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CONDUCT OF OPERATIONS

13.1 SUMMARY DESCRIPTION

The Exelon Generation Company, LLC (EGC) Nuclear Group (NG) is responsible for all plant operations from the start of initial core loading and for utilizing properly licensed personnel to operate the plant. Technical assistance and direction during the initial core loading, startup, and pre-commercial testing was provided by General Electric Company and Bechtel Corporation. Technical assistance will be available as required during plant operation.

The Plant Manager is responsible for the safe, reliable, and efficient operation of the plant. He has a staff of trained and properly licensed personnel to accomplish all of the various plant functions and disciplines. Plant operations are performed in accordance with written and approved operating and emergency procedures. These procedures take into account available pertinent experience encountered in the startup and operation of earlier BWR plants. Significant operations, tests, and information are recorded and a file of these records is maintained.

A training program has been established to provide plant personnel with the knowledge required for safe and efficient operation and maintenance of the station. In addition, a program of continuing training will assure that all operation personnel maintain their level of knowledge and that replacement personnel are fully qualified.

The pre-operational test program was designed to confirm the completeness of construction and to adjust and calibrate the equipment to the extent possible. This program further assured that all process and safety equipment is operational and in compliance with license requirements to the extent possible and necessary to proceed into initial fuel loading and the startup program.

During the startup and the power test program, the reactor fuel was loaded and system temperature, pressure, and power output increased to rated values according to a carefully planned and thoroughly documented procedure.

13.2 ORGANIZATIONAL STRUCTURE

The Exelon Generation Company, LLC (EGC) Nuclear Group (NG) is responsible for the safe, reliable, and efficient operation of the nuclear facilities. In addition, the NG is responsible for appropriate standards, programs, processes, management controls, and support for the nuclear facilities. In keeping with these responsibilities, the NG is committed to providing sufficient personnel having appropriate qualifications to both operate and technically support the nuclear facilities. The NG is under the leadership of a President and Chief Nuclear Officer (CNO).

13.2.1 Offsite Organization

The Exelon Generation Company (EGC) corporate organization and its functions and responsibilities are described in Chapter 1 of the Quality Assurance Topical Report NO-AA-10, as revised.

13.2.2 Operating Organization

The organizational structure and reporting relationships are described in the Quality Assurance Topical Report, NO-AA-10.

13.2.3 Qualification of Nuclear Plant Personnel

13.2.3.1 Qualification Requirements

The qualification requirements for the onsite plant operating staff personnel are as provided in ANSI N18.1-1971, “Standard for Selection and Training of Personnel for Nuclear Plants” and Regulatory Guide 1.8 (Rev. 1). The medical certification of licensed reactor operators and senior reactor operators is in accordance with ANSI/ANS 3.4-1983, “Medical Certification and Monitoring of Personnel Requiring Operator Licenses for Nuclear Power Plants,” as endorsed by Regulatory Guide 1.134 (Rev 2).

A retraining and replacement training program for the facility staff shall be maintained under the direction of the PBAPS Director, Training and shall meet the requirements of Section 5.5 of ANSI N18.1-1971 and 10CFR55.

The Peach Bottom Atomic Power Station staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, with the following exceptions: the Manager, Radiation Protection who shall meet or exceed the qualifications of Regulatory Guide (RG) 1.8, "Personnel Selection and Training," September 1975 and the licensed operators shall comply with the requirements of 10CFR55. Administrative controls exist to ensure that all personnel satisfy those requirements.

Table 13.2.1

Shift Crew Composition - Minimum Requirements

One or Both Units Operating*:	1 Shift Manager (SRO) 1 Shift Supervisor (SRO)** 3 Reactor Operators (RO) 5 Operators 1 Shift Technical Advisor**
Both Units not Operating*	1 Shift Manager or Shift Supervisor(SRO) 2 Reactor Operators (RO) 5 Operators

SRO - NRC licensed senior reactor operator
RO - NRC licensed reactor operator

*For the purpose of this table, a unit is considered not to be operating when it is in a Cold Condition with the reactor mode switch in shutdown or refuel.

**When the STA position is filled by an on-shift SRO, provided the individual meets the 1985 NRC Policy Statement on Engineering Expertise on Shift, the shift crew composition must have a minimum of 3 SROs.

13.3 TRAINING

13.3.1 Plant Personnel Training Program

The Peach Bottom Atomic Power Station (PBAPS) training program incorporates the requirements specified in ANSI N18.1-1971, 10CFR55, and 10CFR50 as promulgated in the NRC Final Rule 10CFR50.120 - "Training and Qualification of Nuclear Power Plant Personnel," which became effective May 26, 1993. The PBAPS training program is based on a Systems Approach to Training (SAT). The training program provides qualified personnel to operate and maintain the facility in a safe manner in all modes of operation. The training program is periodically reviewed by management for effectiveness, and is periodically evaluated and revised, as appropriate, to reflect industry experience as well as changes to the facility, procedures, regulations, and quality assurance requirements. Achievement of this goal is based on philosophy of providing training developed from a systematic analysis of job requirements using job and task analysis where available. This philosophy is consistent with both the Nuclear Regulatory (NRC) requirements and Institute of Nuclear Power Operations (INPO) recommendations necessary for accreditation of training programs.

13.3.1.1 INPO Accredited Training Programs

The following training programs have been accredited by INPO and are continually maintained based on Systems Approach to Training (SAT). Program/Course plans describing the content and details of these programs are maintained by the Training Division - PBAPS.

Program

1. Non-Licensed Operator
2. Reactor Operator
3. Senior Reactor Operator
4. Shift Manager (formerly Shift Supervisor)
5. Shift Technical Advisor
6. Engineering Support Personnel
7. Licensed Operator Requalification
8. Maintenance Technician-Mechanical and Supervisor
9. Maintenance Technician-Electrical and Supervisor
10. Control Technician and Supervisor
11. Chemistry Technician
12. Health Physics Technician

13.3.1.2 Other Training Programs

The following training programs are implemented to support commitments not necessarily related to INPO accreditation.

Program

1. Emergency Preparedness
2. Radwaste
3. Fire Protection
4. Procedure Reviewer
5. Security Force Personnel
6. Industrial Safety and Health
7. General Employee Training
8. General Respiratory Training

13.3.2 Records

Training program records will be maintained by the responsible Training organization as specified in 13.3.1.1, 13.3.1.2, and 13.3.1.3. Records for the plant staff shall be sufficient to show qualification for the position held. As applicable for the training course, records include copies of written examinations and answers, lecture attendance records, results of evaluations, and records of additional remedial training. All quality records indicated pertinent, formalized training of the plant staff shall be entered into the Nuclear Records Management System for proper maintenance and recall, by the plant nuclear training section.

13.3.3 Training Organization

13.3.3.1 Refer to the Quality Assurance Topical Report, NO-AA-10.

13.4 PRE-OPERATIONAL TEST PROGRAM

This section provides HISTORICAL INFORMATION and has not been updated.

13.4.1 Objectives

The objective of the pre-operational test program is to ensure that the plant is ready for fuel loading, low power physics testing, and power testing. Construction acceptance tests described more fully herein, are performed to confirm that system components have been properly installed, calibrated, and adjusted. When complete systems are available, demonstration tests are conducted on safety-related systems. System demonstration tests simulate, as closely as possible, the system design conditions, and include exercising the system in its various operating modes. The completion of the pre-operational test program provides maximum assurance that the startup and power test program will proceed in an orderly and safe manner.

An extensive pre-operational test program is planned for the reactor and its auxiliary systems. This program will start approximately 6 months before initial fuel loading. The actual duration of each test is short relative to the entire 6-month program; the longest test, the control rod drive system checkout, takes approximately 1 month of testing. A typical pre-operational test sequence is shown in Figure 13.4.1.

Key systems are sequenced for completion and testing early enough to provide auxiliary services for testing, for the operation of other systems, or for construction activities, e.g., use of the makeup system for flushing. This results in an early requirement for electrical systems, demineralized water makeup, and cooling water systems.

After nuclear fuel is loaded in the reactor, all interconnected auxiliary systems are treated as potentially radioactive. In time, many of these systems become sufficiently radioactive to impose restrictions and time limitations on maintenance work. During normal plant operations, some components and systems cannot be observed for proper performance. To avoid these limitations during the pre-operational tests, all of the nuclear steam supply system and its auxiliary systems are normally tested before fuel loading.

The pre-operational test period is an important phase in the training of the nuclear system operators. Experience and understanding of plant systems and components is gained with a minimum of risk to the equipment or personnel. Minimal restrictions are imposed on either the operators or the testing.

This gives maximum opportunity to evaluate and train individual operators, and to troubleshoot plant systems. In addition, plant equipment and systems are operated for a sufficient period of time to discover and correct any design, manufacturing, or installation deficiencies and to adjust and calibrate the equipment.

13.4.2 Pre-Operational Test Schedule Considerations

The following key points are considered in developing an adequate sequence and schedule of pre-operational tests:

1. Construction tests are tests performed by Bechtel or subcontractors in the process of constructing systems and preparing systems for acceptance and demonstration testing.

2. Acceptance tests are designed to demonstrate to the licensee that nonsafety-related systems are complete and functional. The extent of the testing and documentation will vary depending on the relative importance of the system to the power generating function of the plant.

3. Demonstration tests are designed to demonstrate to the licensee and the NRC that safety-related systems are complete and functional. These tests meet the requirements of the Guide for Planning of Preoperational Testing Programs, the Guide for the Planning of Initial Start-up Programs, and 10CFR50, Appendix B.

4. Tests are performed in accordance with written procedures which identify, where applicable, the purpose of the test, prerequisite tests, required systems conditions and plant status, applicable limitations and precautions, test method and test instrumentation, acceptance criteria, and approvals.

5. Systems are sequenced for early testing and placed in routine operation to provide necessary auxiliary services for other systems. Examples are plant electrical systems, instrument air systems, and makeup water supply systems.

6. Pre-operational testing is coordinated with construction to permit fuel loading as early as possible, without compromising nuclear safety or impeding construction work. As a result, fuel loading occurs while construction work may be still in progress on unrelated systems and areas.

7. Demonstration testing is performed before fuel loading on all systems which could consequently be exposed to radioactive contamination, to minimize possible contamination problems.

8. Pre-operational tests provide an important phase of the plant operators' training program. They are scheduled on key systems to permit maximum participation by all operators prior to licensing examinations.

9. Temporary construction power is sometimes required for construction tests at the beginning of the pre-operational test program.

10. Electrical jumpers are used to facilitate pre-operational testing in some instances, but their use is minimized and controlled by proper identification of such jumpers in the procedures. All jumpers are removed before fuel loading or exceptions noted and approved.

11. When the plant is ready for fuel loading, construction workers are excluded from the drywell, and strict control is enforced over access to the control room, electrical equipment rooms, reactor building, and radwaste building.

12. Specialized electronic equipment and nuclear instrumentation manufactured by General Electric is checked and pre-operationally tested by General Electric representatives, assisted by licensee personnel.

13. Detailed test specifications are prepared by General Electric, or Bechtel, depending upon system design responsibility for safety-related systems and are reviewed and approved by the licensee. The procedures are specific regarding intent, method, and operating requirements for completing the test without endangering personnel or plant safety, and will include detailed blank data sheets to be completed during the test.

14. In general, tests are performed using permanently installed instrumentation for the required data. Where it is not possible to use permanently installed instrumentation during the system pre-operational tests, it will be necessary to install test instrumentation to ensure safe operation of the

equipment. Special instrumentation is specified in the test procedure. Any test requiring simulation of a plant parameter has the method detailed in the procedure, as well as the means for assuring that the system is returned to normal.

15. The test results will be evaluated and approved to assure test objectives have been met prior to final testing acceptance.

13.4.3 Summary of Pre-Operational Test Content

13.4.3.1 Construction Tests

Construction tests are performed by Bechtel or subcontractors under the technical direction of General Electric or Bechtel (depending on design responsibility). All testing is under the surveillance of the licensee. Certain construction tests are defined by formalized procedures and data sheets and require formal reporting and acceptance but most require limited procedures and reporting and do not require formal acceptance. Data records are kept as required.

Construction testing includes, but is not limited to, the following examples:

1. System hydrostatic tests.

2. Flushing systems.

3. Electrical system tests to and including energizing, e.g., grounding checks, relay checks, circuit breaker operation and control checks, continuity checks, megger tests, phasing checks, high potential measurements, and bus energizing.

4. Initial adjustment and bumping of motors.

5. Checking control and interlock functions of instruments, relays, and control devices.

6. Calibrating instruments and checking or setting initial trip set points.

7. Pneumatic test of instrument and service air system and cleaning of lines.

8. Adjustments of mechanical equipment, such as alignment, greasing, and tightening of bolts.

9. Checking and adjusting relief and safety valves.

10. Checking and adjusting piping system supports and snubbers.

13.4.3.2 Acceptance Test Program

The acceptance test program is performed by the licensee to demonstrate that proper operation is obtained as required on nonsafety-related systems. Bechtel Corporation, General Electric, and/or their subcontractors supply technical direction. Certain acceptance tests are defined by formalized procedures and data sheets but most require simple procedures and do not require formal acceptance. Data records are kept as required.

13.4.3.2.1 Raw Water Treatment System

Demonstrate the ability of the system to provide filtered and chemically treated water to its various services.

Operability of the pumping and filtering units is verified. System operability and performance capability is verified.

13.4.3.2.2 Screen Wash and Chlorination System

Traveling Water Screens and Screen Wash System

Demonstrate the ability of the system to perform its design function of continuously removing and discarding solid materials that pass through the trash racks from the incoming river water.

Operability of the pumping units and the trash removal units shall be verified.

