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14.0 Initial Tests and Operation

[HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED]

The initial nuclear safety-related testing and operation program begins when construction is sufficiently complete to allow testing and/or operation of individual structure, systems or components and extends through the satisfactory performance of each unit's acceptance test at, or near, full power. The objectives of, and methods of achieving, an acceptable initial testing and operation program, as stated herein, are in agreement with the objectives and methods as outlined in the following:

1. *Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors," November 1973.*
2. *Title 10, Code of Federal Regulations, Part 50k Appendix B, Criterion X1, "Tests Controls."*
3. *Standard ANSI N18.7-1972, "Administrative Controls for Nuclear Power Plants."*

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14.1 Test Program

[HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED]

The comprehensive testing program at McGuire Nuclear Station assures that the station:

1. *Has been adequately designed and constructed.*
2. *Meets contractual, regulatory and licensing requirements.*
3. *Does not endanger the health and safety of the public.*
4. *Can be operated in a dependable manner so as to perform its intended function.*

Tests are performed in accordance with approved, written procedures. These written procedures address the purpose and prerequisites of the tests, applicable limits and precautions, required station (or unit) status, test methods and acceptance criteria. Analysis of test results verifies that each test was properly conducted and satisfactorily completed.

The initial testing program is the direct responsibility of the Duke Power Company Nuclear Production Department. The station Manager controls the development of testing procedures, the performance of the specified testing and the evaluation of testing results. The Nuclear Production Department General Office and other departments within Duke, and Westinghouse Electric Corporation and other vendors provide support and assistance to the station staff during the initial testing program, as required.

The Nuclear Production Department General Office staff is responsible for reviewing reactor operating experiences as noted in various NRC and industry publications including:

1. *Edison Electric Institute Nuclear Task Force Abnormal Occurrence Reports*
2. *Atomic Energy Clearinghouse Publication*
3. *Nuclear Safety Information Center PWR Operating Experience*
4. *NRC Reactor Operating Experiences*
5. *NRC Operating Units Status Report (Grey Book)*

Pertinent information from these and other sources is then made available to the station Manager or various members of the station staff for incorporation into or modification of the test program, as appropriate. This is an informal program which has no specific implementing procedures but rather relies on the experience and judgment of both the station and general office personnel.

14.1.1 Test Program Administration

14.1.1.1 Organization

The McGuire station Manager is assigned the responsibility for the proper administration and conduct of the initial testing program. The Superintendent of Operations, Technical Services, and Maintenance are responsible to the station Manager for testing assigned to their cognizance. Other station staff members are assigned initial testing responsibilities as appropriate. The education, experience, training and qualifications of responsible station personnel are given in Chapter 13.

Technical support, coordination, consulatory and advisory services may be provided to the station staff by the Nuclear Production Department General Office and other departments within Duke, and by Westinghouse and other vendors as necessary.

The preoperational and startup test schedule is shown in Table 14-1.

14.1.1.2 Preparation of Procedures

Preoperational and startup test procedures are developed by members of the station staff who have been assigned responsibility for those activities to be controlled by test procedures. The originator of a test procedure has information available from other organizations within Duke Power Company such as the Design Engineering Department and the Nuclear Production Department General Office and outside organizations such as Westinghouse and other vendors. Information available from these organizations includes System Descriptions, drawings, technical manuals and the Final Safety Analysis Report as well as consulatory and advisory services to assist the members of the station staff in developing test procedures and establishing acceptance criteria.

See Section 17.2.6 of DUKE-1, "Quality Assurance Program," for a description of procedure review and approval.

14.1.1.3 Changes to Procedures

Proposed changes to preoperational and startup test procedures may be recommended by any responsible person perceiving the need for such changes. A change to a test procedure is either a minor change or a major change. A minor change is a change which corrects errors of a typographical or editorial nature in an approved procedure. Any other change to an approved procedure is considered a major change.

A major change to an approved procedure receives reviews equivalent to that received by the original procedure; however, the station Manager or a Group Head can make the determination that the safety significance of a proposed change is such that further review is not necessary and can approve the change provided that the appropriate review authority is advised of the approved change.

All changes are documented; major changes by means of a procedure revision notice or by reissuing the procedure and minor changes by entering the change in the control copy of the applicable test procedure.

14.1.1.4 Conduct of Tests

A member of the station staff, normally the individual responsible for the preparation of the applicable procedures is assigned the responsibility for the proper conduct of each required test. Prior to commencement of a test this designated individual is responsible for determining that preparations for the test are complete, which should include, but is not necessarily limited to, the determination that:

1. *An approved procedure is available for use.*
2. *The installation of the structures, systems or components to be tested is correct and complete, or is acceptable for testing without involving major changes to the procedure.*
3. *The responsibility for the structures, systems or components to be tested has been transferred to the Nuclear Production Department station organization.*
4. *Any prerequisites necessary for the test have been satisfied.*

5. *Any special test equipment, that is to be used for the test, is available to test personnel and has been installed and checked out and is functioning properly, as appropriate.*
6. *Appropriate personnel who will assist in the performance of the test, have been briefed and/or have completed a practice test run.*
7. *Any necessary reference materials are available and have been reviewed.*

When preparations are complete the test is initiated. Each test is conducted in accordance with the applicable, approved procedure and if, during the performance of the test, changes to the procedure are necessary they are made in accordance with Section 14.1.1.3. Copies of procedures for conducting tests, fuel loading, initial startup, and operating activities will be made available to NRC inspection personnel for examination at least 30 days prior to the scheduled activity, and not less than 90 days prior to the scheduled fuel loading date.

14.1.1.5 Verification of Test Results

Following completion of a test, the responsible station staff member assembles all necessary documentation and verifies that the acceptance criteria have been met. Further verification that the test has been properly conducted, and that the acceptance criteria have been met is made by the Superintendent of Operations, Superintendent of Technical Services, Superintendent of Maintenance. A complete procedure then becomes an approved test upon approval by the Superintendent of Operations, Superintendent of Technical Services, Superintendent of Maintenance.

14.1.1.6 Documentation of Testing

Initial testing records are retained in sufficient detail to permit adequate confirmation of the testing program. In particular, these records identify the data taker(s), the results of the testing and whether or not the results were acceptable, discrepancies and their cause, and any corrective action resulting from a test. These testing records are retained for the life of the station.

14.1.2 Resolution of Discrepancies

A discrepancy exists when a test has been performed and a requirement of the applicable procedure has not been fulfilled. If, during the conduct of a test or during the verification or approval of test completion, a discrepancy is determined then action is taken to resolve the discrepancy. This action consists of:

1. *Documentation of the discrepancy.*
2. *Determination of the necessary corrective action.*
3. *Initiation and performance of corrective action.*
4. *Documentation of the completion of corrective action.*
5. *Retest or reevaluation of test results, as necessary to verify the adequacy of the corrective action.*

Corrective action(s) may consist of necessary changes to the applicable procedure and/or modifications to structures, systems and components. In the event that procedure revision is required, such revisions are made in accordance with Section 14.1.1.3. If equipment changes are necessary then such changes are executed after the appropriate design review and are properly documented and become a part of the station records.

14.1.3 Testing Prior to Initial Fuel Loading

Testing prior to initial fuel loading begins when construction is sufficiently complete to allow testing and/or operation of individual structures, systems and components and continues until commencement of initial fuel loading. Systems are sequenced for completion, calibration and functional testing in order to provide auxiliary services for the testing and operation of other systems; e.g., service water systems are functionally tested and placed into service relatively early, as they are required for other system tests.

Procedures include appropriate consideration to assure that prerequisite steps for equipment testing, such as completion of necessary construction, prior testing, safety precautions and measures to preserve equipment status, have been or will be performed.

The status of equipment components and facilities is verified to assure readiness for operation. Typical items covered include cleanliness; lubrication; setting of limit switches, torque limiting devices and electrical protective devices; calibration of instruments; and presence of safety devices. Consideration is also given to providing an equipment run-in period to minimize early failures during operation of the station.

Individual system tests establish functional adequacy by operation under prescribed conditions. The tests are designed to permit evaluation of system performance including, for example, the measurement of flow, temperature, pressure, response time and vibration; transfer of power supply to emergency power; and accuracy and response of control devices.

Testing prior to initial fuel loading is intended to demonstrate, as nearly as can be simulated, the overall integrated operation of unit systems at rated conditions including simultaneous operation of auxiliary systems.

Abstracts of significant tests performed prior to initial fuel loading are given in Section 14.4. Deferment of any portion of the preoperational or startup testing program described in the FSAR requires the review and approval of the station Manager.

14.1.4 Initial Startup Testing

Initial startup testing includes initial fuel loading, initial criticality and low power and power escalation testing. The purposes of this program are to:

1. *Accomplish an orderly and safe initial core loading.*
2. *Accomplish that calibration and testing required to assure correct monitoring of necessary parameters during the approach to critical and subsequent power operation.*
3. *Accomplish an orderly and safe approach to critical.*
4. *Conduct low power physics testing sufficient to assure that design parameters are met and safety analysis assumptions are correct.*
5. *Accomplish orderly power escalation, with the required physics and system testing terminating with completion of the power escalation program.*

The startup sequence shown in Table 14-2 depicts the tests performed versus reactor power level. The sequence represents an orderly and organized approach to initial startup testing, assuring that the necessary steps are completed and that a unit may safely proceed to the next power level.

Several power level hold points are required for satisfactory reactor test completion and evaluation, notably at the 25, 50, 75 and 90 percent power levels. Before departure from each *of these levels, specific parameters are measured, evaluated, compared to predicted values and compared with limiting values specified in any applicable Technical Specifications.*

The sequence depicted in Table 14-2 is used as a basis for the planning and scheduling of tests. The existing condition and status of systems and components are the primary factors in determining which tests and operations can be performed at a given time. Therefore, the schedule may be modified to meet particular needs and conditions, but in no event is a test or operation undertaken without satisfying the prerequisites for that test or operation.

Abstracts of significant tests performed during initial startup testing are given in Secion 14.5 and Table 14-2.

Results of startup testing, including initial fuel load testing through full power testing, is documented for each unit's initial fueling and each subsequent reload.

14.1.4.1 Initial Fuel Loading

Fuel loading begins when all prerequisite tests and operations are completed.

The core is assembled in the reactor vessel, submerged in reactor grade water containing sufficient dissolved boric acid to maintain a calculated core effective multiplication factor of 0.95 or lower. The minimum required water level is specified in the procedure. Coolant is circulated by at least one Residual Heat Removal pump, which may be stopped if necessary to ensure proper seating of the fuel assemblies. Core moderator chemistry conditions (particularly boron concentration) are prescribed in the fuel loading procedure and are verified periodically by chemical analysis of moderator samples taken prior to and at a prescribed frequency during core loading operations.

Minimum shift requirements are established in the station technical specifications. Limits are set and administratively controlled to ensure that operating personnel do not receive excessive work duty.

Core loading instrumentation consists of two permanently-installed source range channels and two temporary incore source range channels. The permanent channels are monitored in the Control Room by licensed personnel; the temporary channels are installed in the Containment and are monitored by technically-qualified personnel. At least one permanent channel is equipped with an audible count-rate indicator heard in the Control Room and in the Containment. Constant communication is maintained between the Control Room and fuel handling areas. If fuel loading operations are delayed for a significant amount of time, nuclear instrumentation is response-checked to resumption of loading.

Fuel assemblies and inserted components are received, inspected, and placed in storage in accordance with written, approved procedures. At the time of fuel loading, they are placed in the reactor vessel one at a time according to a previously-established, approved, written sequence which was developed to provide reliable core monitoring with minimum possibility of core mechanical damage. The fuel loading procedure documents include tabular check sheets which prescribe and verify the successive movements of each fuel assembly and its specified inserts from its initial position in the storage racks to its final position in the core. Checks are made of component serial numbers and types at various transfer points to guard against possible inadvertent exchanges or substitutions of components; however, in the event that mechanical damage is sustained during fuel loading operations, to a fuel assembly of a type for which no spare is available onsite, an alternate core loading scheme, whose characteristics closely approximate those of the initial prescribed pattern, is determined and all physics parameters specified for the initial design are verified.

