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14.0 Initial Tests and Operation

The initial test program for structures, systems and components for the Catawba Nuclear Station is described in detail in this chapter. This program begins when construction is sufficiently complete to allow testing and/or operation of the individual structures, systems or components and extends through the satisfactory performance of each unit's acceptance tests at or near full power. The major purpose of this test program is to verify the functional capabilities of structures, systems and components to meet their design objectives in such a manner that the safety of the unit will not be dependent on untested structures, systems or components. Additionally, this test program will allow the demonstration of adequacy of station operating and emergency procedures to the extent practicable and will further familiarize the operating and technical staff in the technical aspects of station design and operation. Adequate numbers of qualified personnel using administrative procedures similar to those required during actual unit operation will assure a successful and timely station initial test program.

The objectives of, and methods for achieving, an acceptable initial testing program are in general agreement with the objectives and methods outlined in the following:

1. Regulatory Guide 1.68, Rev. 2 "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors".
2. Title 10, Code of Federal Regulations, Part 50, Appendix B, Criterion XI "Test Controls".
3. Standard ANSI N 18.7-1976 "Administrative Controls for Nuclear Power".

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14.1 Test Program (PSAR)

Section 14.1 is Historical Information – Not Required to be Revised

Information for this section was submitted in the Catawba Nuclear Station PSAR.

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14.2 Test Program (FSAR)

Sectional 14.2 Is Historical Information – Not Required to be Revised

14.2.1 Summary of Test Program and Objectives

The general objectives of the initial test program at Catawba Nuclear Station is to provide assurance that:

1. *The station has been adequately designed and constructed.*
2. *All contractual, regulatory and licensing requirements are satisfied.*
3. *The station will not adversely affect the public health and safety.*
4. *The station can be operated in a reliable, dependable manner so as to perform its intended function.*
5. *Operating and emergency procedures are appropriate to the extent practicable.*
6. *Personnel have acquired an appropriate level of technical expertise.*

The initial test program at Catawba Nuclear Station is divided into two major portions. The first phase of testing is the preoperational test phase and includes all hot and cold functional testing required prior to fuel loading. The second phase of testing is the initial startup testing phase and includes initial fuel loading and all subsequent testing through the completion of power escalation testing.

Preoperational tests are performed following completion of construction flushing and hydrostatic testing, system turnover and initial calibration of required instrumentation. The major objective of preoperational testing is to verify that structures, systems and components essential to the safe operation of the plant are capable of performing their intended function. Summaries of these individual preoperational tests are provided in Section 14.2.12.

Preoperational testing for satisfying FSAR testing commitments will be completed prior to fuel loading. Tests currently identified which have portions of the test which may be completed following fuel loading are Section 14.4.32, Section 14.4.6, Section 14.4.25, and Section 14.4.5. These tests will be completed prior to initial criticality. Tests currently identified which have portions of the test which may be completed during power escalation testing are Section 14.4.3, Section 14.4.35, Section 14.4.2, Section 14.4.9, and Section 14.4.10.

Other preoperational tests which are not required prior to fuel loading and which are not safety related, such as Administrative Building Ventilation Tests, may be completed following fuel loading. Tests (or portions of tests), for which abstracts are provided, which do not satisfy any regulatory requirement and which are not required by regulatory guides are identified in Section 14.4.

For systems and components which are not nuclear safety related, acceptance criteria will be only to assure reliable and efficient operation of the system.

Initial startup testing will be performed beginning with fuel loading and ending with commercial operation. The purpose of initial startup testing is to assure the safe and orderly loading of fuel, to verify core physics and thermal and hydraulics parameters assumed in the Catawba Safety Analysis, and to demonstrate that the plant is capable of withstanding anticipated transients. Initial startup tests including fuel loading, zero power testing, and power escalation are described in the test summaries provided in Section 14.2.12.

All system features relied on for mitigation or protection for Anticipated Transients Without Scram (ATWS) events as analyzed in Section 15.8 will be verified as a part of the preoperational or startup test programs.

14.2.2 Organization and Staffing

The Nuclear Production Department of Duke Power Company has responsibility for the development and conduct of the initial startup testing program for Catawba Nuclear Station. The Station Manager is ultimately responsible for the development of test procedures, the conduct of testing, and the evaluation of test results. The Nuclear Production Department General Office Staff, other departments within Duke, Westinghouse Electric Corporation, and other vendors provide support and technical assistance to the station staff during the initial testing program, as required. The station staff is responsible for and performs all testing.

Within the Catawba Nuclear Station organization, the Operating Superintendent, Technical Services Superintendent and Maintenance Superintendent are responsible to the Station Manager for the development and conduct of testing assigned as well as for final approval of test results.

Coordination of the overall startup program including scheduling of system turnover from Construction Department to Nuclear Production Department, instrumentation calibration and functional testing is the responsibility of an overall Unit Coordinator who reports to the Station Manager.

Identification and assignment of testing responsibilities, development of administrative procedures to control testing activities, and maintenance of the status of the preoperational and startup testing programs is the responsibility of the Performance Engineer who reports to the Superintendent of Technical Services.

Test Coordinators are assigned to each functional test and shall be responsible for supervising the conduct of the test assigned to them, resolution of discrepancies which arise during the conduct of testing and initial evaluation of test results. Each Test Coordinator shall have a BS in engineering or the physical sciences. In lieu of this requirement, each Test Coordinator shall have a minimum of two years nuclear power plant experience.

Each Test Coordinator reports functionally to his appropriate station section head. Qualifications of individuals in station management responsible for the review and control of test procedures within their jurisdiction are described in Section 13.1.3.

Staffing the station to accomplish the initial startup testing program began more than four years prior to fuel loading. Sufficient personnel will be available by augmenting the normal station staff as required during the startup.

14.2.3 Test Procedures

14.2.3.1 Preparation of Procedures

All preoperational and initial startup tests will be performed in accordance with written approved test procedures. These procedures are developed primarily by degreed Junior, Assistant, and Associate Engineers although other members of the station supervisory staff may be assigned responsibility for test procedure preparation. Insofar as practicable, individuals assigned responsibility for development of test procedures are the same individuals who will be responsible for the conduct of those tests as a test coordinator. 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, the Final Safety Analysis Report, draft Technical Specifications as well as consulatory and advisory services to assist the members of the station staff in developing test procedures and establishing acceptance criteria.

14.2.3.2 Review and Approval of Test Procedures

All preoperational and startup test procedures are subject to appropriate review and approval prior to use. Various individuals or organizations external to the station are afforded the opportunity for review as deemed appropriate. Westinghouse Electric Corporation is given the opportunity to review all test procedures pertaining to their scope of supply. Comments received during such review are given due consideration.

Westinghouse has reviewed Catawba Unit 1 power ascension test procedures and has found them to be consistent with the standard Westinghouse recommended procedures. Since the same basic procedures will be utilized to develop the Unit 2 procedures, a separate review of the Unit 2 procedures by Westinghouse will not be performed.

Prior to use, each procedure is reviewed by a qualified reviewer at the station. Qualified reviewers are designated by the station manager. Each qualified reviewer has a minimum of five years of technical experience, one year of which must be nuclear. A maximum of four years of technical experience may be satisfied by academic or related technical training. This review includes the determination of whether or not additional, cross-disciplinary review is necessary. If deemed necessary, such additional review is performed by the appropriate qualified reviewer(s).

Each procedure is approved prior to use by the Station Manager; or by the Operations, Maintenance or Technical Services Superintendents as previously designated by the Station Manager. Approved safety-related test procedures which satisfy FSAR testing commitments will be made available for review 60 days prior to their intended use.

14.2.3.3 Changes to Procedures

Changes to procedures are classified as two types: minor and major. A minor change is a change to an approved procedure which corrects errors in the applicable approved procedure of a typographical or editorial nature. A major change is any change to an approved procedure determined not to be a minor change.

A minor change may be made by an individual with no special reviews or approvals. Minor changes, by definition, cannot alter the intent or methodology of the test procedure as originally approved. Because of this, minor changes require no additional review or approval. A major change to a procedure is handled in an identical manner as the original review and approval of a procedure-see Section 14.2.3.2.

14.2.3.4 Procedure Format

The format for test procedures will be uniform to the extent practicable and will consist of the following sections: Purpose, references, time required, prerequisite tests, test equipment, limits and precautions, required station (or unit) status, prerequisite system conditions, test method, data required, acceptance criteria, procedure and enclosures. Procedures are written in sufficient detail to permit qualified personnel to perform the required tasks.

Data sheets in procedures used to verify the acceptability of Engineered Safeguards pumps and fans will include all essential information to allow extrapolation of performance from test conditions to post accident design conditions. Adequate documentation is provided by the test procedure to allow determination of system operating configurations at the time test data is obtained.