Chlorination System

Demonstrate the ability of the system to perform its design function of supplying the required chlorination of the circulating water and service water bays.

Operability of the system is verified.

13.4.3.2.3 Makeup Demineralizer System

Demonstrate the operational capability of the system to perform its design function of providing a supply of demineralized water to the demineralized water storage tank.

The operability of the pumping units is verified. System operability is verified under simulated normal conditions before charging resins and using chemicals.

13.4.3.2.4 Auxiliary Boilers

Demonstrate the ability of the plant auxiliary steam supply system to perform its design function of providing a supply of steam to the various auxiliary steam services.

The operability of the pumping units is verified. Operability of the system along with operability of the auxiliary systems is verified.

13.4.3.2.5 Feedwater Heating System

Demonstrate the capability of the system to perform its design function.

Operability of the system is verified. A functional test of the system is deferred until nuclear steam is available during the Power Test Program.

13.4.3.2.6 250-V dc Power Supply System to Turbine- Generator Emergency Bearing Oil Pump

Demonstrate the capability of the system to provide electrical energy for the turbine-generator emergency bearing oil pump motor.

Operability of the 250-V battery and battery charger is verified. Operability of the associated 250-V dc power distribution system is verified.

13.4.3.2.7 Circulating Water System

Demonstrate the ability of the circulating water system to perform its design function of supplying cooling water to the main condenser.

The operability of the pumping units is verified. Operability and performance capability of the circulating water system and the associated screen systems is verified.

13.4.3.2.8 Turbine and Auxiliaries

Demonstrate the capability of the turbine controls (other than the electro-hydraulic control system (EHCS)), and associated auxiliary systems to perform their design functions. A complete functional test is deferred until reactor steam is available during the Power Test Program. A separate demonstration test of the EHCS is conducted (paragraph 13.4.3.3.17).

Operability of the pumping units is verified. Turbine operability including the control system and associated auxiliary systems is verified.

13.4.3.2.9 Generator and Auxiliaries

Demonstrate the capability of the generator and the associated auxiliary systems to perform their design functions.

Operability of the pumping units is verified. Operability of the generator and associated auxiliary systems is verified. A complete functional test is deferred until nuclear steam is available during the Power Test Program.

13.4.3.2.10 Air Conditioning System - Control Room Complex

Demonstrate the ability of the air conditioning system to perform its design function of providing conditioned air in the control and associated rooms, cable spreading room, and computer room.

Operability of fans, pumps, and coolers is verified. Operability and performance capability of the systems is verified.

13.4.3.3 Demonstration Tests

Demonstration tests of safety-related systems and other systems as selected by the licensee are performed by the licensee under technical direction of Bechtel, General Electric, or the licensee as appropriate. Demonstration tests require formalized procedures and data sheets and require formal reporting and approval. Demonstration testing is the surveillance of proper equipment operation under stipulated test conditions to assure its function in anticipated normal and emergency modes. Demonstration tests are characterized by a written approved test specification giving prerequisites, test purpose, system initial conditions, environmental conditions, data required, acceptance criteria, special precautions, and reference material.

The operability of pumping units is verified by checking motor starting and operating currents, temperature, and vibration, and by checking pump discharge pressure shutoff. System operability and performance capability verification includes system flow verification if flow measuring instrumentation is provided as permanent plant instrumentation and operation of remote operated valves from their control stations. It further includes verification of any isolation features (using simulated signals), annunciation, interlocking, automatic and standby starting, remote starting, and automatic transfer features.

13.4.3.3.1 Service Water System

High-Pressure Service Water System

Demonstrate the ability of the high-pressure service water system to perform its design functions of providing (1) a supply of cooling water to the residual heat removal system heat exchangers, and (2) an emergency source of water for flooding the reactor vessel.

The operability of the pumping units is verified. System operability and performance capability for delivery through the normal and alternate flow paths is verified. Pump structure supply and exhaust fans, and unit heaters are verified operable during the testing of the emergency service water system.

Emergency Service Water System

Demonstrate the ability of the emergency service water system to perform its design functions of providing a reliable supply of cooling water to (1) the diesel-generator heat exchangers, (2) the reactor building cooling water heat exchangers upon loss of normal ac power, and (3) the safeguards equipment compartment coolers upon loss of normal ac power.

The demonstration testing of the emergency cooling water pump at the emergency cooling tower for performance of these same functions is included in the testing of the emergency cooling tower system.

Operability of the pumping units and pump structure supply and exhaust fans and unit heaters is verified. System operability and performance capability for water delivery through the normal and alternate flow paths will be verified.

Plant Service Water System

Demonstrate the ability of the service water system to supply cooling water to the various cooling water loops in the system.

Operability of the pumping units is verified. System operability and performance capability for water delivery to the various system cooling water loops are verified.

13.4.3.3.2 Fire Protection Systems

Demonstrate the ability of the fire water system, CO_2 system, and air foam system to perform their design function of providing a supply of fire extinguishing agents to the various systems' services. Demonstrate the ability of the heat and smoke detection system to perform its design function of detecting and annunciating in the main control room, the presence of fire.

The operability of the diesel engine and motor driven fire water pumping units shall be verified. The operability and performance capability of the auxiliary systems for the diesel fire water pump shall be verified. The capability for water delivery up to the various system delivery points shall be verified and a functional test of the deluge system at the transformers shall be performed.

The operability of the CO_2 system shall be verified and a functional CO_2 delivery test at each discharge location will be conducted. The operability of the air foam system shall be verified by determination of proper pressure at the air foam nozzle. The operability of the heat and smoke detection system shall be verified by simulation of the detection parameters.

13.4.3.3.3 Instrument and Service Air Systems

Demonstrate the ability of the compressed air systems to (1) supply dry and oil free, compressed air to the various services in the instrument air system, (2) to supply compressed air to the various services in the service air system, and (3) supply compressed air to various required services upon loss of the normal system line pressure.

13.4.3.3.4 Condensate Filter-Demineralizer System

Demonstrate the operational capability of the condensate filter-demineralizer system to performing the design function of (1) continuously removing dissolved and suspended solids from reactor feedwater, (2) providing a final polishing of makeup water entering the reactor feedwater loop, and (3) maintaining the high purity of water returned to condensate storage and transfer system.

The operability of the pumping units is verified. Operability of the low-pressure air blower is verified by functionally testing the precoat and backwash system, and all phases of operation using water and air flow, but no resin. One change of precoat slurry and mixed powder resins is then pumped to a demineralizer and backwashed to radwaste.

13.4.3.3.5 Reactor Building Cooling Water System

Demonstrate the ability of the reactor building cooling water system to perform its design function of providing cooling water to (1) the reactor recirculating pump seal and motor oil coolers, (2) the reactor water cleanup system non-regenerative heat exchangers, (3) cleanup recirculating pump seal coolers, (4) the sample coolers and, (5) the radwaste building pump coolers during normal operations, (6) the CRD pump oil coolers, air compressor jackets and aftercoolers during loss of turbine building cooling water, (7) the drywell air cooling and the drywell equipment drain sump cooler during a loss of off-site power, (8) the fuel pool heat exchangers via removable spool pieces, in the event of loss of normal cooling water, and to (9) the instrument nitrogen compressor jacket and aftercooler.

The operability of the pumping units shall be verified. System operability and performance capability shall be verified.

13.3.3.6 Turbine Building Cooling Water System

Demonstrate the ability of the turbine building cooling water system to provide cooling water to (1) the air compressor jackets and aftercoolers, (2) the condensate pump thrust bearings and motor oil coolers, (3) the CRD pump lube oil coolers, (4) the isolated phase bus coolers, (5) the filter-demineralizer holding pumps, and (6) the sample coolers during normal operations.

The operability of the pumping units shall be verified. The system operability and performance capability shall be verified.

13.4.3.3.7 Standby ac Power Supply to Emergency System

Demonstrate the ability of the standby ac power and distribution system to provide a reliable source of power for the safe shutdown of the reactors in the event of loss of all off-site AC power, even under design basis LOCA conditions.

Verify the operability of the diesel generators and unit auxiliary system (i.e. starting air system, oil supply and transfer system, diesel lube oil systems, diesel-engine cooling water system, diesel-engine building heating and ventilating system). The testing shall include, where applicable, verification of system operability and performance capability for the unit auxiliary systems. The testing of the diesel-generators shall include automatic starting of the diesel-generators on loss of off-site power, load shedding, diesel-generator circuit breaker operation, normal shutdown loads pickup, and operation at rated load for 4 hr.

In addition, a functional test of all diesels under a simulated design basis LOCA will be conducted to verify the capability of the diesel-generators to pick up in proper sequence the required engineered safeguards loads. A further test of the diesel system capability demonstrating adequacy of the system even with loss of the dc supply to one diesel, for each combination of three diesels will be conducted under simulated LOCA conditions but with the various motor breakers racked out.

13.4.3.3.8 Off-Site AC Power Supplies to the Emergency System

Demonstrate the ability of the off-site ac power supplies to the emergency system to provide reliable power sources for loads which are important to plant safety under accident conditions.

Verify the operability of the off-site power sources, including manually initiated and automatically initiated transfers of power sources from the 4-kV system buses. A functional test will be conducted to verify that, under simulated design basis LOCA conditions and the loss of one off-site power supply, the remaining off-site power supply can carry the emergency system loads. A further functional test will verify that with simulated failure of one off-site AC power supply, the remaining off-site power supply can carry the normal shutdown loads.

13.4.3.3.9 125/250-V dc Power Supply System

Demonstrate the ability of the 125/250-V DC power system to provide electrical energy for the dc utilization apparatus.

Operability of the 125-V batteries and battery chargers is verified. 125/250-V DC power distribution system operability is verified.

13.4.3.3.10 24-V dc Power Supply System

Demonstrate the ability of each ± 24-V DC power system to provide electrical energy for the neutron monitoring system.

Operability of the 24-V DC batteries, battery chargers, and power distribution system is verified.

13.4.3.3.11 Drywell Ventilation System

Demonstrate the ability of the drywell ventilation and chilled water system to perform its design function of (1) providing cooling and ventilating of the drywell air space, (2) providing cooling of auxiliary equipment inside the drywell, (3) providing a path from the alternate source of cooling water during periods of normal power failure, (4) providing means of isolating the drywell cooling and ventilating system.

The operability of the fans and pumping units is verified. System operability and performance capability are verified.

13.4.3.3.12 Standby Gas Treatment System

Demonstrate the ability of the standby gas treatment system to perform its design functions of (1) maintaining negative pressure in the reactor building during secondary containment isolation, (2) filtering of exhaust air to remove any radioactive particles and halogens which may be present, (3) operating with emergency power on failure of normal power, and (4) exhausting drywell purge air to atmosphere when necessary.

The operability of the filters and fans is verified. The operability and performance capability of the standby gas treatment system is verified for various modes of operation.

13.4.3.3.13 Condensate and Feedwater System

Demonstrate the ability of the condensate and feedwater system to perform its design function of providing condensate to:

1. Condensate pumps

2. Steam jet air ejector inter- and after-condensers

3. Steam packing exhauster condenser

4. Feedwater heaters

5. Reactor feedwater pumps

6. Reactor.

Operability of the pumping units is verified. System operability and performance capability is verified. A functional test using the auxiliary boiler to supply steam to the reactor feed pump turbine with the pump disconnected is conducted. A full capacity test of the system is deferred until nuclear steam is available during the Power Test Program.

13.4.3.3.14 Gaseous Radwaste System

Off-Gas System (Without Recombiner Subsystem)

Demonstrate the capability of the gaseous radwaste system to collect, process, monitor, and discharge the potentially radioactive gases from the gland seal exhauster and from the air ejector off-gas.

Operability of the filters, stack dilution fans, and mechanical vacuum pump is verified.

System operability and performance capability are verified. A functional test is deferred until reactor steam is available during the Power Test Program.

Recombiner Subsystem

Demonstrate the ability of the recombiner subsystem of the air ejector off-gas system to provide long-term holdup of the air ejector off-gas.

When installation is complete, operability of the compressors is verified. System operability and performance capability are verified. A functional test is conducted when reactor steam is available.

13.4.3.3.15 Liquid Radwaste Collection System

Demonstrate the capability of the liquid radwaste system to perform its design function of processing liquid radioactive wastes.

Operability of the pumping units is verified. System operability and performance capability are verified. A functional test of all subsystems is conducted using simulated liquid radwaste to verify the system's ability to process wastes.

13.4.3.3.16 Condensate and Demineralized Water Storage and Transfer System

Demonstrate the ability of the system to perform its design function of supplying and storing adequate makeup water to required services.

Operability of the pumping units is verified. System operability and performance capability is verified.

13.4.3.3.17 Turbine Electro-Hydraulic Control System

Demonstrate the capability of the turbine EHCS to accomplish its design function of turbine control and protection.

Operability of the pumping units is verified. System operability and performance capability are verified by simulating various operating conditions using electrical signals.

13.4.3.3.18 <u>Heating and Ventilation - Recombiner Facility Building</u>

Demonstrate the ability of the heating and ventilating system to perform its design function of (1) providing a supply of filtered outside or recycled air for personnel occupancy and equipment protection, and (2) filtering air from potentially radioactive areas through HEPA and charcoal filters before exhausting to the vent stack.

When installed, operability of the fans and filters is verified. System operability and performance capability are verified prior to operation of the recombiner subsystem.

13.4.3.3.19 <u>Heating and Ventilation - Turbine Building</u>

Demonstrate the ability of the turbine building heating and ventilating system to perform its design functions of (1) providing a supply of filtered outside and recycled air for personnel occupancy and equipment protection, (2) filtering of exhaust air from potentially contaminated areas through HEPA filters, and (3) providing supplemental heating for personnel comfort and equipment freeze protection.