An initial nucleus of eight fuel assemblies, the first of which contains an activated neutron source, is the minimum source-fuel nucleus which permits subsequent meaningful inverse count-rate monitoring. This initial nucleus is determined by calculation and previous experience to be markedly subcritical ($k_{eff} \leq 0.95$) under the required conditions of fuel loading. Each subsequent fuel addition is accompanied by detailed neutron count rate monitoring to determine that the just-loaded assembly does not excessively increase the count rate and that the extrapolated inverse neutron count rate ratio is not decreasing for unexplained reasons.

Criteria for safe fuel loading require that loading operations cease immediately if:

1. *An unanticipated increase in the neutron count rate by a factor of two occurs on all responding instrumentation channels during any single loading step after the initial nucleus of eight fuel assemblies is loaded (excluding anticipated changes due to detector and/or source movement), or*
2. *The neutron count rate on any individual instrumentation channel increases by a factor of five during any single loading step after the initial nucleus of eight fuel assemblies is loaded (excluding anticipated changes due to detector and/or source movements).*

An alarm in the Containment and Control Room is coupled to the source range channels with a setpoint at approximately five times the current count rate. This alarm automatically alerts personnel of a high count rate and requires an immediate stop of fuel loading operations until the situation is evaluated.

Upon completion of core loading, the reactor upper internals and the pressure vessel head are installed and additional mechanical and electrical tests are performed prior to initial criticality, in order to assure that the critical operation of the reactor may be conducted in a safe and well-monitored manner.

Mechanical and electrical tests are performed on the rod cluster control assembly drive mechanisms and include an operational checkout of the mechanisms and of the individual rod position indicators. Tests are performed on the reactor trip circuits to test manual trip operation. At all times that the rod cluster control assembly drive mechanisms are being tested, the boron concentration in the reactor coolant is maintained such that criticality cannot be achieved with all rod cluster control assemblies fully withdrawn. Design reactor coolant flow is verified and flow coastdown times are measured to determine conformance with safety analysis.

14.1.4.2 Initial Criticality

Initial criticality is established by sequentially withdrawing the shutdown and control groups of rod cluster control assemblies from the core, leaving the last withdrawn control group inserted far enough in the core to provide effective control when criticality is achieved, and then continuously diluting the heavily borated reactor coolant until criticality is attained. The successive stages of rod cluster control assembly group withdrawal, and of boron concentration reduction, are monitored by observing changes in neutron count rate, as indicated by the normal unit source range nuclear instrumentation, as a function of group position during rod motion and, subsequently, as a function of reactor coolant boron concentration during dilution. Throughout this period samples of the primary coolant are obtained and analyzed for boron concentration. Inverse neutron count rate ratio monitoring is used as an indication of the proximity and rate of approach to criticality during rod cluster control assembly group withdrawal and during reactor coolant boron dilution.

14.1.4.3 Low Power and Power Escalation Testing

A prescribed program of reactor physics measurements is undertaken to verify that the basic static and kinetic characteristics of the core are as expected and that the values of the kinetic coefficients assumed in the safety analysis are satisfactory. The measurements are made at low power and primarily at or near operating temperature and pressure. Measurements include verification of calculated values of rod cluster control assembly group reactivity worth, of isothermal temperature coefficient under various core conditions, of differential boron concentration reactivity worth and of critical boron concentration as a function of rod cluster control assembly group configuration. In addition, measurements of relative power distribution are made.

Concurrent tests are conducted on the instrumentation including the source and intermediate range nuclear channels. The sequence of testing is such that a verification of proper core loading through use of moveable incore detectors, together with determination of required static and kinetic coefficients, is done prior to power escalation. These measurements are sufficient to assure that power escalation can be safely undertaken. Formal zero power physics testing is conducted, including the determination of static and kinetic measurements as called for in the procedures.

On Unit 2 several zero power physics tests and several startup tests are not performed because of the similarity of the Unit 1 and Unit 2 cores. Specific cases are noted in Section 14.5 and Table 14-2.

Procedures specify the tests and measurements to be conducted, and the conditions under which each is to be performed, in order to assure both the safety of operation and the relevancy and consistency of the results obtained. If significant deviations from design predictions exist, unacceptable behavior is revealed or apparent anomalies develop, the testing is suspended and the situation reviewed to determine whether or not a question of safety is involved prior to the resumption of testing.

After the operating characteristics of the reactor and unit are satisfactorily verified by low power testing, a program of power level escalation in successive stages brings the unit to its full rated power level. Both reactor and unit operational characteristics are examined and conformance with the safety analysis is verified before escalation to the next programmed level is affected. Measurements are made to determine the relative power distribution in the core as a function of power level and as a function of rod cluster control assembly group position. Secondary system heat balances assure that the indications of power level are consistent and provide the bases for the calibration of the power range nuclear channels. Also, the response of the unit is determined for the design load changes and unit trips.

The adequacy of radiation shielding is verified by gamma and neutron radiation surveys at selected points throughout the station at various power levels. Unexpected radiation levels in any area are rechecked and additional or temporary shielding may be erected or pertinent access procedures instituted, as appropriate. Periodic sampling is performed to verify the chemical and radio-chemical analysis of the reactor coolant.

14.1.5 Operating Procedure Verification

During the test program, as individual systems and components are tested, applicable permanent station operating procedures, as described in Section 13.5, are implemented and revised, as necessary.

Those procedures which cannot be implemented during the test program are revised, as appropriate, based on initial testing, operating experience and comparison with the as-built *systems. This assures that these procedures are as accurate and comprehensive as practicable.*

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14.2 Supplementary Personnel For Initial Tests and Operation

[HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED]

As described in Section 14.1 above, the McGuire station staff, under the direction of the station Manager, is responsible for the administration and conduct of the initial testing and operation program. This staff is considered adequate to perform the initial tests and operations necessary to verify the functional integrity and operational safety of the station. However, personnel other than the station staff may provide temporary and/or part-time services relating to the initial testing and operations program. These personnel may come from the Nuclear Production Department, other department within Duke or from organizations external to Duke. Such personnel, if assigned, are under the control and supervision of the station management and, for the period of time assigned, function as members of the station organization.

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14.3 Reload Startup Physics Test Program

This section defines the minimum reload physics test program to be performed at the McGuire Nuclear Station. The startup Physics test program was originally submitted to the NRC per H. B. Tucker's April 14, 1988 letter. The NRC issued a SER on May 18, 1988, approving Duke's submittal. The test program has changed since this submittal by adopting new approved methods (Dynamic Rod Worth Measurement) and maintaining consistency with ANSI/ANS 19.6.1. The purpose of the test program is to provide assurance that the reactor core is loaded correctly and can be operated as designed. The testing covered in this section involves core physics measurement only -- mechanical or electrical tests such as the control rod trip time test, the initial calibration of instrumentation, etc. are not addressed.

The startup physics test program is comprised of the following tests (measurements and/or calculations):

1. Zero Power Test Phase
   a. All Rods Out Critical Boron Concentration
   b. Isothermal Temperature Coefficient
   c. Control Rod Bank Worth
2. Power Ascension Test Phase
   a. Flux Symmetry Check (Low Power)
   b. Core Power Distribution (Intermediate Power)
   c. Core Power Distribution (High Power)
   d. Hot Zero Power to Hot Full Power Reactivity Difference

These tests will be performed during each initial startup after refueling. Additional testing may be done as conditions warrant. Routine surveillance monitoring after successfully completing startup testing is not addressed in this document. The initial test conditions, test method, and acceptance criteria for each test are provided.

14.3.1 Definitions

The following terms are defined for the purpose of this section:

RTP.	Rated Thermal Power.
HZP	Hot Zero Power.
HFP	Hot Full Power.
DRWM	Dynamic Rod Worth Measurement.
NC.	Reactor Coolant System.
ARO.	All Rods Out.
ITC.	Isothermal Temperature Coefficient. The reactivity change per unit temperature change in the fuel/moderator, with the fuel and moderator at the same temperature.
PCM.	Percent Milli-Rho; 10^{-5} $\Delta K/K$.

$F^{N}_{\Delta H}$. The ratio of a particular fuel assembly power to the core average fuel assembly power.

Deleted Per 2011 Update.

ADRC Advanced Digital Reactivity Computer calculates core reactivity by using an external signal which is proportional to the core neutron flux.

RMS error.

$$\left[\sum_{i=1}^{N}(\Delta X_i)^2 / N\right]^{1/2}$$

where,

ΔX_i = Measured-predicted $F^{N}_{\Delta H}$ (or normalized reaction rates)for ith operable instrumented location, and

N = Number of operable instrumented location

14.3.2 Zero Power Test Phase

If any acceptance criterion is exceeded, the test results will be reviewed with regard to the impact on applicable safety analyses and subsequent plant operation. This review will be performed by cognizant engineers from the Station Reactor Unit or the General Office Nuclear Engineering Section.

14.3.2.1 All Rods Out Critical Boron Concentration

14.3.2.1.1 Initial Test Conditions

1. Mode 2, below sensible heat
2. NC average temperature 557 ±2°F
3. NC pressure 2235 ±50 psig
4. equilibrium NC boron concentration

14.3.2.1.2 Test Method

The equilibrium NC boron concentration is measured with control bank D near fully withdrawn and core reactivity essentially zero. The fully withdrawn positive reactivity associated with each Shutdown and Control Bank is measured over the course of Dynamic Rod Worth Measurement (DRWM) via the reactivity computer. The measured reactivities from the withdrawals are averaged and converted to an equivalent boron concentration, which is then added to the equilibrium NC concentration to obtain the ARO Boron Concentration.

14.3.2.1.3 Review Criteria

1. Predicted ±50 PPM Boron
2. Predicted ± 500 pcm equivalent boron

14.3.2.1.4 Acceptance Criterion

Predicted ± 1000 pcm equivalent boron

14.3.2.2 Isothermal Temperature Coefficient

14.3.2.2.1 Initial Test Conditions

1. Mode 2, below sensible heat
2. NC average temperature 557 ± 2°F
3. NC pressure 2235 ± 50 psig
4. equilibrium NC boron concentration

14.3.2.2.2 Test Method

Starting with an equilibrium NC system boron concentration, NC system temperature is changed at least 1.1°F. An evaluation of the slope from this change is performed for calculation of the ITC via the reactivity computer. The measurement is repeated with an NC system temperature change in the opposite direction. If measurement results are not within +/- 1 pcm/°F of each other, the test is repeated. The resulting ITCs are averaged and used in conjunction with the doppler coefficient to determine the Moderator Temperature Coefficient.

Deleted Per 2011 Update.

14.3.2.2.3 Review Criteria

1. Measured heatup and cooldown ITC values agree within ± 1 pcm/°F
2. Predicted ±2 PCM/°F

14.3.2.2.4 Acceptance Criterion

Measured MTC within the limit of Tech Spec 3.1.3

14.3.2.3 Control Rod Bank Worth

14.3.2.3.1 Initial Test Conditions

1. Mode 2, below sensible heat
2. NC average temperature 557 ±2°F
3. NC pressure 2235 ±50 psig
4. equilibrium NC boron concentration

14.3.2.3.2 Test Methods

Deleted Per 2011 Update.

The DRWM technique is used to determine the individual rod bank worths. Initially, the control rods are fully withdrawn from the core. The rod bank to be measured is inserted in a continuous motion to near fully inserted. The flux signals from the upper and lower section of an excore detector will be recorded while the bank is being inserted at a increased speed but always equal to or less than 72 steps/minute. The rod bank will be withdrawn to all rods out position while maintaining a stable startup rate. The remaining banks will be measured in a similar manner.

A reactivity computer will use the flux signals recorded during the insertion of the bank to calculate the reactivity worth of the bank.

The DRWM methodology is described in WCAP-13360-P-A, Westinghouse Dynamic Rod Worth measurement technique, DPC-NE-2012A, Dynamic Rod Worth Measurement using CASMO/SIMULATE (SER Feb. 15, 2000), and DPC-NE-1005-P-A, "Nuclear Design Methodology using CASMO-4/SIMULATE-3 MOX" (SER dated August 20, 2004) which have been reviewed and approved by the NRC. DPC-NE-1005-P-A extended the use of the DRWM methodology to CASMO-4/SIMULATE-3 MOX models.