14.2.4 Conduct of Test Program

14.2.4.1 Administrative Procedures

All aspects of the startup test program are conducted under appropriate administrative procedures. The use of properly reviewed and approved procedures are required for all preoperational and startup tests. The results of each preoperational test are reviewed and approved by the responsible group superintendent before they are used as the basis of continuing the test program. The results of startup testing will be reviewed and approved by the Superintendent of Technical Services prior to proceeding to the next significant power plateau. In addition, the results of each individual startup test will receive the same review as that described for preoperational tests. All modifications to safety related systems which are found necessary are reviewed and approved by the responsible group superintendent and the station manager.

The impact of these modifications on future and completed testing is evaluated during this review process and appropriate retesting is conducted if appropriate.

Copies of approved test procedures for satisfying FSAR testing commitments will be made available for review by NRC personnel approximately 60 days prior to their intended use, and not less than 60 days prior to scheduled fuel loading date for startup tests.

14.2.4.2 Conduct of the Preoperational Test Program

The preoperational test program begins when construction is sufficiently complete to allow testing and/or operation of individual structures, systems and components. 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. The Unit Coordinator is responsible for this scheduling to assure that systems are completed and tested as needed.

During construction, a member of the station staff is assigned the responsibility for given systems. He is aware of the system design requirements and those of the preoperational test procedure prerequisites. This individual is responsible for resolving construction and/or design deficiencies, and for readying a system for test. Typical items which are covered during this phase are: design review, construction installation, cleanliness, lubrication, instrument calibration, setting of limit switches, torque limiting devices, electrical and mechanical protective devices, presence of safety devices, etc. Consideration is also given to providing an equipment run-in period to minimize problems encountered during the preoperational test. The Unit Coordinator is kept appraised of the progress of system completion and checkout and schedules the preoperational test when required.

A test coordinator is assigned the responsibility for the proper conduct of each required test. Prior to commencement of a test the test coordinator 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 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 Steam 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 these conditions are met, the preoperational test is conducted according to the approved test procedure. Should major changes to the procedure become necessary, they are subject to the same review and approval as the original procedure. The Test Coordinator documents that acceptance criteria is met or documents any deficiencies in equipment performance, system design, or equipment installation discovered during the test.

14.2.4.3 Conduct of Startup Test Program

With the exception of certain tests, which are identified in Section 14.2.1, all preoperational tests must be completed prior to beginning the startup test program. The Unit Coordinator coordinates day-to-day activities of the plant startup program and establishes that individual systems and equipment are available for specific tests and plant maneuvers. Tests are conducted under the technical direction of the Performance Engineer with assistance from support organizations and Westinghouse as necessary.

The initial fuel-loading and criticality portions of the tests are conducted following detailed instructions as discussed in Section 14.2.10. Tests that result, or may result, in reactivity changes are performed only with the knowledge and consent of an operator or senior operator licensed pursuant to 10CFR 55. Prior to proceeding from one phase of the startup test program to the next, analysis of testing results is conducted as discussed in Section 14.2.5. Startup tests which are not essential and which are not required to be complete prior to the initial escalation to the next power plateau are identified in Section 14.5 and Figure 14-2.

The startup sequence shown in Figure 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 30, 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 Figure 14-1 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.

14.2.5 Review, Evaluation and Approval of Test Results

At the completion of each test, the Test Coordinator conducts a field evaluation of the test results. Following this, a responsible 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 or 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 or Station Manager, as applicable.

If, during the conduct of a test or during the verification or approval process, a discrepancy is identified, action is taken to resolve it. This action could consist 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 re-evaluation of test results, as necessary to verify the adequacy of the corrective action.*
6. *Review and evaluation by the appropriate design organization(s).*

Only if a discrepancy exists is it necessary to involve the responsible design organizations in the review and evaluation of test results.

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 Subsection 14.2.3.3 above. If equipment changes are necessary, then such changes are executed after the appropriate design reviews are properly documented and become a part of the station records.

All work involving maintenance, replacement, or modification of plant systems following final system turnover to the Nuclear Production Department is accomplished under the station work request system or QA Procedure F-13. Each work request/form F-13A review provides for documentation of a retest review. The station Planning Section or Station Superintendents are responsible for documenting a retest review of all work requests/form F-13A processed.

The preoperational and pre-initial test programs are reviewed prior to initial criticality by the Station Manager, and the Operation, Maintenance, and Technical Services Superintendent to assure that all prerequisite testing is complete.

A review, evaluation, and approval program is conducted for startup testing similar to that conducted for the preoperational test program. In addition, the results of zero and low power (<5% F.P.) testing are approved by the Supervisor of Technical Services and the NSSS vendor prior to proceeding with power ascension testing. The results of testing at each major power ascension plateau will be approved by the Superintendent of Technical Services prior to escalation to the next power level.

14.2.6 Test Records

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.*

14.2.7 Conformance of Test Programs With Regulatory Guides

The development and conduct of the initial test program utilizes issued regulatory guides as guidelines for conformance with 10CFR 50, Appendix B, Section XI and Appendix A applicable design criteria which pertain to testing. Where test methods or the scope of testing differs from that specified by the regulatory guide, adequate assurance is provided by other means to assure that the objectives of the regulatory guides are fulfilled. Table 14-1 lists all regulatory guides considered in the development of the initial test program. Where complete compliance with the applicable guides is not indicated by Table 14-1, alternate means of complying with the objectives of the regulatory guide are presented.

14.2.8 Utilization of Reactor Operating and Testing Experiences at Other Reactor Facilities

The Steam 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 Reportable 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.

On-site activities to ensure that feedback from other station's startup testing activities is incorporated at Catawba include:

1. *Review of all McGuire problem reports by system startup engineers for applicability to Catawba.*
2. *Review of completed test reports from McGuire by test coordinators for identification of testing problems encountered at McGuire.*

14.2.9 Trial Use of Plant Operating and Emergency Procedures

The station operating, emergency and surveillance procedures are use-tested during the test program and are also used in the development of preoperational and initial startup procedures to the extent practicable. The trial use of operating procedures serves to familiarize operating personnel with systems and plant operation during the testing phase and also serves to assure the adequacy of the procedures under actual or simulated operating conditions before plant operation begins.

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.

14.2.10 Initial Fuel Loading and Initial Criticality

14.2.10.1 Fuel Loading

Initial fuel loading and criticality are the first portions of the startup test program and are accomplished when the preoperational test program is essentially completed. The purposes of these two steps are:

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 criticality.*

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. Fuel movement is performed under the supervision of a Senior Reactor Operator as required by 10CFR50.54

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 prior to resumption of loading.

Each source range channel is checked for proper neutron response within eight hours of commencement of fuel loading by positioning a portable neutron source near each detector. This source is typically a 1-5 curie Sb-Be source. Should fuel loading be delayed for eight hours or more, each source range channel will be source checked prior to resumption of fuel loading. A minimum count rate of 2 cps will be maintained on each responding source range channels following loading of the primary source assemblies.

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} < 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 fuel 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 permanent source range channels with a setpoint at approximately three 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. Following completion of fuel loading each assembly and its inserted component will be visually checked for proper location and orientation.

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.

14.2.10.2 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. A minimum source range count rate of at least ½ cps on both channels will be verified prior to initiating the initial approach to criticality.*

14.2.11 Test Program Schedule

Figure 14-2 illustrates the existing schedule logic and relative durations of major portions of the preoperational and startup testing programs. The durations shown in Figure 14-2 are approximate and are subject to change as are major portions of the schedule logic. Detailed testing logic and durations are updated monthly and are available on site for review.

Table 14-2 indicates current major schedule milestone dates for Units 1 and 2. Current staffing and training schedules are based on the schedule illustrated by Figure 14-1 and the Unit 2 overlap illustrated by the milestone dates of Table 14-2. Detailed scheduling techniques employed onsite include provisions for inputting manpower resource requirements and for projecting future manpower requirements as well as identifying portions of the schedule where resource requirements are limiting.

A detailed sequence of testing following initial fuel loading is shown on Figure 14-2.

All tests identified in Section 14.4 and Section 14.5 will also be performed on Catawba Unit 2 with the exception of tests or portions of tests provided solely for verification of station operating procedures or initial operator training. Portions of other tests such as the Upper Head Injection high pressure blowdown may be deleted from the Unit 2 test program following review of the Unit 1 test results and subsequent recommendations by the NSSS vendor.

Following completion of testing at each power level, the high flux trip setpoints will be reset to a value no greater than 20% beyond the next intended test plateau power level, prior to increasing power to the next test plateau.

14.2.12 Individual Test Descriptions

14.2.12.1 Testing Prior to Fuel Loading

Refer to Section 14.4 for individual test descriptions.

14.2.12.2 Initial Startup Testing

Refer to Section 14.5 for individual test descriptions.