Operability of the fans is verified. System operability and performance capability is verified.

13.4.3.3.20 <u>Heating and Ventilation - Reactor Building</u>

Demonstrate the ability of the reactor building heating and ventilating system to perform its design function of (1) providing a supply of filtered outside/recycled air for personnel occupancy and equipment protection, (2) providing a supply of filtered and heated outside air for purging the drywell when necessary, (3) filtering of exhaust air from potentially contaminated areas through HEPA filters, (4) providing supplemental heating for personnel comfort and equipment freeze protection, and (5) providing duct isolation and fan trip for isolation signals requiring standby gas treatment system operation.

Operability of the fans is verified. System operability and performance capability are verified.

13.4.3.3.21 <u>Heating and Ventilation - Radwaste Building</u>

Demonstrate the ability of the radwaste building heating and ventilating system to perform its design function of (1) providing a supply of outside air for personnel occupancy and equipment protection, (2) filtering of exhaust air from potentially contaminated areas through HEPA filters, and (3) providing supplemental heating for personnel comfort and equipment freeze protection.

Operability of the fans is verified. System operability and performance capability are verified.

13.4.3.3.22 Emergency Ventilation Air Supply to Control Room

Demonstrate the ability of the control room emergency ventilation air supply system to perform its design functions of (1) filtering outside supply air to prevent contamination of the control room when necessary, and (2) providing filtered outside air to the control room using emergency power on failure of normal power.

Operability of the fans and filters is verified. System operability and performance capability are verified under normal conditions and simulated loss of off-site power and accident conditions.

13.4.3.3.23 Primary Containment Atmospheric Control System

Demonstrate the ability of the primary containment atmospheric control system to perform the design functions of (1) controlling the nitrogen inerting of the primary containment, (2) monitoring the drywell oxygen content, and (3) monitoring the level of airborne radioactivity in the drywell.

Operability of the sampling units is verified. System operability and performance capability are verified.

13.4.3.3.24 Emergency Cooling Tower System

Demonstrate the ability of the emergency cooling tower system to perform its design functions of (1) acting as a heat sink and supplying cooling water to the emergency service water system and high-pressure service water system upon loss of the normal heat sink, and (2) supplying cooling water to the emergency service water system should the emergency service water pumps become inoperable.

Operability of the pumping units and fans is verified. System operability and performance capability are verified for all modes of operation (including emergency cooling water pump operation).

13.4.3.3.25 120-V Instrument ac Power Supply System

Demonstrate the capability of the 120-V ac instrument power supply system and the 120-V uninterruptible ac power supply system to supply power to the various connected loads.

Operability of the power distribution system is verified. A functional test is conducted to verify operability upon loss of the preferred source for the 120-V uninterruptible ac power supply system.

13.4.3.3.26 Containment Isolation Valves Leak Rate Measurement System

Local leakage across the seat of isolation valves in the nuclear steam supply system, and in auxiliary systems that penetrate the primary containment, are measured to ensure compliance with the plant technical specifications.

13.4.3.3.27 Primary Containment Leak Rate Measurement System

The integrated leakage rate test of the primary containment is conducted prior to fuel loading as required by 10CFR50. Penetrations with provisions for leak testing without pressurization of the containment (such as electrical, personnel access hatch, access hatches, and pipe penetrations having bellows seals) are tested prior to conducting the primary containment integrated leak rate test.

13.4.3.3.28 Control Rod Drive Hydraulic System

Demonstrate the ability of the CRD hydraulic system to move all control rods at specified rates and to supply water at design pressures and flow rates.

The operability of the pumping units is verified. System operability and performance capability, including the proper functioning of the CRD mechanisms, control rod position indicator system, logic, and safety and control devices, are verified.

13.4.3.3.29 Control Rod Drive Manual Control System

Demonstrate that the manual rod control system, its associated rod blocks, and indications and alarms function properly.

Verify operability of the CRD manual control system by electrical control and indication checkout without rod motion. Verify operability of rod blocks external to the manual control system by activating end point relays within the system.

13.4.3.3.30 Fuel Pool Cooling and Cleanup System

Demonstrate the capability of the fuel pool cooling and cleanup system to perform its design functions in all modes of operation.

Operability of the pumping units is verified. System operability and performance capability are verified (without resins).

13.4.3.3.31 Fuel Handling and Vessel Servicing Equipment

Demonstrate that, where applicable, the interlocks, limits, and logic associated with the fuel handling and vessel servicing equipment are functioning properly.

Operability of fuel handling and vessel servicing equipment is verified by dry run simulations of separate, required operations, using different pieces of equipment, on the refueling floor, in the fuel storage pool, and both over and in the reactor vessel. Equipment tested in and over the storage pool includes the fuel preparation machinery, inspection scope, lights, the vacuum cleaner, the jib crane, the refueling platform and its associated equipment. Simulated operational tests over the reactor vessel include the service platform and jib crane assembly, the fuel and control element movements, and other procedural methods and tools.

13.4.3.3.32 Reactor Water Cleanup System

Demonstrate the operability of the reactor water cleanup system in as many operational modes as plant conditions permit.

The operability and performance of the reactor water cleanup system are verified by a functional test of the system. Testing shall include the filter subsystems and resin charging and discharging, simulating where necessary and using water from the condensate storage tank if using the reactor vessel for pump suction supply is not feasible.

13.4.3.3.33 Standby Liquid Control System

Demonstrate the operation of the standby liquid control system.

Operability and performance capability of the standby liquid control system are verified. A functional test is conducted of the system under simulated accident conditions by discharging water through the explosive squib valves into the reactor vessel.

13.4.3.3.34 Residual Heat Removal System

Demonstrate the ability of the RHRS to perform its design functions of LPCI, containment cooling, and shutdown cooling, and to provide an additional supply of cooling water to the fuel pool cooling system, through the use of removable spool pieces.

Operability of the pumping units is verified. System operability and performance capability in the various modes of operation are verified. A functional test of the system under simulated accident conditions is conducted for the LPCI and containment cooling modes. A functional test of the shutdown cooling mode under simulated shutdown conditions is conducted. The interconnection with the fuel pool is tested under simulated operating conditions. Operability of the compartment unit coolers is verified.

13.4.3.3.3.35 Feedwater Control System

Demonstrate the capability of the feedwater control system to respond to simulated input signals corresponding to various flow conditions.

Operability of the feedwater control system is verified by proper system response and proper operation of the interlocks and alarms. Operation of the system during the power test program will demonstrate that the system will control flow up to and including rated flow conditions.

13.4.3.3.36 Nuclear Boiler System

Demonstrate the ability to perform the design functions of (1) safety valves, (2) automatic blowdown system, (3) relief valve remote-manual actuation system, (4) relief valve leak detection system, (5) main steam isolation valves, main steam drain valves, and reactor water sample valves for isolation signals, and (6) reactor water level trip and alarm functions.

Safety valve vendor certification for set point and capacity is verified. Operability of the automatic blowdown system and relief valve remote-manual actuation system is verified by functional test. Operability of the relief valve leak detection system is verified. Isolation of the main steam lines, main steam line drain line, and reactor water supply line by appropriate isolation signals is verified by simulation of the various isolation parameters. Operability of all reactor water level trip and alarm functions associated with reactor protection, safeguards, and feedwater systems are verified by actually manipulating the reactor vessel water level to actuate the sensors.

13.4.3.3.37 Reactor Core Isolation Cooling System

Demonstrate the capability of the RCICS to achieve its design function of supplying cooling water to the reactor core when required.

Operability of the RCIC is verified. A functional test of the turbine using steam from auxiliary boiler is conducted. A full capacity test of the system is deferred until nuclear steam is available during the power test program. The operability of the compartment unit coolers is verified.

13.4.3.3.38 Reactor Recirculation and Motor-Generator Sets

Demonstrate the ability of the reactor recirculation system to meet its design function of providing flow up to the maximum volumetric core flow that is consistent with operation precautions.

Operability of the pumping units and M-G sets is verified. System operability and performance capability are verified for cold water conditions under simulated normal and abnormal conditions.

13.4.3.3.39 Core Spray System

Demonstrate the ability of the core spray system to perform its design function of supplying cooling water to the reactor core in the event of a LOCA.

Operability of the system is verified. System operability and performance capability are verified under simulated accident conditions. Operability of the compartment unit coolers is verified.

13.4.3.3.40 High-Pressure Coolant Injection System

Demonstrate the ability of the HPCIS to perform its design function of supplying cooling water to the reactor core in the event of a LOCA.

Operability of the HPCI is verified. A functional test of the turbine using steam from the auxiliary boiler is conducted. A full capacity test of the system is deferred until nuclear steam is available during the power test program. Operability of the compartment unit coolers is verified.

13.4.3.3.41 Emergency Switchgear and Battery Rooms Heating and Ventilating System

Demonstrate the ability of the heating and ventilating system to perform its design function of providing an adequate supply of filtered outside air to the battery room and filtered outside or recycled air to the emergency switchgear room for equipment protection using normal or emergency power.

Operability of the fans and filters is verified. System operability and performance capability are verified under normal conditions and under simulated loss of off-site ac power.

13.4.3.3.42 Solid Radwaste System

Demonstrate the ability of the solid radwaste system to perform its design function of processing solid radioactive waste.

Operability of the solid radwaste system is verified by a functional test of the operation of processing equipment using simulated waste.

13.4.3.3.43 Reactor Protection System

Demonstrate the ability of the RPS to perform its design function of producing reactor scrams in response to reaching various parameter limits.

Scram set points are verified. System operability and performance capability are verified. Functional testing of response to various simulated sensor signals for various conditions is conducted.

13.4.3.3.44 Neutron Monitoring System

Demonstrate the ability of the neutron monitoring system to perform its design function of monitoring the core neutron flux in all modes and supplying proper signals to its various outputs.

Operability of the system is verified. A functional test employing simulated electrical signals verifies system performance.

13.4.3.3.45 Traversing In-Core Probe Calibration System

Demonstrate the capability of the TIP calibration system to provide calibration information to the LPRM's.

Operability of the drive control units is verified. System operability is verified.

13.4.3.3.46 Rod Worth Minimizer

Demonstrate the ability of the rod worth minimizer to perform its design function of limiting rod motion within preselected parameters.

Operability and system performance capability are verified. Functional tests employing random control rod movement and simulated error conditions are conducted.

13.4.3.3.47 Process Radiation Monitoring System

Demonstrate the ability of the following subsystems to perform their design functions:

1. Main steam line radiation monitoring system

2. Off-gas radiation monitoring system

3. Stack gas radiation monitoring system

4. Liquid process radiation monitoring system

5. Reactor building exhaust plenum radiation monitoring system.

Operability of the various subsystems is checked by using appropriate radioactive check sources, thus verifying system interconnection and calibration.

13.4.3.3.48 Area Radiation Monitoring System

Demonstrate the ability of the area radiation monitors to perform their design functions.
The operability of the monitoring system is verified using a portable calibration unit.

13.4.3.3.49 Environs Monitoring System

Demonstrate the ability of the environs monitors to perform their design function.

Operability of the monitoring system is verified using a portable calibration unit.

13.4.3.3.50 Process Computer Interface

Demonstrate the ability of the process computer to perform its design functions of providing alarms for selected monitored parameters. The rod worth minimizer functions are separately demonstrated (paragraph 13.4.3.3.46).

Scan address points for range and correct printout, including alarm set points if applicable, are verified for analog and digital inputs. This system has been retired and replaced by the process computer (PMS).

13.4.3.3.51 Heating and Ventilation - Emergency Cooling Tower Pump Rooms and Switchgear Rooms

Demonstrate the ability of the heating and ventilating system to perform its design function of (1) providing a supply of outside air to maintain suitable temperatures in the pump and switchgear rooms for equipment protection, and (2) providing supplemental heating and recirculation of inside air when necessary for equipment freeze protection.

Operability of the fans is verified. System operability and performance capability are verified.

13.5 STARTUP AND POWER TEST PROGRAM

Sections 13.5.1 and 13.5.2 provide HISTORICAL INFORMATION and have not been updated.

13.5.1 Program Description and Objectives

13.5.1.1 General

The tests comprising the startup and power test program are conducted primarily to show that the overall plant performance is confirmed in terms of the established design criteria at all times starting with fuel loading. These criteria and the associated tests have either a safety or economic orientation, while often both aspects of the design are being explored. A most important result of the startup test program is that the operator has available to him valuable data upon which the future normal and safe operation of the plant can be based. The startup and power test program may be divided into the following discrete and successive test phases:

Phase 1	Fuel Loading and Shutdown Power Level Tests
Phase 2	Initial Heatup to Rated Temperature and Pressure
Phase 3	Power Testing from 25 percent to 100 percent of Rated Output
Phase 4	Warranty Demonstrations

The tests performed can be broadly classified as major plant transients (Table 13.5.1), stability tests (Table 13.5.2), and a residue of tests directed towards demonstrating correct performance of the numerous auxiliary plant systems; clearly certain tests may be identified with more than one class. Each test is discussed later but, at this juncture, the following comments are given by way of outlining the startup and power test program. Table 13.5.3 shows the complete startup and power test program and should be considered in conjunction with Figure 13.5.2, which shows graphically the various test points as a function of core thermal power and flow.

13.5.1.2 Fuel Loading and Shutdown Power Level Tests

Fuel loading requires the movement of the full core complement of assemblies from the fuel pool to the core with each assembly identified by number before being placed in the correct coordinate position. The procedure controlling this movement is arranged so that shutdown margin and subcritical checks are made at predetermined intervals throughout the loading, thus ensuring safe loading increments. Specially sensitive neutron monitors maintained at the loading face as loading progresses serve to provide indication for the shutdown margin measurements and also allow the recording of the core flux level as each assembly is added. A complete check is made of the fully loaded core to ascertain that all assemblies are properly installed, correctly oriented, and occupying their designated positions.

