aligned before filling the system, and during fill the system was used to monitor the level in the steam generator. Before start-up, the feedwater-regulating valve control system was calibrated using simulated signals. During start-up when at power the ability of the system to control level within specified tolerances under transient conditions was also verified
e. Condenser circulating water	Before initial core loading and at power	Before core loading, the main circulating water pumps and circulating water system valves were tested to verify operability. During unit start-up, acceptable condenser operation was verified in accordance with operating procedures.
f. Makeup water and chemical treatment systems	During steam generator fill, hot functional testing, and at power	The makeup system to the steam generators was checked out during fill of the steam generators during hot functional testing and at power. The chemical treatment system was checked out when chemicals were added to the steam generators at heatup and at operating conditions.

a. The preoperational test program for the emergency core cooling system (ECCS) meets the requirements set forth in Regulatory Guide 1.79, June 1974, with clarifications noted in Section 3A.40.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-1 (continued)

LIST OF PREOPERATIONAL TESTS

Test or Measurement	Plant Condition/Prerequisite	Test Objective and Summary of Testing
		VI. Auxiliary Systems
1. Reactor coolant system makeup (Chemical and Volume Control System (CVCS))		See III, item 1.
2. Seal and pump cooling water (CVCS)	Before heatup and at temperature	Before reactor coolant pump operation and with the system pressurized, flow to the pump seals and cooling water flow was adjusted to specified values using installed instruments. During hot functional testing when at operating temperature and pressure, seal and cooling flows and temperatures were checked.
3. Vent and drain system	During initial primary fill and pressurization and during hot functional testing	Venting of the reactor coolant system was done during initial filling by venting the reactor vessel head and pressurizer. During hot functional testing and after core loading, the secondary system was vented while pressurizing the secondary system. Secondary drains were tested for unrestricted flow in accordance with operating procedures.
4. Component cooling system	Ambient and/or hot plant conditions	Component cooling flow to the various components in the system was adjusted, the system operationally checked out, and setpoints adjusted. Data were taken to verify that adequate cooling was provided to each cooled component and, when load was available, that temperature limits were being maintained.

a. The preoperational test program for the emergency core cooling system (ECCS) meets the requirements set forth in Regulatory Guide 1.79, June 1974, with clarifications noted in Section 3A.40.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-1 (continued)

LIST OF PREOPERATIONAL TESTS

Test or Measurement	Plant Condition/Prerequisite	Test Objective and Summary of Testing
		VI. Auxiliary Systems (continued)
5. Residual heat removal system	Before and during hot functional testing	This system was tested by verifying pressure and flow characteristics of the pumps and operation of the isolation valves. During cooldown after hot functional testing, the heat removal capability and cooldown rate of the system was demonstrated. The residual heat removal system piping was checked for excessive vibration while starting and stopping the residual heat removal pumps with normal operation of the valve used to control flow.
6. Purification system (CVCS)	Operating temperature before core loading	During hot functional testing with the demineralizers charged with resin, operation of the purification system was demonstrated by verification of flow, pressure drops, temperatures, and conditioning of ion-exchange resins (see III, item 1).
7. Fire protection system	Before initial core loading	The water fire protection system motor and diesel-driven pumps and pressure maintenance equipment were tested to verify proper operation in conformance with fire insurance requirements. The carbon dioxide fire protection system was tested by individual component checks and by puff tests in various fire-protected areas by simulating system initiating conditions. The Halon 1301 fire protection system was checked for operability and proper installation.
8. Service water system	Before initial core loading	The system was operationally checked out to verify pressure and flow. Service water flow was verified to components in the system.
9. Auxiliary building ventilation	Before initial core loading	The system was operated to test for leaks and air flows to the areas supplied from the system and to verify motor currents and speeds, verify setpoints, and check alarms (see also IX.)

a. The preoperational test program for the emergency core cooling system (ECCS) meets the requirements set forth in Regulatory Guide 1.79, June 1974, with clarifications noted in Section 3A.40.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-1 (continued)

LIST OF PREOPERATIONAL TESTS

Test or Measurement	Plant Condition/Prerequisite	Test Objective and Summary of Testing
	VI. Auxiliary Systems (continued)	
10. Compressed gas system (used for safety-related functions)	Before initial core loading	The instrument air system, including air receivers and compressors, was tested to verify proper operation. A loss-of-instrument-air test was conducted by securing the makeup air to each dedicated air accumulator supplying each safety-related component that is required to operate following a loss of instrument air. The capacity of each dedicated air accumulator was verified by operating the safety-related component a specified number of times over a specified time interval. Air-operated components were tested to ensure that they fail in the safe mode upon loss of operating pressure. Other compressed gas systems were verified for proper operation.
11. Control-rod drive mechanism and rod position indication coil cooling system	Before and/or during hot functional testing	The system was operationally checked out to verify air flow, temperatures, motor current and speed.
12. Neutron shield tank cooling system	Before initial core loading	The system was operationally checked out to verify pump and heat exchanger operability.
13. Leak detection system (sensitivity and accuracy to detect leaks)	Before and during preoperational tests	Temperature detectors and their alarm functions in the drain lines from pressurizer safety valves and the reactor vessel head seal were checked. Pressurizer relief tank level and temperature sensors were calibrated and their associated alarms checked.

a. The preoperational test program for the emergency core cooling system (ECCS) meets the requirements set forth in Regulatory Guide 1.79, June 1974, with clarifications noted in Section 3A.40.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-1 (continued)

LIST OF PREOPERATIONAL TESTS

Test or Measurement	Plant Condition/Prerequisite	Test Objective and Summary of Testing
	VI. Auxiliary Systems (continued)	
14. Primary sampling system	Before and/or during hot functional testing	Operations were performed to: a. Established purge times. b. Demonstrate that liquid and gas samples can be obtained from sample points. c. Demonstrate that valves, instruments and controls function properly. d. Verify proper functioning of the sample cooler. e. Demonstrate that sample vessels can be removed and replaced without problems.
15. Primary pressure relief system	Before hot functional testing and at pressure conditions	The pressurizer relief tank, associated valves, and instrumentation were checked out to verify performance of design functions. (See II, item 4.3, for testing of pressurizer relief and safety valves.)
	VII. Electrical Systems	
1. Normal distribution test (transformers, motor, relay switches, power supplies etc.; phasing and meggering where applicable)	Throughout plant start-up and before applicable equipment operation	The integrity and operation of these components were verified before being energized by meggering, hi-pot testing, continuity checks, and operational verification of controlling devices as applicable. After being energized, phasing and voltage regulation tests were performed and channel and train separation and redundancy features were verified as applicable.

a. The preoperational test program for the emergency core cooling system (ECCS) meets the requirements set forth in Regulatory Guide 1.79, June 1974, with clarifications noted in Section 3A.40.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-1 (continued)

LIST OF PREOPERATIONAL TESTS

Test or Measurement	Plant Condition/Prerequisite	Test Objective and Summary of Testing
		VII. Electrical Systems (continued)
2. Vital bus test (full-load test using all power sources)	Before initial core loading	Verification that the vital bus load could be supplied under normal and power-failure conditions was made. In particular, transfers that take place under loss of power and redundant features function per design were verified.
3. Direct current systems (full-load and duration test)	Before core loading	The redundant features of the battery, battery charger, and inverters were checked out. The capacity of the battery and voltage regulation was verified. The recharging of a discharged battery within a specified period was also verified, The ability of each inverter to maintain design output under varying direct current input was also verified.
4. Communications systems (telephone, public address, intercoms, and evacuation signals)	Before fuel loading and during power operation	To verify proper communications between all onsite stations and interconnection to commercial telephone service. To balance and adjust amplifiers and speakers and verify that evacuation alarms could be heard at all stations throughout the plant. Also, to verify that all temporary communications at the fuel-loading stations and control stations were functioning properly.
5. Emergency power systems (manual start and synchronization, full automatic loading tests, under loss of all alternating current voltage)	Before initial core loading	The automatic starting and loading of the diesel generators was demonstrated under loss of emergency bus alternating current power. The operation of the logic and sequencing of circuit breakers were demonstrated along with the proper safety-related bus stripping and separation of non-vital loads. Load duration tests were demonstrated over several hours of operation along with voltage and frequency regulation tests under transient and steady-state conditions.

a. The preoperational test program for the emergency core cooling system (ECCS) meets the requirements set forth in Regulatory Guide 1.79, June 1974, with clarifications noted in Section 3A.40.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-1 (continued)

LIST OF PREOPERATIONAL TESTS

Test or Measurement	Plant Condition/Prerequisite	Test Objective and Summary of Testing
	VIII. Containment Systems	
1. Reactor containment	Before core loading	A containment structural test was performed. Containment Type A, B, and C leakage tests were performed in accordance with Appendix J to 10 CFR 50.
2. Ventilation system	Before and/or during hot functional testing	The system was operated to balance air flows and to verify ability to maintain temperatures below maximum allowable limits.
3. Post-accident heat removal system (containment sprays)	Before initial criticality	Tests were performed to verify pump operating characteristics and response to control signals, sequencing of the pumps, valves, and controllers, and to ensure that spray nozzles were unobstructed. The time required to actuate the system after a containment high-pressure signal is received was verified.
4. Containment isolation	Before core loading	The operation of actuation systems and components used for containment isolation was verified.
5. Hydrogen removal system	Before initial criticality	Operability of flow paths and heaters associated with the recombiners were verified.
	IX. Gaseous Radioactivity Removal Systems	
Filtration system (testing performed on particulate filter system in containment and auxiliary structures for post-accident and routine release of gaseous effluent)	Before core loading	Testing was performed to verify flows, pressure drops, and effectiveness of these systems in performing their function.