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14.3 Startup Physics Test Program

This section defines the minimum reload physics test program to be performed at the Catawba Nuclear Station. The Startup Physics test program was submitted to the NRC per H. B. Tucker's April 4, 1988 letter. The NRC issued an SER on May 18, 1988, approving Duke's submittal. 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 measurements 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. All Rods Out Critical Boron Concentration (High Power)

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. 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 purposes of this section:

RTP.	Rated Thermal Power.
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.
Reactivity Computer.	A digital or analog device that 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 $F_{\Delta H}^{N}$ (or normalized reaction rates) for its operable instrumented location, and

N = number of operable instrumented locations.

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.

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 fully withdrawn positive reactivity associated with each Shutdown and Control Bank is measured over the course of Dynamic Rod Worth Measurement (DRWM). These reactivities are subsequently averaged and the result is converted to an equivalent boron concentration. This concentration is added to the equilibrium boron concentration obtained via chemistry sampling, conducted concurrently with DRWM, to obtain the ARO Boron Concentration.

14.3.2.1.3 Acceptance Criterion

Predicted ±50 PPM 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 boron concentration, NC temperature is changed at least 1.1°F (with acceptable linearity per Advanced Digital Reactivity Computer (ADRC) analysis). An evaluation of the slope from this change is performed for calculation of the ITC (performed internally by the ADRC). The measurement is repeated with an NC temperature change in the opposite direction and the resulting ITCs are averaged.

14.3.2.2.3 Acceptance Criterion

Predicted ±2 PCM/°F

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 Method

There are two methods used to measure the reactivity worth of each control rod bank, Dynamic Rod Worth Measurement (DRWM) and the rod swap technique. Either method can be used; however, DRWM is the common technique. Rod Swap is an alternate method for infrequent applications.

14.3.2.3.2.1 Dynamic Rod Worth Measurement (DRWM)

The Dynamic Rod Worth Measurement (DRWM) technique is used to determine the individual bank worths. Initially the control rods are fully withdrawn from the core. The bank to be measured will be inserted in a continuous motion from all rods out position to two (or less) steps indicated on the bank demand counters. The flux signals from the upper and lower section of an excore detector will be recorded while the bank is being inserted. The bank will be withdrawn to the all rods out position and the remaining banks will be measured similarly.

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

The Dynamic Rod Worth Measurement (DRWM) methodology is described in Westinghouse 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.

14.3.2.3.2.2 Rod Swap

The "rod swap" technique is used to determine the individual bank worths. Initially, the reference bank is inserted into the core using NC boron dilution, with all other banks withdrawn. The bank's insertion is made in discrete movements, with the associated reactivity changes calculated concurrently using the reactivity computer. The reference bank integral worth is determined by summing the reactivity changes from each insertion until the bank was fully inserted. As a verification, this reference bank worth is then compared to the reference bank worth calculated using the NC boron change and predicted differential boron worth. If these two worths do not agree by ±15%, the reference bank measurement will be repeated before completion of the test.

Beginning with the reference bank at or near 0 steps withdrawn and equilibrium NC boron concentration, the bank to be measured (hereafter the "test bank") is partially inserted into the core. The reference bank is then withdrawn to compensate for the reactivity change. This sequence continues until the test bank is fully inserted. The final reference bank position is noted and the test bank is then withdrawn from the core, in similar fashion, versus the insertion of the reference bank or the next test bank.

The change in the reference bank position is used to infer the test bank's integral worth. The above procedure is repeated until all banks, control and shutdown, are measured. If a test bank is worth more than the reference bank, the remaining withdrawn worth of the test bank is measured, using the reactivity computer, with the reference bank fully out. This additional reactivity is used with the reference bank total worth to infer the test bank integral worth.

14.3.2.3.3 Review Criteria

14.3.2.3.3.1 Dynamic Rod Worth Measurement (DRWM) Review Criteria

1. Individual banks ±15% or ±100 PCM of predicted (whichever is greater). If this review criteria 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. Remeasure the bank using Dynamic Rod Worth Measurement to confirm the measurement.
   c. Prior to exceeding 5% power, a power distribution measurement (full core flux map) must be performed and evaluated.
2. The total rod worth measured (sum of all banks) is within ±8% of the total predicted worth. If this review criteria 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 the boron dilution method (described in the Rod Swap method) and compare the boron difference results (ΔC_B) 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 Criteria on the sum of the worths must be adjusted accordingly (e.g., a 7% allowance will yield a Review Criterion of 5.6%, 7%x8/10).

14.3.2.3.3.2 Rod Swap Review Criteria

1. Reference bank ±10% of predicted,
2. Remaining individual banks ±15% or ±100 PCM of predicted (whichever is greater), and
3. Sum of all banks ≤ 110% of predicted

   Note: If any review criterion is missed, remedial action will be taken per the NRC Safety Evaluation Report for the Duke Power Company Rod Swap Methodology.

14.3.2.3.4 Acceptance Criteria

14.3.2.3.4.1 Dynamic Rod Worth Measurement (DRWM) Acceptance Criteria

1. The total rod worth measured (sum of all banks) is ≥ 90% of total predicted worth. If the acceptance criteria is not met the following actions must be taken.
   a. Perform an overall review of the measurement processes and design constants.
   b. If the error is not found, measure the worth of the heaviest bank (rod swap reference bank) by the boron dilution (described in the Rod Swap method) and compare the boron difference results (ΔC_B) 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% power.

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

14.3.2.3.4.2 Rod Swap Acceptance Criteria

1. Reference bank ±15% of predicted
2. Remaining individual banks ±30% or ±200 PCM of predicted (whichever is greater), and
3. Sum of all banks > 90% of predicted

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.

14.3.3.1 Flux Symmetry Check - Low Power

14.3.3.1.1 Initial Test Conditions

1. reactor power between 0 and 40% 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 $F^{N}_{\Delta H}$ or normalized reaction rates for all operable instrumented locations.

14.3.3.1.3 Acceptance Criteria

1. $F^{N}_{\Delta H}$ 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 40 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 $F^N_{\Delta H}$ or normalized reaction rates for all operable instrumented locations.

14.3.3.2.3 Acceptance Criteria

1. $F^N_{\Delta H}$ 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 $F^N_{\Delta H}$ or normalized reaction rates for all operable instrumented locations.

14.3.3.3.3 Acceptance Criteria

1. $F^N_{\Delta H}$ 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. The predicted boron difference between HZP and HFP is compared to the measured boron difference between HZP and HFP.

14.3.3.4.3 Acceptance Criterion

Predicted ±50 PPM Boron

14.3.4 Startup Report

A summary report of plant startup and power escalation testing will be submitted following (1) receipt of the initial Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.

The Startup Report will address each of the tests identified in Section 14.3.2 and 14.3.3 and include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation will also be described. Any additional specific details required in license conditions based on other commitments will be included in this report.

The Startup Reports will be submitted within (1) 90 days following completion of the Startup Test Program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of Startup Test Program, and resumption or commencement of commercial operation), supplementary reports will be submitted at least every 3 months until all three events have been completed.

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14.4 Preoperational Testing

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14.4.1 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 Hot 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 by Duke Design Engineering and Construction Departments, and approved by Duke Steam Production Department. Engineering acceptance criteria for all movements, limitations, precautions, and corrective actions as applicable are described in detail in the test procedures. 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. Blanket insulation with a similar R valve of mirror insulation will be wrapped around the piping. Small sections will be cut out of the blanket insulation at locations where measurements need to be taken. In general, the amount of exposed piping is at a minimum. All lock devices have been removed from system supports and restraints. Prior to taking measurements above ambient temperatures, temporary insulation has been installed in place of removed mirror insulation, except for localized areas where access to the pipe surface is necessary for measurement purposes.

Test Method

Prior to starting the Reactor Coolant System Hot 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 stopped until an evaluation and 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

Selected points on components and piping of the Reactor Coolant System have no expansion interferences. Analysis of measured movements versus predicted movements indicates that allowable stresses will not be exceeded. The components and piping return to their baseline *cold positions, within the tolerances established by Duke Power Company Design Engineering Department.*

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14.4.2 Reactor Coolant System Hot Functional Test

Abstract

Purpose

To demonstrate and verify the necessary functional responses of components, systems, and instrumentation during plant heatup, cooldown, and normal hot plant operation. The conduct of Hot Functional Testing (HFT) allows the completion of other tests or portions of tests requiring hot, full pressure primary system conditions for their conduct.