At this point in the program, a number of tests are conducted which are best described as initial shutdown power level tests. Phase 1 chemical and radiochemical tests are made in order to check the quality of the reactor water before fuel is loaded and to establish base and background levels which will be required to facilitate later analysis and instrument calibrations. Plant and site radiation surveys are made at specific locations for later comparison with the values obtained at the subsequent operating power levels. Shutdown margin checks are repeated for the fully loaded core and criticality is achieved with each of the two prescribed rod sequences in turn, data being recorded for each rod withdrawn. The reactor is made critical by means of each control rod sequence using the normal SRM in conjunction with the operational sources in order to show that adequate response exists for normal operation. Each CRD is subjected to scram and friction testing at ambient conditions. An initial setting is given to the IRM by comparison with the SRM's. The process computer is checked to see that it is receiving correct values for those process variables which are available. Confirmatory vibration tests will be performed.

13.5.1.3 Initial Heatup to Rated Temperature and Pressure

Heatup follows satisfactory completion of the Phase 1 tests and further checks are made of coolant chemistry together with radiation surveys at the selected plant locations. All CRD's are scram timed at rated temperature and pressure with selected drives timed at intermediate reactor pressures and for different accumulator pressures. Both control rod sequences are further investigated in order to obtain rod pattern versus temperature relationships. The process computer checkout continues as more process variables become available for input. The RCICS and HPCIS will undergo controlled starts at low reactor pressure and at rated conditions with the former tested in the quick start mode at 1,000 psig. Correlations are obtained between reactor vessel temperatures at several locations and the values of other process variables as heatup continues. The movements of drywell piping systems as a function mainly of expansion are recorded for comparison with design data. An intermediate IRM and APRM calibration is made using coolant temperature rise data during nuclear heatup.

13.5.1.4 Power Testing from 25 Percent to 100 Percent of Rated Output

The power test phase comprises the following tests, many of which are repeated several times at the different test levels; consequently, reference should be made to Figure 13.5.1 for the probable order of execution for the full series. It must be appreciated that while a certain basic order of testing is maintained relative to power ascension, there is, nevertheless, considerable flexibility in the test sequence at a particular power level which may be used whenever it becomes operationally expedient.

Coolant chemistry tests and radiation surveys are made at each principal test level in order to preserve a safe and efficient power increase. Selected CRD's are scram timed at various power levels to provide correlation with the initial data. The effect of control rod movement on other parameters, e.g., electrical output, steam flow, and neutron flux level, is examined for different power conditions. The ability to change from one control rod sequence to the other at a reasonable high power level is demonstrated. Following the first reasonably accurate heat balance (25 percent power) the IRM's are reset. At each major power level (25 percent, 50 percent, 75 percent, and 100 percent) the LPRM's are calibrated, while the APRM's are calibrated at each new power level initially and following LPRM calibration. Completion of the process computer checkout is made for all variables and the various options are compared with hand calculations as soon as significant power levels are available. Further tests of the RCICS and the HPCIS are made with and without injection into the reactor pressure vessel. Collection of data from the system expansion tests is completed for those piping systems which had not previously reached full operating temperatures. The axial and radial power profiles are explored fully by means of the TIPS at representative power levels (25 percent, 50 percent, 75 percent, and 100 percent) during the power ascension. Core performance evaluations are made at all test points above the 10 percent power level and for selected flow transient conditions; the work involves the determination of core thermal power, maximum fuel rod surface heat flux, and the minimum critical heat flux ratio (MCHFR).

Overall plant stability in relation to minor perturbations is shown by the following group of tests which are made at all test points:

Flux response to rods
Pressure regulator set point change
Water level set point change
Bypass valve opening

For the first of these tests a centrally located control rod is moved and the flux response noted on a selected LPRM chamber. The next two tests require that the changes made should approximate as closely as possible a step change in demand, while for the remaining test the bypass valve is opened as quickly as possible. For all of these tests the plant performance is monitored by recording the transient behavior of numerous process variables, the principal one of interest being neutron flux. Other imposed transients are produced by step changes in demand core flow, dropping a feedwater heater and failing the operating pressure regulator to permit takeover by the backup regulator. Table 13.5.2 indicates the power and flow levels at which all these stability tests are performed.

The category of major plant transients includes full closure of all the main steam isolation valves, fast closure of turbine-generator control valves, fast closure of turbine-generator stop valves, loss of the main generator and off-site power, tripping a feedwater pump, and several trips of the recirculation pumps. The plant transient behavior is recorded for each test and the results may be compared with the predicted design performance. Table 13.5.1 shows the operating test conditions for all the proposed major transients.

A test is made of the relief valves in which the capacity, leaktightness, and general operability are demonstrated. At all major power levels the jet pump flow instrumentation is calibrated. The as-built characteristics of the recirculation pump drives are investigated as soon as operating conditions permit full core flow. The local control loop performance, based on the drive motor, fluid coupler, generator, drive pump, jet pumps, and control equipment, is checked. The vibration testing is conducted at several power conditions as the operating power level is raised.

13.5.1.5 Warranty Demonstrations

The final test phase consists of a warranty demonstration in which the steaming rate and quality can be shown to comply with contractual obligations.

13.5.2 Discussion of Startup Tests

13.5.2.1 General

All those tests comprising the startup and power test program are discussed in paragraph 13.5.2.2 with reference to the particular test purpose, brief description, and statement of acceptance criteria where applicable.

In describing the purpose of a test an attempt is made to identify those operating and safety-oriented characteristics of the plant which are being explored.

Where applicable, a definition of the relevant acceptance criteria for the test is given and is designated either "Level 1" or "Level 2." A Level 1 criterion normally relates to the value of a process variable assigned in the design of the plant, component systems, or associated equipment. If a Level 1 criterion is not satisfied, the plant will be placed in a suitable hold-condition until resolution is obtained. Tests compatible with this hold-condition may be continued. Following resolution, applicable tests must be repeated to verify that the requirements of the Level 1 criterion are now satisfied.

A Level 2 criterion is associated with expectations relating to the performance of systems. If a Level 2 criterion is not satisfied, operating and testing plans would not necessarily be altered. Investigations of the measurements and of the analytical techniques used for the predictions would be started.

For transients involving oscillatory response the criteria are specified in terms of decay ratio (defined as the ratio of successive maximum amplitudes of the same polarity). The decay ratio must be less than unity to meet a Level 1 criterion and less than 0.25 to meet Level 2.

13.5.2.2 Test Purpose, Description, and Acceptance Criteria

(1) Chemical and Radiochemical

Purpose

The principal objectives of this test are (1) to maintain control of and knowledge about the quality of the reactor coolant chemistry, and (2) to determine that the sampling equipment, procedures, and analytic techniques are adequate to supply the data required to demonstrate that the coolant chemistry meets water quality specifications and process requirements.

Secondary objectives of the test program include data to evaluate the performance of the fuel, operation of the demineralizers and filters, condenser integrity, operation of the off-gas system, and calibration of certain process instruments.

Description

Prior to fuel loading, a complete set of chemical and radiochemical samples will be taken to ensure that all sample stations are functioning properly and to determine initial concentrations. Subsequent to fuel loading during reactor heatup and at each major power level change, samples will be taken and measurements will be made to determine the chemical and radiochemical quality of reactor water and reactor feedwater, amount of radiolytic gas in the steam, gaseous activities leaving the air ejectors, decay times in the off-gas lines, and performance of filters and demineralizers. Calibrations will be made of monitors in the stack, liquid waste system, and liquid process lines.

Criteria

Level 1

Water quality must be known and must conform to the Water Quality Specifications at all times.

The activities of gaseous and liquid effluents must be known and they must conform to license limitations.

Chemical factors defined in the Technical Specifications must be maintained within the limits specified.

(2) Radiation Measurements

Purpose

To determine the background gamma and neutron radiation levels in the plant environs prior to operation in order to provide base data on activity buildup. Also to monitor radiation at selected power levels to assure the protection of personnel and continuous compliance with the guideline standards of 10CFR20 during plant operation.

Description

A survey of natural background radiation throughout the plant site will be made prior to fuel loading. Subsequent to fuel loading, during reactor heatup and at power levels of 25 percent, 50 percent, and 100 percent of rated power, gamma radiation level measurements and, where appropriate, thermal and fast neutron dose rate measurements, will be made at significant locations throughout the plant. All potentially high radiation areas will be surveyed.

Criteria

Level 1

The radiation doses of plant origin and occupancy times shall be controlled consistent with the guidelines of the standards for protection against radiation outlined in 10CFR20 NRC General Design Criteria.

(3) Fuel Loading

Purpose

The purpose of this test is to load fuel safely and efficiently to the full core size.

Description

Prior to fuel loading, control rods will be installed and tested. A neutron source of approximately 10^{-7} neutrons per second will be installed near the center of the core. At least three neutron detectors calibrated and connected in noncoincident mode to high flux scram trips will be located to produce acceptable signals during loading.

Fuel loading will begin at the center of the core and proceed radially to the fully loaded configuration. The following checks will be performed as each cell is loaded.

1. Subcriticality Check - A control rod surrounded by fuel in the vicinity of the cell to be loaded will be completely withdrawn; the core must remain subcritical. Then the rod will be reinserted.

2. Control Rod Function Test - The rod in the cell to be loaded will be completely withdrawn and reinserted.

3. Fuel Loading - Two fuel assemblies will be loaded, the blade guide removed, and the remaining two fuel assemblies loaded to complete the four-assembly cell.

4. The Subcriticality Check will be repeated.

5. The Control Rod Functional Test will be repeated. This also serves as a Subcriticality Check on the loaded fuel cell.

Shutdown margin demonstrations will be performed periodically during fuel loading.

Criteria

Level 1

The criteria for successful completion of this test are that (1) the core is fully loaded, and (2) the core shutdown margin demonstration has been completed.

(4) Shutdown Margin

Purpose

The purpose of this test is to demonstrate that the reactor will be subcritical throughout the first fuel cycle with any single control rod fully withdrawn.

Description

This test will be performed repeatedly during fuel loading and once in the fully loaded core at ambient temperature in the xenon-free condition. Subcriticality will be demonstrated with the strongest rod fully withdrawn and an adjacent rod pulled to a position calculated to be equal to a shutdown margin specified to account for expected reactivity changes during core lifetime. This calculated margin also allows for geometric and materials asymmetries in the core plus a further allowance for calculational errors.

Criteria

Level 1

a) The partially loaded and initially fully loaded core must be subcritical by at least R + 0.25% ΔK with any rod fully withdrawn. (R% ΔK is the amount by which the core reactivity, at any time in the first fuel cycle, is calculated to be higher than the value for new fuel.)

b) The fully loaded core must be subcritical throughout the fuel cycle with any rod fully withdrawn.

(5) Control Rod Drives

Purpose

To demonstrate that the CRDS operates properly over the full range of primary coolant temperatures and pressures from ambient to operating, and particularly that thermal expansion of core components does not bind or significantly slow control rod movements. Also, to determine the initial operating characteristics of the entire CRDS.

Description

The CRD tests performed during Phases 1 through 3 of the startup test program are designed as an extension of the tests performed during the preoperational CRDS tests. Thus, after it is verified that all CRDS operate properly when installed, they are tested periodically during heatup to assure that there is no significant binding caused by thermal expansion of the core components. A list of all CRD tests to be performed during startup testing is in Table 13.5.4.

Criteria

Level 1

Each CRD must have a normal withdraw speed less than or equal to 3.6 in/sec (9.14 cm/sec), indicated by a full 12-ft stroke in greater than or equal to 40 sec.

Upon scramming, the average of the insertion times of all operable control rods, exclusive of circuit response times, must be no greater than:

Percent Inserted	Insertion Time (sec)	
5	0.375	Scram time is measured
20	0.90	from time pilot scram
50	2.0	valve solenoids are
90	5.0	deenergized.

The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two-by-two array shall be no greater than:

Percent Inserted	Insertion Time (sec)	
5	0.398	Scram time is measured
20	0.950	from time pilot scram
50	2.120	valve solenoids are
90	5.300	deenergized.

Level 2

Each CRD must have a normal insert or withdraw speed of 3.0 ± 0.6 in/sec (7.62 ± 1.52 cm/sec), indicated by a full 12-ft stroke in 40 to 60 sec.

With respect to the CRD friction tests, if the differential pressure variation exceeds 15 psid for a continuous drive in, a settling test pressure should not be less than 30 psid nor should it vary by more than 10 psid over a full stroke. Lower differential pressures are indicative of excessive friction.

(6) Control Rod Sequence

Purpose

To achieve initial criticality in a safe and efficient manner for each of the two withdrawal sequences. Also to determine the effect on reactor power of control rod motion at various operating conditions.

Description

Two complementary control rod withdrawal sequences have been calculated which completely specify control rod withdrawals from the all-rods-in condition to the rated power configuration. Each sequence will be used to attain cold criticality. Critical rod patterns will be recorded periodically as the reactor is heated to rated temperature.

Movement of rods in a prescribed sequence is monitored by the rod worth minimizer, which will prevent out-of-sequence withdrawal. Also, not more than two rods may be inserted out of sequence.

(7) Rod Pattern Exchange

Purpose

To perform a representative change in basic rod pattern at a reasonably high reactor power level.

Description

The control rod pattern is exchanged by control rod insertions and withdrawals in a planned sequence. This will be done at a reduced power starting from full power in one control rod withdrawal sequence and will end with the reactor at full power in a rod pattern in the alternate withdrawal sequence.