a. The preoperational test program for the emergency core cooling system (ECCS) meets the requirements set forth in Regulatory Guide 1.79, June 1974, with clarifications noted in Section 3A.40.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-1 (continued)

LIST OF PREOPERATIONAL TESTS

Test or Measurement	Plant Condition/Prerequisite	Test Objective and Summary of Testing
	X. Emergency Core Cooling System[a]	
1. System tests (expansion and restraints, vibration)	Before and/or during hot functional testing	Movement of piping that connects to the reactor coolant system was checked by the test described in II, item 2. Pumps, motors, and piping were observed for excessive vibration.
2. High-pressure safety injection	Before core loading	This system was operationally tested to adjust pressure/flow values. Tests were also conducted to check pump operating characteristics and to verify operation from normal and emergency power sources. More specifically that: a. Valves installed for redundant flow paths operated as designed. b. Pump operating characteristics were verified and the capability of the high-head safety injection pumps to take suction from the low-head pumps was demonstrated with the reactor coolant systems at ambient conditions. c. Valves and pumps operated on operator initiation and/or automatically on initiation of a safety injection signal. d. The fail position on loss of power for each remotely operated valve was as specified. e. Level and pressure instruments were properly calibrated.

a. The preoperational test program for the emergency core cooling system (ECCS) meets the requirements set forth in Regulatory Guide 1.79, June 1974, with clarifications noted in Section 3A.40.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-1 (continued)

LIST OF PREOPERATIONAL TESTS

Test or Measurement	Plant Condition/Prerequisite	Test Objective and Summary of Testing
	X. Emergency Core Cooling System[a] (continued)	
3. Low-pressure safety injection	Before core loading	The low-head safety injection system was checked to verify design flow, flow paths, and pump operating characteristics. Tests were conducted to verify operation from the normal power source. More specifically, that: a. Valves installed for redundant flow paths operated as designed. b. Pump operating characteristics were verified with the reactor coolant system at ambient conditions. c. Valves and motors operated on operator initiation and/or automatically on initiation of a safety injection signal. d. The fail position on loss of power for each remotely operated valve was as specified. e. Level and pressure instruments were properly calibrated.
4. Accumulator	Before core loading	Flow through the accumulator discharge lines was initiated to demonstrate that the motor-operated valves stroked properly and the check valves were free to open. Tests were also made to verify that accumulator pressure could be maintained.

a. The preoperational test program for the emergency core cooling system (ECCS) meets the requirements set forth in Regulatory Guide 1.79, June 1974, with clarifications noted in Section 3A.40.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-1 (continued)

LIST OF PREOPERATIONAL TESTS

Test or Measurement	Plant Condition/Prerequisite	Test Objective and Summary of Testing
	XI. Fuel Storage and Handling System	
1. Spent-fuel pit cooling and refueling purification system	Before core loading	Tests were performed to verify operability of the spent-fuel pit cooling pumps; operability of the refueling purification pumps; flow through the spent-fuel pit heat exchange loops; operation of the skimmer loops; flows through the refueling purification filters and ion exchanger; alarm setpoints; and correct functioning of valves, instruments, and controls.
2. Refueling equipment (hand tools and power equipment, including protective interlocks)	Before storage of new fuel and initial core loading	Tests were performed before core loading to demonstrate the functioning of the fuel transfer system and the fuel handling equipment using a dummy assembly, in accordance with design drawings and instruction manuals. The sections that involve the spent-fuel facility were checked before the storage of new fuel in the spent-fuel storage pool.
3. Operability and leak tests of sectionalizing devices in fuel storage pool and refueling canal	Before initial core loading	During the initial filling of the spent-fuel storage pool, operability and leaking testing of the sectionalizing devices was performed.
4. Spent fuel storage building ventilation system	Before plant start-up	This is part of the auxiliary building ventilation system (refer to VI, item 1).
5. Spent-fuel storage radiation monitoring equipment	Before plant start-up	Refer to XIII, item 1.

a. The preoperational test program for the emergency core cooling system (ECCS) meets the requirements set forth in Regulatory Guide 1.79, June 1974, with clarifications noted in Section 3A.40.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-1 (continued)

LIST OF PREOPERATIONAL TESTS

Test or Measurement	Plant Condition/Prerequisite	Test Objective and Summary of Testing
XII. Reactors Components Handling System		
Reactor components handling system (polar crane)	Before use for installation of components within the containment	Testing was conducted on the polar crane in accordance with standard crane testing procedures during the construction of the station.
XIII. Radiation Protection System		
1. Process, criticality, and area monitors	Before core loading and plant operation	Before core loading, the radiation alarms associated with core loading were checked out and alarm setpoints were verified. Process and area monitor sensors and channels were calibrated and alarm setpoints made.
2. Personnel monitor and survey instruments	Before core loading and/or initial criticality	Before core loading and required equipment use, instruments were calibrated. After this initial calibration, the instruments are periodically checked for recalibration.
3. Laboratory equipment	Before core loading and initial criticality	Laboratory equipment was checked to verify equipment performance and calibration. Chemical analyses performed on standard samples. During start-up the equipment received additional verification by normal usage.

a. The preoperational test program for the emergency core cooling system (ECCS) meets the requirements set forth in Regulatory Guide 1.79, June 1974, with clarifications noted in Section 3A.40.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-1 (continued)

LIST OF PREOPERATIONAL TESTS

Test or Measurement	Plant Condition/Prerequisite	Test Objective and Summary of Testing
XIV. Radioactive Waste System		
Radioactive waste system	Before initial criticality	Tests were performed to establish the satisfactorily performance of pumps and instruments, leaktightness of piping and equipment, and the operation of packaging and waste reduction equipment; and to verify proper operation of alarms and controls. More specifically, to ensure that: a. Manual and automatic valves were operable b. Instrument controllers operated properly. c. Alarms were operable. d. Pumps performed their system function satisfactorily. e. The waste gas compressors operated properly. f. The gas analyzers operated properly. g. The waste evaporator was operational. h. The hydrogen and nitrogen supply packages were sufficient for operation of the system.

a. The preoperational test program for the emergency core cooling system (ECCS) meets the requirements set forth in Regulatory Guide 1.79, June 1974, with clarifications noted in Section 3A.40.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-2

LISTS OF START-UP TESTS

Test or Measurement	Plant Condition/Prerequisite	Test Objective and Summary of Testing
	I. Precritical Tests After Fuel Loading	
1. Mechanical and instrument tests on control rod drive and rod position indicators	Before initial criticality	Operational testing of the rod control systems was conducted to check the controlling features, adjusts setpoints, and verify rod speeds and sequencing of power to the rod drives. After core loading and installation of the rod mechanisms, tests were conducted to verify operation of the rod drive mechanisms, over their full travel, the latching and releasing features were demonstrated, and calibration of the position indicators was performed over the rod full-range travel per tolerances specified in the test procedure.
2. Reactor trip circuit and manual trip tests	Before initial criticality	Operational testing was conducted to verify the reactor protection circuits in the various modes of tripping, including manual reactor trip up to the tripping of the reactor trip breakers. After core loading, the release and insertion of each full-length mechanism was demonstrated.
3. Rod drop measurement cold and hot at rated flow and no flow	Before initial criticality	At cold and hot plant conditions after core loading, the drop times of the full-length rods were measured. The drop time was measured from the beginning decay of the stationary gripper coil voltage until the rod entered the top of the dashpot. This time was verified to be less than the maximum value specified in the Technical Specifications. Ten additional measurements were made for the fastest and slowest rods.
4. Pressure test of reactor coolant system	Before initial criticality	After core loading and installation of the reactor vessel head and torquing of the reactor vessel head studs, pressure testing was performed at 100 psi above operating pressure to verify that no leakage occurred past the head and vessel seal.

a. Reference Drawings 1 through 15 show the logics for initiation of turbine and generator trips. The logics show the sensed variables and all available setpoints. See Figure 7.2-1 for a listing of symbols used in these figures.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-2 (continued)

LISTS OF START-UP TESTS

Test or Measurement	Plant Condition/Prerequisite	Test Objective and Summary of Testing
	I. Precritical Tests After Fuel Loading (continued)	
5. Chemical tests (to establish water quality)	Before heatup	Water for reactor coolant system fill and makeup was analyzed for chloride content, conductivity, total suspended solids, pH, clarity, and fluorides to requirements specified by the chemistry manual for NSSS. During pre-operational testing, hydrazine was added to scavenge oxygen. After core loading and before exceeding 250°F, hydrogen was added to scavenge oxygen during critical operation. After initially establishing chemistry, analysis was performed to verify requirements.
6. Nuclear instrumentation calibration and neutron response	Before core loading	Before core loading, the source range channels were aligned and operational based on data derived from using a neutron source. After a power history had been established on the core, the detector anode and discriminator voltages were reset based on obtained data.
7. Mechanical and electrical tests of incore movable detectors	Before initial criticality and during physics testing	The movable detector systems were checked out in accordance with the operating procedures and ICPs. After core loading and insertion of the detector thimbles, the system was again operationally checked out by ensuring the free passage of detectors into all inserted thimbles. Electrical tests were performed using simulated signals to check out the recorders. During physics measurements the system was operationally checked and limit switches set based on flux mapping data. Incore thermocouples were checked out during hot functional testing (see Table 14.1-1, III, item 4.a).