Prerequisites

1. *The reactor coolant system has been filled and vented.*
2. *Required auxiliary systems are operational to the extent required to support hot, full temperature and pressure primary system operation.*
3. *Sufficient capacity is available in the recycle holdup tanks to accept reactor coolant system letdown during heatup.*
4. *Reactor coolant system hydrotest has been completed.*
5. *Pressurizer Relief Tank Functional Test has been completed.*
6. *Reactor coolant system component and piping supports have been installed in accordance with design drawings.*
7. *The reactor coolant pumps and pressurizer heaters are fully operational.*

Test Method

Reactor coolant pumps are started and water chemistry specifications are established. A pressurizer steam bubble is established and reactor coolant system temperature and pressure are increased. The heatup may be interrupted at various intervals to perform other preoperational tests and to record data as necessary. After achieving hot, no-load temperature, pressurizer pressure control is demonstrated. Following completion of required additional preoperational tests at hot conditions, cooldown is initiated. Proper operation of the residual heat removal system and collapse of the pressurizer steam bubble is demonstrated during cooldown. Concrete temperatures around the main steam line generations will be monitored during HFT to verify that design limits are not exceeded.

During the test, the operation of steam dump system will be verified. An initial roll of the turbine will be performed to demonstrate operability of the Main Steam System. The ability of the systems to maintain condenser vacuum is demonstrated (or may be demonstrated during Power Escalation Testing). The proper operation of the feedwater system is demonstrated (or may be demonstrated during Power Escalation Testing). The closure times of the main steam isolation valves at full pressure and temperatures are measured, and part-stroke operability is verified.

Open and reclosure setpoints of the steam generator power-operated relief valves, the main steam atmospheric dump valves, the main steam safety valves, and the pressurizer power-operated relief valves will be verified at temperature, during the performance of the Reactor Coolant System Hot Functional Test.

Acceptance Criteria

1. *Pressurizer level and pressure control during heatup, hot operation, and cooldown maintains NC system parameters within Technical Specification limits.*
2. *The ability to maintain charging, letdown, and seal injection flow is demonstrated through performance of the Chemical and Volume Control System Functional Test.*
3. *Control of reactor coolant system cooldown rate within the Technical Specification limits is demonstrated.*
4. *Concrete temperature adjacent to main steam line penetrations do not exceed 150°F.*
5. *Main steam, steam dump, and feedwater systems operate within design limits as specified by Duke Power Company Design Engineering Department System Descriptions. This verification may be demonstrated during Power Escalation Testing.*
6. *Condenser vacuum is maintained within normal operating limits, as specified by Duke Power Company Design Engineering Department. This verification may be demonstrated during Power Escalation Testing.*
7. *Feedwater heater controls systems and hotwell level controls function within limits as specified by Duke Power Company Design Engineering Department Specifications. This verification may be demonstrated during Power Escalation Testing.*
8. *The main steam isolation valves close in <5 seconds, and part-stroke capability is successfully demonstrated.*
9. *The main steam safety valve setpoints are within the limits provided by Duke Power Company Design Engineering Department.*

14.4.3 Piping System Thermal Expansion Test

Abstract

Purpose

Verify piping and components of systems identified in Table 3-86 are unrestricted from expanding.

Prerequisites

1. All required snubbers and spring supports are installed and have received Construction's final inspection.
2. Hot Functional Testing is underway. Those tests that cannot be performed as a part of Hot Functional Testing because of the required plant condition will be performed as part of the initial startup and power escalation phase.
3. Cold settings of applicable snubbers and spring supports have been obtained.

Test Method

During Hot Functional Testing and Pre-critical Heatup for power escalation, a visual inspection will be performed to verify that spring supports are within design range (i.e., indicator within spring scale) and recorded. Visual inspection of snubbers will be performed to ensure they have not contacted either stop and are within expected travel range. Snubber piston scales will be read to ensure acceptance criteria for piston to stop gap is met. Also system walkthroughs will be performed during HFT to visually verify that piping and components are unrestricted from moving within their range. Hot displacement measurements of all snubbers will be obtained and motion will be compared with predicted values. Discrepancies will be reviewed and evaluated by Design Engineering.

The Feedwater System and Auxiliary Feedwater System Hot Displacement measurements will be obtained during the initial startup and power escalation phase.

All snubbers and spring supports, which required adjustments during the test, will be reinspected in its hot condition to assure proper adjustments were made. Since the ND System will not be operated above 200°F during initial startup, its snubbers and spring supports which require reinspection will be done at 200°F.

<u>Acceptance Criteria</u> - Initial and Final Inspections

1. Snubbers are not within ½ inch of either piston stop.
2. All Spring Support indicators remain within spring scale at all inspection times.
3. System Piping and components are unrestricted from moving.

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14.4.4 Pressurizer Relief Tank Functional Test

<u>Abstract</u>

<u>Purpose</u>

To demonstrate the functional performance of the pressurizer relief tank and its associated instrumentation and alarms. This system is considered non-safety related.

<u>Prerequisites</u>

The nitrogen gas reactor make-up and waste gas systems are 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.

<u>Test Method</u>

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

<u>Acceptance Criteria</u>

The level and pressure alarms and cover gas system operate at the setpoints designated by Westinghouse and Duke Design Engineering. The pressurizer relief tank spray flow is within limits provided by Westinghouse and Duke Design Engineering. Automatic pressure regulating valves maintains pressure within design limits as outlined in the Westinghouse Limitations and Setpoints Manual, and subsequent Westinghouse transmittals.

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14.4.5 Chemical and Volume Control System Functional Test

<u>Abstract</u>

<u>Purpose</u>

To demonstrate the capability of the Chemical and Volume Control System to provide required flows, pressures, temperatures, and proper flow paths for charging, letdown, seal water, and *make-up to the Reactor Coolant System. To demonstrate operability of the features necessary for sampling, chemical addition and control of the primary system.*

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. Component cooling and Nuclear Service Water Systems are operable to the extent required to operate the system.

The proper functioning of the sampling features may be tested prior to the Hot Functional Test, as the systems are filled and hydro tested.

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. Demineralizer design flow rates and pressure drops will be demonstrated during precritical testing. Charging pumps are tested for design flow rates and pressure drops. Charging pumps are tested for capability to deliver varying flow rates. Volume control tank level and pressure control indications and alarm setpoints are checked. Operational calibration and operation of the different modes of dilution and boration are verified. Flow rates within the charging, letdown, seal water and make-up flow paths are measured and verified. Emergency boration is verified along with boric acid transfer pumps discharge pressure in recirculation. Boric acid tank low level and low temperature alarms are verified. Auto-opening of INV455 (boric acid batching tank temperature control valve) upon a low temperature signal is also verified.

Operability and flow paths for sampling and chemical addition are verified by the use of normal chemistry control procedures, and successful verification is documented as a part of this test.

The demonstration of the fail safe operation of pneumatically and solenoid actuated components in the Chemical and Volume Control System upon loss of power will be performed as a part of the station surveillance program required by Technical Specifications, in compliance with ASME Section XI, part IWV.

Acceptance Criteria

System flows, temperatures, and pressures are within limits specified by Westinghouse, and are conservative with respect to values assured in Chapter 15. Level setpoints and alarms within the flow paths tested actuate at the values specified by Westinghouse.

Sampling and chemical addition components function in accordance with design system descriptions.

14.4.6 Rod Control System Functional Test

"HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED"

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 is available and tested. Nuclear and temperature instrumentation channels are available for input of required test signals.

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 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.

The response of the Rod Control System to a loss of power will be demonstrated as a part of the Station Blackout Test. Please refer to the Abstract, Section 14.5.29.

Acceptance Criteria

Manual and automatic system response is in accordance with the criteria specified by Westinghouse. All interlocks and permissives are verified to function correctly as specified by Westinghouse.

14.4.7 Nuclear Instrumentation System Functional Test

"HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED"

Abstract

Purpose

To assure the proper operation of the Nuclear Instrumentation System prior to initial fuel loading.

Prerequisites

System instrumentation and cabling is installed and tested. Normal electrical power sources are 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 indicators 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

1. *Trip, rod stop, alarms, and control room indications actuate at the proper setpoints as specified by Westinghouse. For setpoints that must be determined after plant operation which are not specified by Westinghouse, initial values shall be based on operating experience of similar plants.*
2. *Proper operation of interlocks with the Reactor Protection System is verified.*
3. *Neutron detectors are verified to be properly positioned.*

14.4.8 Reactor Protection System Functional Test

"HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED"

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 for response time testing. Inputs to the RPS must be energized for logic testing. Response time testing of instrument lines and sensing lines is not included as a part of the station test program since delays introduced by this hardware are considered to be insignificant in comparison with the overall delay times. The response times of individual sensors have been assured to be acceptable by vendor tests.

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 into the sensor and measuring the time from when the process reaches its set point and the Reactor trip breakers open. The actuation times for the active components are also measured. The total response time is assured to be less than the value assumed in the accident analysis through the application of conservative acceptance criteria for sensor response times, channel response times, and actuation times. The response time of the below listed protection channels will be tested.

