XIII - CONDUCT OF OPERATIONS

PAGE

XIII - CONDUCT OF OPERATIONS (Cont'd.)

LIST OF FIGURES

(At end of Section XIII)

LIST OF TABLES

XIII - CONDUCT OF OPERATIONS

1.0 SUMMARY DESCRIPTION

The Nebraska Public Power District (herein referred to as NPPD or the District) has been responsible for all station operations from the start of preoperational testing, and for providing properly licensed and other trained personnel to operate the station. Technical assistance during initial core loading, startup, and precommercial plant testing was provided by Burns & Roe, Inc., General Electric Company (GE), Westinghouse Corporation, and the NPPD General Office Power Supply Department and Generation Engineering Department. Prior to initial fuel loading, GE furnished a sufficient number of their startup personnel to properly assist NPPD-licensed personnel during the period preceding commercial operation.

In accordance with a Support Services Agreement, Entergy Nuclear Nebraska, LLC, provides management support services to NPPD, including providing key management personnel at CNS. However, NPPD remains the sole licensed operator of CNS and retains ultimate control over CNS operations.

The General Manager of Plant Operations, under the direction of the Vice President - Nuclear, has overview and coordination responsibility for the safe, reliable, and efficient operation of CNS. A training program was established to provide and maintain a properly trained staff of technical, maintenance, and licensed operating personnel to accomplish all of the various station functions within the disciplines as shown on the station organization chart (Figure XIII-2-2). Training and licensing have continued after commencement of commercial operation to assure an adequate number of licensed operators and properly trained replacement personnel for other disciplines.

Station operation is performed in accordance with approved written procedures. Activities are conducted in compliance with the Technical Specifications and applicable Federal and State regulations. Significant operations, tests, and other pertinent information is recorded and a record of these documents continues to be maintained.

Changes in physical plant and operating procedures during commercial operation are reviewed and approved prior to implementation of such changes as required by the NPPD Cooper Nuclear Station Quality Assurance Program for Operation Policy Document. In addition, evaluations performed under the provisions of 10CFR50.59 for new procedures and changes in existing procedures and the physical plant are reviewed by the District Safety Review and Audit Board.

2.0 ORGANIZATION AND RESPONSIBILITY

This section contains historical information as indicated by the italicized text. USAR Section I-3.4 provides a more detailed discussion of the purpose of highlighting certain text with italics. The factual information presented has been preserved as it was originally submitted to the Atomic Energy Commission in the CNS FSAR, as amended.

2.1 Nuclear Organization

2.1.1 Overview

The Nuclear Power Group (NPG) is a single, integrated organization responsible for providing safe and efficient operation, maintenance, modification, and support of CNS. This organization is headed by the Vice President - Nuclear.

The overall responsibilities of key management positions in this organization are addressed in USAR Sections XIII-2.1.2 through XIII-2.1.7 and provide the description of the organizational structure required by Technical Specifications. The organizational structure is illustrated in Figure XIII-2-2. USAR Section XIII-9 describes the District's safety review committees.

2.1.2 Vice President - Nuclear

The Vice President - Nuclear reports directly to the President and Chief Executive Officer of Nebraska Public Power District, and is responsible for NPG activities and administration of nuclear affairs of the District. The Vice President - Nuclear has corporate responsibility and oversight of the overall performance of the nuclear operation. He is the Chief Nuclear Officer for the NPG.

Positions/functional areas reporting directly to the Vice President - Nuclear include the General Manager of Plant Operations, Director of Engineering, Director of Nuclear Safety Assurance, and the Safety Review and Audit Board.

2.1.3 General Manager of Plant Operations

The General Manager of Plant Operations reports to the Vice President - Nuclear and is responsible for the safe, reliable, and efficient operation of CNS, and for ensuring that CNS is operated in compliance with the Operating License, Technical Specifications, and applicable Federal and State regulations. He has overall responsibility for operations, maintenance, radiation protection, chemistry, production (planning/scheduling/outages), materials/purchasing/contracts, industrial safety, nuclear projects, and the Station Operations Review Committee.

The radiation protection manager has a direct line of communication to the Vice President - Nuclear for important radiation protection issues.

2.1.4 Director of Engineering

The Director of Engineering reports to the Vice President - Nuclear and is responsible for supporting safe and efficient operation and maintenance of the nuclear facility. He has overall responsibility for design engineering, system engineering, reactor and fuels engineering, and engineering programs and components.

2.1.5 Director of Nuclear Safety Assurance

The Director of Nuclear Safety Assurance reports to the Vice President - Nuclear and is responsible for oversight of emergency preparedness, security, training, and regulatory affairs and compliance (licensing/administrative services/performance improvement).

2.1.6 Director of Nuclear Oversight and Strategic Asset Management

The Director of Nuclear Oversight reports functionally to the President and CEO and maintains a direct line of communication with the Vice President - Nuclear/CNO. The Director is responsible for independent oversight and quality assurance.

The Quality Assurance Manager has a direct line of communication to the President and CEO for the presentation of quality assurance issues.

2.2 Construction Testing Responsibility

Testing of equipment and systems will be required in many instances to assure that the construction contractor has completed his work.

Burns & Roe, Incorporated, will be responsible for the performance of these tests for the entire plant. A flow sheet describing the development, performance, and evaluation of construction tests is included in Figure XIII-2-4. A list of typical tests which were conducted is included in Section XIII-4.3.

2.3 Preoperational Testing Responsibility

Preoperational testing will be the responsibility of NPPD. Technical direction and assistance will be provided by GE and Burns & Roe, as required. A flow sheet describing the development, performance, and evaluation of preoperational tests is included in Figure XIII-2-5.

2.4 Startup and Power Testing Responsibility

During startup and power testing of the station, it will be necessary to assign additional personnel to supplement the District staff.

General Electric Company, the supplier of the nuclear steam supply system, will provide technical advisors capable of being licensed. Personnel from the District's Power Supply, Generation Engineering Departments, and Burns & Roe, Inc., will be assigned, as required, to assist in the startup testing of the station.

On a typical shift, the NPPD Shift Supervisor will be responsible for operations and will receive technical guidance and assistance from the GE and/or Burns & Roe personnel assigned to his shift.

2.5 Nuclear System Supplier Technical Direction

Technical direction during the initial startup of the station will be performed by qualified General Electric Company (GE) personnel with maximum participation by trained personnel from the Nebraska Public Power District staff. In addition to the nuclear system equipment supplied, this technical direction will include:

1. The storage, protection, installation, cleaning, initial calibration, testing, and operation of the nuclear system equipment, instrumentation, and material supplied by GE.

2. The preoperational testing of the nuclear station systems in which GE-supplied equipment is installed. This includes the right of review and comment on the preoperational testing of all station systems that are related to the safety and performance of the nuclear system.

3. All operational checkouts of the nuclear system, from the initial fuel loading, and startup, to the completion of the warranty demonstration test.

4. The onsite training of NPPD personnel during the nuclear system preoperational testing, initial fuel loading, and startup activities.

2.6 Balance of Plant Technical Assistance

Technical assistance during the startup of the balance of plant systems will be performed by Burns and Roe, Incorporated.

This technical assistance will include:

1. Assist in the preparation of detailed test procedures for the balance of plant systems.

2. Coordinate and technically direct the construction testing, functional and instrumentation startup, initial operation and preoperational testing, and performance testing phases for the balance of plant systems.

3. Prepare reports on startup activities and maintain a current startup schedule for all systems.

4. Coordinate the assistance of manufacturers representatives.

3.0 PERSONNEL QUALIFICATIONS AND TRAINING

3.1 Plant Personnel Training Program

The training programs for Cooper Nuclear Station (CNS) are designed to develop and maintain an organization qualified to operate, maintain and support the facility in a safe and reliable manner. Achievement of this goal is based on a philosophy of providing training that is developed from a systematic approach to job requirements using job and task analysis where available. This philosophy is consistent with both Nuclear Regulatory Commission (NRC) requirements and Institute of Nuclear Power Operations (INPO) recommendations necessary for accreditation of training programs. Accredited training programs at Cooper Nuclear Station meet the requirements of 10CFR50.120 and 10CFR55, and encompass NPPD commitments to NUREG-0737 Items I.A.1, I.A.2, and II.B.4. The training program is under the direction of the Training Manager.

3.2 INPO Accredited Training Programs

The following Cooper Nuclear Station training programs have been accredited by INPO and are based on a Systems Approach to Training (SAT):

Programs

1. Non-licensed Operator
2. Reactor Operator
3. Senior Reactor Operator
4. Shift Manager
5. Continuing Training for Licensed Personnel
6. Shift Technical Engineer
7. Instrument and Control Technician
8. Electrical Maintenance Personnel
9. Mechanical Maintenance Personnel and Supervisor
10. Chemistry Technician
11. Radiological Protection Technician
12. Engineering Support Personnel

3.3 Simulator Training

A plant-specific training simulator is used for training licensed operators and licensed operator candidates. It has been certified for operator testing in accordance with 10CFR55.45(b)(5).

3.4 General Orientation Training

All personnel granted unescorted access to the protected area at Cooper Nuclear Station are required to complete Plant Access Training which provides an indoctrination in the general requirements necessary to gain access to the plant, including information on security, emergency planning, industrial safety, radiation protection, fire protection, and quality assurance. Completion of Fitness for Duty training is also required for all personnel granted unescorted access.

In addition, Radiation Worker Training is required for all employees who require access to the Radiologically Controlled Area.

Retraining is provided for Plant Access Training, Fitness for Duty Training, and Radiation Worker Training at a frequency necessary to ensure that personnel remain proficient, as specified in the applicable Training Qualification Descriptions.

3.5 Personnel Qualification Requirements

Qualification requirements for various plant staff are specified in Section 5.3 of the Technical Specifications.
Table XIII-3-1 contains a matrix that correlates the positions delineated in the Technical Specifications with CNS specific titles and with the generic position titles referred to in American National Standards Institute (ANSI) N-18.1-1971, Regulatory Guide (RG) 1.8, Revision 2, and RG 1.8, Revision 3.

Site specific implementation of the ANSI/RG criteria for qualification of station personnel is controlled by station procedure.

3.6 Records

The Training Manager maintains adequate records to document significant training activities for CNS employees. Records of qualifications are kept in the training records for each employee. The effectiveness of training programs is determined, where feasible, by written or oral examinations; the results of which are recorded.

TABLE XIII-3-1

JOB TITLE MATRIX

Tech Spec Position	ANSI/RG Generic Position Title	CNS Specific Title
plant manager	plant manager	General Manager of Plant Operations
Shift Manager	operations shift supervisor	Shift Manager
Senior Operator	senior operator	Control Room Supervisor
Licensed Operator	reactor/licensed operator	Control Room Operator
corporate officer with direct responsibility for the plant	N/A	Vice President-Nuclear
radiation protection technician	technician	Radiological Protection/Chemistry Shift Technician
operations supervisor	operations manager	Assistant Operations Manager-Operating Shift
Shift Technical Engineer	shift technical advisor	Shift Technical Engineer
radiological manager	radiation protection manager	Radiation Protection Manager
operations manager	operations manager	Operations Manager*

* Operations Manager not required to hold current SRO license per Technical Specification 5.3.

4.0 CONSTRUCTION AND PREOPERATIONAL TEST PROGRAM

This USAR section contains historical information as indicated by the italicized text. USAR Section I-3.4 provides a more detailed discussion of the purpose of highlighting certain text with italics. The factual information being presented in this section as historical has been preserved as it was originally submitted to the Atomic Energy Commission in the CNS FSAR, as amended. Satisfactory performance of preoperational testing was a prerequisite to commencement of the Startup Test Program. The Preoperational Test Program was audited through onsite AEC inspection activity, which contributed to the Facility Operating License conclusion that 'The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission'. The specific results of the Preoperational Testing Program are retained with the permanent station records as described in Subsection XIII-8.1.1.1.

4.1 Objectives

The purpose of the preoperational test program is three-fold: 1) Confirm that construction is complete to the extent that equipment and systems can be put into use during completion of other construction; 2) Adjust and calibrate the equipment to the extent possible in the "cold" plant; 3) Assure that all process and safety equipment is operational and in compliance with license requirements, to the extent necessary to proceed into initial fuel loading and the startup program. The foregoing is achieved by written construction and preoperational tests on systems related to nuclear safety. Numerous tests not related to safety are also included in the written preoperational test category.