a. Reference Drawings 1 through 15 show the logics for initiation of turbine and generator trips. The logics show the sensed variables and all available setpoints. See Figure 7.2-1 for a listing of symbols used in these figures.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-2 (continued)

LISTS OF START-UP TESTS

Test or Measurement	Plant Condition/Prerequisite	Test Objective and Summary of Testing
	I. Precritical Tests After Fuel Loading (continued)	
8. Reactor coolant flow measurement	Before initial criticality	After core loading, measurements were made of elbow tap differential pressures to make relative comparison. At hot shutdown conditions after core loading, measurements of loop elbow differential pressure drops were made. Using these data with the reactor coolant pump performance curve, the calculated flow was verified to the design flow. Flow coastdown and transients after reactor coolant pump trips were also determined at shutdown conditions after core loading.
9. Pressurizer effectiveness test	At hot shutdown after core loading	At hot no-load temperature and pressure the effectiveness of the pressurizer heaters in maintaining and increasing system was demonstrated. The heaters were energized and the pressure was compared with an expected pressure rise given in the procedure. The ability of the spray system to reduce pressure was also demonstrated. The spray valves were opened and the pressure decrease compared with the expected pressure decrease given in the procedure.
10. Vibration monitoring on reactor internals	--	No vibration monitoring was done after core loading (refer to test identified in Table 14.1-1, II. item 6)

a. Reference Drawings 1 through 15 show the logics for initiation of turbine and generator trips. The logics show the sensed variables and all available setpoints. See Figure 7.2-1 for a listing of symbols used in these figures.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-2 (continued)

LISTS OF START-UP TESTS

Test or Measurement	Plant Condition/Prerequisite	Test Objective and Summary of Testing
II. Initial Criticality and Low-Power Tests		
1. Initial criticality	Plant at hot shutdown	The objective was to bring the reactor critical for the first time from the plant conditions specified. Before the start of rod withdrawal, the nuclear instrumentation had been aligned and checked, and conservative reactor trip setpoints made per the test procedures. All rods were withdrawn except the last controlling bank, which was left partially inserted for control once criticality was achieved by boron dilution. At preselected points in rod withdrawal and boron dilution, data were taken and inverse count rate ratio pots were made to enable extrapolation to the expected critical point.
2. Radiation surveys	At steady-state conditions during power escalation	Radiation surveys were made during the power escalation to determine dose rate levels at preselected points inside containment due to neutron and gamma radiation. Instruments used were calibrated to known sources, and the calibration rechecked following the survey.
3. Calibration of nuclear instruments with thermal power and determination of overlap	After start-up and during escalation	After initial criticality and during escalation into the intermediate and power ranges, data were taken to verify overlap between the source, intermediate, and power range channels and to verify the alarm and protective functions. These data were collected until the overlaps were firmly established. During low power escalation, the power range detector currents were monitored and compared with the intermediate range currents to verify response of the power range detectors. The power range nuclear channels were calibrated to reactor thermal output based on measurement of secondary plant feedwater flow, feedwater temperature, and steam pressure.

a. Reference Drawings 1 through 15 show the logics for initiation of turbine and generator trips. The logics show the sensed variables and all available setpoints. See Figure 7.2-1 for a listing of symbols used in these figures.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-2 (continued)

LISTS OF START-UP TESTS

Test or Measurement	Plant Condition/Prerequisite	Test Objective and Summary of Testing
	II. Initial Criticality and Low-Power Tests (continued)	
4. Effluent radiation monitors (calibration against known concentration)	Before plant start-up	These instruments were calibrated to a known radiation source or to analog signals which had been calibrated to known radiation sources.
5. Moderator temperature reactivity coefficient	Hot zero power	At normal no-load temperature and no nuclear heating, reactor coolant system cooldown and heatup were accomplished using the steam dump and reactor coolant pumps operation as required. An approximate 5°F change in temperature was initiated, and during these changes the average temperature and reactivity were recorded on an X-Y plotter. From these data the moderator temperature coefficient was determined.
6. Pressure reactivity coefficient measurements	--	Direct measurements of the pressure coefficient of reactivity were not made, since the effects of pressure on reactivity are of second order when compared with other effects.
7. Control rod reactivity worth determination of differential and integral worth and verification of worth for shutdown capability	Hot zero power	Under zero-power conditions at near operating temperature and pressure, the nuclear design predictions for rod cluster control assembly (RCCA) group differential worths were validated. These validations were made from boron concentration sampling data, RCCA bank positions, and recorder traces of reactivity. From this data the integral RCCA group worths were determined, including verification of rod insertion limits to ensure adequate shutdown margin. The minimum boron concentration for maintaining the reactor shutdown with the most reactive rod cluster control assembly stuck in the full-out position was determined for Unit 1. The determination was made from analysis of boron concentration and RCCS worths.

a. Reference Drawings 1 through 15 show the logics for initiation of turbine and generator trips. The logics show the sensed variables and all available setpoints. See Figure 7.2-1 for a listing of symbols used in these figures.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-2 (continued)

LISTS OF START-UP TESTS

Test or Measurement	Plant Condition/Prerequisite	Test Objective and Summary of Testing
	II. Initial Criticality and Low-Power Tests (continued)	
8. Boron reactivity worth measurement	Zero power	Differential boron worth measurements were made by monotonically increasing or decreasing reactor coolant boron concentration. Compensation for the reactivity effect of the boron concentration change was made by withdrawing or inserting respective control rods to maintain moderator average temperature and power level constant and observing the resultant accumulated change in core reactivity corresponding to these successive rod movements.
9. Determination of boron concentration of initial criticality and reactivity allocation	Zero power	These determinations are described under II, item 1 above.
10. Flux distribution measurement with normal rod patterns	Zero power	Flux distribution measurements with normal rod patterns were taken during the zero-power physics tests.
11. Chemical tests to demonstrate ability to control water quality	Before criticality and during power escalation	Before criticality, the procedures and equipment for performing chemical analyses of primary and secondary systems were demonstrated. During power escalation, sampling was performed and analysis done to verify that plant chemistry was within specifications.
12. Pseudo-rod-ejection test, to verify safety analysis (hot)	Zero power	Incore measurements were made for Unit 1 under pseudo-ejected-rod conditions simulating the zero-power accident to determine the hot-channel factors and verify that they were within assumptions made in the accident analysis.

a. Reference Drawings 1 through 15 show the logics for initiation of turbine and generator trips. The logics show the sensed variables and all available setpoints. See Figure 7.2-1 for a listing of symbols used in these figures.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-2 (continued)

LISTS OF START-UP TESTS

Test or Measurement	Plant Condition/Prerequisite	Test Objective and Summary of Testing
		III. Power Ascension Tests
1. Natural circulation test to confirm sufficient cooling capacity	--	The ability of natural circulation to remove decay heat has been demonstrated at the Carolina Power & Light Company's H. B. Robinson Unit 2. Tests have shown natural circulation flow to be more than adequate to remove decay heat, and such a test was not repeated on North Anna Unit 1. However, special tests were conducted for Unit 2, as described in Section 14.1.4.
2. Power reactivity coefficient evaluation and power defect measurements (30, 50, 75 and 100%)	During power escalation	During each power escalation for Unit 1, recorder traces were made of reactor power and reactivity changes. From these traces, the power coefficient of reactivity and power defects were determined.
3. Plant response to load swings, including automatic control system checkout (30, 50, 75 and 100%)	During power escalation	Plant response to the following load changes was demonstrated: a. ±10% step load change from 30, 75 and 100% power. b. 50% load reduction from 75 and 100% power. c. Plant trip from 100% power level.

a. Reference Drawings 1 through 15 show the logics for initiation of turbine and generator trips. The logics show the sensed variables and all available setpoints. See Figure 7.2-1 for a listing of symbols used in these figures.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-2 (continued)

LISTS OF START-UP TESTS

Test or Measurement	Plant Condition/Prerequisite	Test Objective and Summary of Testing
III. Power Ascension Tests (continued)		
		The data collected from the performance of these tests were analyzed for control system behavior and requirements for realignment. Acceptance criteria, such as the plant not tripping (where applicable), relief and safety valves not lifting, and steam dump operating correctly, were identified in the individual procedures. At approximately 30% power, the automatic control systems were checked by initiating a perturbation and observing controller response. During the transient tests these systems were operationally checked under actual design load changing conditions.
4. Chemical analysis (30, 50, 75, 100%)	During power escalation	During low-power physics tests and at 30, 50, 75, and 100% power, samples of reactor coolant were taken and analysis performed to verify that coolant chemistry requirements could be maintained.
5. Effluents and effluent monitoring systems (30, 50, 75, 100%)	During power escalation	Installed effluent monitors were operated continuously at selected locations in the plant to monitor for radioactive constituents in the effluents. Instruments detected any changes in activity and alerted the operator when radiochemical analysis should be performed
6. Evaluation of core performance (30, 50, 75, 100%)	During power escalation	At steady-state power points, incore data were obtained and analysis performed to verify that the core performance margins were within design predictions, for expected normal and abnormal rod configurations.
7. Loss of flow	Before criticality	Reactor coolant system response to loss of flow for various combinations of pump trips was determined from hot shutdown conditions.