1. *Power Range Neutron Flux, High Setpoint and Low Setpoint*
2. *Power Range Neutron Flux, high Negative rate*
3. *OTΔT*
4. *OPΔT*
5. *PZR pressure - low*
6. *PZR pressure - high*
7. *Low reactor coolant flow*
8. *S/G water level lo-lo*
9. *RCP undervoltage*
10. *RCP underfrequency*

Acceptance Criteria

Instrument channels and solid-state logic trains for reactor protection and protection permissives function as specified by Westinghouse. Annunciators, channel status lights and permissive interlock lights function to indicate the correct status of the input signal levels.

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14.4.9 Feedwater and Condensate Systems Functional Test

Abstract

Purpose

To demonstrate the ability of these systems to provide a steady, properly regulated supply of feedwater flow to the steam generators during normal and upset conditions. To demonstrate the operability of the secondary Chemical Addition and Sampling Systems. This test is considered to be non-safety related.

Prerequisites

Support systems necessary to operate the condensate and feedwater systems are sufficiently in service. Steam generators are in service at hot standby temperature and pressure conditions for applicable portions of the procedure.

Test Method

Feedwater flow rates will be varied with the bypass feedwater control valves in manual to demonstrate manual control of steam generator levels during hot functional testing. Feedwater flow rates will be varied with the main feedwater control valves in manual to demonstrate manual control of steam generator levels during Hot Functional Testing and/or power escalation. Manual control of feed-water pump speeds will be demonstrated during Hot Functional Testing and/or power escalation. Operability of the feedwater heaters and feedwater heater drains will be verified during power escalation. The ability to obtain samples at designated points in the system and to add chemicals to control feedwater chemistry are verified by the use of normal station chemistry procedures.

Acceptance Criteria

Valve operations which are required to supply the required flows are demonstrated by operating the required valves from the Control room. The proper response to feedwater isolation as described in Section 10.4.7.2 is verified.

Doghouse high water level alarms actuate in Control room upon simulation of high water level.

Samples are obtained from the feedwater and condensate systems. Chemical Addition capability is verified to be operable.

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14.4.10 Condenser Circulating Water Systems Functional Test

Abstract

Purpose

To demonstrate pre-fuel load, the proper operation of pumps and towers. To initially set flow the balance of the cooling towers.

To demonstrate during Hot Functional Testing and/or power escalation, the capability of the condenser circulating water system to supply cooling water to the main and feedwater pump turbine condensers to condense the turbine exhaust steam and to provide a sufficient heat sink for the steam dump system. This test is considered to be non-safety related.

Prerequisites

The condenser circulating water system is complete and filled. All support systems are operational to the extent necessary to perform the test. Alarms are calibrated and loop checked.

Test Method

Circulating pumps, cooling towers, and instrumentation are tested to demonstrate proper operation. System flow rates are verified where applicable. Initial flow balancing to the cooling towers will be performed by setting inlet valve open limit switches and adjustable weir levels around the cooling tower distribution flumes.

During Hot Functional/Testing and/or power escalation the main and feedwater pump turbine condensers' performance parameters will be monitored to show adequate heat removal capability.

Acceptance Criteria

Circulating pumps can be started remotely and operated. Cooling tower fans can be started remotely and operated. Instrumentation functions and provides remote indication of operating conditions. Initial flows are balanced by adjustment of valve limit switches and adjustable weir levels.

Main and feedwater pump condensers maintain proper vacuum.

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14.4.11 Auxiliary Feedwater System Functional Test

Abstract

Purpose

To demonstrate the capability of the system to deliver design flows to the steam generators under all anticipated conditions. To demonstrate the operability of essential controls, interlocks, and alarms.

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 necessary to accept auxiliary feedwater pump discharge. A temporary steam supply may be required for testing of the turbine-driven auxiliary feedwater pump. The steam generators are required to be at hot shutdown temperature and pressure conditions for portions of the test.

Test Method

Each auxiliary feedwater pump is started and run separately to demonstrate flow from the upper surge tank and the auxiliary feedwater condensate storage tank. Pump performance is verified and the existence of adequate suction head from each of the above sources is verified. Auxiliary feedwater supply from the upper surge tank is verified with this source under vacuum at normal operating temperatures.

Verification is performed of the operability of pump runout protection interlocks, automatic reset of the automatic start defeat circuitry at the P-11 permissive setpoint, and proper automatic valve alignment upon receipt of a simulated auxiliary feedwater start signal. The auxiliary feedwater nozzles will be monitored for indications of water hammer while feeding the steam generators during hot functional testing. At least five successive, cold quick starts of the steam driven auxiliary feedwater pump upon receipt of a start signal will be verified. Steam piping to the steam driven auxiliary feedwater pump will be visually monitored during cold starts for indications of water hammer, flashing, excessive vibration, or interference due to thermal expansion.

Acceptance Criteria

1. *Motor driven pump A develops a total head of 3605 ft., +1%, -3% at a flow of greater than or equal to 400 gpm, and motor driven pump B develops a total head of 3620 ft. +1%, -3% at a flow of greater than or equal to 400 gpm.*
2. *The steam driven pump develops a total head of 3705 ft., +4%, -6% at a flow of greater than or equal to 400 gpm.*
3. *Motor and steam driven pumps start on receipt of the simulated auxiliary feedwater start signal.*
4. *The automatic-start-defeat circuitry resets at the P-11 setpoint.*
5. *Valves required to open automatically to align the auxiliary feedwater pumps to the steam generators open upon automatic start of the pumps.*
6. *Pump runout protection circuitry operates as described in FSAR Section 10.4.9.2.*

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14.4.12 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. For portions of this test, the reactor coolant system must be at hot standby temperature and pressure conditions.

Test Method

Flow paths and flow rates are verified for normal unit conditions for each of the two trains. (Shared equipment will be balanced during the Unit 1 test only)

Flow paths and flow rates are verified for normal unit cooldown conditions with full flow through both trains. (Shared equipment will be balanced during the Unit 1 test only)

The discharge temperature from the KC heat exchanger is verified to be within design limits with the unit at normal operating temperature and pressure.

Automatic starting of the component cooling water pumps and automatic valve alignment is demonstrated for a simulated safety injection signal. This portion of the test is demonstrated during the Engineered Safety Features Actuation System Functional Test. Please refer to the Abstract, Section 14.4.29.

Acceptance Criteria

Automatic valve alignment and pump starts occur in response to engineered safety features actuation signals. Flows to essential components required during modes 1, 3-1, 4, and 5-2 (as defined in FSAR Section 9.2) correspond to the nominal values shown in FSAR Table 9-6.

Temperature of water in the Component Cooling System does not exceed the design temperature shown in FSAR Table 9-8.

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14.4.13 Residual Heat Removal System Functional Test

Abstract

Purpose

To demonstrate the capability of the system to provide design flows, pressures and cooldown rates for removal of residual heat from the Reactor Coolant System during unit shutdown and unit cooldown conditions.

Prerequisites

The Nuclear Service Water System and Component Cooling Water System are operational to the extent required to operate the system. The Reactor Coolant System is less than 350°F and 425 psi for operation in the cooldown mode.

Test Method

The system is operated and performance demonstrated for unit cold shutdown and unit cooldown conditions for each of the following flow paths: the train A flow path taking suction from the loop B hot leg and discharging to loops C and D cold legs and the Train B flow path taking suction from the C loop hot leg and supplying cold legs A and B.

Acceptance Criteria

System interlocks, instruments, and alarms within the specified flow paths are verified to function in accordance with the Duke Power Company Design Engineering Department supplied values. Residual Heat Removal pumps performance equals or exceeds the acceptance head-capacity curve supplied by Duke Power Company Design Engineering Department, adjusted for measurement error. Automatic isolation of Residual Heat Removal System occurs when Reactor Coolant System pressure rises above 600 psig. Residual Heat Removal System isolation valves are prevented from opening when Reactor Coolant System pressure is above 425 psig.

"HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED"

14.4.14 Fire Protection System Functional Test

Abstract

Purpose

To demonstrate the ability of the fire protection system to provide water at acceptable flows and pressures to protected areas.

Prerequisites

The filtered water system is operable to the extent necessary to supply the jockey pumps. LP pressure service water is available to supply the main fire pumps and the 200 gpm jockey pump. Nitrogen is available to the fire system pressurizer tank.

Test Method

The proper starting and operation of the main fire pumps is tested by varying starting switch pressure and by measuring the flow from each pump. The proper starting and operation of the jockey pumps and pressurized tank pressure controls are tested by observing their response to *changes in pressurizer tank level. Flow paths to the major protected areas are verified. The Auxiliary Building and Reactor Building isolation valve operation is verified.*

Fire detection and alarm systems will be tested in accordance with Selected Licensee Commitments Section 16.9, as a part of the normal station surveillance testing program. This testing will be performed prior to fuel loading.