The preparation and implementation of preoperational tests are an important phase in the training of operating personnel. Experience and understanding of station systems and components is gained during this testing program. This gives maximum opportunity to evaluate and train individual operators and to trouble-shoot systems. In addition, 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. Records of all testing will be maintained as indicated in Section 8.1.

Upon receiving the AEC Guide for the Planning of Preoperational Testing Programs, dated March 13, 1971, efforts were made to pattern the CNS preoperational testing program after the Guide. The preoperational testing program as described herein does not necessarily follow the format of the Guide because the original SAR was prepared and submitted prior to issuance of the Guide. However, the preoperational testing program meets the intent of the Guide. An administrative procedure entitled Preoperational Testing Program Description was prepared to further detail the planned testing program as per the aspects of the Guide.[3]

4.2 Preoperational Test Schedule Considerations

The following key points are considered in developing the sequence and schedule of preoperational tests:

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

2. Preoperational testing is coordinated with construction completion to permit fuel loading as early as possible, without compromising nuclear safety or impeding construction work. As a result, fuel loading may occur while construction work is still in progress on unrelated systems and areas.

3. Temporary construction power may be required for initial tests at the beginning of the preoperational test program. However, unnecessary use of temporary power and improvised setups will be avoided because of the possibility of errors and inconsistency with the ultimate objective of proving the final installation.

4. Electrical jumpers are used to facilitate preoperational testing in some instances, but their use is minimized and controlled by proper identification of such jumpers, by tags on the equipment jumpered, and by log book records. All jumpers are removed before fuel loading.

5. When the station is ready for fuel loading, strict control is enforced over personnel access to all critical operating areas of the station including the main control room, electrical equipment rooms, and reactor building.

6. The District's operating personnel will operate the station and equipment during formal preoperational testing. Construction completion tests are performed by construction contractors and are under the surveillance of the District. These construction tests utilize procedures and data sheets and require reporting and acceptance (typical construction tests are listed in Section XIII-4-3).

7. Specialized electronic equipment and nuclear instrumentation manufactured by General Electric is checked and preoperationally tested by the station staff under the technical direction of General Electric representatives.

8. Preoperational test procedures are prepared by the District's personnel with technical direction and review by General Electric and Burns and Roe, depending upon system design responsibility.

9. In general, tests are performed using permanently installed instrumentation for the required data. When it is not possible to use permanently installed instrumentation during the system tests, required test instrumentation will be installed and used to ensure safe operation of the equipment. Any test requiring artificial simulation of a station parameter has the method detailed in the procedure as well as the means for assuring that the system is returned to normal.

10. Prior to performing preoperational tests, equipment and/or systems must be ready to operate and be acceptable for the preoperational testing.[4] A series of tests, checks, and inspections will be completed as prerequisites to the preoperational testing program. These prerequisites are a part of the construction testing program, described in Section 4.3.

4.3 <u>Summary of Construction Test Content</u>

Construction tests include but are not limited to the following examples (activities designated with an asterisk are prerequisites for the associated system pre-operational test):

1. Containment over pressure test.

*2. System hydrostatic tests.**

3. Cleaning, flushing of piping, vessel, and equipment. Leak testing piping and equipment.*

4. Nondestructive testing of field welds.

5. Electrical system tests, including energizing, e.g., checking grounding, checking circuit breaker operation and controls*, continuity checks,* and insulation resistance tests*, phasing check*, high potential measurements*, and energizing of buses*.*

6. Initial adjustment and bumping of motors.*

*7. Check control and interlock functions of instruments, relays, and control devices to verify wiring and pneumatic tubing per design.**

*8. Pneumatic test of instrument and service air system and blow out of lines.**

*9. Adjustments such as alignment, lubrication of equipment, and tightening bolts.**

10. Tests of motor-operated valves including adjusting limitorque switches and limit switches, measuring operating speed, and checking leak tightness of stem packing.

11. Tests of air-operated valves including pilot solenoids, adjusting limit switches, measuring operating speed, checking leak tightness of stem packing and valve seat during hydrotests, and checking leak-tightness of pneumatic operators.

*12. Calibrate instruments (initially) and recheck or set initial trip setpoints.**

4.4 Summary of Preoperational Test Content

Preoperational tests include but are not limited to the following examples:

4.4.1 D-C Power Systems

This test will check and/or calibrate instruments, relays, breakers, interlocks, alarms, and other electrical components. Battery charger capacity and battery discharge rates will be measured.

4.4.2 Electrical Auxiliary Power Systems

This test is intended to supplement construction tests performed on the 4160, 480, and 120/240 volt a-c and the 250, 125, and 48/24 volt d-c auxiliary power systems. Only the functional tests necessary to place these systems in service are included in this test.

4.4.3 Service Water Systems

The objective of this test is to verify the functional capability of the service water system to provide cooling water for the turbine building and reactor building heat exchangers. The test will verify flows, check alarms, instrumentation, and control functions.

4.4.4 Condensate Filter Demineralizer System

This test will check instruments, valves, flow-balancing, alarms, precoat system, backwash system, holding pumps, and effluent strainers. Full flow measurements of water quality will be made and capability to transfer exhausted resins checked.

4.4.5 Plant Makeup Water Treatment System

Check all instrumentation, interlocks, and perform operational check of system components. Place pretreatment plant in operation and check chemistry of makeup system influent. Perform functional test to verify time cycles. Verify water quality of makeup system effluent.

4.4.6 *Turbine Equipment Cooling System*

The objective of this test will be to verify the functional capability, flows, temperature controller response, and instrumentation in the cold condition. Actual ΔT of components at operating temperatures will be measured with the plant at power.

4.4.7 *Reactor Equipment Cooling System*

This test is intended to verify flow to each component serviced by the cooling water system. Instruments, temperature controllers, and alarms will be checked in the cold condition. Final checks of temperatures differential across various components will be made with the plant at power.

4.4.8 *Condensate and Feedwater Systems*

Check/calibrate all valves, instruments, controls, and system interlocks. Pumps will be operated to obtain preliminary performance data, check lubrication, obtain vibration data, and verify correct remote control operation.

4.4.9 *Main Circulating Water System, Screen Wash, and Trash Rake*

This test will verify proper operation of the Trash Rake and Screen Wash System prior to operation of main circulating water pumps. All instruments will be checked/calibrated, controls verified, screen and pump operation tested to assure that a proper spray pattern is developed to minimize fouling. The main circulating water system instruments will be checked, valves operated, controls and interlocks functionally tested, and pump operation verified.

4.4.10 *Control Rod Drive Manual Control*

These tests will include performance checks and/or calibration of pumps, instrumentation, flow control valves, interlocks, alarms, and controls. Control rod selection relays and valves will be tested as well as scram valves and their pilot solenoids, backup scram solenoids, and scram discharge volume vent and drain valves.

4.4.11 *Fuel Pool Cooling and Filter Demineralizer System*

Check and/or calibrate instrumentation, valves, pumps, heat exchangers, filters, demineralizers, and verify operability of system. Spent fuel pool will be filled with demineralized water, checked for leakage, and pool and surge test instrumentation checked.

4.4.12 *Fuel Handling and Vessel Servicing Equipment*

The equipment will be tested with dummy fuel or blade guide assemblies through dry run simulations of the required operations. This test consists of many separate operations to check the different pieces of equipment which are used for service and refueling operations. The tests include checking fuel preparation and inspection equipment, operation of the refueling platform, operation of the service platform and fuel grapple, and operation of equipment used for the installation and removal of internals such as guide tubes and fuel support castings. Refueling interlocks will be included as part of this preoperational test.

4.4.13 *Control Rod Drive Hydraulic System*

These tests will include control rod stroking (both continuous withdrawal and notch by notch withdrawal) and stroke timing, drive speed setting, scram time measurements, proper position indication, control rod scrams to verify safety circuit sensors, and rod withdrawal interlocks exclusive of the rod worth minimizer.

4.4.14 *Reactor Water Cleanup System*

Check and/or calibrate instrumentation, valve and pumps operability, interlocks, and protective devices; perform functional operation to verify time cycle. Verify water quality and flow capacity of filter-demineralizers.

4.4.15 *Standby Liquid Control System*

This test will include operation of pumps and valves using demineralized water. The explosive valves will be tested and the discharge rate to the reactor vessel will be measured. The liquid control solution will then be prepared in the liquid control tank prior to fuel loading. The boron content will be verified by laboratory testing.

4.4.16 *Residual Heat Removal System*

All four subsystems of this system will be tested individually and together where interfaces occur.

1. Suppression Pool Cooling Mode

The objective of this preoperational test is to verify the flow capacity, valve control, and system operation using the standby a-c power system.

2. Low Pressure Coolant Injection Mode

The objective of the preoperational tests will be to verify the automatic initiation and interlocks, valve control, flow rate, and system operation using the standby a-c power system.

3. Containment Spray Mode

The objective of the preoperational tests will be to check the containment spray nozzles by blowing air into the spray headers after the headers have been flushed with water.

4. Shutdown Cooling Mode

The objective of the preoperational tests will be to verify the permissives, interlocks, flow capacity, and valve operation of the system.

4.4.17 *Core Spray System*

The objective of the preoperational tests will be to verify the automatic initiation and interlocks, spray function, flow rate, and system operation using the standby a-c power system.

4.4.18 *High Pressure Coolant Injection System*

The objective of this test will be to check and/or calibrate instrumentation, relays, interlocks, protective devices, and equipment that can be operated. Flow capacity testing of this system will be deferred until adequate nuclear steam was available.

4.4.19 *Reactor Core Isolation Cooling System*

The objective of this test will be to check and/or calibrate instrumentation, relays, interlocks, protective devices, and equipment that can be operated. Flow capacity testing of this system will be deferred until nuclear steam is available.

4.4.20 *Reactor Safety Relief Valves*

Safety and relief valves will be tested at the vendor's facility, where set points will be adjusted, verified, and indicated on the valve. The relief valve controls will be functionally tested during the preoperational phase. The automatic depressurization system control and trip signals will be functionally tested. During subsequent power test program, the relief valves will be actuated to prove their operability.

4.4.21 *Reactor Protection System*

The reactor protection system will be tested after safety system sensors are installed and calibrated and wiring is installed and checked for continuity. The M-G sets will be operated with a resistance load to check capacity and regulation. Each safety sensor will be checked for operation of the proper relay. Using test signals, scram set points will be verified for each sensor. All positions of the reactor mode switch and control rod permissives will be checked to verify interlocks function properly. Automatic closing of reactor vessel isolation valves will be checked and closing times measured. Automatic initiation of core spray, HPCI, RHR (LPCI), RCIC, and Diesel-Generator start will be verified using their proper initiation signals.

4.4.22 *Neutron Monitoring System*

A comprehensive check of each neutron monitor will be made from the chamber to the indicators, recorders, and safety interlocks and trips. This includes the SRM, IRM, LPRM, APRM, RBM, and TIP systems. The proper installation of the chambers, chamber drive, and other components will be verified. Continuity, ground resistance, and noise level of cables will be checked. Chamber installation and removal procedures will be verified.

4.4.23 *Process Computer*

Check all hardware, software, and periphery equipment. Simulate sample calculation for balance of plant and nuclear steam supply system.

4.4.24 *Rod Worth Minimizer*

Simulate control rod movement to verify the response of the rod position information control to position data from all control rod locations and check computer logged position data. Run computer diagnostic programs to verify proper program operation. After the control rod drive system is operational, verify actual rod identification against output readings. In addition, verify that improper rod withdrawal at selected points in the withdrawal sequence result in correct interlock action.

4.4.25 *Feedwater Control System (Reactor Level Control)*

The reactor level control will be tested after sensors are installed and calibrated by simulating flow, level, and reactor feed pump signals to verify the operation of the control system.

The ability of the feedwater control system to maintain reactor vessel level with varying steam flows will be verified with the plant at power.

4.4.26 *Reactor Recirculation System*

The objective of this test will be to check and/or calibrate instrumentation, interlocks, protective devices, and flow control equipment to the degree possible with cold water conditions. Jet pump instrumentation will be checked during these tests.

4.4.27 *Nuclear System Leak Detection*

This test will check the pump and alarm operations of this system. The sump capacities will also be checked and verified.

4.4.28 *Process Radiation Monitoring System*

This test will check the calibration, operation, and alarms on the Main Steam Line Radiation Monitor, Process Liquid Monitors, Stack Gas Monitor, and the Off-Gas Monitor.