Criteria

Level 2

Completion of the exchange of one rod pattern for the complementary pattern with the continued satisfaction of all licensed core limits constitutes satisfaction of the requirements of this procedure.

(8) SRM Performance

Purpose

To demonstrate that the operational sources, SRM instrumentation, and rod withdrawal sequences provide adequate information to the operator during startup.

Description

The operational neutron sources will be installed and source range monitor count-rate data will be taken during rod withdrawals to critical and compared with stated criteria on signal count-to-noise count ratio.

Criteria

Level 1

There must be a neutron signal count-to-noise count ratio of at least 2 to 1 on the required operable SRM's.

There must be a minimum count rate of 3 counts/sec on the required operable SRM's.

(9) IRM Performance

Purpose

The purpose of this test is to adjust the IRMS to obtain an optimum overlap with the SRMS and APRMS.

Description

Initially the IRMS is set to maximum gain. After the APRM calibration, the IRM gains will be adjusted to optimize the IRM overlap with the SRM's and APRM's.

Criteria

Level 1

Each IRM channel must be adjusted so the overlap with the SRM's and APRM's is assured. The IRM's must produce a scram at 120 on a full scale of 125.

(10) LPRM Calibration

Purpose

To calibrate the LPRMS.

Description

The LPRM channels will be calibrated to make the LPRM readings proportional to the neutron flux in the narrow-narrow water gap at the chamber elevation. Calibration factors will be obtained through the use of either an off-line or a process computer calculation that relates the LPRM reading to average fuel assembly power at the chamber height.

Criteria

Level 1

With the reactor in the rod pattern and at the power level at which the calibration is to be performed, the meter reading of each LPRM chamber will be proportional to the average heat flux in the four adjacent fuel rods at the height of the chamber.

(11) APRM Calibration

Purpose

To present the method for calibrating the APRM channels.

Description

A heat balance will be made at least once each shift and after each major power level change. Each APRM channel reading will be adjusted to be consistent with the core thermal power as determined from the heat balance. During heatup a preliminary calibration will be made by adjusting the APRM amplifier gains so that the APRM readings agree with an initial heat balance based on coolant temperature rise data.

Criteria

Level 1

The APRM channels must be calibrated to read equal to or greater than the actual core thermal power.

(12) Process Computer

Purpose

To verify the performance of the process computer under operating conditions.

Description

GE/PAC computer system program verifications and calculational program validations at static and at simulated dynamic input conditions will be preoperationally tested at the computer supplier's site and following delivery to the plant site. Following fuel loading, during plant heatup and the ascension to rated power, the nuclear steam supply system and the balance-of-plant system process variables sensed by the computer as digital or analog signals will become available. Verify that the computer is receiving correct values of sensed process variables and that the results of performance calculations of the nuclear steam supply system and the balance-of-plant are correct. Verify proper operation of all computer functions at rated power operating conditions.

Criteria

Level 2

Program OD-1 and P-1 will be considered operational when (1) the MCHFR calculated by an independent method and the process computer either (a) are in the same fuel assembly and do not differ in value by more than 10 percent, or (b) if two different fuel assemblies are chosen by the two methods, the CHFR calculated by the other method in each assembly agreed with the MCHFR in that assembly by not more than 10 percent, and (2) when the LPRM calibration factors calculated by the independent method and the process computer agree to within 5 percent.

The remaining programs will be considered operational upon successful completion of static testing.

(13) Reactor Core Isolation Coolant System

Purpose

To verify the operation of the RCICS at operation reactor pressure conditions.

Description

A controlled start of the RCICS will be done at a reactor pressure of 150 psig and a quick start will be done at a reactor pressure of 1,000 psig. Verify proper operation of the RCICS and determine time to reach rated flow. These tests may first be performed with the system in the test mode so that discharge flow will not be routed to the reactor pressure vessel. The final demonstration will be made so that discharge flow will be routed to the reactor pressure vessel while the reactor is at partial power.

Criteria

Level 1

The RCICS must have the capability to deliver rated flow, 600 gpm, in less than or equal to the rated actuation time, 30 sec, against rated reactor pressure.

(14) High-Pressure Coolant Injection System

Purpose

To verify the proper operation of the HPCIS throughout the range of reactor pressure conditions.

Description

Controlled starts of the HPCIS will be done at reactor pressures near 150, and 1,000 psig during the heatup phase, and a quick start will be initiated during Phase 3. Verify proper operation of the HPCIS, determine time to reach rated flow, adjust flow controller in HPCIS for proper flow rate and adjust overspeed trip of HPCI turbine. These tests will be performed with the system in the test mode so that discharge flow will not be routed to the reactor pressure vessel. The final demonstration will be made so that discharge flow will be routed to the reactor pressure vessel while the reactor is at partial power.

Criteria

Level 1

The time from actuating signal to required flow must be less than 25 sec with reactor pressure at 1,000 psig. With pump discharge pressure at 1,220 psig the flow should be at least 5,000 gpm. The HPCI turbine must not trip off during startup.

(15) Selected Process Temperatures

Purpose

The purposes of this test are (1) to establish the minimum recirculation pump speed which will maintain water temperature in the bottom head of the reactor vessel within 145°F (80°C) of reactor coolant saturation temperature as determined by reactor pressure, and (2) to provide assurance that the measured bottom head drain temperature corresponds to bottom head coolant temperature during normal operations.

Description

The applicable reactor parameters will be monitored during the initial heatup, the initial cooldown, and after recirculation pump trips in order to determine that adequate mixing of the reactor water is occurring in the lower plenum of the pressure vessel. The adequacy of the bottom-drain-line thermocouple as a means for measuring the bottom reactor vessel temperature will also be determined.

Criteria

Level 1

The reactor recirculation pump flow shall not be increased unless the coolant temperatures in the upper and lower regions of the vessel are within 145°F (80°C) of each other.

The pump in an idle recirculation loop shall not be started unless the temperature of the coolant within the loop is within 50°F (28°C) of the active loop temperature.

(16) System Expansion

Purpose

To verify that the reactor drywell piping system is free and unrestrained in regard to thermal expansion and that suspension components are functioning in the specified manner. The test also provides data for calculation of stress levels in nozzles and weldments.

Description

Observe and record the horizontal and vertical movements of major equipment and piping in the nuclear steam supply system and auxiliary systems to assure components are free to move as designed. Adjust as necessary for freedom of movement.

Criteria

Level 1

There shall be no evidence of blocking of the displacement of any system component caused by thermal expansion of the system.

Hangers shall not be bottomed out or have the spring fully stretched.

Level 2

Displacements of instrumented points shall not vary from the calculated values by more than ±50 percent or ±0.5 in, whichever is larger.

(17) Core Power Distribution

Purpose

To (1) confirm the reproducibility of the TIPS readings, (2) determine the core power distribution in three dimensions, and (3) determine core power symmetry.

Description

A check of the reproducibility of the TIP traces is made twice; the first time the TIPS is used, and once again at a later date after the TIPS has been used a number of times and is "broken in." The check is made with the plant at steady-state condition by producing several TIP traces in the same location with each TIP machine. The traces are evaluated to determine the extent of deviations between traces from the same TIP machine.

Core power distribution including power symmetry will be obtained during the power ascension program. Axial power traces will be obtained at each of the TIP locations. Several TIPS's have been provided to obtain these traces a common location can be traversed by each TIP chamber to permit intercalibration. The results of the complete set of TIP traces will be evaluated to determine core power symmetry.

Criteria

Level 2

In the TIP reproducibility test, the TIP traces shall be reproducible within ±3.5 percent relative error or ±0.15 in (3.8 mm) absolute error at each axial position, whichever is greater.

(18) Core Performance

Purpose

To evaluate the core performance parameters of the core flow rate, core thermal power level, maximum fuel rod surface heat flux, and MCHFR.

Description

Core power level, maximum heat flux, recirculation flow rate, hot channel coolant flow, MCHFR, fuel assembly power, and steam qualities will be determined at existing power levels and assumed overpower conditions. Plant and in-core instrumentation, conventional heat balance techniques, and core performance worksheets and nomograms will be used. This will be performed above 10 percent power and at various pumping conditions and can be done independent of the process computer (now PMS) functions.

Criteria

Level 1

Reactor power, maximum fuel surface heat flux, and MCHFR must satisfy the following limits:

Maximum fuel rod surface heat flux shall not exceed 428,300 Btu/hr-sq ft during steady-state conditions when evaluated at the operating power level.

MCHFR shall not be less than 1.9 when evaluated at the operating power level. The basis for evaluation of MCHFR shall be "Design Basis for Critical Heat Flux Condition in BWR's." APED-5286, September 1966.

Steady-state reactor power shall be limited to values on or below the licensed flow control line (maximum power of 3,293 MWt with flow of at least 102.5 x 10^6 lb/hr).

(19) Steam Production

Purpose

To demonstrate that the reactor steam production rate is satisfied.

Description

Operate continuously for 100 hr at rated reactor conditions. When it is determined that all plant conditions are stabilized, the steam production rate will be measured during a 2-hr period at conditions prescribed in the nuclear steam generating system warranty.

Criteria

Level 1

The nuclear steam supply system must produce 13,380,000 lb/hr of steam of not less than 99 percent quality at a pressure of 985 psia at the second isolation valves, when operating at specified warranty conditions.

(20) Flux Response to Rods

Purpose

To demonstrate stability in the power reactivity feedback loop with increasing reactor power and to determine the effect of control rod movement on reactor stability.

Description

Rod movement tests will be made at chosen power levels to demonstrate that the transient response of the reactor to a reactivity perturbation is stable for the full range of reactor power. A centrally located rod will be moved, and the neutron flux signal from a nearby LPRM chamber will be measured and evaluated to determine the dynamic effects of rod movement.

Criteria

Level 1

The decay ratio must be less than 1.0 for each process variable that exhibits oscillatory response to control rod movement.

Level 2

The decay ratio is expected to be less than or equal to 0.25 for each process variable that exhibits oscillatory response to control rod movement when the plant is operating above the lower limit setting of the master flow controller.

(21) Pressure Regulator

Purpose

To (1) determine the reactor pressure control system responses to pressure regulator set point changes, (2) demonstrate the stability of the reactivity-void feedback loop to pressure perturbations, (3) demonstrate the control characteristics of the bypass and control valves, (4) demonstrate the "take-over" capabilities of the backup pressure regulator, and (5) optimize the pressure regulator settings to give the best combination of fast response and small overshoot.

Description

The pressure set point will be decreased rapidly and then increased rapidly by about 10 psi and the response of the system will be measured in each case. The backup regulator will be tested by increasing the operating pressure regulator set point rapidly until the backup regulator takes over control. The load reference set point will be reduced, and the test repeated with the bypass valve in control. The response of the system will be measured and evaluated and regulator settings will be optimized.

Criteria

Level 1

The decay ratio must be less than 1.0 for each process variable that exhibits oscillatory response to pressure regulator changes.

Level 2

In all tests of the simulated failure of the operating pressure regulator, the decay ratio is expected to be less than or equal to 0.25 for each process variable that exhibits oscillatory response to pressure regulator changes when the plant is operating above the lower limit setting of the master flow controller.

During the simulated failure of the operating pressure regulator the backup regulator is expected to control the transient such that the reactor does not scram.

Steady-state hunting or limit cycle characteristics due to control and valve deadband must be less than ±0.5 percent at rated steam flow.

(22) Feedwater System

Purpose

To (1) demonstrate acceptable reactor water level control, (2) evaluate and adjust feedwater controls, (3) demonstrate capability of auto. flow runback feature to prevent low water level scram following trip of one feedwater pump, (4) demonstrate adequate response to feed heater loss, and (5) demonstrate general reactor response to inlet subcooling changes.

Description

Reactor water level set point changes of approximately ±6 in will be used to evaluate and acceptably adjust the feedwater control system settings for all power and feedwater pump modes.
One of the three operating feedwater pumps will be tripped at 75 percent and full power while the automatic flow runback circuit acts to drop power to within the capacity of the remaining two pumps. One feedwater heater extraction will be closed and the resulting transients recorded.

Criteria

Level 1

The decay ratio must be less than 1.0 for each process variable that exhibits oscillatory response to feedwater system changes.

Level 2

The decay ratio is expected to be less than or equal to 0.25 for each process variable that exhibits oscillatory response to feedwater system set point changes when the plant is operating above the lower limit of the master flow controller. System response for large transients should not be unexplainably worse than preanalysis. The automatic flow runback feature will prevent a scram from low water level following a trip of one feedwater pump.

(23) Bypass Valves

Purpose

To demonstrate the ability of the pressure regulator to minimize the reactor disturbance during an abrupt change in reactor steam flow and to demonstrate that a bypass valve can be tested for proper functioning at rated power without causing a high flux scram.

Description

One of the turbine bypass valves will be tripped open by a test switch. The pressure transient will be measured and evaluated to aid in making final adjustments to the pressure regulator.

Criteria

Level 1

The decay ratio must be less than 1.0 for each process variable that exhibits oscillatory response to bypass valve changes.

Level 2

The decay ratio is expected to be less than or equal to 0.25 for each process variable that exhibits oscillatory response to bypass valve changes when the plant is operating above the lower limit setting of the master flow controller.

The maximum pressure decrease at the turbine inlet should be less than 50 psig to avoid approaching low steam line pressure isolation or cause excessive water level swell in the reactor.

(24) Main Steam Isolation Valves

Purpose

To (1) functionally check the MSIV's for proper operation at selected power levels, (2) determine reactor transient behavior during and following simultaneous full closure of all MSIV's and following closure of one valve, and (3) determine isolation valve closure time.