Acceptance Criteria

System flow paths are verified to be open. Each pump develops ≥ 331 ft. of head at a flow of ≥ 2500 gpm. Jockey pumps are capable of maintaining system pressure. Main fire pumps start automatically on low pressure.

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14.4.15 Nuclear Service Water System Functional Test

Abstract

Purpose

To verify acceptable pump performance by obtaining at least three points on the head/capacity curve and verifying against acceptance criteria.

Balance system flows to individual components with manually balanced flows to assure minimum acceptable cooling flow to each essential component in each of the following modes of operation:

Sump recirculation after containment spray (limiting mode, essential flows) shutdown with blackout (limiting mode, non-essential flows)

Balance return flows to each finger of the Standby Nuclear Service Water Pond (SNSWP) during the Unit 2 functional test.

Verify Nuclear Service Water System (RN) pump motor cooler inlet isolation valve interlocks.

Verify strainer backwash on simulated high strainer ΔP and associated alarms.

Verify proper dynamic response (including setpoints) of the RN System to lake isolation and resulting low level swapover to the SNSWP - generic demonstration to be performed for one train only (not performed on Unit 2).

The following alarms are verified during the course of the test:

RN pit level alarms

RN System low flow alarms

RN essential header pressure alarms

Proper system response at the proper setpoint is verified for a simulated low intake pit level for the train not used for the actual dynamic swapover at low intake pit level (this verification is performed for both trains of Unit 2).

Prerequisites

All components and essential instrumentation of the Nuclear Service Water System are installed and operational. Portions of the components served by the Nuclear Service Water System are installed and operational.

Test Method

The Nuclear Service Water Pumps will be run singularly to allow data to be collected in order to evaluate their performance.

With each RN Pump in operation with its respective train of components, manual throttling valves and control valve travel stops will be set. The RN System will be lined up for its Sump Recirculation After Containment Spray mode. Then, flows will be verified and others set with the RN System lined up for the shutdown with blackout.

The RN System will be lined up with its return flow to the SNSWP. Verification that the flow to each finger of the pond is balanced will be performed during the Unit 2 functional test. With each RN train in normal operation, the RN pump motor cooler inlet isolation valves will be verified to have opened. Also, a strainer simulated high ΔP will be given to verify initiation of an automatic strainer backwash.

The Lake Wylie source of cooling water will be isolated from the RN Pump Pit. The Unit 1 RN Pump in operation will pump the pit level down. A dynamic low level swapover to the SNSWP will be verified. For the other Unit 1 RN Pump and both Unit 2 RN pumps, a simulated low pit level will be given to verify proper system response.

Essential alarms and annunciators initiated during any of the above tests will be verified.

Acceptance Criteria

Each nuclear service water pump develops less than or equal to 226 feet of head after adjustment for instrumentation error at a minimum flow of 9000 gpm ± 1.9%. Flows to essential components comply with detailed acceptance criteria provided by Design Engineering.

Each nuclear service water pump motor cooler inlet isolation valve interlock allows valve to open upon pump start.

Dynamic swapover is accomplished as described in FSAR Section 9.2.1.2.1, for the pump and pit tested. (Unit 1 only)

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14.4.16 Loss of Instrument Air Test

Abstract

Purpose

To demonstrate that a reduction and loss of instrument air pressure causes fail-safe operation of safety-related 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 penumatic loads are cut-in to the extent possible at the time test begins.

Test Methods

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. Systems or partial systems to be tested are air operated containment isolation valves, pressurizer relief and spray valves and other air operated valves in the Reactor Coolant System, main feedwater control valves, main steam atmospheric and condenser dump valves, control valves in the *turbine gland sealing system, air operated valves in the Safety Injection Containment Spray Systems, and main steam isolation system.*

Section 14.4.47 in conjunction with this test satisfies the requirements of Regulatory Guide 1.80, Regulatory Position C.1-C.7.

Acceptance Criteria

All valves fail to positions as shown on Duke Power Company Design Engineering Department system mechanical drawings.

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14.4.17 Control Room 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

The Control Room Air Conditioning and Ventilation System, normal power, and emergency power (Class 1E) 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

An integrated test is performed on each train to verify proper temperature and humidity can be maintained during all normal modes of operation and proper temperature can be maintained during post-accident conditions. Proper flow through 1CRA-PFT and 2CRA-PFT-1 is demonstrated. Proper operation of each filter train is demonstrated. The refrigeration units are tested to demonstrate their proper operation and cooling capacity. Instrumentation required for safety is verified for proper sequencing and function. The Control Room is pressurized to ≥ 1/8" W.G. and the flow rate required to maintain this pressure is recorded.

The Control Room Ventilation System duct leak tightness is demonstrated by a pressure test conducted prior to turnover of the system from the Duke Power Company Construction Department to the Nuclear Production Department. These tests assure the leak tightness of the ducts as a prerequisite to acceptance of the system.

Acceptance Criteria

1. *The Control Room Ventilation System is capable of achieving a system flow of 6000 cfm ± 10% through 1CRA-PFT-1 and also 2CRA-PFT-1 when tested per the requirements of ANSI N510-1980.*
2. *Valves and dampers align as described in FSAR Section 9.4.1 in normal automatic operating mode, and realign upon receipt of simulated high radiation and high chlorine alarm signals.*
3. *HEPA filter banks demonstrated an efficiency of greater than or equal to 99.95% when they are tested in-place in accordance with ANSI N510-1980 while operating the system at a flow rate of 6000 cfm ± 10%.*

4. *Laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Rev. 2 meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Rev. 2.*
5. *Charcoal absorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in accordance with ANSI N510-1980 while operating at 6000 cfm ± 10%.*

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14.4.18 Diesel Building Ventilation System Functional Test

Abstract

Purpose

To demonstrate the operability of the Diesel Building Ventilation System's fans and dampers in the AUTO, TEST, and PURGE modes.

To demonstrate that the system's dampers fail to their safety position.

To demonstrate that the system's dampers and fans perform properly upon receiving a simulated fire protection signal.

To demonstrate that an environment suitable for personnel access is maintained in one Diesel-Generator Room while the Diesel-Engine is in operation.

Prerequisites

The Diesel-Engine will be required to be in operation to provide a design heat load when verifying that a suitable environment can be maintained in one Diesel-Generator Room.

Test Method

The Diesel Building Ventilation System will be operated in the AUTO, TEST, and PURGE modes to verify proper alignment of the system's fans and dampers.

The electrical supply to the Normal and Emergency Ventilation Dampers will be interrupted to determine if each damper will move to its fail safe position.

A relay jumper will be used to simulate a fire protection signal to determine that the system's fans will shutdown and their respective dampers will move to the proper position.

For one of the four station emergency diesel generators, the Diesel-Engine will be operated at times when the external conditions are expected to approach the two (2) external design day conditions, 10°Fdb and 95°Fdb. The Emergency Ventilation Fans will be in operation at the same time. Data will be recorded to verify that the internal environment is maintained within its acceptance criteria. If the exterior design day conditions are not reached, the internal vs external temperature data taken during the test will be used to extrapolate to find the internal temperature which would have been reached at the design external conditions.

The above will be performed on "A" train diesel ventilation. Unit One "B" train and Unit Two "A" and "B" trains will have fan performance tested instead of Design Hot Day testing.

Acceptance Criteria

The system's fans align as shown in FSAR Figure 9-128 in each mode of operation.

The system's dampers move to their fail safe position at a simulated failure.

The system's fans stop and dampers close upon receipt of a simulated fire protection signal.

For the Diesel Unit under test, the diesel-Generator Room environment is maintained within 60°F minimum and 120°F maximum with the Diesel-Engine in operation at both of the external design day conditions of:

High - 95°F db temperature

Low - 10°F db temperature

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14.4.19 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, and other equipment necessary to maintain the plant in a safe condition during loss of power 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. The portion of this test involving automatic starting and loading tests performed immediately following the 24-hour run will be run concurrently with the Engineered Safeguards Functional Test.

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 full-load-carrying capability is demonstrated by loading the diesel generator to > 5600kW but < 5750kW for an interval of not less than 24 hours. During the 24 hour run demonstration, the ability to swap from one fuel oil filter to the other is demonstrated along with the ability to swap from one fuel oil strainer to the other. Automatic diesel generator starting and loading is again demonstrated for a combined safety injection and for a loss of normal power simulated signals, immediately following the 24 hour run. Simultaneous starting of both diesel generators will be demonstrated.

Diesel generator load rejection will be demonstrated at ≥ 5600 kW but ≤ 5750kW load and at a load equivalent to the largest sequenced load. The ability to transfer the emergency load to offsite power will be demonstrated. Diesel generator emergency response is demonstrated not to be impaired during testing. Diesel generator emergency reliability is demonstrated by means of any 35 consecutive valid tests with no failures.