4.4.29 *Area Radiation Monitoring System*

This test will verify the operation and alarm settings of the Area Radiation Monitors and the Reactor Building Ventilation Monitor.

4.4.30 *Radioactive Waste Disposal System*

1. *Liquid Radwaste*

Testing will assure proper instrument operation and capacity of sumps, tanks, filters, pumps, and demineralizers. Required portions of the radwaste disposal system will be available to receive wastes from the reactor building drains, fuel pool, and interconnecting auxiliary systems required for fuel loading.

2. *Solid Radwaste*

This test will be to gain experience in operating the equipment before it becomes significantly contaminated. Equipment will be checked for proper operation.

4.4.31 *Instrument and Service Air Systems*

The objective of this test will be to check/calibrate system instrumentation, and perform functional test of system. Dew point of the air dryer outlet will be measured and capacity of compressor checked.

4.4.32 *Plant Fire Protection System*

Check/calibrate instrumentation, pumps, diesel engine. Automatic features of system will be verified and deluge sprinkler operation checked.

4.4.33 *Plant Heating Boiler*

Check instruments, controls, valves, pumps, blowers, heaters, and other equipment associated with the heating steam system.

4.4.34 *Building Heating, Ventilating, and Air Conditioning*

These tests are a series of thirteen preops for various areas, performed to check and/or calibrate instrumentation, valves, dampers, controls, pumps, fans, coolers, and other equipment associated with the various systems. Balancing of ventilating systems will also be performed.

4.4.35 *Drywell Cooling and Ventilation*

Verify proper operation of drywell fans and coolers. Instruments, controls will be checked and/or calibrated.

4.4.36 *Standby Gas Treatment System/Reactor Building Leak Rate*

Check/calibrate instrumentation and controls. Blowers will be operated to check flow capacity. Verification of automatic isolation of the reactor building and initiation of the standby gas treatment system will be performed.

With the normal reactor building ventilation system isolated, the standby gas treatment system will be operated to measure reactor building leak rate under a negative pressure. The efficiency of the off-gas filter media will also be measured.

4.4.37 *Turbine Lube Oil (Transfer & Oil Purification) System*

This test is intended to verify operability of system pumps, interlocks, instrumentation, and place filtration units in service.

4.4.38 *Turbine Control and Instrumentation*

This test is performed to demonstrate that the electrohydraulic control system is operating properly prior to operating the turbine - generator on reactor steam. A check will be made of relays, limit switches, pressure switches, interlocks, overspeed trips, position indicators, supervisory instrumentation. Verify all trip inputs.

4.4.39 *Turbine Drains, Extraction, and Steam Valves*

This test is intended to demonstrate that all remote control devices and alarms associated with the main steam system, steam seal system, and the turbine drain system are in operation.

4.4.40 *Generator Cooling*

This test will check the functional operation of the hydrogen (including the CO2 purge) gas and seal oil system. Included are the checking and/or calibrating of instruments, pumps, regulators, alarms, and other related equipment.

4.4.41 *Generator Excitation System*

This test will verify that the generator excitation system is ready for operation prior to actually rotating the turbine-generator with reactor steam. The excitation system, its controls and protective devices were checked.

4.4.42 *Condenser and Auxiliaries*

This test will be performed to check/calibrate instruments, valves, and controls associated with the condensers, mechanical vacuum pumps, steam jet air ejectors, and vacuum priming system.

4.4.43 *Condensate and Demineralized Water Storage and Transfer System*

This test is intended to check operation of pumps, valves, controls, and ability of system to deliver water to outlets in the various areas of the plant.

4.4.44 *Containment Isolation*

This test is to check operation of the primary containment isolation valves. This includes the main steam isolation valves as well as other system and instrument isolation valves. Control logic, closing times, and valve indication will be among the items tested.

4.4.45 *Standby Diesel Generators*

This test is performed to verify operability of the diesel-generator sets and standby a-c power system. Each diesel generator will be individually tested unloaded to assure its readiness to be placed on the line. Each diesel-generator will then be tested for load carrying capability and ability to pick-up all emergency loads in proper sequence.

4.4.46 *Plant Communications System*

The operability of the system will be verified. Individual components will be tested and adjusted to assure system provides proper coverage and audibility. Check proper operation and alarm function on loss of a-c power.

4.4.47 *Isolated Phase Bus Duct Cooling System*

This test is intended to check functional operation of the cooling system - including checking instrumentation dampers, blowers, and interlocks.

4.4.48 *Drywell and Suppression Leak Rate Test*

In this test the total containment system, which includes the drywell, suppression chamber, isolation valves and all penetrations, are leak tested at the required pressures. Integrated leak rate measurement tests will be performed at two pressures, one at the maximum calculated peak pressure after an accident and the other at a reduced pressure, not less than 50% of peak accident test pressure. The absolute pressure-temperature method shall be employed.

4.4.49 *Containment Inerting System*

This test will be performed to demonstrate that valves, alarms, instruments and the nitrogen makeup and purge systems operate properly. The actual containment inerting will be performed after the power test program.

4.4.50 *TIP Calibration System*

This test will check the ability of the TIP to traverse all required core positions as well as the control logic. The input capabilities to the process computer will also be verified.

5.0 STARTUP AND POWER TEST PROGRAM

This USAR section contains historical information as indicated by the italicized text. USAR Section I-3.4 provides a more detailed discussion of the purpose of highlighting certain text with italics. The factual information being presented on the proposed Startup and Power Test Program has been preserved as it was originally submitted to the Atomic Energy Commission in the CNS FSAR, as amended. Satisfactory completion of the CNS Startup and Power Test Program was reported to the AEC/NRC in letters dated November 1, 1974, January 30, 1975, April 29, 1975, and May 14, 1975. The specific results of the Startup and Power Test Program are retained with the permanent station records as described in Subsection XIII-8.1.1.1. Although historical in nature, the following subsections provide the criteria (as applicable) for future startup and power testing.

A summary report (Startup Report) of plant startup and power escalation testing is submitted following:

1. Receipt of an operating license.

2. Amendment to the license involving a planned increase in power level.

3. Installation of fuel that has a different design or has been manufactured by a different fuel supplier.

4. Modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

The Startup Report addresses each of the tests identified in the USAR and includes a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation is also described. Any additional specific details required in license conditions based on other commitments is included in this report.

A Startup Report is submitted within (1) 90 days following completion of the startup test program, (2) 90 days following commencement of commercial power operation or subsequent resumption of power operation, or (3) 9 months after criticality is reached, whichever is earliest. Supplementary reports are submitted every 3 months until all three of these events are complete.

5.1 General Objectives

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. 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 groups of tests.

A. Fuel Loading and Low Power Physics Tests
B. Initial Heatup to Rated Temperature and Pressure
C. Power Testing from 25% to 100% of Rated Output
D. Warranty Demonstrations

The tests performed can be broadly classified as Major Plant Transients, Stability Tests, 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 XIII-5-1 shows the *complete startup and power test program and should be considered in conjunction with Figure XIII-5-1 which shows graphically the various test points as a function of core thermal power and flow. It is expected that in the elapsed time before embarking upon the startup test program, it may well be necessary to modify the scope of the program and/or the content of individual tests in order to utilize experience gained from earlier startups. If such modifications do not affect the safety or the safety analysis of the plant, the program may be altered from the program presented here. If such modifications affect safety they will be treated as standard amendments to the FSAR.*

The prerequisites for the startup and power test program consist of satisfactory completion of the preoperational test and cold functional test programs. The preoperational test cold functional test programs expose systems to abnormal condition testing. Purchase specifications for components require that these components operate under abnormal environmental conditions.[4]

In comparing the Startup and Power Test Program with the AEC issued Guide For the Planning of Initial Startup Programs[3], the startup test program for Cooper is tailored to the design of the particular plant; whereas the AEC guide is general and in some cases based on BWR test programs of many years ago. The test program will comply with the following items taken from the twenty-six examples listed under Power Ascension Tests for BWR's. The justification for each deviation is given below:

a) 2e Moderator Temperature Reactivity Defect Measurement

b) 2f Power Defect Measurement

These two tests were once performed to verify design models and/or obtain physics information. Design Engineering no longer requires these data on the Cooper type core. The parameters are of no direct operational interest but their values would affect the dynamic behavior of the reactor and its ease of control. It is this dynamic behavior and ease of control which the test program demonstrates rather than the specific values of the individual parameters.

c) 2h Measurement of Moisture Carryover and Steam Carryunder

Carryover measurements are made as part of the Chemical and Radiochemical test, but a complete set of carryover, carryunder measurements to verify efficiency of the steam separators, and steam dryers are only done on first of a kind and one of a kind separator-dryer designs. Once verified, there is no reason to expect that this efficiency of the separator and dryers of the same design should vary from plant to plant. The separator design used at Cooper was first used and tested on Brown's Ferry I.

d) 2n Emergency Condenser Performance and Capacity

There is no emergency condenser on the Cooper plant.

e) 2p Rod Pattern Exchange Demonstration

A rod pattern exchange demonstration was to be a demonstration to familiarize plant personnel with acceptable methods of pattern exchange while still under close direction of experienced GE personnel. This demonstration was to be done at the highest prudent power level (75-85%) to observe typical values of maximum heat flux and MCHFR variations. Current GE recommendations are that rod pattern exchanges be made while at a low power or as a consequence of a shutdown and restart. In either case no rod pattern exchange demonstration is necessary.

f) The Startup Test Program for Cooper does include tests not listed in the AEC guide. These are described briefly in Section 5.2.2 under the following headings:

Test Number 13 Process Computer
Test Number 14 RCIC System
Test Number 18 Core Power Distribution

TABLE XIII-5-1
STARTUP AND POWER TEST PROGRAM

TEST NO.	Power, %[1] / Flow, %[2] / TEST CONDITION (See Figure XIII-5-1)	LOW POWER OR COLD TEST	HEAT UP	20-30 ≈37 1	50% LOAD LINE ≈25 NC 2A	32-37 ≈52 2C	35-45 ≈74 2D	45-55 ≈107 2E	≈52 ≈114 2F	75% LOAD LINE ≈33 NC 3A	43-53 ≈52 3C	55-65 ≈72 3D	70-80 ≈104 3E	100% LOAD LINE ≈45 NC 4A	60-70 ≈52 4C	75-85 ≈70 4D	95-100 ≈100 4E	95-100 ≈105 4F
1	Chemical & Radiochemical	X	X	X				X					X				X	
2	Radiation Measurements	X	X	X				X					X				X	
3	Fuel Loading	X																
4	Full Core Shutdown Margin	X																
5	CRD	X	X	X		X		X									X	
6	SRM Perf. & Control Rod Seq.	X	X	X		X												
9	Water Level Measurement		X								X						X	
10	IRM Performance	X	X	X														
11	LPRM Calibration			X				X					X				X	
12	APRM Calibration		X	X		X		X					X				X	
13	Process Computer	X	X	X	X[6]			X									X	
14	RCIC		X	L														
15	HPCI		X					M										
16	Selected Process Temperatures		X		X					X				X				
17	System Expansion	X	X	X														
18	Core Power Distribution		X	X				X		X			X				X	
19	Core Performance		X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
20	Steam Production																X	
21	Flux Response to Rods			L			M	M		X	M	M	M	X	M	M	M	
22	Press. Reg :Set Point Changes			L			M	M		X	M	M	M	X	M	M	M	
	Backup Regulator			L				M					M				M	
23	FW System: FW Pump Trip												M				M,SP	
	Water Level Stpt Chg			L			M	M,A		X	M	M	M,A	X	M,A	M,A	M,A	M
	Heater Loss																M[12]	
24	Bypass Valves			L			M,A	M,A		X	M,A	M,A	M,A	X	M,A	M,A	M,A	M
25	MSIVs: Each Valve		X[9]					M,SP					M,SP			M,SP		
	Full Isolation																	
26	Relief Valves: Capacity			L														M,SE[5]
	Actuation		X[9]	L									M					
27	Turbine Trip and							M,SE[5]								A,SE	M,SE	
	Generator Load Rejection			L,SP													A,SE	
28	Shutdown From Outside C. Rm.			X														
29	Flow Control					A	A	A	A		A	A	A		A	A	M,A	
30	Recirc Sys: Trip 1 Pump							M[7],M[8]					M[7],M[7]				M[7],M	
	Trip 2 Pumps							M[7]					M				M	
	Sys Perf.			X	X		X	X	X	X			X	X			X	X
	Non-Cavit. Verif.							X[10]										
31	Loss of T/G & Offsite Power			L,SE[5]														
32	Recirc MG Set Speed Control			L		M	M	M	M		M	M	M		M	M	M	
33	Turbine Stop Valve Surveillance			L				M			M	M	M		M	M	M,SP[11]	
34	Vibration Measurements[4]	X		X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
35	Recirc System Flow Calibration	X						X					X				X	
70	Reactor Water Cleanup System		X															
71	Residual Heat Removal System		X	X														
72	Drywell Atmosphere Cooling Sys.		X	X													X	
73	Cooling Water Systems		X	X													X	
100	Hot Functional Test		X															

1 Percent of rated power 2,380 MWt
2 Percent of rated flow, 73.5 x 106 lb/hr
3 Also obtain data with Tests 25, Full Iso & Test 27
4 Obtain data with Test 30
5 Perform Test 5, timing of 4 slowest control rods in conjunction with these scrams
6 Perform the Dynamic System Test Case

7 Included only to meet Test 34 requirements
8 Trip the Generator Field Breaker
9 Heat up tests of the MSIVs & Relief Valves are to check operation only
10 From Test Condition 2E to 5
11 Determine maximum power without scram
12 At 90% of rated power

L Local Manual Flow Control Mode
M Master Manual Flow Control Mode
A Automatic Flow Control Mode
X Test Independent of Flow Control Mode
SP Scram Possibility
SE Scram Expected
NC Natural Circulation

5.1.1 *Fuel Loading and Low Power Physics 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 core 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 situated at suitable locations within the reactor vessel serve to provide indication for the shutdown margin demonstrations 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.