14.3.2.3.3 Review Criteria

1. Individual banks ±15% or ± 100PCM of predicted (whichever is greater). If this review criterion is not met the following actions must be taken:
   a. The test results will be reviewed to determine if any bank specific measurement process error has occurred.
   b. For confirmation, remeasure the bank using DRWM.
   c. If confirmed, a power distribution measurement (full core flux map) must be taken and evaluated prior to exceeding 5% RTP.
2. Total rod worth measured (sum of all banks) ± 8% of total predicted worth. If this review criterion is not met the following actions must be taken:
   a. Perform an overall review of the measurement process and design constants.
   b. If the error is not found, measure the worth of the heaviest bank (rod swap reference bank) by boron dilution and compare the boron difference results with the prediction. Use the results of this measurement to assess the impact on the other DRWM bank measurements.
   c. Perform an evaluation on the impact to the shutdown margin within 60 days by taking into account the effects of the design error on the measurement results.

Note: If the core designer uses a number different than 10% for the bank worth allowance in the shutdown margin calculation, then the review criterion on the sum of worths must be adjusted accordingly (e.g., a 7% allowance will yield a review criterion of 5.6%, 7%x8/10).

14.3.2.3.4 Acceptance Criteria

Total rod worth measured (sum of all banks) ≥ 90% of total predicted worth. If this acceptance criterion is not met the following actions must be taken:

a. Perform an overall review of the measurement process and design constants.
b. If the error is not found, measure the worth of the heaviest bank (rod swap reference bank) by boron dilution and compare the boron difference results with the prediction. Use the results of this measurement to assess the impact on the other DRWM bank measurements.
c. Perform an evaluation on the adequacy of the current (beginning of life) shutdown margin prior to exceeding 5% RTP.

Note: If the core designer uses a number different than 10% for the bank worth allowance in the shutdown margin calculation, then the acceptance criterion on the sum of worths must be adjusted accordingly (e.g., a 7% allowance will yield a acceptance criterion of 93%).

14.3.3 Power Ascension Test Phase

If any acceptance criterion is exceeded, the test results will be reviewed with regard to the impact on applicable safety analyses and subsequent plant operation. This review will be performed by cognizant engineers from the Station Reactor Unit or the General Office Nuclear Engineering Section.

14.3.3.1 Flux Symmetry Check - Low Power

14.3.3.1.1 Initial Test Conditions

1. reactor power between 0 and 30% RTP
2. NC average temperature T_{ref} ±2°F
3. NC pressure 2235 ±50 psig

14.3.3.1.2 Test Method

A full incore flux map is taken, maintaining reactor power and control bank D position stable - power changing ≤1%/hr, ±5 steps rod motion. The map analysis includes a comparison of predicted to measured $F_{\Delta H}^{N}$ or normalized reaction rates for all operable instrumented locations.

14.3.3.1.3 Acceptance Criteria

1. $F_{\Delta H}^{N}$ or normalized reaction rates ±10% of predicted, and
2. Root Mean Square error ≤0.05

14.3.3.2 Core Power Distribution - Intermediate Power

14.3.3.2.1 Initial Test Conditions

1. reactor power between 50 and 80% RTP
2. NC average temperature T_{ref} ±2°F
3. NC pressure 2235 ±50 psig

14.3.3.2.2 Test Method

A full incore flux map is taken, maintaining reactor power and control bank D position stable - power changing ≤1%/hr, ±5 steps rod motion. The map analysis includes a comparison of predicted to measured $F_{\Delta H}^{N}$ or normalized reaction rates for all operable instrumented locations.

14.3.3.2.3 Acceptance Criteria

1. $F_{\Delta H}^{N}$ or normalized reaction rates ±10% of predicted, and
2. Root Mean Square error ≤0.05

14.3.3.3 Core Power Distribution - High Power

14.3.3.3.1 Initial Test Conditions

1. reactor power above 90% RTP
2. NC average temperature T_{ref}±2°F
3. NC pressure 2235 ±50 psig

14.3.3.3.2 Test Method

A full incore flux map is taken, maintaining reactor power and control bank D position stable - power changing ≤1%/hr, ±5 steps rod motion. The map analysis includes a comparison of predicted to measured $F_{\Delta H}^{N}$ or normalized reaction rates for all operable instrumented locations.

14.3.3.3.3 Acceptance Criteria

1. $F_{\Delta H}^{N}$ or normalized reaction rates ±10% of predicted, and
2. Root Mean Square error ≤0.05

14.3.3.4 HZP to HFP Reactivity Difference

14.3.3.4.1 Initial Test Conditions

1. reactor power above 90% RTP
2. NC average temperature T_{ref} ±2°F
3. NC pressure 2235 ±50 psig
4. Xenon worth changing ≤0.1 PCM per minute
5. Control bank D ≥200 steps withdrawn or positioned, as necessary, for axial flux difference control or control rod withdrawal limits
6. equilibrium NC boron concentration

14.3.3.4.2 Test Method

The NC boron concentration is measured with control bank D near fully withdrawn. The measured value is corrected to account for any reactivity effects due to deviations from the conditions the predicted boron concentration is based on. It is also adjusted for the measured-to-predicted difference determined at the Zero Power ARO Critical Boron measurement.

14.3.3.4.3 Acceptance Criterion

Predicted ±50 PPM Boron

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14.4 Testing Prior to Initial Fuel Loading

[HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED]

14.4.1 Component Cooling Water System Functional Test

Abstract

Purpose

To demonstrate the capability of the Component Cooling Water System to provide cooling water during normal unit operation, during unit cooldown and during an emergency situation; and to demonstrate proper system response to a simulated engineered safety features actuation signal.

Prerequisites

Systems and components supplied by the Component Cooling Water System are available to the extent required to conduct this test.

Test Method

The system is operated and performance demonstrated under normal unit conditions for each of the two flow paths.

System performance is demonstrated for normal unit cooldown conditions with full flow through both flow paths.

Automatic starting of the component cooling water pumps and automatic valve alignment is demonstrated for a simulated safety injection signal. Automatic Containment isolation is verified for receipt of a simulated high-high Containment pressure signal.

Acceptance Criteria

System flow requirements are met for all modes of operation. System instrumentation functions properly. Automatic valve alignment and pump start occur in response to engineered safety features actuation signals.

14.4.2 Emergency Diesel Generator Functional Test

Abstract

Purpose

To demonstrate that the emergency diesel generators are capable of automatically providing the required power to equipment vital to safe reactor shutdown under emergency conditions.

Prerequisites

Diesel generator support systems are operational. Switching relays are calibrated and all normal bus protection is operational. Diesel generator area ventilation and fire protection are functional.

Test Method

Automatic diesel generator starting is demonstrated for a simulated safety injection signal and for a simulated loss of normal power signal. Diesel generator loading is demonstrated for a simulated loss of normal power signal and diesel generator sequenced loading is demonstrated for the combined safety injection and loss of normal power simulated signals.

Diesel Generator Sequenced Loading is demonstrated for each of the two generators, utilizing only one at a time. In addition, the manual operation of the generators is verified by loading each generator to its continuous operation rating.

Acceptance Criteria

Automatic start and loading sequence are accomplished for simulated emergency conditions. During loading, frequency is maintained at not less than 95 percent of nominal and generator voltage is maintained at not less than 75 percent of nominal. Support system performance is sufficient to operate the diesel generator at its continuous rated load for the time required to reach temperature equilibrium plus one hour.

14.4.3 125 VDC Vital Instrumentation And Control

Abstract

Purpose

To demonstrate that the 125 VDC Vital Instrumentation and Control batteries are capable of providing power during normal operation and under abnormal conditions.

Prerequisites

Battery area ventilation must be adequate.

Test Method

The system is energized for normal operation and normal dc loads applied. The capability of each battery charger to individually maintain a float charge on its associated battery, while concurrently maintaining the normal bus dc loads, is demonstrated. The capability of each charger to individually provide an equalizing charge is demonstrated.

The capability of the system to transfer each bus from battery charger to battery power is demonstrated by de-energizing the chargers while the applicable bus is carrying its normal station loads.

A battery performance test is performed in accordance with IEEE 450-1972.

The actual load on the batteries/chargers recorded during the performance of the Safety Injection System function test, is compared with the design loads for the system.

The operability of vital loads is verified at reduced system voltage by the operation of selected equipment.

Acceptance Criteria

All battery chargers provide satisfactory float charge while concurrently maintaining normal bus loads. The system responds properly to loss of normal unit power. Batteries are capable of supplying dc power upon de-energization of their chargers.

14.4.4 Diesel Generator Fuel Oil System Functional Test

Abstract

Purpose

To demonstrate the capability of the system to provide an adequate fuel supply to the emergency diesel generators for operation under load conditions as described in the FSAR.

Prerequisites

The electrical and fire protection systems are in service to the extent of conducting fuel oil transfer demonstrations in a safe manner. The emergency diesel generators must be available to operate loaded to the capacity required under the accident conditions as described in the FSAR.

Test Method

With both diesel fuel oil day tanks at a known acceptable operating level, the diesel generators are started and brought to the accident load condition. The units are operated for a sufficient length of time to measure the fuel consumption rate from each day tank.

The day tanks are filled with the units running with the transfer pumps. During transfer pump operation flow rates and discharge pressures are demonstrated for each pump.

During operation, a day tank low-level alarm is actuated to verify automatic starting of the fuel oil transfer pumps.

Acceptance Criteria

The demonstrations are conducted for a sufficient length of time to verify flow rates, discharge pressures, fuel oil consumption rates and modes of operation as described in the system design section of the FSAR.

14.4.5 Radiation Monitoring System Functional Test

Abstract

Purpose

To demonstrate the capability of the Radiation Monitoring System to detect, indicate and record radiation levels in process systems, effluents and various station areas, and to alarm when high radiation levels are present or upon system circuit failure.

Prerequisites

Containment isolation valves associated with the system are operational, ventilation systems are operational in areas where samples are withdrawn or exhausted at other than atmospheric pressure, and sample tubing routing is verified.

Test Method

Sample system flowrates are verified where applicable, alarm setpoints are verified, high radiation and circuit malfunction alarms are demonstrated, and channel calibrations are verified utilizing the transfer sources provided with the system.

Acceptance Criteria

System channels respond to transfer sources in agreement with primary calibration data. Sample flow rates meet design requirements. Alarms function properly.

14.4.6 Nuclear Instrumentation System Functional Test

Abstract

Purpose

To assure the proper operation of the Nuclear Instrumentation System prior to Reactor Protection System and Engineered Safety Features Actuation System testing and initial fuel loading.

Prerequisites

System instrumentation and cabling is installed and tested. Electrical power is available and verified.

Test Method

Nuclear instrumentation channels are operationally checked and aligned using test signals. All channels are checked to verify that trip, rod stop and alarm setpoints are in accordance with the test documents. Proper operation of associated auxiliary equipment and recorders is also verified by the use of simulated test signals. Source range detectors and channels are operationally checked and aligned, with the detector in the presence of a test neutron source.

Acceptance Criteria

Nuclear instrumentation channels respond properly to test signals and operate in accordance with system design requirements.

14.4.7 Reactor Protection System Functional Test

Abstract

Purpose

To demonstrate the capability of the Reactor Protection System to respond properly to logic initiation signals prior to initial fuel loading.

Prerequisites

The instrument and protection systems are energized, calibrated and aligned in accordance with the test documents.

Test Method

Proper operation of the Reactor Protection System is verified under various logic conditions. Testing is performed utilizing signals or simulated signals on each of the nuclear and process protection system analog inputs in accordance with the applicable manufacturer's instruction manual. Response timing of channels is verified through insertion of signal at detector and timing response. In order to verify, on a selective basis that initial overall channel response times are consistent with response times assumed in the safety analysis, the response time of one of each type sensor listed below is measured either separately or as a part of the channel response time.

Pressurizer Pressure
Reactor Coolant Pump Bus Undervoltage
Reactor Coolant Pump Bus Underfrequency
Reactor Coolant Loop Flow
Steam Generator Water Level

The methods which may be utilized in performing these measurements are described in Section 7.2.2.2.3. Each individual protection channel and logic train is tested, timed and verified through tripping of the reactor trip breakers.

Acceptance Criteria

Instrument channels and solid-state logic trains for reactor protection and protection permissives function in accordance with the test documents and the functional logic diagrams. Annunciators, channel status lights and permissive interlock lights function properly.

14.4.8 Engineered Safety Features Actuation System Functional Test

Abstract

Purpose

To demonstrate the capability of the system, when actuated, to protect against significant accidental release of radioactivity to the atmosphere and to provide automatic sequencing of engineered safety features as required for emergency core cooling.

Prerequisites

All system and components actuated by the Engineered Safety Features Actuation System are operable to the extent required to conduct this test. All components to be actuated are in their normal power operation position.