a. Reference Drawings 1 through 15 show the logics for initiation of turbine and generator trips. The logics show the sensed variables and all available setpoints. See Figure 7.2-1 for a listing of symbols used in these figures.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-2 (continued)

LISTS OF START-UP TESTS

Test or Measurement	Plant Condition/Prerequisite	Test Objective and Summary of Testing
	III. Power Ascension Tests (continued)	
8. Turbine trip (100%)[a]	At power	This test verified that the pressurizer safety valves did not lift and that the plant could be maintained in a hot shutdown condition. The turbine trip from 100% power was conducted as an integral part of the generator trip from 100% power.
9. Generator trip (100%)[a]	At power	Generator trip was performed at 100% power to verify the plant's capability of withstanding an instantaneous reduction in load from 100 to 0%. A generator trip was initiated by manually opening the main generator breakers. This would be automatically initiate a turbine trip.
10. Shutdown from outside the control room	Greater than or equal to 10% generator power	The ability to bring the plant to and maintain the plant in hot shutdown conditions after a trip from greater than or equal to 10% power was demonstrated using instrumentation and controls outside the control room.
11. Loss of offsite power	Greater than or equal to 10% generator power	Tests were performed in which loss of voltage was simulated. Starting of the diesels and connecting of the emergency loads on the emergency bus was demonstrated.
12. Radiation surveys and shielding effectiveness (50 and 100%)	At power	The surveys to determine the effectiveness of the shielding have been discussed under Radiation Survey (Item II.2). These surveys were conducted up to and including 100% power.
13. Part-length rod insertion/withdrawal (75%)	Approximately 75% power	Technical Specifications required part-length rods for Unit 1 to remain fully withdrawn; therefore no testing was performed. No part-length rods were installed in Unit 2.

a. Reference Drawings 1 through 15 show the logics for initiation of turbine and generator trips. The logics show the sensed variables and all available setpoints. See Figure 7.2-1 for a listing of symbols used in these figures.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-2 (continued)

LISTS OF START-UP TESTS

Test or Measurement	Plant Condition/Prerequisite	Test Objective and Summary of Testing
		III. Power Ascension Tests (continued)
14. Dropped rod-effectiveness of instruments to detect dropped rod and verification of associated automatic action	Greater than or equal to 50% reactor power	Automatic turbine runback and rod withdrawal stop is not necessary as a result of a dropped rod, and the circuitry required for such action does not exist in this plant. Rod drop tests based on common failure criteria were performed dynamically to demonstrate the negative rate trip function from greater than or equal to 50% reactor power.
15. Vibration measurements on reactor internals (30, 50, 75, and 100%)	--	These measurements were not performed at power. Refer to II, item 6, above for measurements before operation.
16. Pseudo-rod-ejection test to verify safety analysis	30% power	Incore measurements were made for Unit 1 with individual rods withdrawn out of bank position to determine the resulting hot-channel factors and verify that they are within expected limits. These determinations were made from movable detector and thermocouple data. This measurement was not performed on Unit 2 because of the negligible worth of a control rod withdrawn from its full-power insertion limit, and the large magnitude of margin remaining to hot-channel factor limits in the ejected-rod configuration (see Section 14.1.3). This was verified by the test performed for Unit 1.
17. Evaluation of flux asymmetry	50% power	Incore flux measurements were made with a single rod assembly moving partially below bank position and fully inserted to demonstrate that core limits are not exceeded.

a. Reference Drawings 1 through 15 show the logics for initiation of turbine and generator trips. The logics show the sensed variables and all available setpoints. See Figure 7.2-1 for a listing of symbols used in these figures.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-2 (continued)

LISTS OF START-UP TESTS

Test or Measurement	Plant Condition/Prerequisite	Test Objective and Summary of Testing
	III. Power Ascension Tests (continued)	
18. Process computer (30, 50, 100%)	During power escalation	When available during power escalation, the process computer was checked out and comparisons made between process signals and those assessed by the process computer. (No safety-related functions are performed by the computer.)
19. Moisture carryover measurement	At power	Radioactive tracer of sodium-24 was injected into the steam generators. Samples obtained from the steam generator upper shell, main steam line taps, and the feedwater system were analyzed.

a. Reference Drawings 1 through 15 show the logics for initiation of turbine and generator trips. The logics show the sensed variables and all available setpoints. See Figure 7.2-1 for a listing of symbols used in these figures.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-3

UNIT 2 START-UP PHYSICS TESTS

I. Hot Zero Power Tests

1. Reactivity computer checkout.
2. Isothermal temperature coefficient at ARO and D bank in (also D, C banks in if MTC for D bank in is greater than or equal to 0 pcm/°F).
3. Boron endpoints at ARO; D bank in; D, C banks in; D, C, B banks in; D, C, B, A banks in; shutdown bank B in with all other rods out; and shutdown bank A in with all other rods out.
4. Reactivity worths of all control and shutdown rod banks.
5. Boron worth over the range of control banks A through D moving during rod insertion and withdrawal.
6. Power distribution measurements for ARO and D bank in.

II. Power Ascension Tests

1. 30% power flux map.
2. 50% power flux map.
3. Pseudo-dropped-rod test (RCCA D-10) and associated power distribution measurements at 50% power.
4. Incore/ex-core detector calibration flux maps at 75% power.
5. APDMS flux maps at or below 95% power.
6. Flux maps at 90% and 100% power (equilibrium conditions).

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-4

PHYSICS TESTS THAT HAVE BEEN DELETED FOR UNIT 2

I. Hot Zero Power Tests

1. Isothermal temperature coefficient at D, C banks in; D, C, B banks in; and D, C, B, A banks in.
2. Boron endpoint for the N-1 rods-in configuration.
3. Reactivity worth of N-1 rods.
4. Pseudo-rod-ejection and associated power distribution measurements.

II. Power Ascension

1. Pseudo-rod-ejection and associated power distribution measurement at 30% power.
2. Pseudo-dropped-rod test (RCCA H-6) and associated power distribution measurement.
3. Power coefficients.
4. Integral power defect.
5. Doppler-only power coefficients.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-5

SUMMARY OF UNIT 1 MEASURED VALUES, DESIGN VALUES, DESIGN TOLERANCE, AND ACCIDENT ANALYSIS CRITERIA FOR PHYSICS TESTS THAT HAVE BEEN DELETED FOR UNIT 2

Test Description	Core Condition	Parameter	Unit 1 Measured Value	Design Value (beginning of life (BOL), Best Estimate)	Design Tolerance	Accident Analysis Criterion
1. Isothermal temperature coefficient	Banks D, C in	α_T	-7.86 pcm/°F	-8.9 pcm/°F	±3 pcm/°F	≤ -2.107 pcm/°F
	Banks D, C, B in	α_T	-13.48 pcm/°F	-14.1 pcm/°F	±3 pcm/°F	≤ -2.134 pcm/°F
	Banks D, C, B, A in	α_T	-14.07 pcm/°F	-13.8 pcm/°F	±3 pcm/°F	≤ -2.135 pcm/°F
2. Boron endpoint	N-1 rods inserted	C_B	601 ppm	580 ppm	±50 ppm	$\alpha \times C_B \leq 24000$ pcm where = α 11.08 pcm/ppm
3. Rod worth	N-1 rods	I_{N-1} [a]	8015	7893 pcm	±789 pcm	$(I_{N-1})/1.04 \geq 5780$ pcm
4. Pseudo-ejected control rod	hot zero power (HZP), Bank C at 120 steps, Bank D at 0 steps, RCCA B-8 at 228 steps.	F_Q	6.85	10.8	NA	13.0
		I_{B-8} [a]	443 pcm	464 pcm	±46 pcm	$(I_{B-8}) \times 1.04 \leq 785$ pcm
	30% power, Bank D at 194 steps, RCCA B-8 at 228 steps.	F_Q	2.1	2.1	NA	7.07
		I_{B-8}	3 pcm	7 pcm	±1 pcm [b]	$(I_{B-8}) \times 1.04 \leq 200$ pcm
5. Pseudo-dropped control rod	50% power, RCCA H-6	$F^N_{\Delta H}$	1.62	1.70	NA	1.69 [c]
		I_{H-6} [a]	138 pcm	146 pcm	±22 pcm	$(I_{H-6}) \times 1.04 \leq 250$ pcm

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-5 (continued)

SUMMARY OF UNIT 1 MEASURED VALUES, DESIGN VALUES, DESIGN TOLERANCE, AND ACCIDENT ANALYSIS CRITERIA FOR PHYSICS TESTS THAT HAVE BEEN DELETED FOR UNIT 2