A small AM-BE Neutron source will be at near-center and four Boron-10 coated proportional counters will be used during fuel loading at Cooper to gather data on neutron flux level.[5] *This same arrangement has been used in all recent GE-BWR's. The data will be available for making inverse multiplication plots. However, this provides very little quantitative information for large core loadings. An improved data-analysis method has been identified and is undergoing evaluation. A formal topical report on this subject is being planned and will be available in late 1972.*

At this point in the program, a number of tests are conducted which are best described as low power physics tests. 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 prior to initial criticality 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.

Each control rod drive is subjected to scram, friction, and functional testing at ambient conditions. An initial setting is given to the Intermediate Range Monitors (IRM) and the process computer is checked to see that it is receiving correct values for those process variables which are available.

The cold flow vibration tests will be run during this period in the test program for comparison with the prototype data taken on the Fitzpatrick reactor. Initial data will also be taken on jet pump flow instrumentation and system expansion instruments.

The low power physics tests (called "open vessel" physics tests by the CNS Technical Specifications) will be conducted with the vessel head off or with the vessel head on and the head vent open and reactor power level limited to 5 MWt. The low power physics tests consist of measuring the full core shutdown margin by taking the fully loaded core critical; taking the reactor critical using normal rod sequences and the normal Source Range Monitors (SRM) in conjunction with the operational sources to show that adequate response exists for normal operation. The source range and intermediate range instrumentation (SRM/IRM) overlap will also be demonstrated.

5.1.2 *Initial Heatup to Rated Temperature and Pressure*

Heatup follows satisfactory completion of the low power physics tests and further checks are made of coolant chemistry together with radiation surveys at the selected plant locations. All control rod drives are scram timed at rated temperature and pressure with selected drives timed at intermediate reactor pressures and for different accumulator pressures. The control rod sequences are further investigated in order to demonstrate the ability to change sequences at power. The process computer checkout continues as more process variables become available for input. The RCIC and HPCI systems will undergo controlled starts at low reactor pressure and at rated conditions with the former tested in the quick start mode at 1000 psig. Correlations are obtained between selected process 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 thermal expansion are recorded for comparison with design and installation data. A preliminary APRM calibration is made using coolant temperature rise data during nuclear heatup.

5.1.3 Power Testing from 25% to 100% 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 Table XIII-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 there is, nevertheless, considerable flexibility in the test sequence 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, while maintaining reactor water quality and local radiation levels within specified limits. Selected control rod drives 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. Following the first reasonably accurate heat balance (25% power) the power range nuclear instruments are calibrated. At each major power level (25%, 50%, 75%, and 100%) the LPRMs are calibrated, while the APRMs are calibrated at each new power level initially and following each LPRM calibration. Completion of the process computer checkout is made for all variables and the various options are compared with other proven methods of calculation as soon as significant power levels are available. Further tests of the RCIC and HPCI systems 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 Traversing Incore Probe (TIP) system at representative power levels (25%, 50%, 75%, and 100%)* during the power ascension. Core performance evaluations are made at all test points above the 10% 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 most test points:

Flux response to rods
Pressure regulator setpoint change
Water Level setpoint change
Bypass valve opening
Flow Control

For the first of these tests a centrally located control rod is moved and the flux response is 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 two remaining tests a bypass valve is opened as quickly as possible and flow control setpoint changes are made, respectively. For all tests the plant performance is monitored by recording the transient behavior of numerous process variables, the principle one of interest being neutron flux. Other imposed transients are produced by dropping of feedwater heaters and failing the operating pressure regulator to permit takeover by the backup regulator. Table XIII-5-1 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, full closure of one MSIV at selected power level, fast closure of the turbine generator control valves, closure of turbine generator stop valves, 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 XIII-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 and general operability is demonstrated. At all major power levels, the jet pump flow instrumentation is calibrated. The as-built characteristics of the

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** These levels are nominal and variations of 5% in power are common and sufficient in defining the actual test level with the qualification that 100% power will not intentionally be exceeded.*

recirculation pump drives are investigated as soon as operating conditions permit full core flow. The local recirculation speed control loop performance, based on the drive motor, fluid coupler, generator, drive pump, jet pumps, and control equipment is checked. The vibration testing conducted at the cold flow condition is extended to measurements at several power conditions as the operating power level is raised.

5.1.4 *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.

5.2 *Discussion of Startup and Power Tests*

5.2.1 *General*

All the tests in the startup and power test program are discussed in Subsection XIII-5.2.2 with reference to the particular test purpose, a brief description, and a 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 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.

5.2.2 *Test Purpose, Description, and Acceptance Criteria*

Test Number 1 - Chemical and Radiochemical

Purpose

The principal objectives of this test are a) to maintain control of and knowledge about the quality of the reactor coolant chemistry, and b) 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.

Test Number 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%, 50%, and 100% 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, AEC General Design Criteria and applicable Health Physics procedures.

Test Number 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. The five operational neutron sources of approximately 10^7 neutrons per second will be installed in their normal positions in the core. At least three neutron detectors calibrated and connected in a 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 50% density loaded configuration using a checkerboard pattern. The second 50% of the fuel will be loaded starting at the center and proceeding outward. The following checks will be performed.

1. Control Rod Function Test - The rod will be completely withdrawn and reinserted to check operability, indication, and coupling. This test will be done within one week of the start of fuel loading.

2. Fuel Loading - Two fuel assemblies will be loaded around the blade guides. The remaining two fuel assemblies will be loaded to complete the four assembly cell starting from the center of the core as the blade guides are removed.

3. 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 re-inserted.

4. 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 a) the core is fully loaded b) the partial core shutdown margin demonstration has been completed as required, and c) the full core shutdown margin demonstration has been completed.

Test Number 4 - Full Core 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 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 asymmetrics in the core plus a further allowance for calculational and analytical errors.

Criteria

Level 1

a. The fully loaded core must be subcritical with the strongest control rod fully withdrawn and a shutdown margin $\geq 0.35\% \Delta K/K$.

b. If (a) cannot be satisfied, then the shutdown margin of the fully loaded core is satisfied if the reactor remains subcritical by $\geq 0.25\% \Delta K/K$ *during the sequential, complete withdrawal and insertion of every control rod within the core.*

Test Number 5 - Control Rod Drives

Purpose

To demonstrate that the Control Rod Drive (CRD) System 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 CRD System.

Description

The CRD tests performed during Phases II through IV of the startup test program are designed as an extension of the tests performed during the preoperational CRD System tests. Thus, after it is verified that all control rod drives 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 control rod drive tests to be performed during startup testing is given below.

CONTROL ROD DRIVE SYSTEM TESTS

		Reactor Pressure, Psig (With Core Loaded)			
Test Description	Preop Tests	0	600	800	1000
Position Indication	All	All			
Normal Insert/Withdraw 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 (Zero Accum. Press.)					4*
Scram Times (Scram Discharge Volume High Level	All				
Scram Times, Rated Power (Norm. Accum. Press.)					4**

* *Value refers to the four slowest drives as determined from the normal accumulator pressure scram test at ambient reactor pressure. The time to 5% insertion is used in this determination.*

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

Criteria

Level 1

Each CRD must have a normal insert or withdraw speed of 3 ± 0.6 inches per second, indicated by a full 12 foot stroke in 40 to 60 seconds.

The mean control rod insertion times after the de-energization of the scram solenoid valves based on the average of all control rods shall not exceed the following:

Percent Inserted	*Insertion Time (Seconds) Vessel Dome Pressure ≥950 psig*	*Insertion Time (Seconds) Vessel Dome Pressure <950 psig*
5	*0.375*	*0.475*
20	*0.90*	*1.100*
50	*2.0*	*2.0*
90	*5.0*	*5.0*

The mean control rod insertion times after the de-energization of the scram solenoid valves, based on the average of the three fastest out of four, of any 2 x 2 group of control rods shall not exceed the following:

Percent Inserted	*Insertion Time (Seconds) Vessel Dome Pressure ≥950 psig*	*Insertion Time (Seconds) Vessel Dome Pressure <950 psig*
5	*0.398*	*0.504*
20	*0.954*	*1.166*
50	*2.120*	*2.120*
90	*5.300*	*5.300*

Level 2

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

The insert and withdraw speed of each CRD should be 3 ± 0.6 inches per second as indicated by a full 12-foot stroke in 40 to 60 seconds.

Scram times with normal accumulator charge should fall within the time limits indicated in Figure 5.3-1 of STI-5.

Test Number 6 - SRM Response and Control Rod Sequence

Purpose

The purpose of this test is to demonstrate that the operational sources, SRM instrumentation, and rod withdrawal sequences provide adequate information to achieve criticality and increase power in a safe and efficient manner. The effect of typical rod movements on reactor power will be determined.

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 and signal count-to-noise count ratio.

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. Sequence A 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.

As the withdrawal of each rod group is completed during the power ascension, the electrical power, steam flow, control valve position, and APRM response will be recorded.

A control rod sequence swap will be made at rated power to assure the capability and to check out the B Sequence.

Criteria

Level 1

There must be a neutron signal count-to-noise count ratio of at least two to one on the required operable SRMs or Fuel Loading Chambers.

There must be a minimum count rate of three counts/second on the required operable SRMs or Fuel Loading Chambers.

The IRMs must be on scale before the SRMs exceed the rod block set point.

Test Number 9 - Water Level Measurement

Purpose

To verify the calibration and agreement of the GEMAC and YARWAY water level instrumentation under various conditions.

Description

The test is divided into two parts. The first part will be done at rated temperature and pressure and steady-state conditions and will verify that the reference leg temperature of the YARWAY instrument is the value assumed during initial calibration. If not, the instrument will be recalibrated using the measured value. After the recalibration, the GEMAC and YARWAY indications should be in reasonable agreement. The second part of the test consists of determining the agreement of the water level instrumentation at two core flow rates and various heights.

Criteria

Level 1

The narrow range GEMAC calculated error will be ≤2" and the narrow range YARWAY calculated error will be ≤4.2".

Test Number 10 - IRM Performance

Purpose

To adjust the Intermediate Range Monitor system to obtain an optimum overlap with the SRM and APRM systems.

Description

The IRM system will initially be set at maximum gain prior to heatup. Calibration of the IRM's will be made on the APRM-IRM power overlap region subsequent to calibration of the APRM's.

Criteria

Level 1

Each IRM channel must be adjusted so that overlap with the SRM and APRMs is assured.

The IRM's must produce a scram at ≤96% of full scale (≤120/125 of indicated scale).

Test Number 11 - LPRM Calibration

Purpose

To calibrate the Local Power Range Monitoring System.

Description

The LPRM channels will be calibrated to make the LPRM readings proportional to average heat flux in the four corner fuel rods surrounding each chamber at the chamber elevation. The calibration factors are obtained from either an off-line or process computer calculation.

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 neutron flux in the narrow-narrow water gap at the height of the chamber.