Test Method

Appropriate signals are inserted at the various detectors to verify proper system logic and measure channel response time through the process and logic subsystems. In order to verify, on a selective basis, that initial overall channel response times are consistent with the response times assumed in the safety analysis, the response time of one of each type sensor listed below is measured either separately or as a part of the channel response time.

Pressurizer Pressure
Containment Pressure
Steam Generator Water Level
Main Steam Flow
Steam Generator Pressure

The methods which may be utilized in performing these measurements are described in Section 7.2.2.2.3.

Proper response of appropriate systems and components to a containment isolation signal ("T" signal), safety injection signal ("S" signal) and containment spray signal ("P" signal) is demonstrated by manual actuation of the appropriate pushbuttons. These testing provides response timing of valve and pump operation (using test flow lines). The test will demonstrate the ability of the Engineered Safety Features Actuation System to function properly with offsite power available. Electrical load shedding and subsequent loading of the diesel generators are individually demonstrated verifying independence of the various redundant power sources and load groups as outlined in Regulatory Guide 1.41. The test will also demonstrate unit response to a complete loss of offsite power including diesel generator loading with appropriate blackout loads. Blackout will be initiated by tripping switchyard breakers for both units for the Unit 1 test. For the Unit 2 test, the blackout will be initiated by isolating all bus ties between Units 1 and 2 and then tripping the Unit 2 switchyard breakers.

Acceptance Criteria

Upon initiation of each actuating signal, proper system lineup and sequence of valves, pumps and electrical systems are accomplished within the required time interval.

14.4.9 Rod Control System Functional Test

Abstract

Purpose

To demonstrate the operation of the Rod Control System in the automatic and manual modes of control. To assure proper interfacing between the Rod Control System and signals from other systems. To verify proper operation of rod control permissives.

Prerequisites

Electrical power available and tested. Nuclear instrumentation channels available. Temperature instrumentation channels available.

Test Method

The manual mode of control is checked for each applicable position of the bank selector switch and the response of the system is checked into the logic cabinet. The automatic mode is operationally checked by inserting simulated nuclear instrumentation signals and temperature signals into the Rod Control System. Logic cabinet rod speed and direction signal are verified to be in accordance with the test documents and manufacturer's instruction manual as the simulated input signals are varied. Automatic rod control permissives and permissive status lights are monitored for proper operation during the use of the simulated test signals.

Acceptance Criteria

Manual and automatic system responses are in accordance with the manufacturer's instruction manual and the test documents.

14.4.10 Residual Heat Removal System Functional Test

Abstract

Purpose

To demonstrate the capability of the system to provide design flows, pressures and cooldown rates for unit shutdown and unit cooldown conditions.

Prerequisites

The Nuclear Service Water System and Component Cooling Water System are operational. The Reactor Coolant System is sufficiently complete to accept the Residual Heat Removal System flows and pressures.

Test Method

The system is operated and performance demonstrated for unit cold shutdown and unit cooldown conditions for each of the flow paths.

Acceptance Criteria

System flows, pressures and cooldown rates are verified, and setpoint adjustments made in compliance with the test documents, for the various flow paths. System interlocks, instruments, alarms and automatic valve operation are verified to function properly.

14.4.11 Safety Injection System Functional Test

Abstract

Purpose

To demonstrate the capability of the system to provide design flows during each of the injection phases using centrifugal charging pumps, safety injection pumps, accumulators and residual heat removal pumps. To demonstrate proper operation of all pumps and valve motors when supplied from normal offsite power or emergency power sources.

Prerequisites

Reactor Coolant System flushing and hydrostatic testing have been completed. For the ambient temperature portion of the test, the system is cold and the vessel head is removed. The hot temperature portion of the test is conducted during the hot functional test program. The refueling water storage tank contains sufficient water to perform the required testing, and the refueling canal is available to accept excess water drained from the Reactor Coolant System. Normal and emergency power sources are available to all safety injection equipment.

Test Method

Each pump is tested separately with water drawn from the refueling water storage tank. The overflow from the reactor vessel passes into the refueling canal. Pump head and flow are determined during this period. Pumps are then operated to determine a second point on the head/flow characteristics curve.

Each accumulator is filled and partially pressurized with the motor operated discharge valve closed. The valve is opened and the accumulators allowed to discharge into the reactor vessel. Additionally, the capability to operate the valve under maximum differential pressure conditions of maximum expected accumulator precharge pressure and zero RCS pressure is verified.

The Safety Injection System is aligned for normal power operation, with the exception that the boron injection tank is filled with refueling water instead of concentrated boric acid and the accumulators are not pressurized. A safety injection signal ("S" signal) is manually initiated, allowing all affected equipment to actuate. Proper system alignment, flow capability and acceptable net positive suction head performance under maximum system flow conditions are demonstrated. The Safety Injection System is operated in its various modes of operation, using the Refueling Water Storage Tank as the source of water. This testing is repeated with the breakers supplying offsite power to the 4160 volt Auxiliary Power System tripped so that operation of the emergency diesels is tested in conjunction with the Safety Injection System. Proper system and component response times are demonstrated in the Engineered Safety Features Actuation System Functional Test.

The ability of the charging pumps to supply flow through the injection lines while the Reactor Coolant system is at operating conditions (greater than 500°F) is verified. Operation of injection line check valves and accumulator check valves is also verified at this time.

Acceptance Criteria

Pump and system head and flow performance is verified. Automatic valve operation and proper sequencing are verified. Level, flow and pressure instruments are set at the specified points in accordance with the test documents, and provide alarm, reset and control signals as required.

Proper check valve operation is verified. Satisfactory safety injection signal generation and transmission, including operation of the emergency diesel generators and sequential load pickup is verified.

14.4.12 Upper Head Injection Functional Test

Abstract

Purpose

To demonstrate that the upper head injection portion of the Safety Injection System is capable of performing as required.

Prerequisites

The Reactor Coolant System is cold and the reactor vessel head installed with the upper internals removed. The Reactor Coolant system water inventory is sufficiently low and the reactor coolant piping vented to minimize pressure buildup in the Reactor Coolant System during injection.

Test Method

Blowdown tests are performed by filling and pressurizing the upper head injection water and nitrogen accumulators with the isolation valves closed. The isolation valves are subsequently opened and the accumulator is allowed to discharge into the reactor vessel.

Two blowdown tests are performed - one with low accumulator pressure (about 100 psi) and one with gas pressure in the normal operating range. The low pressure test provides piping resistance information utilized in determining the level set points for isolation valve closure. The high pressure test provides verification of isolation valve operation under maximum differential pressure and verification that the required volume of water is injected into the Reactor Coolant System prior to isolation valve closure.

During Reactor Coolant System cooldown from hot conditions, check valves operability is demonstrated by injection of a small flow of water upstream of the valve.

Acceptance Criteria

The required volume of water is delivered to the reactor vessel. Isolation valves and associated interlocks operate in accordance with design. Check valves are demonstrated operable at elevated temperatures.

***Note:** The high pressure blowdown test is conducted on Unit 1 only.*

14.4.13 Containment Spray System Functional Test

Abstract

Purpose

To demonstrate the capability of the system to respond to an actuation signal and to provide the required flows.

Prerequisites

The refueling water storage tank is available and contains sufficient water for demonstration tests. The system is aligned to isolate the spray nozzles, obtain suction from the refueling water storage tank and recirculate water back to the refueling water storage tank.

Test Method

With the spray nozzles bypassed, the system is operated with suction from the refueling water storage tank to demonstrate design flow rates. This includes system testing with both loops operating concurrently, and with each of the flow trains operating independently. Proper operation of the controls and interlocks associated with valves relied on to effect a transfer to the recirculation mode is demonstrated. Automatic startup of the system is demonstrated by simulating high-high containment pressure logic.

Proper spray nozzle performance is demonstrated by blowing air or smoke through the spray ring headers and nozzles and observing the flow from the nozzles.

Acceptance Criteria

System flow requirements are met for the tested modes of operation, the system responds automatically to high-high containment pressure logic, flow nozzles are unrestricted and all instrument indications, interlocks and alarms function properly.

14.4.14 Chemical And Volume Control System Functional Test

Abstract

Purpose

To demonstrate the capability of the Chemical and Volume Control System to perform as required during various modes of operation.

Prerequisites

The Reactor Coolant System Hot Functional Test is in progress. Chemical and Volume Control System components and piping are cleaned, flushed and hydro tested. System instrumentation and controls are available and calibrated. The reactor coolant filter is installed.

Test Method

The capacities of the letdown paths and the reactor coolant filter differential pressure are measured. Letdown temperature and pressure controller responses are demonstrated. Proper operation of the excess letdown flow path is verified and the demineralizer is tested for design flow rates and pressure drops. Charging pumps are tested for capability to deliver varying flow rates. Volume control tank level control indications and alarm setpoints are checked. Operational calibration and testing of the different modes of dilution and boration are accomplished. Flow rates of the various subsystems are measured and verified.

Acceptance Criteria

The system operates properly and in accordance with the applicable system description. Flows, temperatures, pressures, levels and alarms are in accordance with the test documents.

14.4.15 Containment Initial Integrated Leak Rate Test And Structural Integrity Test

Abstract

Purpose

To verify the structural integrity of the Containment and to verify that the integrated leak rate from the Containment does not exceed the maximum allowable leakage.

Prerequisites

The Containment is operational and penetration local leak rate testing has been satisfactorily completed. All systems inside Containment which have containment isolation valves identified as "thru line leak class 2" in Table 6-114 are vented and drained except for the following:

System	Reason
Ice Condenser glycol supply and return	*Ice Condenser is in operation*
Steam Generator wet layup recirculation	*Steam Generators are in recirculation*

Test Method

Closure of containment isolation valves is accomplished by the means provided for normal operation of the valves. The Containment is strength tested at 110 to 115 percent of the design *internal pressure (15 psig), and an integrated leak rate test is conducted at not less than the calculated peak accident pressure. Testing is performed in accordance with 10CFR 50, Appendix J. A minimum of 4 hours is allowed for stabilizations of containment conditions (temperature, pressure, humidity) prior to commencement of the leak rate test. The test duration is at least 24 hours. A supplement leak rate test is performed by imposing a known leak rate on the containment utilizing the methods outlined in ANSI N45.4-1972.*

Acceptance Criteria

The structural integrity of the Containment is verified and the measured Containment integrated leak rate does not exceed .15 percent by weight of the containment volume per day. The difference between the supplemental test data and the initial leak rate data is within .05 percent by weight of the containment volume per day.

14.4.16 Containment Isolation Functional Test

Abstract

Purpose

To demonstrate the functional performance of the valves and dampers that are provided in those lines which must be isolated immediately following an accident.

Prerequisites

All valves that are to be operated as a result of an isolation signal are operable in the automatic mode. The necessary support systems for isolation actuation are in service to the extent required for this test.

Test Method

With the applicable components and support systems operable, Containment isolation signals are simulated to demonstrate automatic Containment isolation. Response times of all power operated valves are measured.

Acceptance Criteria

Proper operation of all containment isolation valves and equipment necessary to effect containment isolation is verified as demonstrated by proper valve closures and system lineups upon the initiation of the Containment isolation signals. Operating times for all power operated valves are within the required limits.

14.4.17 Nuclear Service Water Functional Test

Abstract

Purpose

To demonstrate the capability of the system to provide flow during normal unit operation, during cold shutdown, during unit cooldown and during an emergency situation. To demonstrate proper component response to simulated engineered safety features actuation signals.

Prerequisites

All support systems and components supplied by the Nuclear Service Water System are available to the extent required for conducting this test.

Test Method

The system is started and performance demonstrated under normal operating conditions for each of the flow paths.

System performance is verified for normal unit cooldown conditions by full flow through each flow path.

Automatic flow path isolation and component performance is verified for simulated engineered safety features actuation signals.

System recirculation to the Standby Nuclear Service Water Pond is verified.

Acceptance Criteria

System flow is verified for all modes of operation. Instrumentation functions properly and automatic valve alignment and pump startup occur in response to simulated engineered safety features actuation signals.

14.4.18 Pressurizer Functional Test

Abstract

Purpose

To establish the continuous spray flow rate and to determine the effectiveness of the pressurizer normal control spray and of the pressurizer heaters.

Prerequisites

The Reactor Coolant System is in a hot condition. The Reactor Coolant System is lined up for normal operation in accordance with applicable operating procedures. All reactor coolant pumps are operating. Each bank of pressurizer heaters is operable.