Description

Functional checks (10 percent closure) of each isolation valve will be performed at selected reactor power levels. A test of the simultaneous full closure of all MSIV's will be performed at about

100 percent of rated thermal power. Correct performance of the RCIC and relief valves will be shown. Reactor process variables will be monitored to determine the transient behavior of the system during and following full isolation. MSIV closure times will be determined. The maximum power conditions at which individual valve full closure tests can be performed without a reactor scram is to be established.

Criteria

Level 1

MSIV stroke time will be between 3 and 5 sec, exclusive of electrical delay time. Reactor pressure shall be maintained below 1,230 psig, the set point of the first safety valve, during the transient following closure of all valves. During full closure of individual valves, scram should not occur.

Level 2

The maximum reactor pressure should be about 1,200 psig; 30 psi below the first safety valve set point following closure of all valves. This is a margin of safety for safety valve weeping. During full closure of individual valves, pressure must be 20 psi below scram, neutron flux must be 10 percent below scram, and steam flow in individual lines must be below the trip point.

(25) Relief Valves

Purpose

To verify the proper operation of the dual purpose relief safety valves, to determine their capacity, and to verify their leaktightness following operation.

Description

The main steam relief valves will each be opened manually so that at any time only one is open. Capacity of each relief valve will be determined by the amount the bypass or control valves close to maintain reactor pressure. Proper reseating of each relief valve will be verified by observation of temperatures in the relief valve discharge piping.

Criteria

Level 2

Each relief valve is expected to have a capacity of at least 800,000 lb/hr at a pressure setting of 1,080 psig. Relief valve leakage must be low enough that the temperature measured by the thermocouples in the discharge side of the valves falls to within 10°F of the temperature recorded before the valve was opened. Each valve must move from fully closed to fully opened in 0.3 sec.

(26) Turbine Stop and Control Valve Trips

Purpose

The purpose of this test is to demonstrate the response of the reactor and its control systems to protective trips in the turbine and generator.

Description

The turbine stop valves will be tripped at selected reactor power levels and the main generator breaker will be tripped in such a way that a load imbalance trip occurs. Several reactor and turbine operating parameters will be monitored to evaluate the response of the bypass valves, relief valves, RPS, and the effect of recirculation pump overspeed, if any, during the control valve trip. Additionally, the peak values and change rates of reactor steam pressure and heat flux will be determined. The ability to ride through a load rejection at 25 percent power without a scram will be demonstrated.

Criteria

Level 1

Reactor pressure shall be maintained below 1,230 psig, the set point of the first safety valve during the transient following fast closure of the turbine stop and/or control valves.

Reactor thermal power, as indicated by the simulated heat flux readout, must not exceed the safety limit line.

The turbine control valves must begin to close before the stop valves during the stop valve trip.

Feedwater system settings must prevent flooding of the steam line following these transients.

Level 2

The maximum reactor pressure should be less than 1,200 psig, 30 psi (2.1 kg/sq cm) below the first safety valve set point, during the transient following fast closure of the turbine stop and control valves. This pressure margin should prevent safety valve weeping.

The measurement of simulated heat flux must not be significantly greater than preanalysis.

The trip at 25 percent power must not cause a scram. The trip scram function for higher power levels must meet RPS specifications. The pressure regulator must regain control before a low-pressure reactor isolation occurs.

Feedwater control adjustments shall prevent low level initiation of the HPCIS and main steam isolation as long as feedwater flow remains available.

(27) Shutdown From Outside the Control Room

Purpose

The purpose of this test is to demonstrate that the reactor can be brought from a normal initial steady-state power level to the point where cooldown is initiated and under control with reactor vessel pressure and water level controlled from outside the control room.

Description

The test will simulate the reactor shutdown following a control room evacuation. The reactor will be scrammed from a normal steady-state condition and the MSIV's will be closed. Following this event, the vessel water level and pressure will be controlled from outside the control room.

Criteria

Level 2

During a simulated control room evacuation, the reactor must be brought to the point where cooldown is initiated and under control, and the reactor vessel pressure and water level are controlled using equipment and controls outside the control room.

(28) Flow Control

Purpose

To determine the plant response to changes in recirculation flow and thereby adjust the local control loops. Also to examine the plant overall load following capability in order to establish correct interfacing of the pressure and flow control systems including final settings for the master and local flow controllers.

Description

Various process variables will be recorded while step changes are introduced into the recirculation flow control system (increased and decreased) at chosen points on the 50 percent, 75 percent, and 100 percent load lines. Up to 30 percent/min change in recirculation flow will be made from the 100 percent flow condition down to the lower limit of the master flow controller and return on the 100 percent load line. Load following capability will be demonstrated in the automatic flow control mode.

Criteria

Level 1

The decay ratio must be less than 1.0 for each process variable that exhibits oscillatory response to flow control changes.

Level 2

The decay ratio is expected to be less than or equal to 0.25 for each process variable that exhibits oscillatory response to flow changes when the plant is operating above the lower limit setting of the master flow controller. Scram must not occur. Automatic flow control range must be at least 80 percent to 100 percent power along the full power load line. Load response to a 20 percent load demand step must be at least 0.5 percent/sec; limit cycles in the steam flow shall not be greater than ±0.5 percent.

(29) Recirculation System

Purpose

To (1) determine transient responses and steady-state conditions following recirculation pump trips at selected reactor power levels and to obtain jet pump performance data, and (2) calibrate the jet pump flow instrumentation.

Description

Both recirculation pumps will be simultaneously tripped at power levels of 50 percent, 75 percent, and 100 percent of rated power. Single pump trips will be performed at the same power levels. The single pump trips will be initiated by opening the generator field breaker. Two pumps trips will be initiated by tripping the M-G set drive motors. Reactor pressure, steam and feedwater flow, jet pump delta P, and neutron flux will be recorded during the transient and at steady-state conditions. MCHFR evaluations will be made for conditions encountered during the transient. The jet pump instrumentation will be calibrated to read total core flow, based on data obtained at the various test levels.

Criteria

Level 1

Transient MCHFR shall be greater than 1.0 at all times.

Level 2

For each pump trip test, the minimum transient MCHFR based on operating data divided by the minimum transient MCHFR evaluated from design values is expected to be equal to or greater than 1.0.

Flow instrumentation has been calibrated such that the reactor jet pump total flow recorder provides correct flow indication.

(30) Loss of Turbine-Generator and Off-Site Power
Purpose

To demonstrate proper performance of the reactor and the plant electrical equipment and systems during the loss of auxiliary power transient.

Description

The loss of auxiliary power test will be performed at 25 percent of rated power. The proper response of reactor plant equipment, automatic switching equipment, and the proper sequencing of the diesel-generator load will be checked. Appropriate reactor parameters will be recorded during the resultant transient.

Criteria

Level 1

All test pressure transients must have maximum pressure values below 1,230 psig which is the set point of the first safety valve. All safety systems, such as the RPS, the diesel-generator, RCIC, and HPCI, must function properly without manual assistance.

Level 2

Normal reactor cooling systems should be able to maintain adequate torus water temperature, adequate drywell cooling and prevent actuation of the auto depressurization system. The maximum reactor pressure should be 30 psi below the first safety valve set point. This is a margin of safety for safety valve weeping.

(31) Recirculation Loop Control

Purpose

To determine the as-built characteristics of the recirculation control system including the drive motor, fluid coupler, generator, drive pump, and jet pumps.

Description

During heatup, check for functional instabilities in the fluid coupler. When power level allowing 100 percent flow has been reached, record a steady-state gain curve of generator speed versus function generator to make the curve as linear as possible. Make several small step changes in speed at the input to the function generator at various pump speeds and record appropriate recirculation loop transient signals.

Make appropriate speed controller settings after closing each speed control loop and demonstrate performance over the full speed range with small speed demand step tests.

Criteria

Level 2

It is expected that at the completion of this test speed control and function generator settings will be found which will give adequate loop response with a decay ratio of 0.25 or less over the entire range from about 20 percent to 100 percent speed.

(32) Vibration

Purpose

Determine the vibration characteristics of reactor internals and recirculation loops induced by hot two-phase forces.

Description

A confirmatory type vibration program will be performed. Vibratory responses will be recorded at various recirculation flow rates.

Criteria

Level 1

The criteria by which the results of the vibration tests will be judged involve complex, precalculated relationships among spatial locations, vibrational amplitudes, and vibrational frequencies as related to stress and limited by ASME Code, Section III. A complete set of these relationships will be used on site by the shift test engineer and the vibration test consultant to judge the adequacy of the results of the tests.

Description

With the reactor in a cold flow condition, measure the water level profile at various water levels as a function of core flow using eight water level probes installed along a radius of the steam separator. Since individual steam separator efficiency is a function of local water level, measurement of the water level profile will give an indication of overall steam separator assembly performance. Compare the experimental water levels with those indicated by plant instrumentation.

Criteria

Not Applicable.

13.5.3 EPU Power Ascension Test Program

13.5.3.1 General

The purpose of the EPU power ascension test program is to demonstrate that SSCs will perform satisfactorily in service at the new reactor thermal power (RTP) level (3951 MWt). The test program also provides additional assurance that the plant will continue to operate in accordance with all design criteria at new RTP conditions. The PBAPS EPU power ascension test program was developed using information from several sources.

a. The startup and power ascension testing that was done during initial plant startup as discussed in the previous sections of this chapter.
b. The testing that was done at the time of PBAPS's stretch power uprate in 1994 and 1995 and the Measurement Uncertainty Recapture (MUR) Power Uprate in 2002 and 2003. Information on this testing was obtained from the test plans and procedures used at those times.
c. The guidance for extended power uprates in GE topical report NEDC-32424P-A "Generic Guidelines for General

Electric Boiling Water Reactor Extended Power Uprate" (ELTR-1), including the NRC's Requests for Additional Information, GE responses and the NRC's staff position on the ELTR-1 documented in NRC letter dated February 8, 1996.

d. The guidance for extended power uprate in GE topical report NECD-32523P-A "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate" (ELTR-2) including Supplement 1, Volumes 1 and 2 and the NRC's staff position on the ELTR-2 documented in NRC letter dated September 14, 1998.
e. The guidance for constant pressure power uprates (CPPU) in GE Topical Report NEDC-33004P-A "Constant Pressure Power Uprate" (CLTR), including the NRC's Requests for Additional Information, GE responses and the NRC's safety evaluation of the CLTR documented in NRC letter dated March 31, 2003.
f. Information and data from plant transients that have occurred during PBAPS's operating history as applicable.
g. The NRC Standard Review Plan (SRP), NUREG-0800, Section 14.2.1 "Generic Guideline for Extended Power Uprate Testing Programs."
h. Experience from other BWR plants that have implemented EPUs.

13.5.3.2 Power Ascension Testing

Standard Review Plan 14.2.1 (NUREG-0800) requires a comparison of the EPU power ascension tests against the original power ascension tests. The scope of the comparison includes the tests performed at greater than 80% original power and those performed at less than 80% power that would be invalidated by the increase in rated thermal power. The tests identified were evaluated and either justified as not required or included in the PEU power ascension test plan. Table 13.5.5 shows the power ascension tests performed for the original startup, stretch uprate, MUR uprate and the EPU.

The aggregate impact of modifications required to achieve the new rated thermal power is also included in the final power ascension test procedure. The complete power ascension test procedure was approved by the Plant Operational Review Committee (PORC). All new modifications are tested at various intermediate power levels to assure there is no impact on the overall impact on operation of PBAPS at the new rated thermal power.

Large transient testing such as closure of all MSIVs and turbine trip which were performed during the initial power ascension test program were evaluated in accordance with SRP 14.2.1 guidance. Elimination of these large transient tests was justified based on the following:

a. Previous PBAPS and industry operating experience
b. No new thermal-hydraulic phenomena introduced
c. No new system interactions
d. PBAPS conformance to limitations associated with analysis results
e. Plant staff familiarization with PBAPS operation and use of operating procedures and EOPs
f. Guidance contained in GEH topical reports
g. Risk implications of performing the test

The initial power ascension to the new rated thermal power (3951 MWt) was conducted in a deliberate planned approach. Power was increased to the prior rated thermal power (3514 MWt), stabilized, and then reduced to approximately 80% of the new rated thermal power. Tests were conducted at this level then again after power was increased to approximately 84.5% and again at 89% (3514 MWt). The data from these tests was used to verify baseline performance with prior operation. Power was increased to the new rated thermal power in increments of approximately 4% power. The tests were performed at each level and results reviewed and approved by plant management prior to increasing power to the next level. In addition, the NRC imposed license conditions for supplemental testing and reviews for FIV and steam dryer testing (EPU License Amendment 293 and 296). The test results from FIV accelerometers and the replacement steam dryer testing were transmitted to the NRC for their review at each testing plateau. The replacement steam dryer test data included an evaluation of strain gauge data and the comparison to the data taken from the Unit 2 steam dryer instrumentation. The power ascension tests were performed at the power levels shown in Table 13.5.5.