Test Number 12 - APRM Calibration

Purpose

To present the method for calibrating the Average Power Range Monitor System.

Description

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 the results of a constant heatup rate heat balance. The APRMs should be recalibrated in the power range by a heat balance as soon as adequate feedwater indication is available.

Criteria

Level 1

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

Technical Specification and Fuel Warranty Limits on APRM scram and Rod Block shall not be exceeded.

In the startup mode, all APRM channels must produce a scram at ≤15% of rated thermal power.

Recalibration of the APRM System will not be necessary from safety considerations if at least two APRM channels per RPS trip circuit have readings greater than or equal to core power.

Level 2

If the above criteria are satisfied then the APRM channels will be considered to be reading accurately if they agree with the heat balance to within ± 7% of rated power.

Test Number 13 - 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%, or b) if two different fuel assemblies are chosen by the two methods, the CHFR calculated by the other method in each assembly agrees with the MCHFR in that assembly by not more than 10%, and 2) when the LPRM calibration factors calculated by the independent method and the process computer agree to within 5%.

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

Test Number 14 - RCIC System

Purpose

To verify the operation of the Reactor Core Isolation Cooling (RCIC) system over its expected operating pressure range.

Description

A controlled start (of the RCIC system) and a quick start will be done at several reactor pressures between 150 psig and 1000 psig. Verify proper operation of the RCIC system and determine time to reach rated flow. These tests may first be performed with the system in the test mode and then in the auto mode. Discharge flow will be routed to the emergency condensate storage tank. A vessel injection test will also be run at rated pressure.

Criteria

Level 1

The reactor will be allowed to operate at all conditions, including 100% power, if the RCIC System can deliver rated flow, 400 gpm, in less than or equal to the rated actuation time, 30 seconds, against rated reactor pressure. If this criteria cannot be met, the reactor will be allowed to operate at a lower power level until a resolution is obtained.

Test Number 15 - HPCI System

Purpose

The purpose of this test is to verify the proper operation of the High Pressure Coolant Injection (HPCI) system over its expected operating range.

Description

Controlled and quick starts of the HPCI system will be done at reactor pressures between 150 psig and rated. Verify proper operation of the HPCI system, determine the time to reach rated flow, and adjust the HPCI flow controller for proper flow rate of the 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 HPCI System must be capable of pumping rated flow into the reactor vessel. The time from actuating signal to required flow must be less than 25 seconds at any reactor pressure between 150 psig and rated. With pump discharge at any pressure between 150 psig and 1120 psig, the flow should be at least 4250 gpm. The HPCI turbine must not trip off during startup.

Test Number 16 - Selected Process Temperatures

Purpose

The purposes of this test are a) to establish the minimum reactor recirculation pump speed which will maintain water temperature in the bottom head of the reactor vessel within 145 °F of reactor coolant saturation *temperature as determined by reactor pressure, and b) 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 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 shall not be operated unless the coolant temperatures in the upper and lower regions of the vessel are within 145 °F of each other.

Level 2

The bottom head coolant temperature as measured by the bottom drain line thermocouple should be within 50 °F of reactor coolant saturation temperature.

Test Number 17 - 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 that the 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

Final displacements of instrumented points shall not vary from the calculated values by more than ±50% or ±0.5 inches, whichever is smaller. Displacements of less than 0.25 inch shall be considered negligible, since 50% of this value is contingent on the accuracy of measurements.

Test Number 18 - Core Power Distribution

Purpose

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

Description

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 TIP systems have been provided to obtain these traces; a common location can be traversed by each TIP chamber to permit intercalibration.

A check of the reproducibility of the TIP traces is made at least twice; once near the first time the TIP system is used, and again after the TIP system has been used several times. The check is made with the plant at steady-state conditions by producing several TIP traces in the same location with each TIP machine. Traces from the same TIP machine are evaluated to determine the extent of deviations between them. The results of the complete set of TIP traces may also be evaluated to determine core power symmetry.

Criteria

Level 2

In the TIP reproducibility test, the TIP traces shall be reproducible within ± 3.5% relative error or ± 0.15 inch absolute error at each axial position (whichever is greater) in the non-boiling region.

Test Number 19 - Core Performance

Purpose

To evaluate the core performance parameters of the core flow rate, core thermal power level, maximum fuel rod surface heat flux, core minimum critical heat flux ratio (MCHFR), maximum average planar heat flux, and core minimum bundle power ratio (MBPR).

Description

Core power level, maximum heat flux, recirculation flow rate, hot channel coolant flow, minimum critical heat flux ratio, fuel assembly power, and steam qualities will be determined at existing power levels. Plant and in-core instrumentation, conventional heat balance techniques and core performance worksheets, and monograms will be used. This will be performed above 10% power and at various pumping conditions and can be done independently of the process computer functions.

Level 1

The maximum fuel rod heat flux during steady-state conditions shall not exceed allowable heat flux. The design allowable heat flux is 135 W/cm^2.

MCHFR shall be maintained at or above the flow dependent limit line which passes through a MCHFR limit of 1.9 for full power and full flow.

Steady-state reactor power shall be limited to values on or below the licensed flow control line (Maximum power of 2,381 MWt with core flow of at least 73.5 x 10^6 lbs/hr).

The minimum bundle power ratio (MBPR) shall not be maintained greater than or equal to 1.0. The maximum average planar linear heat generation rate (MAPLHGR) shall not exceed the Technical Specification limits.

Test Number 20 - Steam Production

Purpose

To demonstrate that the reactor steam production warranty rate is satisfied.

Description

Operate continuously for 100 hours at rated reactor conditions. When it is determined that all plant conditions are stabilized, the steam production rate will be measured during a two-hour period at conditions prescribed in the Nuclear Steam Supply System warranty.

Level 1

The NSSS parameters as determined by using normal operating procedures shall be within the appropriate license restrictions.

The Nuclear Steam Supply System must produce 9,558,312 lb/hr of steam of not less than 99.7% quality at a pressure of 985 psig at the second isolation valves. This output is contingent upon the feedwater flow being 9,523,312 lb/hr at 367.1 °F with 0.015 ppm maximum oxygen content, less than 0.5 μmho/cm maximum conductivity, and pH 6 to 8, measured at 25 °C.

Test Number 21 - Flux Response to Rods

Purpose

The purpose of this test is to demonstrate the stability of the core power reactivity feedback mechanism with regard to small perturbations in reactivity caused by rod movement.

Description

Rod movement tests will be made at chosen power levels to prove that the transient response of the reactor to a reactivity perturbation is sufficiently stable over the full range of reactor power and flow conditions. The signal from a nearby LPRM string will be measured and evaluated to determine the local core dynamic effects of the 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.

Test Number 22 - Pressure Regulator

Purpose

The purposes of this test are 1) to determine the optimum settings for the pressure control loop by analysis of the transients induced in the reactor pressure control system by means of the pressure regulators, 2) to demonstrate the takeover capability of the backup pressure regulator upon failure of the controlling pressure regulator and to set spacing between the set points at an appropriate value and 3) to demonstrate smooth pressure control transition between the governor valves and bypass valves when reactor steam generation exceeds steam used by the turbine.

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. It is desirable to accomplish the set point change in less than one second. The load reference set point will be set so that the transient is handled by governor valves, bypass valves, and both. The backup regulator will be tested by increasing the operating pressure regulator set point rapidly until the backup regulator takes over 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 except 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.

Pressure control deadband, delay, etc., shall be small enough that steady-state limit cycles, if any, shall produce turbine steam flow variations no larger than ± 0.5% of rated flow as measured by the gross generated electrical power.

Optimum gain values for the pressure control loop shall be determined to give the fastest return from the transient condition to the steady-state condition within the limits of the above criteria.

During the simulated failure of the controlling pressure regulator, the backup regulator shall control the transient such that the reactor does not scram.

Following a ± 10 psi (0.7 kg/cm^2) pressure setpoint adjustment, the time between the set point change and the occurrence of the pressure peak shall be 10 seconds or less.

Test Number 23 - Feedwater System

Purpose

The purposes of this test are 1) to adjust the feedwater control system for acceptable reactor water level control, 2) to demonstrate stable reactor response to subcooling changes, 3) to demonstrate the capability of *the automatic core flow runback feature to prevent low water level scram following the trip of one feedwater pump and 4) to demonstrate adequate response to feed heater loss.*

Description

Reactor water level set point changes of approximately ± 4 inches will be used to evaluate and adjust the feedwater control system settings for all power and feedwater pump modes. The level set point changes will also demonstrate core stability to subcooling changes.

One of the two operating feedwater pumps will be tripped, and the automatic flow runback circuit will act to drop power to within the capacity of the remaining pump. One group of feedwater heaters will be bypassed 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 changes when the plant is operating above the lower limit of the Master Flow Controller.

Following a 3-inch level set point step adjustment in three element control, the time from the set point step change until the water level peak occurs shall be less than 35 seconds without excessive feedwater swings (changes in feedwater flow greater than 25% of rated flow).

The automatic core flow runback feature will prevent a scram from low water level following a trip of one of the operating feedwater pumps.

Test Number 24 - 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. Steam pressure should reach a steady-state within 25 seconds after a bypass valve has been opened or closed.

Test Number 25 - Main Steam Isolation Valves

Purpose

To a) functionally check the main steam line isolation valves (MSIV) for proper operation at selected power levels, b) to determine reactor transient behavior during and following simultaneous full closure of all MSIV and following closure of one valve, and c) to determine isolation valve closure time.

Description

During Hot Standby at rated pressure, both slow and fast single valve closure will be performed. A test of the simultaneous full closure of all MSIVs will be performed at about 100% of rated thermal power. Correct performance of the RCIC System and relief valves will be shown. Reactor process variables will be monitored to determine the transient behavior of the system during and following the Main Steam Line (MSL) isolation. The maximum power conditions at which individual valve full closure tests can be performed without a reactor scram is to be established, and one individual valve full closure test will be performed on the 100% power load line to check ability to perform surveillance tests on this load line.

The MSIV closure times will be determined from the MSL isolation data by multiplying 1.1 times the time increment between closure initiation and activation of the 90% closed light.

Criteria

Level 1

MSIV closure time must be between 3 and 5 seconds, exclusive of electrical delay time. Reactor pressure shall be maintained below 1,240 psig, the setpoint of the first safety valve, during the transient following closure of all valves.

Level 2

The maximum reactor pressure should be about 1,200 psig, 40 psi below the first safety valve setpoint 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% below scram, and steam flow in individual lines must be below the trip point.

Test Number 26 - Relief Valves

Purpose

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

Description

The main steam relief valves will each be opened using the "manual" control mode so that at any time only one is open. During heatup at 250 psig, each valve will be opened and closed to demonstrate proper functioning. Capacity of each relief valve will be determined at rated pressure by the amount of bypass or governor valve closure required to maintain reactor pressure. Proper reseating of each relief valve will be verified by observation of temperatures in the relief valve discharge piping. At selected test conditions each valve will be manually actuated.

Criteria

Level 1

Each relief valve shall 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 valve falls to within 10 °F of the temperature recorded before the valve was opened.

Test Number 27 - Turbine Stop and Control Valve Trips

Purpose

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 breakers 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, reactor protection system (RPS), and the effect of recirculation pump overspeed, if any, during the governor 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% power without a scram will be demonstrated.

Criteria

Level 1

Reactor pressure shall be maintained below 1,240 psig, the setpoint of the first safety valve, during the transient following fast closure of the turbine stop and governor valves.

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

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

Level 2

The maximum reactor pressure should be less than 1,200 psig, 40 psi below the first safety valve setpoint, during the transient following fast closure of the turbine stop and governor valves. This pressure margin should prevent safety valve weeping.

The measurement of simulated heat flux must not be significantly greater than pre-analysis.

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

Test Number 28 - Shutdown from Outside the Control Room[6]

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

If the control room becomes uninhabitable for any reason, it is desirable to be able to shut the reactor down from outside the control room. During the test, operators and observers will be stationed in the control room.

The operators will be available in case of emergency and the observers will be taking test data.

The general procedure is to scram the reactor by simulating a loss of power to the APRM system. This will be accomplished by tripping circuit breakers 9-14A and B in Reactor Protection Power Panels 1A and 1B. This procedure utilizes equipment that is presently installed in the cable spreading room elevation 918'-0". The above referenced equipment and circuits are part of the Reactor Protection system and as such, are safety grade equipment and also meet the requirements of IEEE-279.