Test Method

While maintaining pressurizer level constant, spray bypass valves are adjusted until a minimum flow is achieved which maintains less than a 200°F temperature difference between the spray line and the Reactor Coolant System, and pressurizer heater cycling is minimized.

To determine pressurizer heater and spray capability, normal pressurizer spray valves are closed. All pressurizer heaters are then energized and the time to reach a 2300 psig system pressure is measured and recorded. Full spray is initiated through the spray valves. Pressure versus time is recorded for each transient. The transient is terminated at a Reactor Coolant System pressure of 2000 psig by shutting the spray valves.

Acceptance Criteria

For setting of continuous spray flow, the flow through each bypass valve is such that the temperature difference between the spray line and the Reactor Coolant system is less than 200°F.

For spray and heater response tests, the response to induced transients is within the band assumed in the FSAR Safety Analysis.

14.4.19 Pressurizer Relief Tank Functional Test

Abstract

Purpose

To demonstrate the functional performance of the pressurizer relief tank and associated equipment.

Prerequisites

This test is performed after the hydrostatic leak test of the Reactor Coolant System and prior to the start of the initial unit heatup. Support systems and components supplied by the pressurizer relief tank must be available to the extent necessary to demonstrate pressurizer relief tank performance. The pressurizer relief tank is ready for service and empty. Associated instrumentation and control equipment checkout has been completed.

Test Method

The pressurizer relief tank is isolated, filled and pressurized. Data are recorded during level and pressure increases. Associated instrumentation and control equipment setpoints are verified and/or adjusted as necessary. The tank is drained and backfilled using nitrogen as a cover gas.

Acceptance Criteria

The level and pressure alarms and cover gas system operate at the setpoints designated in the test documents. The pressurizer relief tank spray flow is verified to meet design requirements. Automatic pressure regulating valves and valve interlocks are verified to function properly.

14.4.20 Reactor Coolant System Heatup Functional Test

Abstract

Purpose

To perform the operational checks required on the Reactor Coolant System and associated valves, instrumentation and equipment necessary to bring the unit from cold shutdown conditions to normal operating temperature and pressure for the first time.

Prerequisites

Required support systems and components are available to the extent necessary to perform Reactor Coolant System heatup. The Pressurizer Relief Tank Functional Test has been completed. Reactor Coolant System component and piping supports have been installed in accordance with design drawings. The pressurizer continuous spray valves are fully open. The Reactor Coolant System hydrostatic test has been completed. The Reactor Coolant System is filled and vented. The Reactor Coolant System pumps and pressurizer heaters are operational.

Test Method

With a specified charging, letdown and seal water flow and the pressurizer heaters on, nitrogen is vented from the pressurizer and a pure steam space established as reactor coolant temperature and pressure are increased. The reactor coolant pumps are then started. Water chemistry specifications are established. The heatup may be interrupted at various intervals to perform other preoperational tests and to record data as necessary. The heatup is continued in steps to, or near, the normal no-load operating temperature. Steam generator level is maintained throughout this procedure as required.

Acceptance Criteria

Satisfactory performance of components and systems which are exposed to Reactor Coolant System temperatures is demonstrated. Proper operation of instrumentation controllers and alarms is verified to be in accordance with the test documents. Design charging, letdown and seal water flows are verified. Pressurizer level control is verified.

14.4.21 Reactor Coolant System Hot Functional Test

Abstract

Purpose

To demonstrate the performance and instrumentation required during normal hot operations. To demonstrate proper pressurizer pressure control and operation of reactor coolant flow trips and alarms. To provide normal operating temperatures and pressures for performing or completing the preoperational tests dependent upon these conditions.

Prerequisites

The Reactor Coolant System heatup test is completed and the system is at no-load pressure and temperature conditions. Required support systems are available to the extent necessary to perform this test.

Test Method

Pressurizer pressure control is demonstrated by manually controlling the heaters and/or the spray valves. The system is stabilized and the reactor coolant flow trips and alarms are demonstrated by stopping the individual reactor coolant pumps. The system is again stabilized at no-load conditions in order that other peroperational tests may be performed as required.

Acceptance Criteria

Operation and setpoints for the pressurizer heaters, sprays, pressure control and the reactor coolant flow trips and alarms are verified to be in accordance with test documents.

14.4.22 Reactor Coolant System Cooldown Functional Test

Abstract

Purpose

To demonstrate proper control of the Reactor Coolant System cooldown rate. To demonstrate pressurizer cooling, introduction of pressurizer nitrogen blanket and maintenance of pressurizer to reactor coolant loops temperature differential requirements.

Prerequisites

The preoperational hot functional tests are complete and the Reactor Coolant System is at the hot shutdown condition. All support systems required for the planned cooldown and depressurization are in service, or available to the extent necessary to perform this test.

Test Method

The pressure and level of the volume control tank are varied to perform Reactor Coolant System degassification. All but one reactor coolant pump is shutdown and the primary system cooldown is started. The pressurizer is cooled by de-energizing the heaters and operating the sprays as necessary. The residual heat removal system is placed in service, a nitrogen blanket is introduced into the pressurizer to replace the steam space and cooldown continued observing the pressurizer to Reactor Coolant system differential temperature requirements until cold conditions are reached. The system is depressurized, drained and vented in preparation for maintenance and/or initial fuel loading.

Acceptance Criteria

Proper Reactor Coolant System degassification and Reactor Coolant System pressurizer cooldown methods and procedures are verified. Venting and draining of the Reactor Coolant System can be accomplished properly.

14.4.23 Reactor Coolant System Thermal Expansion And Restraint Test

Abstract

Purpose

To determine the movement of the Reactor Coolant System and demonstrate that the thermal expansion confirms predicted analytical design movements and thermal stresses within the system. To assure that the Reactor Coolant System can expand without obstruction during the initial system heatup from the cold condition to operating conditions. To confirm the design travel of system supports and restraints, operability and acceptability of same. To demonstrate that the Reactor Coolant System piping and components return to their baseline cold position after the initial cooldown to ambient conditions.

Prerequisites

This test is carried out in conjunction with the Reactor Coolant System Heatup Functional Test and the conditions required for the performance of that test must be established. Supports, restraints and hangers have been installed and expansion clearances set to the proper clearances in accordance with design and construction drawings. Reference points and predicted maximum movements have been established and identified in the detailed test procedure which is jointly developed and approved by Duke Design Engineering and Nuclear Production Departments. Engineering acceptance criteria for all movements, limitations, precautions, and corrective actions as applicable are described in detail in the test procedure. Measurement devices or fixtures installed for this test have been firmly secured. Insulation at points of anticipated interference has been removed to allow measurements to be taken. All lock devices have been removed from system supports and restraints.

Test Method

Prior to starting the Reactor Coolant System Heatup Functional Test, with the Reactor Coolant System at ambient temperature, a complete set of position measurements at selected points is taken and the data recorded. During the Reactor Coolant system heatup, position measurement data are recorded at specified intervals. If at the specified interval predetermined movements are exceeded or do not take place, the system heatup is held constant until necessary corrective actions have been taken. After the successful heatup and on completion of the plant cooldown, a complete set of position measurements is again taken at ambient temperature.

Acceptance Criteria

Unrestricted expansion and acceptable predicted movements are verified for selected points on components and piping of the Reactor Coolant System in accordance with the detailed procedure. The components and piping are verified to return to their approximate baseline cold position. Any movement exceeding the established maximum is referred to the Design Engineering Department for evaluation, acceptance or corrective action. Design Engineering evaluates all recorded results in the event a critical measurement does not conform to the established criteria and resolves any discrepancy found to exist.

14.4.24 Auxiliary Feedwater System Functional Test

Abstract

Purpose

To demonstrate the capability of the system, when actuated, to supply water to the steam generators if normal feedwater sources are unavailable.

Prerequisites

All support systems are in service to the extent necessary to operate the Auxiliary Feedwater System. The normal and alternate supplies of water are available to the pump suctions. The steam generators are in service to the extent of accepting auxiliary feedwater pump discharge. A temporary steam supply may be required for testing of the turbine-driven auxiliary feedwater pump.

Test Method

With the steam generators in the hot shutdown temperature and pressure condition, each auxiliary feedwater pump is started separately to demonstrate flow from the upper surge tank, condenser hotwell, and auxiliary feedwater condensate storage tank. Pump suction is also lined up to the Nuclear Service Water System and pumps are individually started to demonstrate flow. This service water need not be pumped to the steam generators, however.

Automatic start signals are simulated for each auxiliary feedwater pump to assure proper operation.

Acceptance Criteria

Proper system flows and flow paths are verified. All system interlocks, alarms and logic function properly. All required valve operations take place as specified.

14.4.25 Control Room Air Conditioning And Ventilation System Functional Test

Abstract

Purpose

To demonstrate the capability of the Control Room Air Conditioning and Ventilation System to provide and maintain a satisfactory environment during normal and emergency operations.

Prerequisites

All support systems are operational to the extent necessary to perform the test. Access to the Control Room is limited while the test is being performed.

Test Method

Each control room air handling unit is operated in conjunction with its respective filter train and dampers to demonstrate air capacity, direction of flow, static pressure and proper operation, for both normal conditions and emergency conditions. The refrigeration units are tested to demonstrate their proper operation and cooling capacity. Instrumentation is verified for proper sequencing and function.

Acceptance Criteria

Satisfactory environment can be maintained during the normal mode and emergency mode of operation. All alarms, control and interlocks perform their design function.

14.4.26 Containment Purge And Ventilation System Functional Test

Abstract

Purpose

To demonstrate the capability of the system to provide proper temperature control within Containment.

Prerequisites

Cooling water is available and all filter elements are installed and clean. Personnel access control is in effect.

Test Method

The system fans are operated and temperature measurements are taken at selected points in the containment. In place filter tests are conducted to verify system flows and pressure drops are as designed. Adequate removal capability of the filtering elements will be verified.

A high radiation signal is simulated to verify isolation of the purge penetrations.

Acceptance Criteria

The system is capable of maintaining the Containment within specified temperature limits. Proper system response to a simulated high radiation signal is demonstrated. All interlocks instruments alarms and filters function properly.

14.4.27 Loss Of Instrument Air Test

Abstract

Purpose

To demonstrate that a reduction and loss of instrument air pressure causes fail-safe operation of pneumatically-operated equipment.

Prerequisites

The Instrument Air System is in service at rated pressure with support systems operational to the extent necessary to conduct the test. All pneumatic loads are cut-in to the extent possible at the time test begins.

Test Method

Where safe to personnel and equipment, a total loss of air test is performed on integrated systems by venting down instrument air to all the components in the systems. Where deemed necessary, components are depressurized individually and their response noted.

Acceptance Criteria

Proper fail-safe operation of systems subjected to a reduction and loss of instrument is verified.

14.4.28 Containment Air Return And Hydrogen Skimmer System Functional Test

Abstract

Purpose

To demonstrate the capability of the system to operate and to provide design air flows.

Prerequisites

The ice condenser inlet doors are blocked closed to prevent operation.

Test Method

Each containment air return fan and hydrogen skimmer fan is operated. Tests are performed to demonstrate the proper head and flow characteristics of each fan. Automatic operation of the Containment air return fans is verified for a simulated high-high containment pressure signal (S_p). System interlocks are also verified.

Acceptance Criteria

Each fan provides air flow as specified in the FSAR and the Containment air return fans respond properly to a simulated high-high containment pressure signal (S_p).

14.4.29 Ice Condenser System Functional Test

Abstract

Purpose

To accomplish initial system ice loading and to establish the operability of the system.

Prerequisites

System components are installed and installation checkout of the various package units is completed.

Test Method

The operability of associated subsystems is demonstrated and initial adjustment of air handling units is made to achieve design ambient conditions in the ice condenser. The lower door opening forces will be tested to insure that operation is in accordance with design. The drains will be tested to insure proper operation. Initial ice loading is then accomplished and the proper quantity and quality of the ice is verified. Proper operation of the air handling units is assured by verification that desired temperature conditions are achieved throughout the ice condenser and by inspection of individual air handling units. system alarm and operational setpoints are verified during the preceding steps.

Acceptance Criteria

Initial system ice loading is properly accomplished and system operability is verified.

14.4.30 Containment Divider Barrier Leakage Area Verification Test

Abstract

Purpose

To verify that the available leakage area for ice condenser steam bypass is within the limits assumed in the safety analysis.