Test Description	Core Condition	Parameter	Unit 1 Measured Value	Design Value (beginning of life (BOL), Best Estimate)	Design Tolerance	Accident Analysis Criterion
6. Power coefficient	30% power	$(\partial\rho/\partial Q)_{power}$	-15.24 pcm/% power	-14.02 pcm/% power	±4.57 pcm/% power	NA
	50% power	$(\partial\rho/\partial Q)_{power}$	-12.74 pcm/% power	-13.75 pcm/% power	±3.82 pcm/% power	NA
	75% power	$(\partial\rho/\partial Q)_{power}$	-13.57 pcm/% power	-13.39 pcm/% power	±4.07 pcm/% power	NA
	90% power	$(\partial\rho/\partial Q)_{power}$	-10.70 pcm/% power	-13.31 pcm/% power	±3.21 pcm/% power	NA
7. Power defect	0 - 100% power	Reactivity worth	1270 pcm	1299 pcm	±191 pcm	NA
8. Doppler-only power coefficient	30% power	$(\partial\rho/\partial Q)_{Doppler}^{inferred}$	-13.62 pcm/% power	-11.35 pcm/% power	±4.09 pcm/% power	Inferred value ±30% uncertainty must overlap allowance range of Figure 15.1-3
	50% power	$(\partial\rho/\partial Q)_{Doppler}^{inferred}$	-10.77 pcm/% power	-10.75 pcm/% power	±3.23 pcm/% power	Inferred value ±30% uncertainty must overlap allowance range of Figure 15.1-3

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 14.1-5 (continued)

SUMMARY OF UNIT 1 MEASURED VALUES, DESIGN VALUES, DESIGN TOLERANCE, AND ACCIDENT ANALYSIS CRITERIA FOR PHYSICS TESTS THAT HAVE BEEN DELETED FOR UNIT 2

Test Description	Core Condition	Parameter	Unit 1 Measured Value	Design Value (beginning of life (BOL), Best Estimate)	Design Tolerance	Accident Analysis Criterion
8. Doppler-only power coefficient (continued)	75% power	$(\partial\rho/\partial Q)^{inferred}_{Doppler}$	-11.08 pcm/% power	-9.96 pcm/% power	±3.32 pcm/% power	Inferred value ±30% uncertainty must overlap allowance range of Figure 15.1-3
	90% power	$(\partial\rho/\partial Q)^{inferred}_{Doppler}$	-7.59 pcm/% power	-9.38 pcm/% power	±2.28 pcm/% power	All inferred values fell within range of Figure 15.1-3, as shown on Figure 14.1-3

a. I_{N-1} = integrated reactivity worth of all control rods except the most reactive rod (N-1).
I_{B-8} = integrated reactivity worth of rod cluster control assembly (RCCA) B-8.
I_{H-6} = integrated reactivity worth of RCCA H-6.

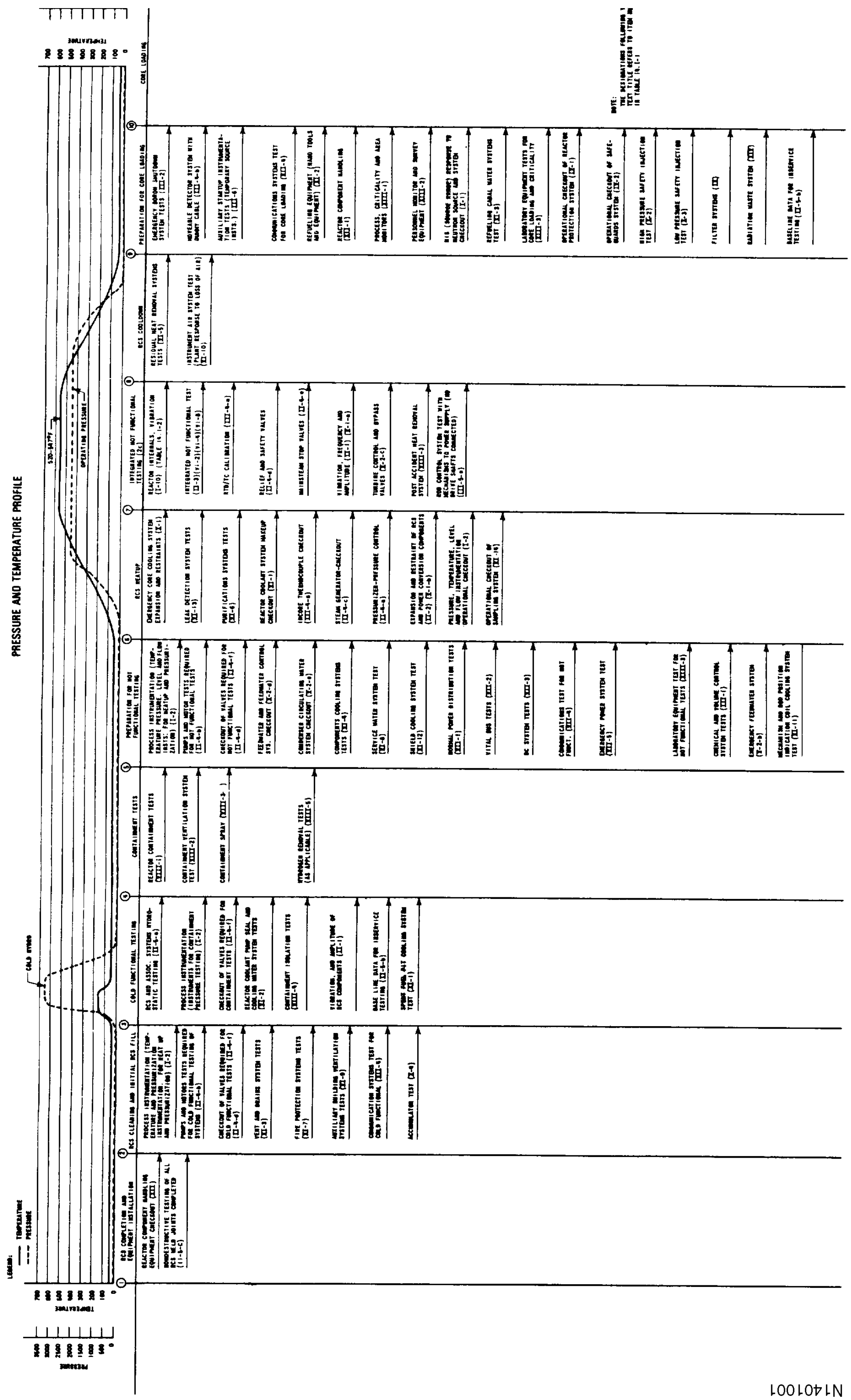
b. Violation of design tolerance was evaluated to be insignificant due to low value of reactivity worth.

c. Accident analysis referenced to hot full power.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Figure 14.1-1