These demonstrations are accomplished for each of the two generators utilizing only one at a time, except as noted above. In addition, the manual operation of the generators is verified by loading each generator to its qualified load rating.

Acceptance Criteria

Automatic start and loading sequence are accomplished for all simulated emergency conditions. During loading, frequency is maintained at not less than 95% of nominal and generator voltage *is maintained at not less than 75% of nominal. The diesel generator operates at its qualified rated load for the time required to reach temperature equilibrium plus one hour, with engine water and oil temperatures within manufacturer's recommended bounds. The fuel oil can be changed from one fuel oil filter to the other, with the diesel operating, without a drop or loss of fuel oil pressure. The fuel oil can be changed from one fuel oil strainer to the other, with the diesel operating, without a drop or loss of fuel oil pressure. The transient following the complete loss of load should not cause the diesel generator to reach 500 RPM. The diesel generator is capable of rejecting a load of ~825 KW while maintaining voltage at 4160 ± 416 volts and frequency at 60 ± 1.2 Hz. The diesel generator can transfer the emergency load to offsite power, when available. Diesel generator emergency response is not impaired by testing. The diesel generator has 35 consecutive valid tests (as defined in Regulatory Guide 1.108, c.2.e) without a failure.*

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14.4.20 125 VDC Vital Instrumentation and Control Power Test

Abstract

Purpose

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

Prerequisites

Battery area ventilation must be adequate. Sufficient DC loads are available to allow testing of the system.

Test Method

The system is energized for normal operation and a load equal to the maximum accident-condition steady-state dc load as measured during the Engineered Safety Features Actuation System Functional Test is applied. The capability of each battery charger to individually maintain a float charge on its associated battery, while concurrently maintaining the maximum bus dc loads, is demonstrated.

The capability of each charger to supply sufficient current to recharge a completely discharged battery within 24 hours while supplying the steady-state loads of its own load group is verified.

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 service test is performed in accordance with IEEE 450-1975.

The actual load on the vital buses recorded during the performance of the Engineered Safety Features Actuation System functional test is compared with the design loads for the system. This applies to Unit 1 only.

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

Acceptance Criteria

All battery chargers provide float charge while concurrently maintaining maximum bus loads. The system responds properly to loss of normal unit power by maintaining power to the normal loads from the batteries. Batteries are capable of supplying dc power upon de-energization of *their chargers. The battery capacities as determined in the battery service tests are greater than or equal to the capacity necessary to carry the vital loads during the critical period of the accident analysis.*

No individual cell voltages shall reach a level of +1 volt or less during a discharge test.

The battery chargers provide sufficient current to recharge a fully discharged battery within 24 hours while supplying the steady-state loads of their own load group, as described in the test method.

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14.4.21 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 loaded conditions.

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 post accident conditions as described in Chapter 8 of the FSAR.

Test Method

With both diesel fuel oil day tanks at least one-half full, the diesel generators are started and brought to the qualified rated load condition. The units are operated for a sufficient length of time to measure the fuel consumption rate from each day tank.

Acceptance Criteria

System operation including alarms and controls function as specified in the manufacturer's operating manual. Fuel oil consumption rates at qualified rated load are such that the fuel oil day tank would provide greater than one hour's fuel supply.

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14.4.22 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 completed to the greatest extent possible (see Section 14.4.29). All systems inside Containment which have containment isolation valves identified as Potential Bypass Leakage Paths in Table 6-87 are vented and drained except for the following:

System	***Reason***
Ice Condenser glycol supply and return (M372 and M373)	*Ice Condenser is in operation (Unit 1 only - Unit 2 test is performed prior to ice load)*
Containment Air Release Line (VQ M204)	*Turbine Flowmeter (For Imposed Leak Rate Test) is installed on this penetration.*
Containment Hydrogen Sample and Purge (VY M346)	*Containment Relief Valve is installed on this penetration.*
ILRT Test Pressure sensing lines (3 penetrations)	*Lines are open to monitor containment pressure during test.*
Chemical and Volume Control System (NV M256) Nuclear Service Water System (RN M230, M308) Liquid Waste System (WL M345, M221, M374, M359) Component Cooling Water System (KCM321)	*These penetrations are exempted from the venting and draining requirements since the containment isolation valves are supplied by the Seal Water Injection (NW) System. Exemption was approved in Supplement 3 to the Catawba SER.*

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 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. The test duration is at least ten (10) hours preceded by a period for stabilization of containment conditions. In order to verify the test methods, a supplemental leak test is performed by imposing a known leak rate on the containment.

Acceptance Criteria

The Containment vessel shows no signs of structural degradation following the 110% strength test. The measured Containment integrated leak rate does not exceed .15 percent by weight of the containment volume per day. The sum total of the initial containment leak rate and the supplemental imposed leak rate shall not differ from the composite leak rate by more than 0.05% by weight of the containment volume per day. The composite leak rate is defined as the total containment leak rate, as measured by the containment leakage measuring system, during the supplemental imposed leak rate test.

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14.4.23 Containment Air Return and Hydrogen Skimmer System Functional Test

Abstract

Purpose

To demonstrate the capability of the system to respond to an actuation signal as designed. To demonstrate the containment recirculation capability.

Prerequisites

The Containment Air Return and Hydrogen Skimmer System, solid state protection system, and associated support systems are functional to the extent required to test the system. The ice condenser inlet doors are blocked closed to prevent operation.

Test Method

Each containment air return fan and hydrogen skimmer fan is operated. Automatic operation of the Containment air return fans is verified for a simulated high-high containment pressure signal (Sp). Proper operation of the 0.25 psid permissive interlock is verified.

Acceptance Criteria

Containment air return fan motor current and hydrogen skimmer fan current are within the limits of Technical Specification 4.6.5.6.1. Automatic opening of the containment air return fan damper and interlocks that prevent containment air return fan from starting with low containment pressure function as required by Technical Specification 4.6.5.6.2.

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14.4.24 Annulus Ventilation System Functional Test

Abstract

Purpose

To demonstrate the capability of the Annulus Ventilation System to produce and maintain a negative pressure in the annulus following a LOCA and to minimize the release of radioisotopes following a LOCA by recirculating a large volume of filtered annulus air relative to the volume discharged for negative pressure maintenance.

Prerequisites

All essential system components, including fans, filter trains, dampers, and Class 1E power systems are operational to the extent necessary to perform the test.

Test Method

Each ventilation train is operated in conjunction with its respective fan, filter train, dampers, and associated ductwork to demonstrate required capacity per ANSI N510-1980. Essential electrical components, switchovers, and starting controls are demonstrated to be functional. The ability to obtain and maintain the required negative pressure inside the annulus will be demonstrated. The acceptability of the annulus ventilation system HEPA and charcoal filters will be demonstrated per use of test procedures as specified in Regulatory Guide 1.52 Rev. 2.

Acceptance Criteria

1. *Each train of the annulus ventilation system, operating independently of the other train, is capable of achieving a system flow of 9000 cfm ± 10% when tested per the requirements of ANSI N510-1980.*

2. *HEPA filter banks demonstrated an efficiency of greater than or equal to 99.0% when they are tested in-place in accordance with ANSI N510-1980 while operating the system at a flow rate of 9000 cfm ± 10%.*

3. *Laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Rev. 2 meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Rev. 2.*

4. *Charcoal absorbers remove greater than or equal to 99.0% of a halogenated hydrocarbon refrigerant test gas when they are tested in accordance with ANSI N510-1980 while operating at 9000 cfm ± 10%.*
5. *The annulus ventilation system demonstrates the ability to achieve a negative pressure of greater than or equal to 0.5 in W.G. within the time period assumed by the station safety analysis. (This criteria may be verified during the Integrated ESF Test).*

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14.4.25 Ice Condenser Region Functional Test

Abstract

Purpose

To assure that adequate ice inventory is available in the ice condenser region. The test will also verify the proper installation and operation of top deck doors, intermediate deck doors, lower inlet doors, and floor drains. The test will verify that flow passages in the ice condenser are open and unblocked. The ability to maintain proper ice bed temperatures is also verified.

Prerequisites

System structures and components are complete. Initial cooldown in preparation for initial ice loading has been completed.

Test Method

As initial ice loading progresses, initial ice basket weights will be obtained. Basket weights will be analyzed in order to verify basket minimum weights and total ice inventory. The operability of top deck doors, intermediate deck doors, lower inlet doors, and floor drains will be verified by use of appropriate exercise and inspection procedures. Flow passages will be verified to be clear and free of excessive ice accumulation. Ice bed and wear slab temperatures will be verified using installed RTDs.

Acceptance Criteria

Average ice bed temperatures is <20°F with no individual temperature higher that 27°F.