Procedure 544 of the CNS Operations Manual is the basis from which this test is written. While the instructions in the Operations Manual cover the general situation, there will probably not be enough decay heat in the core for this particular test to cause the pressure to remain high after the scram. It is therefore interpreted that the test will be complete when the reactor is scrammed, isolated, reactor pressure is decreasing and under control, and water level is under control. Should there be significant core decay heat at the time this test is performed, the test will be deemed complete when the pressure is under control if the other above conditions are met.

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.

Comment: *If a standby reactor operator in the control room makes a minor adjustment on the pressure or level control system in order to keep the test going, this shall not necessarily invalidate the test. If it is judged that the adjustment could have been made outside the control room but at some risk (by leaving his station), the test will still be acceptable.*

Installation Instructions

A positive means of communication will be provided for each operator outside the control room who will participate in the test.

Initial Conditions

The reactor is operating normally at about 20% power.

The shift supervisor is stationed at Boards 25-5A and 25-6A on the 931'6" level of the Reactor Building.

The unit operator is stationed in the RCIC area of northeast corner of the Reactor Building.

The assistant unit operator is stationed southeast corner of the Reactor Building.

The equipment operator is stationed in the controlled corridor outside the reactor feed pump rooms.

An operator will be assigned to the CRD hydraulic control area to close the charging water header valve when required. This operator will be directed from Panel 25-5A to close the valve when water level reaches +45" on the yarway.

Before the test is initiated, a cleanup reject flow of 59 GPM will be established to cancel the effect of the CRD pump input since the CRD pump would be tripped in an actual event.

If the cleanup system isolates for any reason during the test, it will be re-established by the standby control room operator.

Reactor pressure should be adjusted to 980-1000 psig before the test is initiated.

Procedure Outline

1. When all operators are in position and emergency communications have been established, the shift supervisor directs the assistant unit operator to scram the reactor by tripping the scram discharge volume high level switches. The scram will cause the turbine to trip after a time delay. The assistant unit operator will verify the reactor is shut down by observing that the CRD accumulator low pressure alarm lights are lit.

2. The shift supervisor will monitor the reactor water level and pressure on Boards 25-5A and B and 25-6A and B.

2.1 During the scram and turbine trip transient, the water level will drop to about zero, then rapidly increase to above normal. The feedwater control system should stop the water level rise before the yarways go above +45".

2.2 If the water level indication on the yarways reaches +44" the operators at the switchgear should be directed to trip the operating feedwater pump.

2.3 If the feedwater pump is tripped, the unit operator should be directed to start the RCIC system and control level by manually throttling the test bypass valve.

2.4 The pressure will be controlled automatically by the pressure regulator for several minutes after the scram. The pressure will gradually drop until 850 psig is reached at which time the main steam isolation valves close on low steam line pressure in the run mode.

2.5 When conditions are stable, the equipment operator will assure that the turbine generator is put on the turning gear.

3. Immediately after the MSIVs close, perform the following:

3.1 One of the standby control room operators should break vacuum by opening the vacuum breaker (150 MV).

3.2 The standby control room operator will close the SJAE air valves 161 MV and 162 MV.

3.3 The standby control room operator will start a mechanical vacuum pump and open 157 MV, 158 MV, 159 MV, and 160 MV.

3.4 One of the standby control room operators should verify that the turbine-generator remains on turning gear once it has engaged.

Note: The above items would not be necessary during a real event, but are done during this test for the protection of equipment and to prevent possible contamination of the Turbine Building. The above noted valves are found on B&R Drawing 2009, Revision 13.

3.5 The operating feed pump should be tripped by the equipment operator. The unit operator should be requested to start the RCIC System and manually control level by throttling the test bypass to the emergency condensate storage tanks.

4. When the plant in in the isolated condition, with pressure and water level under control and the reactor scrammed, the test is complete.

5. A temperature versus pressure curve is attached for use by the shift supervisor.

6. Detailed procedures of RCIC system local operation are contained in the CNS Operations Manual Procedure 544, "Abandonment of the Control Room."

Test Number 29 - 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 (increase and decrease) are introduced into the recirculation flow control system at chosen points on the 50%, 75%, and 100% load lines. Ramp changes will be made with the concurrence of the District at rates within the range of 10% - 30% power per minute. Load following capability will be demonstrated in all flow control modes.

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 control changes when the plant is operating above the lower limit setting of the Master Flow Controller.

Following a + 10% load demand step from approximately 80% power on the 100% load line, the reactor must not scram, and the 10% load change must be achieved within 40 seconds.

The automatic flow control range must be at least 80% to 100% power along the full-power load line. The load change resulting from a maximum ramp increase in load reference from 80% to 100% load must be achieved within 40 seconds without reactor scram. (Contingent on PCIOMR).

Steady-state limit cycles, if any, shall produce turbine steam flow variations no larger than ± 0.5% of rated flow as measured by the gross generated electrical power.

Test Number 30 - Recirculation System

Purpose

The purposes of this test are a) to evaluate the recirculation flow and power level transients following trips of one or both of the recirculation pumps, b) to obtain recirculation system performance data, and c) to verify that no recirculation system cavitation will occur on the operable region of the power-flow map.

Description

Single and both recirculation pumps will be tripped at various power levels. Two pump trips will be initiated by tripping the M-G set drive motors. A single pump trip at 50% power will be initiated by opening the generator field breaker. The remaining single pump trips are to be initiated by tripping the M-G Set drive motor. Reactor operating parameters will be recorded during the transient and steady-state conditions. MCHFR evaluations will be made for conditions encountered during the transient. With the recirculation pumps operating at the upper limit of the Master Flow Controller, power will be reduced by inserting rods to ~ 20% power where the recirculation pumps will automatically run back to 20% speed. A check will be made to determine if recirculation or jet pump cavitation occurs.

Level 1

MCHFR shall be greater than 1.0 during the transient.

Level 2

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

Test Number 31 - Loss of Turbine-Generator and Offsite Power

Purpose

The purpose of this test is to determine the reactor transient performance during the loss of the main generator and all offsite power, and to demonstrate the correct performance of the station electricity supply system.

Description

The loss of auxiliary power test will be performed at 25% 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

Reactor pressure shall be maintained below 1,240 psig, the set point of the first safety valve, during the transient following the loss of the main generator and all offsite power. All safety systems, such as the Reactor Protection system, the diesel-generator, RCIC and HPCI, must function properly without manual assistance.

Level 2

The maximum reactor pressure should be less than 1,200 psig, 40 psi below the first safety valve setpoint, during the transient following the loss of the main generator and all offsite power. This pressure margin should prevent safety valve weeping. Normal reactor cooling systems should be able to maintain adequate suppression pool water temperature, maintain adequate drywell cooling, and prevent actuation of the auto-depressurization system.

Test Number 32 - Recirculation M-G Set Speed Control

Purpose

The purpose of this test is to determine the individualized characteristics of the recirculation control system (i.e., Drive Motor, Fluid Coupler, Generator, Drive Pump, and Jet Pumps), and to obtain acceptable speed control system performance by the adjustment of linear and nonlinear controller elements.

Description

During the initial startup testing, data will be collected to optimize the loop gains. The cams in the scoop tube positioner feedback loops will be programmed to reduce the effect of abrupt nonlinearities in the coupler characteristics. The time response of the individual recirculation pump speed loops will be optimized by adjusting the gains of the speed controllers. The response of the speed loops will then be checked by step changes in speed demand at all test conditions.

Criteria

Level 1

The decay ratio must be less than 1.0 for each process variable that exhibits oscillatory response to recirculation M-G set speed changes.

Level 2

The decay ratio should be be less than or equal to 0.25 for each process variable that exhibits oscillatory response to recirculation M-G set speed changes over the entire range from 20% to 100% speed.

Following a 10% step change in speed demand from any speed in the speed control range, the time from the step demand until the generator speed peak occurs shall be greater than 10 but less than 25 seconds.

Steady-state limit cycles, if any, shall cause turbine steam flow variations no larger than ± 0.5% of rated flow as measured by the gross generator electrical power output.

Test Number 33 - Main Turbine Stop Valve Surveillance Test

Purpose

The purpose of this test is to demonstrate an acceptable procedure for weekly turbine stop valve surveillance tests at a power level as high as possible without producing reactor scram.

Description

Individual main turbine stop valves must be closed weekly during plant operation as required for plant surveillance testing. At several test points the response of the reactor will be recorded, and the maximum possible power level for performance of this test along with the 100% power flow control line will be established. Each stop valve closure is manually initiated and reset. Rate of valve stroking and timing of the close-open sequence will be chosen to minimize the disturbance introduced.

Criteria

Level 2

Peak neutron flux must be at least 5% below the scram trip setting. Peak vessel pressure must remain at least 10 psi below the high pressure scram setting.

Peak steam flow in the high-flow lines must remain 10% below the high flow isolation trip setting.

Test Number 34 - Vibration Measurements

Purpose

To obtain vibration measurements on various reactor components to demonstrate the mechanical integrity of the system to flow induced vibration and to verify the accuracy of the analytical vibration model. Testing is in response to AEC Safety Guide 20.

Description

Vibration sensors, including strain gauges, LVDTs and accelerometers, are installed on reactor internal components to monitor vibration responses. The extent of the testing depends on the degree of uniqueness as compared to previously instrumented plants. All tests include monitoring of the jet pump, with other components being included as deemed necessary based on previous tests and analytical evaluations. Cold flow tests at elevated pressure will be performed, and hot power tests at normal reactor pressure will be performed during steady-state and transient conditions.

Criteria

The vibration criteria, used to judge the results of the vibration measurements, is the precalculated vibration amplitude at each sensor when the maximum stress in any one of the internal's structures or components equals 10,000 psi, including stress concentration factors.

This stress represents approximately one half the stress limit given in ASME Code Section III for 40-year life. Because of their complexity the criteria are not presented here but will be administered on site by the *vibration test engineer conducting the test. (See Section 8 of the Startup Test Instruction 34 for more detail). The results of the vibration analysis are equal to or less than the precalculated amplitudes.*

Test Number 35 - Recirculation System Flow Calibration

Purpose

The purpose of this test is to perform a complete calibration of the installed recirculation system flow instrumentation.

Description

During the testing program at operating conditions which allow the recirculation pumps to be operated at the speeds required for rated flow at rated power, the jet pump flow instrumentation will be adjusted to provide correct flow indication based on the jet pump flow. After the relationship between drive flow and core flow is established, the flow biased APRM/RBM system will be adjusted to match this relationship.

Criteria

Level 1

Not applicable.

Level 2

Jet pump flow instrumentation shall be adjusted such that the jet pump total flow recorder will provide a correct core flow indication at rated conditions.

The APRM/RBM flow-bias instrumentation shall be adjusted to function properly at rated conditions.

Test Number 70 - Reactor Water Cleanup System

Purpose

The purpose of this test is to demonstrate specific aspects of the mechanical operability of the Reactor Water Cleanup System. (This test, performed at rated reactor pressure and temperature, is actually the completion of the preoperational testing that could not be done without nuclear heating).

Description

With the reactor at rated temperature and pressure, process variables will be recorded during steady-state operation in three modes as defined by the System Process Diagrams: Blowdown, Hot Standby, and Normal.

Level 1

Not applicable.

Level 2

The temperature at the tube side outlet of the nonregenerative heat exchangers shall not exceed 130 °F in any mode.

The pump available NPSH will be 10 feet or greater during the hot standby mode defined in the process diagrams.

The cooling water supplied to the nonregenerated heat exchangers shall be within the flow and outlet temperature limits described in the process diagrams. (This is applicable to "normal" and "blowdown" modes).

Test Number 71 - Residual Heat Removal System

Purpose

The purpose of this test is to demonstrate the ability of the Residual Heat Removal (RHR) System to remove residual and decay heat from the nuclear system so that refueling and nuclear system servicing can be performed and to condense steam while the reactor is isolated from the main condenser.

Description

With the reactor at 75 psig (5.3 kg/cm2) or less, the shutdown cooling mode of the RHR System will be demonstrated. With the reactor operating at ≤10% power, the steam condensing mode will be demonstrated.

Criteria

Level 1

Not applicable.

Level 2

The RHR System shall be capable of operating in the steam condensing mode (with both one and two heat exchangers) at the flow rates indicated on the process diagrams.