TYPICAL PRE-OPERATIONAL TEST SEQUENCE



The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

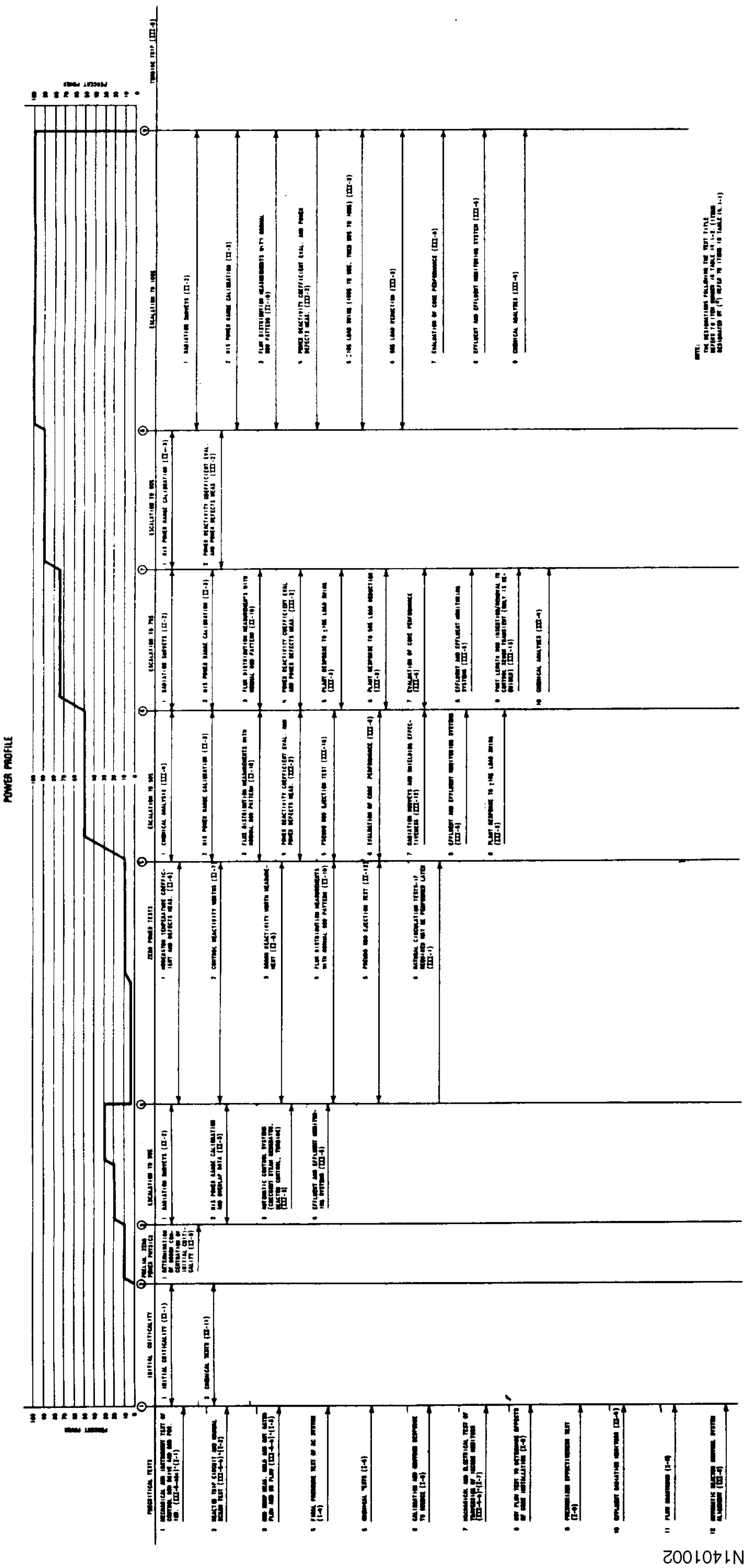
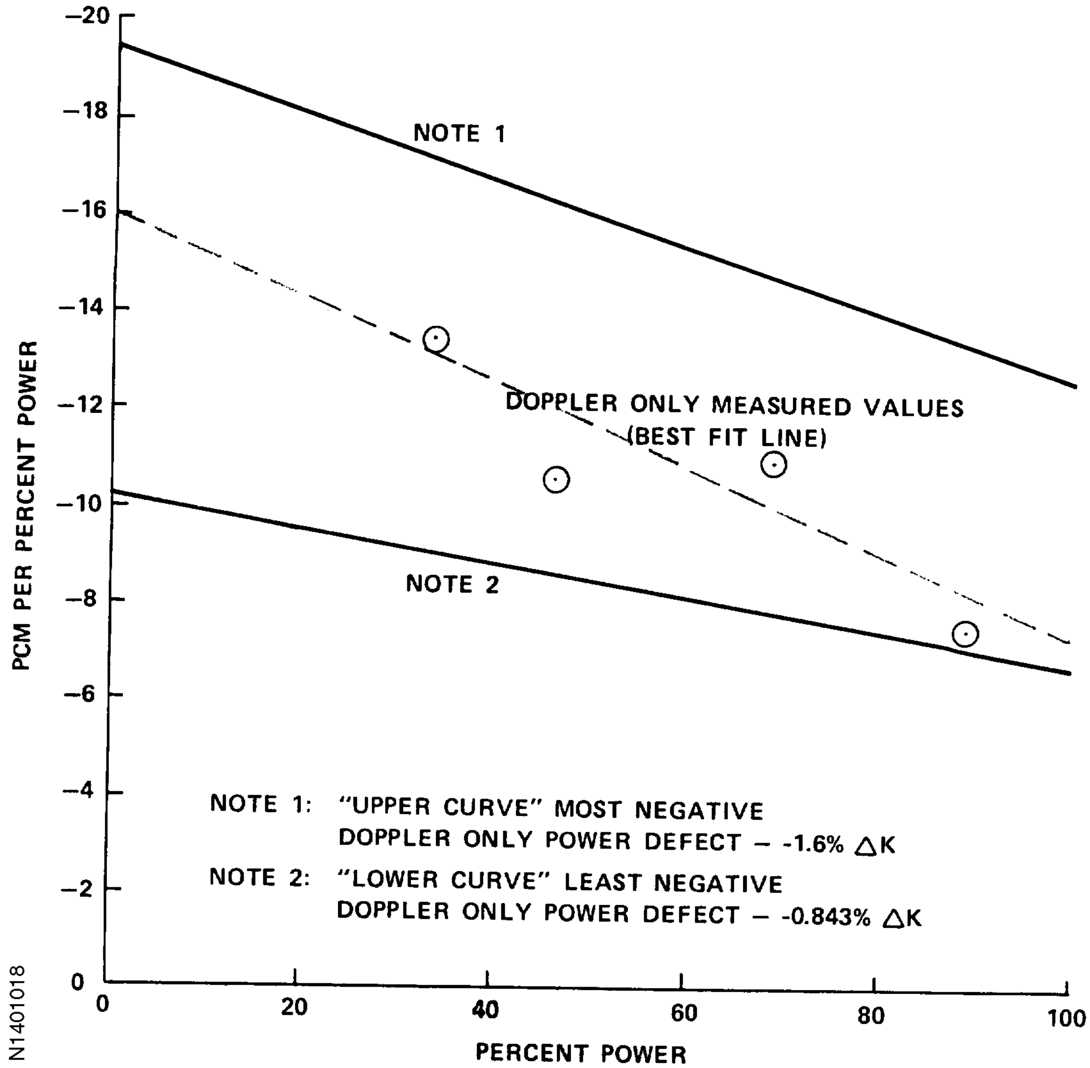


Figure 14.1-2
TYPICAL START-UP TEST SEQUENCE

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Figure 14.1-3
CYCLE 1 BOL PHYSICS TEST DOPPLER POWER COEFFICIENT USED IN ACCIDENT ANALYSIS UNIT 1



14.2 AUGMENTATION OF VEPCO'S STAFF FOR INITIAL TEST AND OPERATION

The start-up organization used during the period of initial operation consisted of all personnel in the Station Operations Department, with additional support provided by the Engineering Services Department and the Chemistry and Health Physics Department.

In operations, there were, for one-unit operation, five senior licensed shift supervisors, five licensed control room operators, and a combined group of 14 assistant control room operators and auxiliary operators with a minimum of two licensed assistant control room operators. For two-unit operations, there were five senior licensed shift supervisors, five senior licensed assistant shift supervisors, 10 licensed control room operators, and a combined group of 18 assistant control room operators and auxiliary operators with a minimum of six licensed assistant control room operators. Shifts were scheduled to ensure that a minimum of three licensed reactor operators and two licensed senior reactor operators are on duty at all times during two-unit operations.

Technical support was provided during start-up using the services of graduate-level engineers. A trained power engineer was assigned through the architect-engineer to assist in preliminary operations for both units. In addition, engineering representatives were assigned at the station by the supplier of the nuclear steam supply system to render start-up support.

VEPCO had overall responsibility during plant start-up, including precriticality tests, approach to criticality, and postcriticality operations. The station staff was assisted by the architect-engineer and the supplier of the nuclear steam supply system. The Stone & Webster start-up engineer was assigned to the station from the start of flushing operations through commercial operation. The start-up engineer reported directly to the Station Manager and received instructions from him.

Experienced Westinghouse reactor engineers were also assigned to the station for fuel loading, initial criticality, and physics testing. These reactor engineers were qualified and knowledgeable in reactor operations.

They reported directly to the VEPCO reactor engineers and received instructions from them. The Westinghouse reactor engineers acted in an advisory capacity only; VEPCO retained responsibility for, and control of, the unit. Reactor specialists (e.g., control engineers) were available and utilized as required.

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Appendix 14A
NRC Questions and VEPCO's Responses
Regarding the North Anna Power Station Unit 2
Modified Startup Physics Testing Program

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APPENDIX 14A
NRC QUESTIONS AND VEPCO'S RESPONSES REGARDING THE NORTH ANNA POWER STATION UNIT 2 MODIFIED STARTUP PHYSICS TESTING PROGRAM

Question 1

With respect to Item B.3 on Table 14.1-3 and Item B.2 on Table 14.1-4, what is the reason for performing the pseudo-dropped-rod test with RCCA D-10 instead of RCCA H-6?

Response

This test was performed twice for Unit 1, once using RCCA D-10 and once using RCCA H-6. The evaluation of the results associated with these tests indicated that of the two rods, RCCA D-10 resulted in the more limiting radial power distribution and consequently had the minimum margin to departure from nucleate boiling. This was the basis for choosing RCCA D-10 for the Unit 2 test.

Question 2

Describe any known differences between the fuel and core of North Anna Unit 1 and the fuel and core of North Anna Unit 2.

Response

Three differences have been identified between the Unit 1 and Unit 2 fuel and core. They are:

1. The location of the secondary sources within the core.
2. The part-length control rods have been removed.
3. The fuel rods have been prepressurized to a different pressure.

The core locations of the secondary sources for Unit 1 and Unit 2 are shown in Figure 14A-1. It is not expected that this change will lead to a measurable difference in the physics characteristics between the Unit 1 and Unit 2 cores.

The core location of the part-length control rods for Unit 1 is shown in Figure 14A-2. This change will not lead to a measurable difference in the physics characteristics between the Unit 1 and Unit 2 cores because the use of the part-length control rods is not permitted during Unit 1 core operation. It is planned that these rods will be removed from Unit 1 following the end of Cycle 1 operation.

Unit 2 fuel has a prepressurization value that is approximately 50 psi lower than that used for Unit 1 fuel. This difference will have no perceptible effect on the physics characteristics of the

core. In addition, an evaluation has shown that there will be no adverse impact on fuel or core performance.

Question 3

Describe any differences between the physics test methods that were used for Unit 1 and the physics test methods that will be used for Unit 2.

Response

For Unit 1, the reactivity worth of shutdown bank B was determined with control banks A through D fully inserted into the core, using the dilution-boration technique. The worth of shutdown bank A was determined with control banks A through D and shutdown bank B fully inserted into the core. Shutdown bank A underwent an exchange with the most reactive rod (RCCA B-8) followed by a dilution of the reactor coolant system in order to fully insert shutdown bank A.