Prerequisites

All normal divider barrier seals are installed. The Reactor Coolant System is at hot conditions. The Upper and Lower Containment Ventilation Systems are operable as required. Refueling canal drains are operable.

Test Method

Measurements are taken of all identified leakage paths. The total known leakage area is computed and compared to the value assumed in the safety analysis.

All known leakage paths at the operating deck level are then temporarily sealed. Operation of the Upper and Lower Containment Ventilation Systems is adjusted to obtain a sufficient temperature difference between the upper and lower compartments to induce a natural circulation air flow. The open refueling canal drains are then inspected to detect air flow. If any significant air flow is detected, an investigation is made of the operating deck area and seals to identify the return flow leakage path. Repairs and modifications are made as necessary.

Acceptance Criteria

Total divider barrier leakage area is consistent with the value assumed in the safety analysis.

14.4.31 Transformer Voltage Tap Setting Verification

Abstract

Purpose

To verify that all transformer voltage tap settings are adequate to provide optimum voltage at the safety-related buses for the full load and minimum load conditions that are expected throughout the anticipated range of voltage variation of the offsite power source.

Prerequisites

The 7KV, 4KV, and 600VAC bus loading must be adequate and relatively stable for the duration of the test. All sources of power to Class 1E buses must be operational during the test.

Test Method

Voltage, current, watts, vars and/or power factor and time will be measured at various predetermined points in the power system with the plant auxiliary load held constant. The measured load will be input in the computer program which will calculate an expected voltage based on load measured. The calculated voltages will be compared to the measured voltages to confirm the accuracy of the computer model. Selection of the voltage taps of the intervening transformers is based on the optimized voltage levels at the safety-related buses for the full load and minimum load conditions, as modeled by the computer program, that are expected throughout the anticipated range of voltage variation of the offsite power source.

Acceptance Criteria

The expected voltages on the safety-related buses calculated by the computer model based on field measured load data compare favorably with the actual measured voltages on these buses.

THIS IS THE LAST PAGE OF THE TEXT SECTION 14.4.

14.5 Initial Startup Testing

[HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED]

14.5.1 Initial Fuel Loading

<u>Abstract</u>

<u>Purpose</u>

To accomplish initial fuel loading in a safe and orderly manner.

<u>Prerequisites</u>

Testing prior to initial fuel loading is completed sufficiently to demonstrate the operability of required systems and components. Temporary and permanent source range channels are operable. At least one path for boron addition to Reactor Coolant System is available. Uniform boron concentration in the Reactor Coolant System is maintained by recirculation with at least one Residual Heat Removal Pump and is sufficient to assure $K_{eff} \leq 0.95$ *during fuel loading. Containment integrity is established in accordance with station operating license.*

<u>Test Method</u>

Fuel and, where appropriate, fuel inserts are inserted into the reactor vessel in accordance with the prespecified loading sequence. Neutron count rate is monitored on temporary and permanent source range detectors. Core reactivity is monitored through plots of inverse neutron count rate ratio.

<u>Acceptance Criteria</u>

The core is assembled in accordance with the prespecified configuration.

14.5.2 Moveable Incore Detector Functional Test

<u>Abstract</u>

<u>Purpose</u>

To assure proper alignment, indexing and operation of the moveable incore detectors. To demonstrate the satisfactory response of each of the channels.

<u>Prerequisites</u>

Moveable incore detector thimbles are inserted into the core, upper internals are installed in the reactor vessel, and the reactor vessel head is installed with studs tensioned. The Reactor Coolant System is in the cold shutdown condition.

<u>Test Method</u>

The system is operated manually and automatically in all modes after setting the indexing and limit switches. The response of each channel to simulated detector outputs is verified.

<u>Acceptance Criteria</u>

A minimum number of detectors and thimbles, as defined by the plant operating license, are operable as required to perform the following functions:

1. *Determination of axial flux offset.*
2. *Determination of quadrant power tilt ratio.*

14.5.3 Incore Thermocouple Functional Test

Abstract

Purpose

To verify the continuity of thermocouple circuitry and the correct identification of thermocouple output signals.

Prerequisites

Reactor vessel upper internals are installed and the reactor vessel head is installed with studs tensioned. The Reactor Coolant System is in the cold shutdown condition. Thermocouple connections are made through the reactor vessel head.

Test Method

An identification check of each thermocouple is made and the resistance and the continuity of each is measured and recorded.

Acceptance Criteria

The results of resistance and continuity measurements satisfy design requirements and thermocouple output signals are verified and properly identified at terminal points.

14.5.4 Incore Thermocouple And RTD Cross-Calibration

Abstract

Purpose

To determine the installation correction factors for each Reactor Coolant System resistance temperature detector (RTD) and incore thermocouple.

Prerequisites

The reactor is in the shutdown condition. Incore thermocouple functional test is satisfactorily completed. All RTD's are installed. Thermocouple reference junction box RTD's and controllers are satisfactorily calibrated.

Test Method

As the unit is heated up, isothermal conditions are established at selected intervals. At these isothermal plateaus, resistance is measured and recorded for all Reactor Coolant System RTD's. Temperature is then calculated using the manufacturer's calibration sheets. The average of these temperature calculations is considered to be the actual reactor coolant temperature. The variations between individual RTD's and the average temperature are calculated for use as installation correction factors. Temperature readings for each incore thermocouple are also recorded at each plateau to generate individual isothermal correction factors.

Acceptance Criteria

Response characteristics for each RTD are consistent with vendor calibration data. Isothermal correction factors are obtained which, when applied to the applicable incore thermocouple, give thermocouple output consistent with RTD data.

14.5.5 Rod Position Indication Check

Abstract

Purpose

To verify that the Digital Rod Position Indication System satisfactorily performs the required indication and alarm functions for each individual rod under hot shutdown conditions, and to demonstrate that all full length rods operate satisfactorily over their entire range of travel.

Prerequisites

The reactor is at hot shutdown, no-load operating temperature and pressure with at least one reactor coolant pump running. All full length Rod Control System equipment has been installed and all preliminary testing and calibrations have been complete. Preliminary tests on the Digital Rod Position Indication System must be completed. Pulse-to-Analog converters must have been aligned. Plant source range channels shall be in operation and monitored at all times when rods are being moved.

Test Method

Each full length rod cluster control assembly is pulled to its fully withdrawn position in increments. Indication and alarms are observed for proper operation.

Acceptance Criteria

Satisfactory performance of the rod position indication system for each rod cluster control assembly over its entire length of travel.

14.5.6 Rod Cluster Control Assembly Drop Time Test

Abstract

Purpose

To verify the drop time for each full-length rod cluster control assembly under no-flow and full-flow conditions, with the reactor in the cold shutdown condition and at normal operating temperature and pressure.

Prerequisites

Initial fuel loading is completed and the unit is in the cold or hot condition with full-flow or no-flow as required for the particular phase of the test to be performed. Containment integrity has been established in accordance with applicable Technical Specifications. All rod cluster control assembly drive mechanisms have had the preliminary checkout completed. Checkout and preliminary adjustment of the rod position indicators has been completed.

Test Method

Each rod cluster control assembly for each unit condition is individually withdrawn, then the drop time is determined by monitoring the rod position indication signal following de-energization of the stationary winding of the rod cluster control assembly drive mechanism.

Acceptance Criteria

The time from release of the rod cluster control assembly until it reaches the top of the dashpot is less than 2.2 seconds (Unit 1) and 3.3 seconds (Unit 2) in the hot, full-flow condition.

14.5.7 Rod Control System Alignment Test

Abstract

Purpose

In the cold shutdown condition, to assure proper connection, identification and continuity of Rod Control System power and control cabling. In the hot shutdown condition, to adjust Rod Control System bank-overlap setpoints and to demonstrate proper system control and indication.

Prerequisites

The reactor vessel upper internals are installed, the reactor vessel head is installed with the studs tensioned, the full-length rods are latched, and the Reactor Coolant System is filled and vented. The reactor is in the cold or hot shutdown condition as dictated by the specific test requirements. Containment integrity is established in accordance with applicable Technical Specifications.

Test Method

With the reactor in the cold shutdown condition, the connection and identification of each power and control cable are visually checked and the resistance of each measured. With the reactor in the hot shutdown condition, the Rod Control System is operated in various modes and indications and alarms observed. Bank start and stop positions during insertion, withdrawal and overlap operations are recorded. Setpoint adjustments are made as required.

Acceptance Criteria

Cold resistance and identification checks are satisfactory. Bank overlap setpoints are in accordance with those specified and step counters perform properly. The Rod Control System performs in accordance with design requirements.

14.5.8 Full-Length Rod Drive Mechanism Timing Test

Abstract

Purpose

To demonstrate proper operation and timing of each rod drive mechanism.

Prerequisites

The reactor vessel upper internals are installed, the reactor vessel head is installed with studs tensioned, each full-length rod is latched and the Reactor Coolant System is filled and vented. The reactor is in the cold or hot shutdown condition, as dictated by the test requirements. Containment integrity is established in accordance with applicable Technical Specifications. Cold condition Rod Control System alignment has been completed.

Test Method

With the reactor in the cold shutdown condition, the timing for each slave cycler is set, measured and reset as necessary. Each full-length rod drive mechanism is manually operated with a rod cluster control assembly attached, checking the latching and releasing features of each. The test is repeated for each rod drive mechanism with the reactor in the hot shutdown condition.

Acceptance Criteria

The final settings for each slave cycler with the reactor in the hot shutdown condition are in accordance with the rod drive mechanism design requirements. The free latching and releasing of each rod drive mechanism is verified under both cold and hot conditions.

14.5.9 Reactor Coolant System Flow Test

Abstract

Purpose

To verify predicted Reactor Coolant System flow rates at normal no-load operating temperature and pressure. To align the Reactor Coolant system flow instruments.

Prerequisites

The reactor is in the hot standby condition with all rod cluster control assemblies at their fully inserted position. All four reactor coolant pumps are operable. Pressure damping devices are installed in the elbow tap differential pressure cell sensing lines.

Test Method

Unit 1:

In each reactor coolant flow configuration, pump power, pump rotational speed and loop elbow differential pressure are measured and recorded. These measurements are compared with design pump performance curves to establish the point at which the pump is operating. From this information, in-service pump characteristic flow curves for each configuration are established. The flow transmitters are adjusted for 100 percent flow at normal operating conditions and zero output at zero flow.

Unit 2:

With the Reactor Coolant System at pressure and temperature, loop elbow differential pressure, temperature, and pressure are measured and recorded. From these measured parameters Reactor Coolant System flow is calculated. This test reflects the latest Westinghouse method for measurement of Reactor Coolant System flow.

Acceptance Criteria

Reactor coolant system flow is greater than the thermal design flow and less than the mechanical design flow as stated in Chapter 5. Flowrate for any loop as compared to the average flowrate of loops under the same conditions is within 10 percent. Flow transmitters yield proper output at full flow and no flow conditions.

14.5.10 Reactor Coolant System Flow Coastdown Test

Abstract

Purpose

To measure the rate at which reactor coolant flow rate decreases, subsequent to reactor coolant pump trips, from various flow configurations. To measure various delay times associated with assumptions made in the analysis of the loss of flow accident.

Prerequisites

The reactor is in the hot standby condition with all rod cluster control assemblies at their fully inserted position, all four reactor coolant pumps are operating. The Reactor Coolant System Flow Test has been completed with instrumentation calibrated accordingly, and pressure damping devices installed for the flow test have been removed.

Test Method

The control rod with the slowest drop time is withdrawn, while maintaining the hot standby condition. One reactor coolant pump tripped, measuring and recording the time from breaker opening to first rod motion. A second reactor coolant pump is tripped and the first reactor coolant pump restarted. The three running reactor coolant pumps are then tripped *simultaneously. All reactor coolant pumps are restarted and then tripped simultaneously. On a high-speed strip chart recorder, for each transient, one elbow tap differential pressure cell for each loop, each reactor coolant pump breaker position, rod position indication signal for one rod and reactor trip breaker position are measured and recorded.*

Acceptance Criteria

The core flow decrease for the first ten seconds of the transient is, in each case, closer than that assumed in the FSAR. Time delays from actuation to low flow trip, undervoltage trip and underfrequency trip actuation are less than or equal to those assumed in the safety analysis section of the FSAR.