The heat removal capability of each RHR heat exchanger in the shutdown cooling mode shall be at least 170 x 10^6 *Btu/hr when the inlet flows and temperatures are as indicated on the process diagrams.*

The time to place the RHR heat exchangers in the steam condensing mode with the RCIC system using the heat exchanger condensate flow for suction shall be one-half hour or less.

Test Number 72 - Drywell Atmosphere Cooling System

Purpose

The purpose of this test is to verify the ability of the Drywell Atmosphere Cooling System to maintain design conditions in the drywell during operating conditions and scram conditions.

Description

During heatup and power operation, data will be taken to ascertain that the drywell atmospheric conditions are within design limits.

Criteria

Level 2

The drywell cooling system shall maintain drywell air temperatures and humidity at or below the design values during normal operation.

Test Number 73 - Cooling Water Systems

Purpose

To verify that the performance of the Reactor Building Closed Cooling Water (RBCCW), Turbine Building Closed Cooling Water (TBCCW), and Service Water Systems are adequate with the reactor at rated temperature.

Description

With the reactor at rated pressure, following initial heatup, data will be obtained to verify that the flow rates in the RBCCW and TBCCW heat exchangers are adequate and properly balanced, and that the heat exchanger outlet temperatures are balanced within design values. Flow rate adjustments will be made as necessary to achieve satisfactory system performance. The test will be repeated at selected power levels to verify continued satisfactory performance with higher plant heat loads.

Criteria

Level 2

Verification that the system performance meets cooling requirements constitutes satisfactory completion of this test.

6.0 NORMAL OPERATIONS

All normal operations are performed in accordance with written procedures which are set forth in the CNS Operations Manual. The CNS Operations Manual includes the following sections:

0. Administrative Procedures
1. Site Services Procedures
2. Operations Procedures
3. Engineering Procedures
4. Instrumentation Operations Procedures
5. Emergency Procedures
6. Surveillance Procedures
7. Maintenance Procedures
8. Chemistry Procedures
9. Radiological Protection Procedures
10. Nuclear Performance Procedures
11. Station Computer Procedures
12. Quality Control Procedures
13. Performance Evaluation Procedures
14. Instrument and Control Procedures
15. Surveillance (Non-Technical Specification) Procedures

Procedures are in compliance with CNS Technical Specifications. Also, procedures call for the reporting of abnormal operations or unusual incidents to the Station Operations Review Committee. This committee investigates such conditions and recommends any procedural or design changes necessary to prevent unsafe conditions.

6.1 Administrative Procedures

The administrative procedures include various program requirements, station organization and responsibilities, records requirements, reporting requirements, and routine work assignments.

6.2 Site Services Procedures

Security and procurement administrative requirements are specified in the site services procedures.

6.3 Operations Procedures

Station operations are conducted in accordance with the operations procedures. These procedures cover normal and foreseeable abnormal operating conditions.

These procedures include the following:

1. Detailed check lists for all major systems, including safety and instrumentation systems, to ensure that all necessary equipment is functioning and in the proper mode for startup.

2. Detailed procedures for startup, normal operation, and shutdown of major pieces of equipment, systems, and integrated station operation.

3. Alarm procedures to define the meaning of alarms and specify the action required.

4. Procedures for abnormal operations.

5. Radwaste procedures.

6.4 Engineering Procedures

Procedures provide instructions for the conduct of engineering activities such as the station modification process, equipment environmental qualification (EQ) documentation control, and the revision of plant drawings.

6.5 Instrumentation Operations Procedures

Operating procedures have been prepared for the following systems:

1. Neutron Monitoring
2. Rod Worth Minimizer
3. Reactor Manual Control
4. Reactor Protection
5. Reactor Vessel Instrumentation
6. Process Radiation Monitoring
7. Area Radiation Monitoring
8. Primary Containment and Reactor Vessel Isolation
9. Rod Position Information
10. Seismic Instrumentation
11. Drywell and Suppression Chamber Instrumentation
12. Safety System Status Panel
13. Meteorological System
14. Ventilation System Radiation Monitoring

6.6 Emergency Procedures

Emergency procedures have been written which would be followed under the following conditions:

1. Following acts of nature, such as an earthquake or flood, and during a tornado watch.

2. During or after Special Events, such as Shutdown From Outside the Control Room and Station Blackout.

3. Lost or degraded system functions.

4. Following postulated accidents.

5. During fires in various plant areas.

6. During a civil disturbance or security threats.

7. Emergency Plan Implementation.

8. Following plant conditions that require entry into the Emergency Operating Procedures.

9. Following plant conditions that require entry into the Severe Accident Management Guidelines.

The NRC in Supplement 1 to NUREG-0737 (Generic Letter 82-33) required licensees to develop a set of human factored, symptom-based, emergency operating procedures to improve human reliability and the ability to mitigate the consequences of a broad range of initiating events and subsequent multiple failures or operator errors to respond to potential accident situations. The BWR Owner's Group (BWROG) developed a set of Emergency Procedure Guidelines (EPGs) which could be utilized by individual licensees in their development of plant-specific EOPs. The BWROG EPGs are symptom-based, and address accident scenarios which are well beyond the CNS original design and licensing basis.

The current Revision 4 of the BWROG EPGs, which have been reviewed and approved by the NRC, meet the requirements outlined in Supplement 1 to NUREG-0737 and NUREG-0899. Subsequent to BWROG EPG Revision 4, BWROG has released EPG/Severe Accident Guideline (SAG) revisions and approved EPG Issues that address severe accident guidelines and discrepancies/enhancements noted from previous revisions. The current CNS EOPs have been developed from Revision 3 of the BWROG EPG/SAG with additional changes from approved BWROG EPG Issues.

6.7 Surveillance Procedures

The Surveillance Program includes written procedures to control the testing and inspection of components and systems as required by CNS Technical Specifications, the CNS Technical Requirements Manual, and the Cooper Nuclear Station Offsite Dose Assessment Manual for Gaseous and Liquid Effluents (ODAM).

Certain Technical Specification surveillance intervals are controlled under the Surveillance Frequency Control Program (SFCP). Refer to Technical Specification 5.5.14 for a description of the program and the method for changing frequencies listed in the SFCP.

6.8 Maintenance Procedures

Maintenance programs, involving removal and replacement, have been devised for various mechanical, electrical, and refueling equipment.

6.9 Chemistry Procedures

Schedules and procedures have been developed for the chemical and radiochemical analysis of plant system samples, plant effluents, and environmental samples.

6.10 Radiological Protection Procedures

Radiological protection procedures have been developed to provide for protection of station personnel against unnecessary exposure to radiation and radioactive materials in conformance with the requirements of 10CFR50, Appendix I. These procedures provide for area control, protective clothing and equipment, personnel monitoring, radiation and contamination plant surveys and controls.

6.11 Nuclear Performance Procedures

Procedures have been implemented for gathering data for evaluation of reactor core performance. Written procedures also include receiving and handling of unirradiated fuel, inspection and channeling of unirradiated fuel, refueling, and handling and storage of irradiated fuel.

6.12 Station Computer Procedures

Procedures have been established to support and control the configuration of computer software and databases.

6.13 Quality Control Procedures

Procedures implement the Quality Control (QC) Program by establishing the requirements for QC activities such as inspector qualification and certification, and the performance of inspections.

6.14 Performance Evaluation Procedures

Procedures ensure standard conditions are established when taking data for trending and evaluating the performance of plant components.

6.15 Instrument and Control Procedures

A calibration program has been developed for various instrumentation and control systems.

6.16 Surveillance (Non-Technical Specification) Procedures

Procedures provide instructions for the performance of surveillance on systems which are not within the scope of the CNS Technical Specifications.

6.17 Shift Crew Requirements

1. A licensed senior reactor operator (SRO) is present at the station at all times when there is any fuel in the reactor.

2. Minimum crew size during reactor operation consists of four licensed reactor operators (two of whom are licensed SRO) and three unlicensed operators. Minimum crew size during reactor cold shutdown conditions consists of two licensed reactor operators (one of whom is licensed SRO) and one unlicensed operator. In the event that any member of a minimum shift crew is absent or incapacitated due to illness or injury, a qualified replacement is designated to report onsite within two hours.

3. A licensed senior reactor operator (SRO) with no other concurrent duties is directly in charge of core alterations, as defined in Technical Specifications.

4. A licensed reactor operator (RO) with no other concurrent duties is directly in charge of operations involving the handling of irradiated fuel other than core alterations.

5. The Shift Manager and the Control Room Supervisor hold a senior reactor operator license.

6. The Unit Operators hold, at a minimum, a reactor operator license.

7. Two licensed reactor operators are in the control room during all startup, shutdown and other periods involving significant planned control rod manipulations.

7.0 EMERGENCY PLANNING

Information related to Emergency Planning is contained in the NPPD Emergency Plan for Cooper Nuclear Station, which is incorporated into the USAR by reference.

8.0 RECORDS

8.1 Initial Testing and Operations

Detailed records of the preoperational and startup test program(s) were developed and are retained with the station records to document initial station performance for reference and comparison with subsequent test results. These records include the results of preoperational testing, initial fuel loading, and low level core tests, and startup and power tests prior to commercial operation.

8.2 Normal Operations and Maintenance

Complete records of operations, tests, and maintenance activities are collected, stored, and maintained as prescribed by the NPPD Cooper Nuclear Station Quality Assurance Program For Operation Policy Document.

9.0 OPERATIONAL REVIEW AND AUDITS

9.1 Administrative Control

Administrative control of plant operations is exercised through the General Manager of Plant Operations. ANSI N18.7-1976, "American National Standard Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants," and the associated Regulatory Guide 1.33 (Safety Guide 33), apply to the CNS Operational QA Program, with exceptions, as described in the NPPD Cooper Nuclear Station Quality Assurance For Operation Policy Document.

9.2 Routine Reviews

Review and audit of facility operations is performed by the Station Operations Review Committee (SORC) and the NPPD Safety Review and Audit Board (SRAB). The responsibilities and authorities of those bodies are contained in ANSI N18.7-1976 as invoked in the NPPD Cooper Nuclear Station Quality Assurance Program For Operation Policy Document, which is incorporated into the USAR by reference.

9.3 NPPD Safety Review and Audit Board

The NPPD Safety Review and Audit Board (SRAB) has been organized to be responsive to the recommendations presented in ANSI N18.7-1976, "American National Standard Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants." Details of SRAB are defined in the NPPD Cooper Nuclear Station Quality Assurance Program For Operation Policy Document. All of the Board members will have either degrees in engineering, or a physical science, or possess equivalent nuclear experience as determined by the Vice President - Nuclear, and have been actively associated with the nuclear energy field for 10 or more years. Alternate membership on the Board will be kept to a minimum and the alternates selected shall be required to have either degrees in engineering, or a physical science, or possess equivalent nuclear experience as determined by the Vice President - Nuclear, and have a minimum of at least five years experience in the design, analysis, or operation of a nuclear facility.

10.0 FIRE PROTECTION PROGRAM

10.1 Purpose and Importance of Fire Protection

The different fire protection systems, including prevention, detection, suppression, preservation of property, and the safeguarding of life are fundamental to the reduction of fire hazards to an acceptable level.

The CNS Fire Protection Program is comprised of the fundamental fire protection features, systems and components, administrative controls, and fire equipment that are provided to satisfy the Fire Protection Operating License Condition 2.C.(4).

10.2 Fire Protection Regulatory Compliance

The fire protection program is based on the NRC requirements and guidelines, Nuclear Electric Insurance Limited (NEIL) Property Loss Prevention Standards and related industry standards. With regard to NRC criteria, the fire protection program meets the requirements of 10 CFR 50.48(c), which endorses, with exceptions, the National Fire Protection Association's (NFPA) 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants - 2001 Edition." Cooper Nuclear Station has further used the guidance of NEI 04-02, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program under 10 CFR 50.48(c)" as endorsed by Regulatory Guide 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants."

Adoption of NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants", 2001 Edition in accordance with 10 CFR 50.48(c) serves as the method of satisfying 10 CFR 50.48(a) and General Design Criterion 3. Prior to adoption of NFPA 805, General Design Criterion 3, "Fire Protection" of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Licensing of Production and Utilization Facilities," was followed in the design of safety and non-safety related structures, systems, and components, as required by 10 CFR 50.48(a).