For Unit 2, the reactivity worths of shutdown banks A and B will be determined individually with all other control rod banks out of the core. The worth of shutdown bank B will be determined using the conventional dilution/boration technique. A boron endpoint determination will be made for this control rod configuration. The worth of shutdown bank A will be determined by using rod exchange with one of the rod banks, and if necessary, dilution/boration of the reactor coolant system in order to reach the desired state point, i.e., shutdown bank A fully inserted with all other rods out. A boron endpoint determination will be made for this control rod configuration.

Question 4

For each of the start-up physics tests that were performed for Unit 1 but are not going to be performed for Unit 2, give the technical basis for not performing those tests.

Response

The tests that are not going to be performed for Unit 2 are listed in Table 14.1-4. Deletion of these physics tests from the start-up program is justified for the following reasons:

1. The successful performance of the abbreviated program is sufficient to
   a. Verify that the core was correctly loaded and that there are no anomalies present that could cause problems later in the cycle.
   b. Verify that the calculational model that has been used will correctly predict core behavior during the cycle.
   c. Verify the reactivity worth of the control rod banks.

   d. Provide data for nuclear instrumentation calibration.

   e. Demonstrate the sensitivity of this instrumentation to abnormal core conditions.

2. The calculation model was verified as a result of the Unit 1 start-up.

3. The fuel and core characteristics of Unit 2 are virtually identical to those of Unit 1, and the results obtained for these tests during the Unit 1 start-up demonstrated that a large margin exists between the measured parameter values and the design values used in the safety analyses.

Each of the tests that are not going to be performed for Unit 2 is listed below, together with the specific technical basis for not performing these tests as part of the Unit 2 start-up physics testing program.

A1. Isothermal Temperature coefficient at D, C banks in, D, C, B banks in, and D, C, B, A banks in

The Core Operating Limits Report will require that a nonpositive value for the moderator temperature coefficient be maintained during normal operation. Based on the results of design calculations and the Unit 1 moderator temperature coefficient test results, it is expected that performance of the moderator temperature coefficient tests with all rods out and with control bank D in will be adequate to demonstrate a nonpositive moderator temperature coefficient value and also provide enough data to establish control rod withdrawal limits, should they be necessary. The successful completion of these tests will verify the design model used to predict the isothermal temperature coefficient values. Additionally, review criteria have been developed for the Unit 2 isothermal temperature coefficient tests that are based on the Unit 1 test results (Table 14A-1). The acceptability of the Unit 2 tests with respect to these review criteria will further demonstrate the similarity between the Unit 1 and Unit 2 cores. All of the Unit 1 measured temperature coefficient values were acceptable. The Unit 1 test results and review criteria for the isothermal temperature coefficients at D, C banks in, D, C, B banks in, and D, C, B, A banks in are listed in Table 14.1-5, Item 1.

A2. Boron Endpoint for the N-1 rods-in configuration

Boron endpoint measurements will be made following each of the rod worth tests. The successful completion of these tests will verify the design model used to predict the boron endpoint values. Additionally, review criteria have been developed for the Unit 2 boron endpoint measurements that are based on the Unit 1 test results (Table 14A-1). The acceptability of the Unit 2 tests with respect to these review criteria will further demonstrate the similarity between the Unit 1 and Unit 2 cores. All Unit 1 measured endpoint values were acceptable. The Unit 1 test results and review criterion for the boron endpoint for the N-1 rods-in configuration are listed in Table 14.1-5, Item 2.

A3. Reactivity worth of N-1 rods

As described in the response to Question 3, the reactivity worth of the control and shutdown rod banks will be measured as part of the physics testing program. The successful completion of these measurements will verify the design models used to predict the reactivity worth of the rod banks. Additionally, review criteria have been developed for the Unit 2 rod bank reactivity worth tests that are based on the Unit 1 test results (Table 14A-1). The acceptability of the Unit 2 tests with respect to these review criteria will further demonstrate the similarity between the Unit 1 and Unit 2 cores. All Unit 1 measured rod worth values (including the reactivity worth of N-1 rods) were acceptable, and demonstrated that a large margin existed with respect to the shutdown margin limit. The Unit 1 test results and review criterion for the reactivity worth of N-1 rods are listed in Table 14.1-5, Item 3.

A4. Pseudo-rod-ejection and associated power distribution measurements (HZP)

The successful completion of the rod bank reactivity worth measurements for the four control banks and the two shutdown banks for Unit 2 will verify the design model used to calculate rod worths. Since the same design model is used to predict all rod worths, including the worth of an ejected rod, additional verification of the design model is not required. Additionally, review criteria have been developed for the Unit 2 rod bank reactivity worth tests that are based on the Unit 1 test results (Table 14A-1). The acceptability of the Unit 2 tests with respect to these review criteria will further demonstrate the similarity between the Unit 1 and Unit 2 cores. The Unit 1 test results and review criteria for the pseudo-rod-ejection and associated power distribution measurements (HZP) are listed in Table 14.1-5, Item 4. These results indicated that the measured rod worth value and the heat flux hot-channel factor value were acceptable, and demonstrated a large margin with respect to the values used in the safety analysis.

B1. Pseudo-rod-ejection and associated power distribution measurement at 30% power

The successful completion of the rod bank reactivity worth measurements for the four control banks and the two shutdown banks for Unit 2 will verify the design model used to calculate rod worths. Since the same design model is used to predict all rod worths, including the worth of an ejected rod, additional verification of the design model is not required. Additionally, review criteria have been developed for the Unit 2 rod bank reactivity worth tests that are based on the Unit 1 test results (Table 14A-1). The acceptability of the Unit 2 tests with respect to these review criteria will further demonstrate the similarity between the Unit 1 and Unit 2 cores. The Unit 1 test results and review criteria for the pseudo-rod-ejection and associated power distribution measurement at 30% power are listed in Table 14.1-5, Item 4. These results indicated that the measured rod worth value and the heat flux hot-channel factor value were acceptable, and demonstrated a large margin with respect to the values used in the safety analysis.

B2. Pseudo-dropped-rod test (RCCA H-6) and associated power distribution measurement

As described in the response to Question 1, this test was performed twice for Unit 1, once using RCCA D-10 and once using RCCA H-6. For Unit 2, this test will be performed using the limiting rod, RCCA D-10. The successful completion of this test will verify the design models and demonstrate margin to the values used in the safety analysis. Additionally, the use of review criteria that are based on Unit 1 test results for other Unit 2 tests (Table 14A-1) will further demonstrate the similarity between the Unit 1 and Unit 2 cores. The Unit 1 results for both dropped rod tests were acceptable and demonstrated margin with respect to the values used in the safety analysis. The Unit 1 test results and review criteria for the pseudo-dropped-rod test and associated power distribution measurement using RCCA H-6 are listed in Table 14.1-5, Item 5.

B3. Power coefficient tests

B4. Integral power defect

B5. Doppler-only power coefficients

The successful completion of the isothermal temperature coefficient tests, the boron endpoint measurements, and the rod bank reactivity tests, together with the acceptability of these tests with respect to their respective review criteria, will service to further demonstrate the similarity between the Unit 1 and Unit 2 cores. As indicated in Table 14.1-5, Items 6, 7, and 8, the Unit 1 measured values for the total power coefficient, the integral power defect, and the Doppler-only power coefficient verified the design models used to predict the values of these parameters and were acceptable with respect to the values used in the safety analyses. An additional description of the Unit 1 test results and their evaluation has been provided in a letter from Mr. C. M. Stallings, VEPCO, to Mr. H. R. Denton, USNRC, Ser. No. 169, dated March 20, 1979.

In summary, this information provides a sufficient technical basis for the deletion of these tests from the Unit 2 physics testing program.

Question 5

It is suggested that review criteria be established, where appropriate, to compare the Unit 2 test results with the Unit 1 test results for the isothermal temperature coefficient measurements, the boron endpoint measurements, the rod bank reactivity worth measurements, and the boron worth measurement. For each of these tests, list the specific review criteria that will be used. Also, indicate the action that will be taken if the review criteria are not met.

Response

The tests and the specific test review criteria that will be used are listed on Table 14A-1. As described in the response to Question 3, the reactivity worth and boron endpoint measurements associated with shutdown banks A and B that will be performed for Unit 2 are not direct

duplicates of the tests that were performed during the Unit 1 testing program. Therefore, review criteria based on Unit 1 test results would be inappropriate. The results of these tests will be reviewed, instead, with respect to design values (best-estimate predictions) and the standard design tolerances.

As stated in the main body of the FSAR and as required by the VEPCO Nuclear Power Station Quality Assurance Manual, test results will be reviewed and evaluated by the Station Nuclear Safety and Operating Committee. Should the results of any of these tests fail to meet the review criteria, the Committee may decide to perform additional testing. This additional testing may be a repeat of the original test or may be the performance of a test that had been deleted from the Unit 2 physics testing program. In addition, the NRC Region II Resident Inspector will be notified verbally in a timely manner, and a report will be sent to Nuclear Reactor Regulation (NRR).