TABLE 13.5.1

MAJOR PLANT TRANSIENTS

TEST TITLE		TEST CONDITION				
	Approximate Power (% rated)	25	50	75	80	100
	Approximate Core Flow (% rated)	37	100	100	70	100
Feedwater Pump Trip				X		X
MSIV's (one valve)			X	X	X	
MSIV's (all valves)						X
T-G Stop Valve Fast Close			X			X
T-G Control Valve Fast Close		X		X	X	
Recirc Pump Trip (one)			X	X		X
Recirc Pump Trip (two)			X	X		X
Loss of Gen & Off-site Power		X				

TABLE 13.5.2

STABILITY TESTS

TEST CONDITION

TEST TITLE												
Approximate Power (% rated)	25	40	50	27	50	60	75	37	60	72	100	50
Approximate Core Flow (% rated)	37	75	100	NC	52	70	100	NC	51	68	100	NC
Flux Response to Rods	X	X	X	X	X	X	X	X	X	X	X	X
Press Reg Set Point	X	X	X	X	X	X	X	X	X	X	X	X
Press Reg Backup Reg	X		X				X				X	
FW System Set Point	X	X	X	X	X	X	X	X	X	X	X	X
FW System Drop Heater											X	
Bypass Valve	X	X	X	X	X	X	X	X	X	X	X	X
Flow Control	X	X	X		X	X	X		X	X	X	

TABLE 13.5.3

STARTUP TEST PROGRAM

	Open Vess. or Cold Test	Heat Up	50% Flow Line				75% Flow Line				100% Flow Line				WARRANTY
Power[1], %			15-25	35-40	45-55	~27	45-55	55-65	70-80	~37	55-65	67-77	95-100	~50	
Flow[2], %			~37	75	~100	NC	52	70	~100	NC	~51	~68	100	NC	
TEST CONDITION (Figure 13.5.2)			1	4	5	6	8	9	10	11	13	14	15	16	
Chemical	X	X	X		X				X				X		
Radiation Measurement	X	X	X		X								X		
Fuel Loading	X														
Shutdown Margin	X														
CRD	X	X	X		X							X	X		
Control Rod Sequence	X	X	X	X			X				X		X		
Rod Pattern Exchange[3]															
SRM Performance	X	X													
IRM Performance	X	X	X												
LPRM Calibration			X		X				X				X		
APRM Calibration		X	X	X	X		X		X				X		X
Process Computer	X	X	X												
RCIC		X	L												
HPCI		X			M										
Selected Process Temperatures		X													
System Expansion		X	X												
Core Power Distribution			X		X				X				X		X
Core Performance		X	X	X	X	X	X	X	X	X	X	X	X	X	X
Steam Production															X
Flux Response to Rods			L	M	M	X	M	M	M	X	M	M	M	X	
Press. Reg.:															
:Set Point Changes			L	M	M	X	M	M	M	X	M	M	M	X	
:Backup Regulator			L		M				M				M		
FW System: FW Pump Trip									M				M		
:Water Level Stpt. Chg.			L	M	MA	X	M	M	MA	X	MA	MA	MA	X	
:Heater Loss[3]															
Bypass Valves			L	MA	MA	X	MA	MA	MA	X	MA	MA	MA	X	
Main Steam Iso. Valves:															
:Each Valve		X			M,SP				M,SP			M,SP			
:Full Iso.													M,SE		
Relief Valves		X	L						M						
Turbine Stop and					M,SE								M,SE		
Control Valve Trips			L,SP						M,SE			A,SE			
Flow Control			L	MA	MA		MA	MA	MA		MA	MA	MA		
Recirc. System:															
:One Pump Trip					M				M				M		
:Two Pump Trip					M				M				M		
:Flow Calibration			X		X				X				X		
Loss of T-G & Off-Site Power			L,SE												

TABLE 13.5.3 (Continued)

	Open Vess. or Cold Test	Heat Up	50% Flow Line				75% Flow Line				100% Flow Line				WARRANTY
Power[1], %			15-25	35-40	45-55	~27	45-55	55-65	70-80	~37	55-65	67-77	95-100	~50	
Flow[2], %			~37	75	~100	NC	52	70	~100	NC	~51	~68	100	NC	
TEST CONDITION			1	4	5	6	8	9	10	11	13	14	15	16	
Recirc Loop Control		X		X	X										
Vibration					M				M				M		
Shutdown Outside Control Room			X												

KEY: X = Test independent of flow control mode
M = Master manual control mode
A = Automatic control mode
L = Local manual control mode
SE = Scram expected
SP = Scram possibility
NC = Natural circulation

(1) Power is in percent of rated power, 3,293MWt.

(2) Flow is in percent of rated flow, $102.5x10^6$lb/hr.

(3) Conditions for performing this test to be finalized prior to actual startup testing.

TABLE 13.5.4

CONTROL ROD DRIVE SYSTEM TESTS

Test Description	Pre-op Tests	Reactor Pressure (psig) (With Core Loaded)			
		0	600	800	1000
Position Indication	All	All			
Normal Insert/ Withdrawn Times	All	All			4*
Coupling	All	All			
Friction	All				4*
Scram Times (Norm. Accum. Press.)	All	All	4*	4*	All
Scram Times (Min. Accum. Press.)		4*			
Scram Times (Scram Discharge Volume High Level)	All				
Scram Times, Rated Power (Norm. Accum. Press.)					4**

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* Value refers to the four slowest drives as determined from the normal accumulator pressure scram test at ambient reactor pressure.

** Scram times of the four slowest rods will be determined at 25 percent, 50 percent, and 100 percent of rated power during planned reactor scrams at these power levels.

TABLE 13.5.5
EPU POWER ASCENSION TEST PROGRAM

Original Test Description	Initial Test Power Level % OLTP (Original Licensed Thermal Power)	Stretch Uprate Power Level % OLTP (3458 MWt)	MUR Uprate Power Level % OLTP (3514 MWt)	Test Power Level % RTP (EPU) (≈MWt)					
				≤ 80 (3165)	84.5 (3340)	88.9 (3514)	92.6 (3660)	96.3 (3805)	100% (3951)
SUT-1 Chemical: A complete set of chemical and radiochemical samples were taken to ensure that all sample stations are functioning properly and to determine initial concentrations.	20, 50, 75, 100	100, 105	99, 105, 106.5		X	X	X	X	X
SUT-2 Radiation Measurement: A survey of natural background radiation throughout the plant site was made prior to fuel loading, during reactor heatup and at power levels of 25, 50, and 100 percent of rated power. Gamma radiation level measurements and, where appropriate, thermal and fast neutron dose rate measurements, were made at significant locations throughout the plant. All potentially high radiation areas were	25, 50, 100	95, 100, 102.5, 105	106.5		X	X	X	X	X

Original Test Description	**Initial Test Power Level % OLTP (Original Licensed Thermal Power)**	**Stretch Uprate Power Level % OLTP (3458 MWt)**	**MUR Uprate Power Level % OLTP (3514 MWt)**	**Test Power Level % RTP (EPU) (≈MWt)**					
				≤ 80 (3165)	**84.5 (3340)**	**88.9 (3514)**	**92.6 (3660)**	**96.3 (3805)**	**100% (3951)**
surveyed.									
SUT-3 Fuel Loading: To load fuel safely and efficiently to the full core size	<15% Note 1	None	None						
SUT-4 Shut down Margin: To demonstrate that the reactor will be subcritical throughout the fuel cycle with any single control rod fully withdrawn.	<15% Note 1	None	None						
SUT-5 CRD: CRDs are tested periodically during power ascension to verify proper operation over the full range of primary coolant temperatures and pressures.	25, 50, 75, 100	<85	Operator rounds at 105 & 106.5			X	X	X	X
SUT-6 Control Rod Sequence: To verify the acceptability of the specified control rod withdrawal sequence.	25, 40, 50, 60, 100	None	None						
SUT N/A Steam Dryer: (1) Perform the station test to verify that moisture carryover is within limits. (2) Obtain strain gauge data and confirm stresses are	None	None	None	X	X	X	X	X	X

Original Test Description	Initial Test Power Level % OLTP (Original Licensed Thermal Power)	Stretch Uprate Power Level % OLTP (3458 MWt)	MUR Uprate Power Level % OLTP (3514 MWt)	Test Power Level % RTP (EPU) (≈MWt)					
				≤ 80 (3165)	84.5 (3340)	88.9 (3514)	92.6 (3660)	96.3 (3805)	100% (3951)
with limit curves									
SUT N/A WRNM Calibration: To calibrate the WRNM system to EPU power level and confirm overlap with the APRM system.	Startup	Startup	Startup	X					
SUT N12 APRM/ PRNM Calibration: To calibrate the APRM/ PRNM system to the EPU power level	20, 35, 50, 75, 100	<85, 90, 95, 100, 102.5 & 105	105, 106.5	X	X	X	X	X	X
SUT 13 Process Computer: To verify the performance of the process computer under plant operating conditions.	20	None	None						
SUT 14 RCIC: To demonstrate the ability of the RCIC system to provide the required flow at various turbine steam supply and pump discharge pressures and to start from cold standby conditions.	20 (Manual)	<85, 105	106.5	X					X
SUT 15 HPCI: To demonstrate the ability of the HPCI system to provide the required flow at various turbine steam supply and pump discharge pressures	50(Manual)	<85, 105	106.5	X					X

Original Test Description	Initial Test Power Level % OLTP (Original Licensed Thermal Power)	Stretch Uprate Power Level % OLTP (3458 MWt)	MUR Uprate Power Level % OLTP (3514 MWt)	Test Power Level % RTP (EPU) (≈MWt)					
				≤ 80 (3165)	84.5 (3340)	88.9 (3514)	92.6 (3660)	96.3 (3805)	100% (3951)
and to start from cold standby conditions.									
SUT 17 System Expansion: The purpose of this test is to verify that: (1) The drywell piping is free and unrestrained in regard to thermal expansion and that suspension components are functioning in the specified manner. (2) Provides data for calculation of stress levels in nozzles and weldments.	20	None	None						
SUT 18 Core Power Distribution: The purpose of this test is to: (1) Obtain axial power distributions at various conditions of rod patterns, power levels, recirculation flow rate and sub cooling. (2) Verify the reproducibility of each TIP system. (3) Determine power distribution symmetry of octant-symetric control rod patterns.	20, 50, 75, 100	105	Satisfied during startup	X					X
SUT 19 Core	20, 40, 50,	95,100,	99,105,	X	X	X	X	X	X

Original Test Description	**Initial Test Power Level % OLTP (Original Licensed Thermal Power)**	**Stretch Uprate Power Level % OLTP (3458 MWt)**	**MUR Uprate Power Level % OLTP (3514 MWt)**	**Test Power Level % RTP (EPU) (≈MWt)**					
				≤ 80 (3165)	**84.5 (3340)**	**88.9 (3514)**	**92.6 (3660)**	**96.3 (3805)**	**100% (3951)**
Performance: The purpose of this test is to evaluate the core and thermal hydraulic performance to ensure parameters are within limits.	60, 75, 100	102.5,105	106.5						
SUT 21 Flux Response to Rods: The purpose of this test is to demonstrate the relative stability of the power-reactivity feedback loop with regard to small perturbations in reactivity caused by rod movement with increasing power.	20, 40, 50, 60, 75, 100	None	None						
SUT 22 Pressure Regulator: Verify the system response to (1) Set point changes and (2)verify the take-over of the Backup Regulator	(1) 20, 40, 50, 60, 75, 100 (2) 20, 50, 75, 100	(1) <85, 102.5 (2) None	(1) 99,105, 106.65 (2) None	X X	X X	X X	X X	X X	X ≤100%
SUT 23 FW System: Verify the system response to: (1) A RFP Trip. (2)Level setpoint changes. (3) Confirm the feedwater flow calibration. (4) Determine if the maximum feedwater runout capability is	(1) 75,100 (2) 20, 40, 50, 60, 75, 100	(1) Not performed (2) 102.5	(1) Not performed (2) 99, 105, 106.5	X	X	X	X	X	X ≤100%

Original Test Description	Initial Test Power Level % OLTP (Original Licensed Thermal Power)	Stretch Uprate Power Level % OLTP (3458 MWt)	MUR Uprate Power Level % OLTP (3514 MWt)	Test Power Level % RTP (EPU) (≈MWt)					
				≤ 80 (3165)	84.5 (3340)	88.9 (3514)	92.6 (3660)	96.3 (3805)	100% (3951)
compatible with the licensing assumptions for EPU conditions.									
SUT 24 Bypass Valves: Demonstrate: (1) The ability of the pressure regulator to minimize the reactor pressure disturbance during a small step change in reactor steam flow. (2) That the bypass valves can be tested at rated power without causing a high flux scram.	20, 40, 50, 60, 75, 100	None	None	X Note 2					
SUT 25 Main Steam Isolation Valves: (1) Single valve closure to functionally check MSIV operation at different power levels. (2) Determine reactor transient behavior following closure of all valves.	(1) 50, 75 (2)100	(1) 75 (2) Not performed	(1) and (2) Not performed since there was no pressure or control changes	X Note 2					
SUT 26 Relief valves: The purpose of this test is to: (1) Demonstrate operability of the MSRVs. (2) Verify capacity of the MSRVs. (3) Demonstrate leak	100	None	None	Note 3					

Original Test Description	Initial Test Power Level % OLTP (Original Licensed Thermal Power)	Stretch Uprate Power Level % OLTP (3458 MWt)	MUR Uprate Power Level % OLTP (3514 MWt)	Test Power Level % RTP (EPU) (≈MWt)					
				≤ 80 (3165)	84.5 (3340)	88.9 (3514)	92.6 (3660)	96.3 (3805)	100% (3951)
tightness of the MSRVs after operation.									
SUT 27 Turbine Valve Testing: (1) Stop valve trip. (2) Control valve trip.	50, 100 20,75	(1) ST partial closure 100, 102.5, 105	(1) and (2) Not performed since there was no pressure or control changes	X	X	X	X	X	X
SUT 29 Flow Control (Recirc): This test determines: (1) The plant response to changes in Recirc. flow. (2) Plant load following capability.	20, 40, 50, 60, 75, 100	None	Not performed since there was no pressure or control changes						
SUT 30 Recirc. System: This test evaluates plant response following: (1) 1 pump trip. (2) 2 pump trip. (3) Calibrate the reactor core flow measurement system.	50, 75, 100 50, 75, 100 20, 50, 75, 100	None	(1) and (2) Not performed since there was no pressure or control changes (3) 99, 105, 106.5	X	X	X	X	X	X
SUT 31 Loss of TG & Offsite Power: The purpose of this test is to demonstrate that the reactor can safely with stand a loss of the turbine-generator and all	25	Not performed	Not Performed						

Original Test Description	Initial Test Power Level % OLTP (Original Licensed Thermal Power)	Stretch Uprate Power Level % OLTP (3458 MWt)	MUR Uprate Power Level % OLTP (3514 MWt)	Test Power Level % RTP (EPU) (≈MWt)					
				≤ 80 (3165)	84.5 (3340)	88.9 (3514)	92.6 (3660)	96.3 (3805)	100% (3951)
offsite power.									
SUT 32 Recirc Loop Control: The purpose of this test is to determine the as-built characteristics of the recirculation control system and to adjust control systems parameters for optimum performance.	40, 50	None	None						
SUT 90 Vibration: The purpose of this test is to monitor vibration in key components and piping systems.	50, 100		106.5 (key components only	X	X	X	X	X	X
SUT Various Plant Parameter Monitoring: This test performs routine component and system performance monitoring of power dependent parameters.	Various	Various	Various	X	X	X	X	X	X

Note 1: These tests were performed with the vessel open or during initial heat-up only.
Note 2: The surveillance test was performed to determine the maximum power level that the test can be performed without a scram or isolation.
Note 3: There are no changes to the pilot operated relief valves required for EPU. The additional spring safety valve added for EPU is identical to the existing two and is bench tested prior to installation.