NFPA 805 does not supersede the requirements of GDC 3, 10 CFR 50.48(a), or 10 CFR 50.48(f). Those regulatory requirements continue to apply. However, under NFPA 805, the means by which GDC 3 or 10 CFR 50.48(a) requirements are met may be different than under 10 CFR 50.48(b). Specifically, whereas GDC 3 refers to SSCs important to safety, NFPA 805 identifies fire protection systems and features required to meet the Chapter 1 performance criteria through the methodology in Chapter 4 of NFPA 805. Also, under NFPA 805, the 10 CFR 50.48(a)(2)(iii) requirement to limit fire damage to SSCs important to safety so that the capability to safely shut down the plant is satisfied by meeting the performance criteria in Section 1.5.1 of NFPA 805.

A Safety Evaluation was issued on April 29, 2014 by the NRC, that transitioned the existing fire protection program to a risk-informed, performance-based program based on NFPA 805, in accordance with 10 CFR 50.48(c).

10.3 Design Basis Summary

10.3.1 Defense-in-Depth

The fire protection program is focused on protecting the safety of the public, the environment, and plant personnel from a plant fire and its potential effect on safe reactor operations. The fire protection program is based on the concept of defense-in-depth. Defense-in-depth shall be achieved when an adequate balance of each of the following elements is provided:

1. Preventing fires from starting,

2. Rapidly detecting fires and controlling and extinguishing promptly those fires that do occur, thereby limiting fire damage,

3. Providing an adequate level of fire protection for structures, systems, and components important to safety, so that a fire that is not promptly extinguished will not prevent essential plant safety functions from being performed.

10.3.2 NFPA 805 Performance Criteria

The design basis for the fire protection program is based on the following nuclear safety and radiological release performance criteria contained in Section 1.5 of NFPA 805:

1. Nuclear Safety Performance Criteria. Fire protection features shall be capable of providing reasonable assurance that, in the event of a fire, the plant is not placed in an unrecoverable condition. To demonstrate this, the following performance criteria shall be met.

    a. Reactivity Control - Reactivity control shall be capable of inserting negative reactivity to achieve and maintain subcritical conditions. Negative reactivity inserting shall occur rapidly enough such that fuel design limits are not exceeded.

    b. Inventory and Pressure Control - With fuel in the reactor vessel, head on and tensioned, inventory and pressure control shall be capable of maintaining or rapidly restoring reactor water level above top of active fuel for a BWR such that fuel clad damage as a result of a fire is prevented.

    c. Decay Heat Removal - Decay heat removal shall be capable of removing sufficient heat from the reactor core or spent fuel such that fuel is maintained in a safe and stable condition.

    d. Vital Auxiliaries - Vital auxiliaries shall be capable of providing the necessary auxiliary support equipment and systems to assure that the systems required under (a), (b), (c), and (e) are capable of performing their required nuclear safety function.

    e. Process Monitoring - Process monitoring shall be capable of providing the necessary indication to assure the criteria addressed in (a) through (d) have been achieved and are being maintained.

2. Radioactive Release Performance Criteria - Radiation release to any unrestricted area due to the direct effects of fire suppression activities (but not involving fuel damage) shall be as low as reasonably achievable and shall not exceed applicable 10 CFR, Part 20, Limits.

Chapter 2 of NFPA 805 establishes the process for demonstrating compliance with NFPA 805.

Chapter 3 of NFPA 805 contains the fundamental elements of the fire protection program and specifies the minimum design requirements for fire protection systems and features.

Chapter 4 of NFPA 805 establishes the methodology to determine the fire protection systems and features required to achieve the nuclear safety performance criteria outlined above. The methodology shall be permitted to be either deterministic or performance-based. Deterministic requirements shall be "deemed to satisfy" the performance criteria, defense-in-depth, and safety margin and require no further engineering analysis. Once a determination has been made that a fire protection system or feature is required to achieve the nuclear safety performance criteria of Section 1.5, its design and qualification shall meet the applicable requirement of Chapter 3.

10.3.3 Codes of Record

The codes, standards and guidelines used for the design and installation of plant fire protection systems are as follows: (for specific applications and evaluations of codes refer to NEDC 10-080, Fundamental Fire Protection Program and Design Elements (B-1 Table)). The NFPA Codes of Record for CNS are:

NFPA 10, Standard for Portable Fire Extinguishers
NFPA 12, Standard on Carbon Dioxide Extinguishing Systems
NFPA 12A, Standard on Halon 1301 Fire Extinguishing Systems
NFPA 13, Standard for the Installation of Sprinkler Systems
NFPA 14, Standard for the Installation of Standpipe, Private Hydrant, and Hose Systems
NFPA 15, Standard for Water Spray Fixed Systems for Fire Protection
NFPA 20, Electric-Drive Controllers and Accessories
NFPA 22, Standard for Water Tanks for Private Fire Protection
NFPA 24, Standard for the Installation of Private Fire Service Mains and Their Appurtenances
NFPA 30, Flammable and Combustible Liquids Code
NFPA 50A, Gaseous Hydrogen Systems
NFPA 72, National Fire Alarm Code
NFPA 80, Standard for Fire Doors and Fire Windows
NFPA 80A, Recommended Practice for Protection of Buildings from Exterior Fire Exposures
NFPA 90A, Installation of Air Conditioning and Ventilating Systems
NFPA 241, Standard for Safeguarding Construction, Alteration, and Demolition Operations
NFPA 256, Standard Methods of Fire Tests of Roof Coverings

10.4 NFPA 805 System Description

10.4.1 Required Systems

10.4.1.1 Nuclear Safety Capability Systems, Equipment, and Cables

Section 2.4.2 of NFPA 805 defines the methodology for performing the nuclear safety capability assessment. The systems, equipment, and cables required for the nuclear safety capability assessment are contained in NEDC 11-019, Nuclear Safety Capability Assessment (NSCA).

10.4.1.2 Fire Protection Systems and Features

Chapter 3 of NFPA 805 contains the fundamental elements of the fire protection program and specifies the minimum design requirements for fire protection systems and features. Compliance with Chapter 3 is documented in NEDC 10-080, Fundamental Fire Protection Program and Design Elements (B-1 Table).

Chapter 4 of NFPA 805 establishes the methodology and criteria to determine the fire protection systems and features required to achieve the nuclear safety performance criteria of Section 1.5 of NFPA 805. These fire protection systems and features shall meet the applicable requirements of NFPA 805 Chapter 3. These fire protection systems and features are documented in CNS Fire Safety Analysis.

10.4.1.3 Radioactive Release

Structures, systems, and components relied upon to meet the radioactive release criteria are documented in NEDC 10-062.

10.4.2 Definition of "Power Block" Structures

Where used in NFPA 805 Chapter 3 the terms "Power Block" and "Plant" refer to structures that have equipment required for nuclear plant operations. For the purposes of establishing the structures included in the fire protection program in accordance with 10 CFR 50.48(c) and NFPA 805, the plant structures listed in Table XIII-10-2 are considered to be part of the Power Block.

TABLE XIII-10-2

POWER BLOCK BUILDINGS

Building/Structure	Fire Area(s)
Reactor Building	RB-A, RB-B, RB-CF, RB-DI, RB-E, RB-FN, RB-J, RB-K, RB-M, RB-N RB-P, RB-T, RB-V, TB-C, DW
Control Building	CB-A, CB-A-1, CB-B, CB-C, CB-D
Turbine Generator Building	TB-A
Diesel Generator Building	DG-A, DG-B
Water Treatment Building	TB-A
Intake Structure	IS-A
Radwaste Building	TB-A
Fire Pump House	YD
Offgas Building	YD
Optimum Water Chemistry Building	YD
Hydrogen Storage Building	YD
Offsite power distribution equipment (i.e., main transformers, emergency transformer, and start-up transformer) and the diesel generator oil storage transfer pumps	YD

10.5 Regulatory Evaluation

The CNS Fire Safety Analysis documents the achievement of the nuclear safety and radioactive release performance criteria of NFPA 805 as required by 10 CFR 50.48(c). This document fulfills the requirements of Section 2.7.1.2 "Fire Protection Program Design Basis Document" of NFPA 805. The document contains the following:

1. Identification of significant fire hazards in the fire area. This is based on NFPA 805 approach to analyze the plant from an ignition source and fuel package perspective.

2. Summary of the Nuclear Safety Capability Assessment (at power and non-power) compliance strategies.

    a. Deterministic compliance strategies

    b. Performance-based compliance strategies (including defense-in-depth and safety margin)

3. Summary of the Non-Power Operations Modes compliance strategies.

4. Summary of the Radioactive Release compliance strategies.

5. Summary of the Fire Probabilistic Risk Assessments.

6. Key analysis assumptions to be included in the NFPA 805 monitoring program.

10.6 Fire Protection Program Documentation, Configuration Control and Quality Assurance

In accordance with Chapter 3 of NFPA 805 a fire protection plan documented in CNS Procedure 0.23, CNS Fire Protection Plan, defines the management policy and program direction and defines the responsibilities of those individuals responsible for the plan's implementation. The CNS Fire Protection Plan:

1. Designates the senior management position with immediate authority and responsibility for the fire protection program.

2. Designates a position responsible for the daily administration and coordination of the fire protection program and its implementation.

3. Defines the fire protection interfaces with other organizations and assigns responsibilities for the coordination of activities. In addition, the CNS Fire Protection Plan identifies the various plant positions having the authority for implementing the various areas of the fire protection program.

4. Identifies the appropriate authority having jurisdiction for the various areas of the fire protection program.

5. Identifies the procedures established for the implementation of the fire protection program, including the post-transition change process and the fire protection monitoring program.

6. Identifies the qualifications required for various fire protection program personnel.

7. Identifies the quality requirements of Chapter 2 of NFPA 805.

Detailed compliance with the programmatic requirements of Chapters 2 and 3 of NFPA 805 are contained in NEDC 11-019, Nuclear Safety Capability Assessment (NSCA), NEDC 10-080, Fundamental Fire Protection Program and Design Elements (B-1 Table), and Procedure 0.23, CNS Fire Protection Plan.

11.0 IODINE MONITORING PROGRAM

In response to NUREG-0578, Section 2.1.8.c, CNS established a method to ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This monitoring method, along with the training of appropriate personnel in its calibration and use under special procedures, provides reasonable assurance that CNS has the capability to accurately detect and thereby obtain an initial estimate of the presence of iodine to determine if the use of respiratory protection equipment by plant personnel is warranted or required.

12.0 TECHNICAL REQUIREMENTS MANUAL

The Technical Specification amendment to convert the CNS Current Technical Specifications (CTS) to Improved Technical Specifications (ITS) per NUREG-1433 (Amendment #178) includes relocation of certain CTS requirements to the Technical Requirements Manual (TRM). The TRM is a licensee controlled document, and the Technical Requirements Manual Limiting Conditions for Operation (TLCOs) are a part of the USAR.

13.0 PROCESS CONTROL PROGRAM

The CNS Process Control Program (PCP) establishes the processing conditions for assuring the solidification, dewatering or stabilization of CNS radioactive waste streams produced from the CNS liquid radioactive waste treatment system and from activities producing radioactive waste requiring solidification, dewatering or stabilization such as decontamination system resins, irradiated components and highly contaminated equipment. The PCP ensures that processing of radioactive waste containing liquid, which is subject to the requirements of 10CFR61, is consistent with the requirements specified in the Cooper Nuclear Station Offsite Dose Assessment Manual for Gaseous and Liquid Effluents (ODAM).

The CNS PCP is comprised of the Dewatering Process Control Program (DPCP) and Vendor Process Control Programs. The DPCP utilizes the CNS dewatering system to process solid wet waste streams from the CNS liquid radioactive waste treatment system or from chemical decontamination resins. Vendor Process Control Programs utilize NRC or Agreement State approved PCP's, stabilization processes and High Integrity Containers to process various forms of solid and liquid radioactive wastes at CNS.

District initiated changes to the PCP become effective upon review and acceptance by SORC. Changes are reported to the NRC in the Radioactive Effluent Release Report, as described in the Cooper Nuclear Station Offsite Dose Assessment Manual for Gaseous and Liquid Effluents (ODAM). The report includes sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information, a determination that the change did not reduce the overall conformance of the solidification waste product to existing criteria for solid wastes, and documentation of the fact that the change has been reviewed and found acceptable by the SORC.

14.0 REFERENCES FOR CHAPTER XIII

1. FSAR Amendment 32.
2. Deleted.
3. Q/A 13.7; Amend. 11.
4. Q/A 13.5; Amend. 11.
5. Q/A 13.6; Amend. 11.
6. Q/A 13.9; Amend. 15.
7. Deleted.
8. Deleted.
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