Table 14A-1
UNIT 2 ISOTHERMAL TEMPERATURE COEFFICIENT, BORON ENDPOINT, ROD WORTH REACTIVITY, AND BORON WORTH TESTS AND REVIEW CRITERIA

	Review Criteria	
Test Description	Unit 1 Measured Value	Tolerance
Isothermal temperature coefficient		
ARO	0.98 pcm/°F	±2 pcm/°F
D bank in	-4.29 pcm/°F	±2 pcm/°F
Boron endpoint		
ARO	1322 ppm	±24 ppm
D bank in	1193 ppm	±24 ppm
D, C banks in	1075 ppm	±24 ppm
D, C, B banks in	884 ppm	±24 ppm
D, C, B, A banks in	781 ppm	±24 ppm
Shutdown bank A in	1220 ppm (design)	±21 ppm
Shutdown bank B in	1224 ppm (design)	±20 ppm
Control rod worth		
D bank	1463 pcm	±100 pcm
C bank	1303 pcm	±98 pcm
B bank	2036 pcm	±153 pcm
A bank	1309 pcm	±98 pcm
Total D through A	6111 pcm	±306 pcm
Shutdown bank A	1114 pcm (design)	±111 pcm
Shutdown bank B	1043 pcm (design)	±104 pcm
Boron worth		
ARO through A bank	$11.08 \frac{\text{pcm}}{\text{ppm}}$	$\pm 0.55 \frac{\text{pcm}}{\text{ppm}}$

Figure 14A-1
SECONDARY SOURCE LOCATIONS

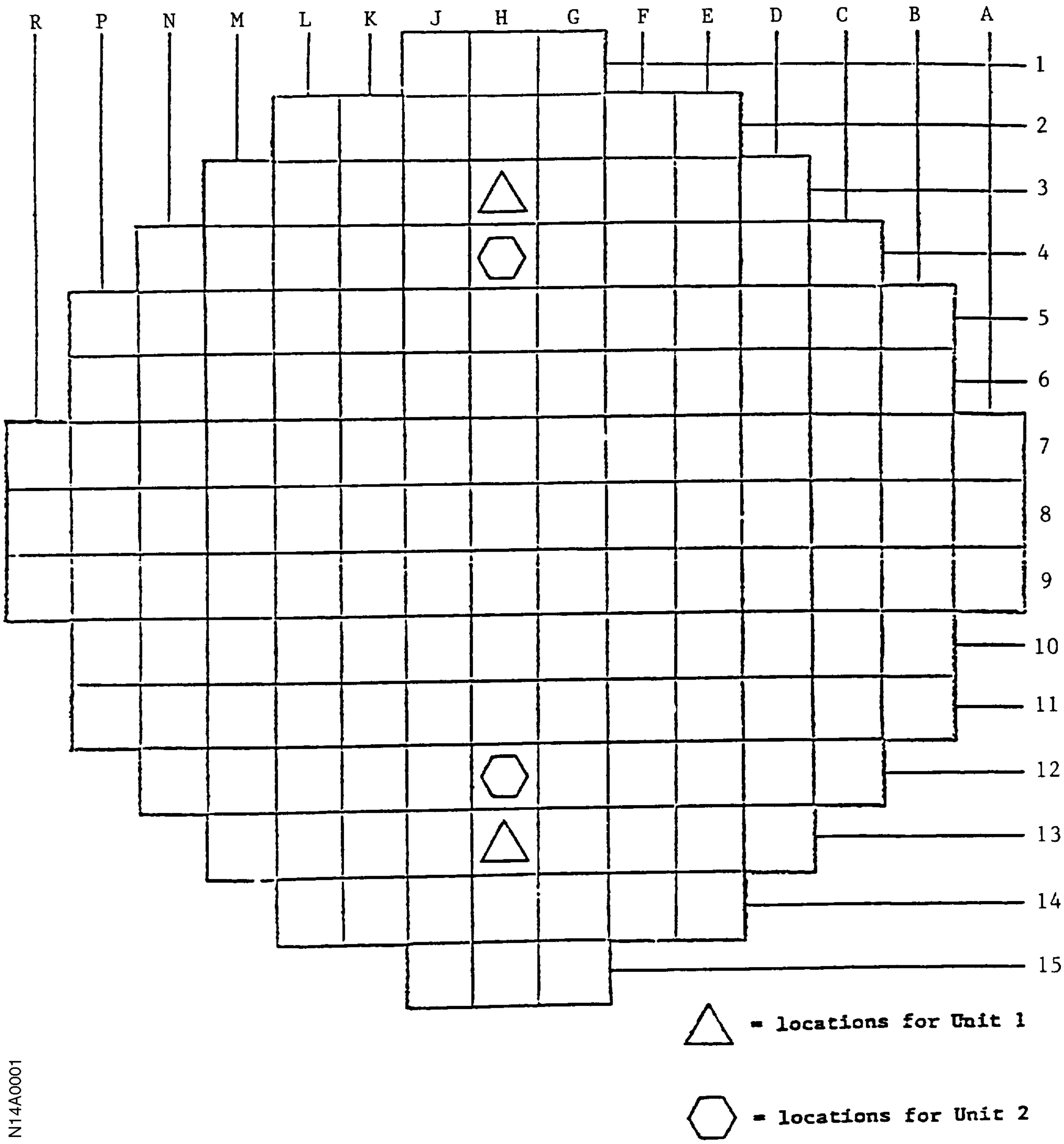
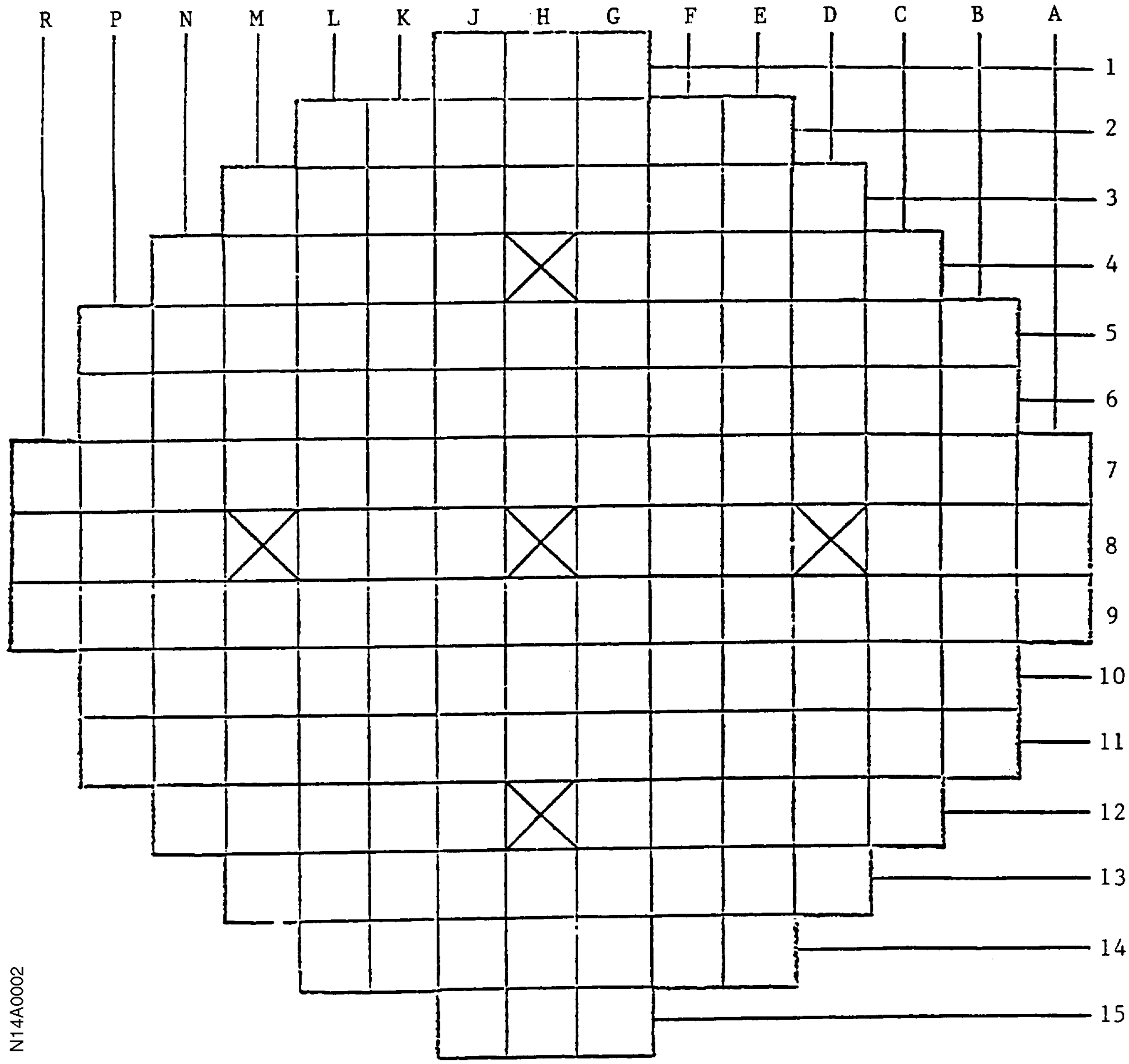


Figure 14A-2
PART LENGTH CONTROL ROD LOCATIONS (UNIT 1)



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