14.5.11 RTD Bypass Flow Verification

Abstract

Purpose

To determine the flowrate necessary to achieve the required reactor coolant transport time in each RTD bypass loop, to verify that the coolant transport times are acceptable and to verify the low flow alarm setpoint and reset for the total RTD bypass flow in each reactor coolant loop.

Prerequisites

The reactor is in the hot shutdown condition with all reactor coolant pumps running. All RTD bypass loop flow measurement channels are calibrated and in service.

Test Method

The flow required to achieve the required reactor coolant transport time is determined by accurately measuring and recording the lengths of installed piping from the bypass loop inlet connections on each reactor coolant loop to the last downstream RTD of both the cold and hot leg bypass loops, and then calculating the flow necessary to achieve less than 1.0 second transport time. Total bypass flowrate for each reactor coolant loop is measured and recorded, then the actual bypass loop transport time is calculated. The low flow alarm setpoint is verified by sequentially throttling the hot and cold leg manifold isolation valves in each loop and noting the flow when the alarm point is reached.

Acceptance Criteria

The RTD bypass loop transport time is less than 1.0 second or, if greater, is noted for comparison with results from unit trip testing at 100 percent power. Measured bypass flowrates show no major deviation between loops in the full flow configuration. The low flow alarm actuates at 90 percent of full bypass loop flow.

14.5.12 Reactor Protection System Trip Circuits Test

Abstract

Purpose

To verify that initial trip setpoint adjustment have been made prior to initial unit startup and to specify which trip setpoint adjustments will require readjustment during startup. To obtain a record of all trip setpoints. To verify proper operation of trip circuitry.

Prerequisites

Reactor trip instrumentation has been aligned and calibrated with setpoints adjusted to values given in the Technical Specifications or the unit test documents. Reactor trip instrumentation *has been energized for a time sufficient to achieve stability. Containment integrity is established in accordance with applicable Technical Specifications.*

Test Method

Inputs for each automatic and manual trip are simulated and proper trip response into the protection logic is noted. All combinations of logic are simulated and proper response noted. During startup and test operations, specific setpoints noted for readjustment on the data sheets are readjusted and final setpoint values recorded.

Acceptance Criteria

Initial reactor trip setpoints are verified to be within design criteria and in conformance with or more conservative than values in the Technical Specifications. Setpoints readjusted during startup and testing are noted and a final record of all setpoints is obtained.

14.5.13 Initial Criticality

Abstract

Purpose

To achieve initial reactor criticality.

Prerequisites

The boron concentration in the Reactor Coolant System is verified to be within ±50 ppm of that which existed at the termination of initial fuel loading. The Reactor Coolant System is at hot no-load pressure and temperature with a steam bubble in the pressurizer. All full-length rod cluster control assembly banks are fully inserted. All source and intermediate range channels are operable.

Test Method

After establishing baseline count rates, the shutdown and control banks are withdrawn in normal sequence until Bank D is to the point that it has approximately 0.1% $\Delta k/k$ worth remaining. Reactor Coolant System boron dilution is commenced at a rate of approximately 1% $\Delta k/k$ per hour and Reactor Coolant System boron concentration is sampled at fifteen minute intervals. When the inverse neutron count rate ratio is approximately 0.05, dilution is terminated and the Reactor Coolant System is allowed to mix. If criticality is not achieved during mixing, the withdrawal of Bank D will be commenced at approximately fifteen step intervals. If criticality is not achieved when Bank D reaches 228 steps, Bank D is reinserted to its original position and dilution is commenced at approximately 0.3% $\Delta k/k$ per hour until criticality is achieved. When criticality is achieved, the startup rate is limited to one decade per minute. During rod withdrawal, a plot of inverse neutron count rate ratio versus bank position is maintained and, during dilution, plots of inverse neutron count rate ratio versus time and inverse neutron count rate ratio versus charging water integrator values are maintained.

Steady-state reactor conditions are achieved at hot zero power conditions.

Acceptance Criteria

The reactor achieves a critical configuration in an orderly and safe manner.

14.5.14 Zero Power Physics Test

Abstract

Purpose

To verify the basic nuclear characteristics of the reactor core through the following measurements:

1. *Nuclear instrumentation overlap verification.*
2. *Onset of nuclear heat.*
3. *All rods out critical boron concentration.*
4. *Isothermal temperature coefficient.*
5. *Differential and integral worth of the sequenced control banks.*
6. *Differential boron worth at hot zero power.*
7. *Integral control rod worth with one stuck rod.*
8. *Ejected rod cluster control assembly worth at hot zero power. (Unit 1 only)*

Prerequisites

The Reactor Coolant System is in the hot zero power condition with the reactor critical with the neutron flux level in the source range as established in the initial criticality sequence. Reactor Coolant System temperature is being maintained. Required signals for data collection and recording are available.

Test Method

1. *The neutron flux level will be increased by outward control rod motion and the nuclear instrumentation overlap recorded. Adjustments will be made as necessary to insure minimum overlap as described in the Technical Specifications.*

2. *The neutron flux level will be increased by outward control rod motion until temperature feedback effects are noted. The upper limit for zero power physics testing is defined as approximately one decade below this level.*

3. *The all rods out, critical boron concentration is determined by measuring the just critical boron concentration with Bank D near the fully withdrawn position. The amount of reactivity held down by Bank D is then dynamically determined by withdrawal of Bank D, noting the amount of reactivity inserted and converting this value to an equivalent amount of boron.*

4. *The isothermal temperature coefficient for various boron concentrations is obtained by dynamically measuring the reactivity change due to a temperature change in the primary system.*

5. *The sequenced bank differential rod worth is determined by either borating the Reactor Coolant System while withdrawing the control banks or by diluting the Reactor Coolant System while inserting the control banks to maintain nominal system criticality. Integral worth is then determined from the differential reactivity data.*

6. *Differential boron worth at hot zero power is determined by obtaining and analyzing reactor coolant samples for boron content in conjunction with control bank movement to maintain nominal criticality during dilution/boration. Boron concentration as a function of time in combination with integrated reactivity as a function of time is used to plot reactivity versus boron concentration, the slope of which yields differential boron worth.*

7. *Integral control rod worth with one stuck rod is measured by achieving a configuration in which all banks are fully inserted except the most reactive rod cluster control assembly. Incremental rod worth measurements are made as the banks are inserted during boron dilution. Integral control rod worth is the sum of the incremental reactivity measurements made in obtaining this configuration.*

8. *Ejected rod cluster control assembly worth at hot zero power is determined by obtaining a critical configuration with the sequenced rod banks at their insertion limit as defined in the Technical Specifications. The most reactive inserted rod is withdrawn to maintain nominal criticality during boration. The reactivity addition is determined by summing the differential reactivity insertions as the rod is withdrawn to its out limit.*

Acceptance Criteria

1. *Nuclear instrumentation overlap meets the minimum requirements as defined in the Technical Specifications.*
2. *The onset of nuclear heat occurs within one decade of the predicted value.*
3. *The all rods out, critical boron concentration is within ±50 ppm of design prediction.*
4. *The moderator temperature coefficient is negative under the observed conditions of critical power operation.*
5. *Differential boron worth, over the range measured, is within ±10% of the predicted value.*
6. *Control rod worth measurements verify that the insertion limits defined in the Technical Specifications provide a 1.6% $\Delta k/k$ shutdown margin under hot shutdown conditions with the most reactive rod cluster control assembly stuck in the withdrawn position.*
7. *The worth of an ejected rod cluster control assembly at hot zero power is less than or equal to the value used in the safety analysis.*

14.5.15 Rod Control System AT-Power Test

Abstract

Purpose

To verify the performance of the Rod Control System.

Prerequisites

The reactor is at an equilibrium condition at not less than 30 percent power. The Rod Control System is in manual control and rod cluster control assemblies are in the maneuvering band for the existing power level.

Test Method

Signals from parameters affecting automatic reactor control are connected to recorders. Recorder traces are compared to control board indications to assure correspondence. With the average reactor coolant temperature within ±2°F of the reference reactor coolant temperature, the Rod Control System is placed in automatic. System response is observed during a period sufficient to assure proper control during steady state conditions. The Rod Control System is placed in manual and the average reactor coolant temperature is elevated to 6°F greater than the reference reactor coolant temperature. The Rod Control System is returned to automatic and system response is observed and recorded. With the average reactor coolant temperature initially 6°F lower than the reference reactor coolant temperature, the test is repeated. Setpoints are adjusted as necessary and the test repeated.

Acceptance Criteria

With final setpoints, the Rod Control System maintains acceptable stability under steady state conditions and no unacceptable overshoot occurs during induced transients.

14.5.16 Pressurizer Pressure And Level Control System Test

Abstract

Purpose

To demonstrate that the pressurizer controllers are operative and to verify pressurizer pressure and level setpoints.

Prerequisites

The reactor is at an equilibrium condition at approximately 30 percent power. The pressurizer safety and relief valves are operable. Pressurizer level and pressure instrumentation has been tested and calibrated and is operable.

Test Method

The Pressurizer Pressure and Level Control System is operated in various modes. After demonstrating satisfactory parameter control in manual, small perturbations are initiated in pressure and level, then automatic control is selected and system response measured and recorded. Setpoints are adjusted as required and tests repeated where setpoint adjustments were made.

Acceptance Criteria

With final setpoints, the controllers respond to induced transients in pressurizer pressure and level in accordance with setpoint documents.

14.5.17 Core Power Distribution Test

Abstract

Purpose

To obtain and analyze core power distributions for various control rod configurations.

Prerequisites

Reactor is critical at a steady state power level as specified by procedure (5%, 25%, 50%, 75%, 100%). Incore instrumentation system functional test is complete and the system is operable. Computer systems are operable as necessary for incore map processing.

Test Method

Reactor power level is stabilized and complete incore flux maps are obtained and processed.

Acceptance Criteria

Core peaking factors are consistent with those predicted in the core design report.

14.5.18 Unit Load Steady State Test

Abstract

Purpose

To measure NSSS steady parameters as a function of power to compare with design predictions, and equipment and system limits.

Prerequisites

The unit is at a steady state power level and reactor coolant pump combination as specified in the procedure. Specified parameters are available to be recorded.

Test Method

With stable conditions established, applicable parameters are recorded and averaged over the specified time period. Averaged values are compared to design predictions and adjustments are made as indicated.

Acceptance Criteria

NSSS exhibits stable operation; no oscillatory or unusual behavior is detected. Recorded parameters fall within MIN/MAX values as specified.

14.5.19 Radiation Shielding Survey

Abstract

Purpose

To measure radiation dose levels at preselected points throughout the station to verify shielding effectiveness.

Prerequisites

Radiation survey instruments to be used are calibrated against known sources. The reactor is critical at various power levels from zero to 100 percent, as specified by the test procedure.

Test Method

In accordance with procedures for radiation surveys, dose levels are measured at points throughout the station. At specified reactor power levels, measurements are repeated.

Acceptance Criteria

Measured radiation levels are within the limits for the zone designation of each area surveyed.

14.5.20 Nuclear Instrumentation Initial Calibration

Abstract

Purpose

1. *To determine the linearity and uniformity of power range detector output.*
2. *To calibrate the power range channels to reflect actual power levels.*
3. *To obtain Nuclear Instrumentation System channel overlap data.*

Prerequisites

The reactor is at the power level specified by the test procedure and in a stable condition. Precritical nuclear and temperature instrumentation calibration has been successfully completed.

Test Method

The tests described below are repeated at various power levels, as required by the test procedure:

1. *Acceptable power range output is determined by measuring and plotting power range detector currents versus power level. From these plots, the linearity of each power range channel and the degree of uniformity between power range channels are determined.*
2. *The gain of each power range channel is adjusted to correspond to the results of heat balance calculations.*
3. *Intermediate and power range channel outputs during power level changes are measured, recorded and plotted to establish channel overlap.*

Acceptance Criteria

1. *Power range detectors display linear output over the range of normal power operation. Power range channels, after gain settings from thermal power measurement, give uniform power level indication.*
2. *The power range channel gains accurately reflect actual power levels.*
3. *Consistent overlap data are obtained on power level changes.*

14.5.21 Effluent Radiation Monitor Test

Abstract

Purpose

To verify the performance of the effluent monitors under actual discharge conditions.

Prerequisites

The reactor has been operating for a time sufficient to generate representative effluents. The effluent monitors have been checked against known sources.