13.6 NORMAL OPERATIONS AND PROCEDURE CONTROL

13.6.1 General

All operations are performed by licensee personnel in accordance with written procedures. Reactor operations are directed by personnel with a Senior Reactor Operator's License. Procedure changes and approval are discussed in Section 13.6.6 and 13.6.7 respectively.

13.6.2 Power Operation

An operating procedure is a detailed written instruction covering a specific operation in a step-by-step sequence. The procedures include all limitations and restrictions on the subject system and, where applicable, on related systems.

These procedures are made known to, discussed with, and copies made available to the necessary personnel. These procedures cover all normal and abnormal, reasonably foreseeable operating conditions, through station startup, operation, shutdown, and refueling.

Reactor, operating or maintenance procedures and changes thereto are approved by a Station Qualified Reviewer authorized by the Plant Manager to review and approve the procedure involved. If the change is only of an editorial nature, SQR review and approval is not required. The procedure or procedure change is authorized by the Site Function Area Manager or by the Plant Manager. Reactor operating or maintenance procedures and changes thereto that require a 10CFR50.59 evaluation are reviewed by PORC and approved by the Plant Manager or his/her designated alternate.

13.6.3 Maintenance and Testing Procedures

Planned and routine maintenance and testing programs are performed in accordance with established written procedures. Procedures covering non-routine maintenance may not be available, requiring preparation of temporary procedures which will receive appropriate approval.

These procedures will include radiation protection program and radiation control standards which will be applicable to all testing and maintenance functions.

A quality assurance program for use during maintenance work will be developed.

13.6.4 Emergency Response Procedures

13.6.4.1 Introduction

Although it is extremely improbable that an accident might occur during the life of the plant which could affect people offsite, it is necessary that well formulated and practiced emergency procedures be established. The emergency response procedures are formulated with appropriate public agencies so that problems which may arise during any emergency involving personnel, onsite or offsite, can be handled in an orderly, effective manner. These procedures are written to include the services of the State Police, the local fire department, a local hospital, and other public agencies as appropriate.

13.6.4.2 Training

Plant personnel receive instructions in emergency procedures and the use of emergency equipment. Periodic training drills are held to ensure maximum efficiency in the event of an actual emergency.

13.6.4.3 Communications and Alarms

A distinctive alarm system is used for emergencies. Interior and exterior alarms are installed in locations that might be occupied by personnel.

A site communication system alerts all onsite personnel to an emergency. The emergency plan includes provisions for communicating with offsite personnel and public agencies.

13.6.4.4 Medical Assistance

A medical assistance plan has been developed in cooperation with local medical facilities and personnel.

In addition, an emergency medical assistance program for the handling of accidents/incidents has been developed to provide and arrange for specialized medical-health-physics services for the planning, evaluation, diagnosis, and complete treatment of radiation accident casualties.

13.6.5 Security Procedures

These procedures establish the administrative requirements and responsibilities for the plant security program that has been developed to implement the industrial security plan and to supplement features and physical barriers designed to control access to the plant and, as appropriate, to vital areas within the plant. These procedures are distributed to appropriate personnel in accordance with current distribution lists, which reflect the confidentiality of security provisions, to ensure that outdated or in appropriate procedures are not used.

13.6.6 Procedure Revisions

13.6.6.1 Procedure and Administrative Policy

Each procedure and administrative policy of Technical Specification 5.4 and changes thereto, and any new procedure or procedure change that the Plant Manager determines to affect nuclear safety shall be reviewed and approved prior to implementation.

13.6.6.2 Temporary Changes to Procedures

Temporary procedures are implemented through the normal procedure revision process. Reviews and approval are performed in accordance with administrative procedures. Temporary changes are approved by an SQR and an SRO and do not change the procedure intent.

13.6.7 Procedure Review and Approval

13.6.7.1 New Procedure and Procedure Change

Each new procedure and procedure change shall be reviewed and approved by a Station Qualified Reviewer (SQR) and then authorized by a Site Functional Area Manager (SFAM) designated as responsible for the procedure. Procedures requiring a 50.59/72.48 Evaluation shall be reviewed by PORC prior to authorization by the Plant Manager or his designated alternate. The SQR shall not be the individual who prepared the new procedure or procedure change, he may however be from the same organization as the preparer. In addition to performing procedure authorization, the SFAM shall ensure that a sufficient complement of SQR's for their functional area is maintained.

13.6.7.2 Reviewing, Approving and Authorizing New Procedure and Procedure Change

The SQR shall review and approve, if appropriate, each new procedure or procedure change. The SQR shall render a determination in writing of whether or not a cross-disciplinary review of the new procedure or procedure change is necessary. If such a review is determined to be necessary, it shall be performed by the appropriate personnel. The SQR shall verify that the procedure package is appropriately reviewed including Regulatory reviews. Upon completion of the SQR review and approval, each new procedure or procedure change shall be reviewed by the SFAM designated for that procedure. Unless the procedure requires PORC review, the new procedure or procedure change shall be authorized, if appropriate, by the responsible SFAM or the Plant Manager prior to implementation.

13.6.7.3 Not Used

13.6.7.4 Performing Functions of SQR

Personnel recommended to perform the function of SQR shall be approved and designated as such by the Plant Manager. The SQR shall be an individual from the responsible organization and knowledgeable in the functional area affected. The individual shall meet the qualifications of ANSI/ANS-3.1-1981, Selection, Qualification and Training of Personnel for Nuclear Power Plants, Section 4, excluding subsections 4.3.2 and 4.5. Individuals who do not possess the formal educational requirements specified shall not be automatically eliminated where other factors provide sufficient demonstration of their abilities. These factors, as listed in subsection 4.1 of ANSI/ANS-3.1, shall be evaluated on a case-by-case basis and approved and documented by the Plant Manager.

13.6.7.5 Recording Documentation

Records documenting procedure reviews and approvals shall be maintained in accordance with the QA Program.

13.6.8 Requirements Relocated Out of Technical Specifications

On September 29, 1994, the licensee submitted Technical Specifications Change Request (TSCR) 93-16 to the NRC. This TSCR proposed an overall conversion of the PBAPS custom Technical Specifications (TS) to the Improved Technical Specifications (ITS), as contained in NUREG 1433, "Standard Technical Specifications, General Electric Plants BWR/4." This TSCR was approved by the NRC on August 30, 1995 via issuance of Amendment Nos. 210 and 214 for Units 2 and 3, respectively.

As a result of the ITS, many of the custom TS requirements were relocated out of TS and into various licensee controlled documents. The Technical Requirements Manual (TRM) and Appendix A of the Offsite Dose Calculation Manual (ODCM) were established to house those relocated TS requirements to which the operators need to refer frequently. In accordance with NEI 98-03, Revision 1, the TRM and Appendix A of the ODCM are treated in a manner consistent with procedures fully or partially described in the UFSAR, and therefore, are controlled as procedures in accordance with the applicable established procedure process. For the purposes of 10CFR50.59, information presented in the TRM and Appendix A of the ODCM is considered to be information described in the UFSAR, as updated, and therefore, is to be treated at the same level as information presented in the UFSAR. Relocated TS requirements which do not involve frequent reference by the operators were incorporated into other documents such as plant procedures, the ITS Bases or the UFSAR.

The Relocated Items Matrix was created to identify those custom TS requirements relocated as part of the ITS project, their proposed new location and the change control mechanism (10CFR50.59, 10CFR50.54(a)), which will be used to process future changes to the relocated requirement. The Relocated Items Matrix was included in the NRC's August 30, 1995 Safety Evaluation. Accordingly, the matrix is part of the Safety Analyses Report (SAR), as defined in LR-C-13, "50.59 Reviews." To assist in the retrievability of this matrix, it is included in the TRM.

13.7 RECORDS

13.7.1 Initial Tests and Operation

The records of the pre-operational phase of plant operation will be compiled and filed with the plant records and will be available for reference. These records will include the results and analysis of pre-operational testing, initial fuel loading, low power level tests, and high power tests prior to commercial operation.

13.7.2 Normal Operations

The logbook records constitute an important part of the operating record. The record in the logbooks is regarded as the authentic record of occurrences. Entries are made promptly, accurately, and completely and each entry initialed by the person making the record. The logbooks which will be maintained include at least the following:

1. Reactor logbook

2. Control room logbook

3. Shift supervisor logbook

4. Radwaste logbook.

A digital computer is provided for making calculations pertaining to plant performance, core power distribution, and reactor power output. The computer provides a printed output including a log of operating data. The plant records are maintained to indicate unusual or significant occurrences. The records of operation are reviewed daily by supervisory or engineering personnel and unusual occurrences are noted and corrections made as required. Operating records are filed and retained in accordance with administrative procedures.

13.7.3 Maintenance and Testing

A recorded history of equipment maintenance will be kept to provide reference information and to assist in scheduling maintenance. The maintenance records will include spare parts replacement and the cause of the failure.

Records of maintenance and testing will include at least the following:

1. Periodic testing of engineered safeguards

2. Routine tests of standby equipment

3. Mechanical and electrical maintenance testing

4. Instrumentation calibration, testing, and maintenance

5. Maintenance Log - This will contain records of any maintenance work and/or modifications performed on all station equipment.

13.7.4 Other Records

In addition to the station operation, maintenance, and test records, personal radiation exposure records and environmental monitoring records will be maintained including at least the following:

1. Daily record of dosimeter readings

2. Continuous and cumulative records of exposures as required by 10CFR20

3. Radiation levels

4. Contamination levels

5. Airborne activity.

13.8 OPERATIONAL REVIEW AND AUDITS

13.8.1 GENERAL

The licensee implements review and audit functions to ensure that activities important to safety are properly accomplished in accordance with company policies and rules, approved operating procedures, and operating license provisions during the operations phase. The review and audit functions are implemented in accordance with the provisions described in Appendix D.

13.8.2 PLANT OPERATION REVIEW COMMITTEE (PORC)

13.8.2.1 The Plant Operations Review Committee (PORC) provides, as part of the normal duties of plant supervisory personnel, timely and continuing monitoring of operating activities to advise the Plant Manager on all matters related to nuclear safety. PORC functions in accordance with an administrative procedure.

13.8.2.2 PORC is composed of regular members collectively having experience in the areas of plant operations, engineering, maintenance, instrumentation and controls, planning, radiation safety, chemistry, and regulatory assurance.

13.8.2.3 PORC members meet the requirements of ANSI N18.1-1971, sections 4.2, 4.4, or 4.6 for applicable required experience and are appointed in writing by the Plant Manager. The Chairmen and alternate Chairman of PORC are drawn from PORC members and are appointed in writing by the Plant Manager.

13.8.2.4 All alternate members are appointed in writing by the Plant Manager to serve on a temporary basis; however, no more than two alternates shall participate as voting members in PORC activities at any one time.

13.8.2.5 PORC meets regularly as convened by the PORC Chairman or his designated alternate. A quorum necessary for the performance of PORC responsibilities and authorities consists of the PORC Chairman or his designated alternate, and four members or their alternates.

13.8.2.6 Administrative procedure(s) shall control appropriate items that are required to be reviewed by the PORC. Changes to the following items require PORC approval prior to implementation:

a. Deleted.

b. 10CFR50.59 and 10CFR72.48 evaluations.

c. Units 2 and 3 Technical Specifications.

d. Emergency Preparedness Plan.

e. Security Plan.

13.8.2.7 PORC shall maintain written minutes of each meeting and copies shall be provided to the Vice President, PBAPS, Plant Manager, and to the NSRB.

13.8.3 INDEPENDENT REVIEW

The composition and responsibilities of the Nuclear Review Board are described in the Quality Assurance Topical Report, NO-AA-10.

13.8.4 ASSESSMENT PROGRAM

See the Quality Assurance Topical Report, NO-AA-10.

13.8.5 OPERATING EXPERIENCE ASSESSMENT PROGRAM (OPEX)

The OPEX Program is described in Section 13.2.2.4.

13.8.6 INDEPENDENT TECHNICAL REVIEW PROGRAM

The Independent Technical Review Function is described in the Quality Assurance Topical Report, NO-AA-10.