Test Method

Following standard procedures, the suitability of effluents for discharge is verified by radiochemical analysis. Discharge is commenced and the response of effluent monitors is observed and recorded. Effluent is sampled in accordance with established procedures and effluent monitor performance is verified through radiochemical analysis.

Acceptance Criteria

The installed effluent monitors perform in accordance with design standards and properly indicate the radioactive content of the effluent.

14.5.22 Doppler Only Power Coefficient Verification (Unit 1 only)

Abstract

Purpose

To verify the nuclear design predictions of the Doppler-only power coefficient.

Prerequisites

The reactor is at the required power level with rods in the specified maneuvering band. The instrumentation necessary for collection of data is installed, calibrated and operable.

Test Method

At the required power levels, turbine load is cycled and various system parameters are measured. Using these parameters a Doppler coefficient verification factor is calculated and *compared to a predicted doppler coefficient verification factor obtained from core design parameters.*

Acceptance Criteria

The predicted and calculated Doppler coefficient verification factors are within acceptable limits of each other. This verifies that the Doppler-only power coefficient assumed in the FSAR accident analysis is conservative.

14.5.23 Incore And Nuclear Instrumentation Systems Detector Correlation

Abstract

Purpose

To determine power distribution and power range detector response. To form a relationship between incore and nuclear instrumentation generated axial offsets and f(Δ1) functions.

Prerequisites

The unit is stable at the power level specified in the test procedure. The controlling bank is near the all-rods-out configuration. The Incore Instrumentation System is tested and operable.

Test Method

To form the incore nuclear instrumentation axial offset relationship, the controlling bank is inserted, compensating with boron dilution/addition as required. Steady state negative axial offset values are generated by this partial insertion of the controlling bank. Positive axial offsets are then generated by xenon redistribution following controlling bank withdrawal. The required incore/excore correlation data are obtained during the transient. The f(Δ1) functions are generated by utilizing the relationship between the calculated incore distribution and the full power, power range detector currents extrapolated from the detector current versus core thermal power relationship.

Acceptance Criteria

The f(Δ1) function for overpower and overtemperature differential temperature setpoints is set in accordance with Technical Specification limits, based on data obtained.

14.5.24 Below-Bank Rod Test

Abstract

Purpose

To obtain the differential and integral worth of the most reactive below-bank rod cluster control assembly. To demonstrate the response of the nuclear and incore instrumentation to a rod cluster control assembly below the nominal bank position and to determine hot channel factors associated with this misalignment.

Prerequisites

All power range nuclear instrumentation channels are operable. The moveable incore detectors are operable. Power escalation testing is completed to approximately the 50 percent reactor power level.

Test Method

Single rod movement is accomplished by disconnecting the lift coils of all rods in the affected bank except the selected rod. The differential worth of the rod cluster control assembly is determined by making a series of stepwise adjustments in rod position to maintain nominal system criticality during a continuous, controlled Reactor Coolant System dilution. The flux level response to the step change in reactivity is translated to equivalent reactivity. Differential and integral worths are calculated from this reactivity. During rod cluster control assembly insertion, power range detector currents, thermocouple maps and moveable incore detector traces are periodically recorded. The power range detector data provides information to relate core quadrant tilt to rod cluster control assembly position. The thermocouple maps, in conjunction with the moveable incore detector traces, provide data necessary to determine hot channel factors and core axial and radial power distributions as a function of rod cluster control assembly position.

Acceptance Criteria

Hot channel factors are within FSAR design safety limits. No significant radial or axial power maldistribution exists with the rod cluster control assembly less than or equal to fifteen inches below its bank. Incore and/or nuclear instrumentation is demonstrated to detect any significant power maldistribution caused by the misaligned rod cluster control assembly.

14.5.25 Pseudo Rod Ejection Test (Unit 1 only)

Abstract

Purpose

To verify ejected rod worth and hot channel factors assumed in the safety analysis. To demonstrate the response of nuclear and incore instrumentation to a rod cluster control assembly above the nominal bank position and to an ejected rod.

Prerequisites

All power range nuclear instrumentation channels are functional. The moveable incore detectors are operable. Power escalation testing is completed to approximately the 50 percent reactor power level. Reactor is at steady state power with the controlling bank at the full power insertion limit.

Test Method

Single rod movement is accomplished by disconnecting the lift coils of all rods in the affected bank except the selected rod. The differential worth of the rod cluster control assembly is determined by making a series of stepwise adjustments in rod position. Nominal system criticality is maintained via the negative reactivity feedback of the moderator temperature coefficient. The average temperature response to the step change in rod position is translated to an equivalent reactivity increment. Differential worth is defined as the change in reactivity per unit change in rod cluster control assembly position about an average rod cluster control assembly position between the endpoints of the step change. Integral rod cluster control assembly worth is determined from the differential reactivity data. During the rod cluster control assembly withdrawal, periodic power range detector currents, thermocouple maps and moveable incore detector traces are recorded. The power range detector data provides information to relate core quadrant tilt to rod cluster control assembly position. The thermocouple maps, in conjunction with the moveable incore detector traces, provide the data necessary to determine hot channel factors and core axial and radial power distributions as a function of rod cluster control assembly position.

Acceptance Criteria

The worth of the ejected rod and the hot channel factors, with measurement uncertainty, are less than or equal to those assumed in the safety analysis. No significant radial or axial power maldistribution exists with the rod cluster control assembly less than or equal to fifteen inches above its bank. Incore and/or nuclear instrumentation is demonstrated to detect any significant power maldistribution caused by the misaligned rod cluster control assembly.

14.5.26 Unit Load Transient Test

Abstract

Purpose

To demonstrate satisfactory unit response to a 10 percent load change.

Prerequisites

The various control systems have been tested and are in automatic. All pressurizer and main steam relief and safety valves are operable. The control rods are in the maneuvering band for the power level existing at the commencement of the test. Unit conditions are stabilized and all pertinent parameters to be measured are connected to high speed recorders or to the transient monitors.

Test Method

Output is manually reduced at a rate sufficient to simulate a step load change equivalent to approximately a 10 percent load decrease. After stabilization of all systems, output is manually increased at a rate sufficient to simulate a step load change equivalent to approximately a 10 percent load increase. Pertinent parameters affected by a load change are measured and recorded. At various power levels, as required by the test procedure, the test is repeated.

Acceptance Criteria

Neither the turbine nor the reactor trips, and no initiation of safety injection is experienced. No pressurizer or main steam relief or safety valves lift. No operation action is required to restore conditions to steady state. Parameters affected by the load change do not incur sustained or divergent oscillations.

14.5.27 Dynamic Rod Drop Test (Unit 1 only)

Abstract

Purpose

To demonstrate the operation of the negative rate trip circuitry in detecting the simultaneous insertion of two rod cluster control assemblies, provided that their summed rod worth equals or exceeds 400 PCM. For summed rod worths of less than 400 PCM, the test records data for the dynamic reactor response to the simultaneous drop of two rods into the core.

Prerequisites

All power range nuclear instrumentation channels are operable. The reactor is at the steady state power level specified in the procedure wit the controlling bank near the full power insertion limit. Pertinent parameters to be measured are connected to recording devices.

Test Method

Two of the most reactive rods from the group most difficult to detect by excore detectors due to low worth and core location are simultaneously dropped by removing voltage to both the *moveable and stationary gripper coils of the designated rods. Following the transient, recorded data is evaluated for system and instrumentation response.*

Acceptance Criteria

The reactor trips as a result of the negative rate trip, provided the summed rod worth is equal to or greater than 400 PCM. If the summed rod worth is less than 400 PCM, accurate recording of the dynamic reactor response is required for later analysis. Steam Generator and pressurizer safety valves do not lift. Safety injection is not initiated.

14.5.28 Unit Loss Of Electrical Load Test

Abstract

Purpose

To demonstrate the ability of the unit to sustain a net load loss at elevated power. To evaluate the interaction between control systems and to evaluate system responses to the transient. To verify the proper response of the Steam Dump Control System.

Prerequisites

The various control systems are in the automatic mode and functioning properly. The reactor is at steady state full power with the rods in the maneuvering band. Pressurizer and main steam safety and relief valves are operable. Pertinent parameters to be measured are connected to recording devices.

Test Method

Both main generator breakers are manually placed in the tripped position to simulate a loss of generator load. Pertinent parameters are recorded on recording devices. Following the transient, recorded data is evaluated for system and controller response and possible abnormalities.

Acceptance Criteria

Safety Injection is not initiated. Pressurizer safety valves do not lift. The steam generator power operated relief valves (PORV) may lift in this test. Pressurizer power-operated relief valves do not lift (48% power). No operator action is required until reactor power is less than approximately 20 percent. No safety limits are exceeded.

14.5.29 Turbine Trip Test

Abstract (Unit 1 only)

Purpose

To demonstrate the ability of the unit to sustain a trip of the main turbine generator at elevated power. To evaluate the interaction between control systems and to evaluate system responses to the transient.

Prerequisites

The various control systems are in the automatic mode and functioning properly. The reactor is at steady state full power with the rods in the maneuvering band. Pressurizer and main steam safety and relief valves are operable. Pertinent parameters to be measured are connected to recording devices.

Test Method

The main turbine generator is tripped. Pertinent parameters are recorded on recording devices. Following the transient, recorded data is evaluated for system and controller response and possible abnormalities.

Acceptance Criteria

Safety injection is not initiated. Main steam and pressurizer safety valves do not lift. System instrumentation responds as predicted. No safety limits are exceeded.

Note: *For Unit 2, Unit Loss of Electrical Load Test satisfied this Turbine Trip Test.*

14.5.30 Loss Of Offsite Power Test

Abstract

Purpose

To demonstrate the ability of the turbine-generator to sustain an isolation of the offsite power distribution system and to subsequently act as the onsite power source. To evaluate the interaction between control systems and to evaluate system responses to the transient.

Prerequisites

The various control systems are functioning properly. The unit is at a steady state power level greater than 10% of rated generator load. Pressurizer and main steam safety and relief valves are operable. Pertinent parameters to be measured are connected to recording devices.

Test Method

Switchyard circuit breakers connecting the unit to the offsite power distribution system are manually placed in the tripped position. Pertinent parameters are recorded on recording devices. Following the transient, recorded data is evaluated for system and controller response and possible abnormalities.

Acceptance Criteria

Station auxiliaries are maintained with the main generator as the power source.

14.5.31 Shutdown From Outside Control Room Test

Abstract

Purpose

To demonstrate the capability to shutdown the unit from outside the control room.

Prerequisites

Various control systems are functioning properly. Unit generator output ≥10%.

Test Method

Evacuation of the main control room is simulated by dispatching personnel to their assigned stations while additional operators occupy the control room to observe unit behavior. The reactor is tripped at the local reactor trip switchgear. The unit is maintained in the hot standby condition by manipulation of local controls and observation of local indications.

The unit is then cooled down approximately 20°F by local operation of steam dump valves. The capability to cooldown to cold shutdown conditions from outside the control room is then demonstrated by a simulation of the remainder of the procedure.

Acceptance Criteria

The reactor trips. The turbine generator trips. A stable hot standby condition is established and maintained. The capability to cooldown to cold shutdown is demonstrated.

14.5.32 Steam Generator Water Hammer Test (Unit 1) Abstract

Abstract

Purpose

To verify that the Feedwater Bypass System prevents any bubble collapse pressure pulses from occurring which could damage the steam generator preheater during feedwater flow switchover.

Prerequisites

Reactor power at less than or equal to 30%. Feedwater supply to the steam generator is through the Feedwater Bypass System with the feedwater isolation valve in the main feedwater line shut. Special instrumentation shown in Figure 14-1, Figure 14-2 and Figure 14-3 is installed in one steam generator and associated feedline. Provisions are made for recording the following process signals from the same loop as the instrumented steam generator during the test:

Feedwater Flow
Steam Flow
Feedwater Header Temperature
Feedwater Temperature (at steam generator main feed inlet)
Primary Average Temperature
Primary Temperature Difference
Steam Pressure
Steam Generator Water Level
NRC Licensing personnel are notified 24 hours prior to performing the test.

Test Method

Feedwater temperature is lowered to approximately 250°F. Feedwater flow is then switched from the auxiliary feedwater nozzle to the main feedwater nozzle while maintaining reactor power constant. Signals from the special instrumentation and the process instrumentation identified above are recorded during the transient.

Acceptance Criteria

Any pressure pulses recorded during the test by the transducers mounted in the preheater or in the feedline shall be less than 50 psi in magnitude.

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