Average wear slab temperature is <20°F.

Ice condenser inventory must meet or exceed the requirements of Technical Specification 3.6.5.1.

Ice condenser top deck doors are operational.

Ice condenser intermediate deck doors are operational.

Ice condenser lower inlet doors are operational.

Ice condenser floor drains are operational.

Ice condenser flow channels are open and unblocked.

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14.4.26 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

Reactor building structures are complete with all normal divider barrier seals installed.

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.

Acceptance Criteria

Total divider barrier leakage area is less than or equal to the maximum allowable bypass leakage area specified in FSAR Section 6.2.1.1.3.

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14.4.27 Electric Hydrogen Recombiner Functional Test

Abstract

Purpose

To demonstrate the capability of each electric hydrogen recombiner to achieve recombination temperatures at an air flow equal to or greater than the minimum air flow assumed in Chapter 6 of the FSAR. The test also demonstrates the proper functioning of controls, instrumentation, and indications necessary for post-accident operation.

Prerequisites

The hydrogen recombiners and associated controls are functional to the extent required to test the system.

Test Method

The electric hydrogen recombiners will be energized. Minimum acceptable heater sheath heatup rate required in order to satisfy Technical Specifications surveillance requirements will be verified. The capability of the heaters to maintain a temperature in excess of the recombination temperature as measured on the heater sheath will be verified. Air flow to each recombiner will then be measured. Following completion of the heatup test, heater resistance to ground will be verified. The results of the heatup test will be used to establish a reference power setting for use in station operating procedures.

Acceptance Criteria

A flow rate greater than or equal to the value assumed in the FSAR analysis is verified. Heater sheath heatup rate satisfies the surveillance requirement of Technical Specifications. The ability to achieve and maintain heater sheath temperatures above the hydrogen recombination temperature is verified. All controls and indications which performs a safety-related function are verified to operate as specified in Duke Power Company Design Engineering Department system descriptions, and post-heatup continuity and resistance to ground checks are satisfactory.

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14.4.28 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. To demonstrate the capability to obtain the necessary balanced flows to the Reactor Coolant System loops with the safety injection pumps running in hot leg or cold leg recirculation.

Prerequisites

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 characteristic curve. The safety injection pumps are each run in both hot leg and cold leg recirculation modes. Flows to each branch are balanced, with each branch flow within the required range.

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 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. 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 is also verified at this time. Accumulator check valve operation is verified during cooldown from Hot Functional Testing.

Acceptance Criteria

Pump flow vs. head performance meets or exceeds the error adjusted acceptance head-capacity curve values assumed in the FSAR Chapter 15 analyses. Automatic valve operation and proper sequencing are verified. Level, flow and pressure instruments are set at the specified points in accordance with Westinghouse specifications, and actuate at those setpoints.

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

Safety injection pump branch injection line flows in both hot leg and cold leg recirculation are within the bounds assumed in the Chapter 15 analysis.

Centrifugal charging pump cold leg injection flow is within the range assumed in the Chapter 15 analysis.

Residual heat removal pump flow is less than the upper bound assumed in the Chapter 15 analysis, for hot leg recirculation and cold leg recirculation, in order to assure minimum NPSH.

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14.4.29 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 systems 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 sub-systems. 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 which originates an ESF signal is measured either separately or as a part of the channel response time.

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. This testing provides response timing of valve and pump operation (using permanently installed 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 essential load groups as outlined in Regulatory Guide 1.41. The test will be performed so that each diesel generator will be required to supply the most severe startup loads in the loading sequence, to assure the capability to carry full accident loads. Verification that all ESF components respond as designed upon reset of the initiating signal is performed.

Acceptance Criteria

Upon initiating of each actuating signal, proper system lineup and sequencing of valves, is accomplished within the time requirements of Catawba Technical Specification 4.6.3.2. All components start and operate as required to fulfill their ESF function. All components remain in their safety position following reset of the initiating signal.

Response times of active components are within the limits assumed in the plant safety analysis.

Buses not under test are absent of voltage.

14.4.30 Deleted

"HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED"

14.4.31 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. Also, Containment Pressure Control Cabinet annunciator is verified on loss of control power.

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 to the spray headers and to verify the pump head curve. Proper operation of the controls and interlocks associated with valves relied on to effect a transfer to the recirculation mode is demonstrated. Interlocks associated with the 0.25 psid permissive are verified to function as designed.

Proper spray nozzle performance and orientation is visually verified by blowing air through the spray ring headers and nozzles and observing the flow from the nozzles.

An unobstructed flow path is verified by the overlapping of the water flow test and the air test at the headers. Power is isolated to both trains of the Containment Pressure Control Cabinets to verify Control Room annunciators.

Acceptance Criteria

Flow nozzles are unrestricted.

Pump head vs. flow performance meets or exceeds the manufacturer's performance curve, within the error of the measurement. Pump performance in recirculation mode meets or exceeds the requirements of Technical Specification 4.6.2.b.

Interlocks which operate or prohibit operation of valves or components based upon the position of valves or containment pressure are verified to operate as designed.

System response to high-high containment pressure logic is verified during the ESF Functional Test.

Control Room annunciators actuate when control power is isolated to the Containment Pressure Control Cabinets.

"HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED"

14.4.32 Spent Fuel Cooling System

Abstract

Purpose

To demonstrate the capability of the system to provide the proper flow paths and flow rates required to remove decay heat from the Spent Fuel Pool. The purification capability of the system is verified by demonstrating the proper purification flow paths and flow rates.

Prerequisites

The component cooling water system is operational to the extent required to operate the Spent Fuel Cooling System.

Test Method

The spent fuel cooling pipe anti-suction holes are visually verified to be free of obstructions. Flow Paths and Flow Rates are verified for each of two cooling paths from the fuel pool through the pumps, heat exchangers, and returning to the Spent Fuel Pool. Proper operation of the Spent Fuel Pool purification and skimmer loops is also demonstrated by verifying proper flow paths. Operation of the spent fuel pool low level alarm at the proper setpoint is verified.

Acceptance Criteria

The specified flow paths are verified.

Spent fuel cooling pump performance meets or exceeds design values listed in FSAR Section 9.1.3. Spent fuel pool low level alarm actuates at a level equal to or higher than the value assumed in FSAR Section 15.7.4.

The anti-siphon holes are free of obstructions.

"HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED"

14.4.33 Fuel Handling Area Ventilation System Functional Test

Abstract

Purpose

To demonstrate the ability of the system to maintain the fuel handling and storage building at slightly less than atmospheric pressure, to control airborne activity, and to maintain a suitable temperature in the area.

Prerequisites

The system is operable to the extent required to conduct this test. The unit vent is capable of receiving air flow from the system.

Test Method

The system is operated in the normal filter train bypass mode. The ability of the system to automatically direct air flow through the filter trains upon a high radiation level in the exhaust duct system is demonstrated. The pressure in the fuel handling area is measured. The ability of the system to provide cooling and heating of the area is demonstrated by changing the temperature error signal.

Acceptance Criteria

1. *Each train operating independently of the other train, is capable of achieving a flow rate of 33,130 cfm ± 10% when tested per the requirements of ANSI N510-1980.*

2. *Satisfactory performance of all components, controls, alarms, and interlocks required in order for the system to fulfill its required function, as described in FSAR Section 9.4.2, is demonstrated.*
3. *HEPA filter banks demonstrated an efficiency of greater than or equal to 99.0% when they are tested in-place in accordance with ANSI N510-1980 while operating each train at a flow rate of 33,130 cfm ± 10%.*
4. *Laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Rev. 2 meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Rev. 2.*
5. *Charcoal absorbers remove greater than or equal to 99.0% of a halogenated hydrocarbon refrigerant test gas when they are tested in accordance with ANSI N510-1980 while operating each train at a flow rate of 33,130 cfm ± 10%.*
6. *The Fuel Handling Area Ventilation System demonstrates the ability to achieve a negative pressure of greater than or equal to 0.25 in W.G. within the Spent Fuel Storage Pool area relative to the outside atmosphere.*
7. *The Fuel Handling Area Ventilation System responds to changes in the temperature error signal by providing heating or cooling as appropriate, to maintain the set temperature in the fuel handling area.*

14.4.34 Radiation Monitoring System Functional Test

"HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED"

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 check sources provided with the system. Isolation features are verified to operate upon simulated high radiation signals. Ability to obtain samples is verified.

Acceptance Criteria

System channels respond to check sources in agreement with primary calibration data. Alarms function in accordance with Duke Power Company Design Engineering Department System Descriptions. Sample flow rates of the Reactor Coolant Radiation Monitor are verified to be low enough to allow adequate decay time of sample liquid as specified by Duke Power Company Design Engineering Department. Isolation features operate upon simulation of high radiation signals.

"HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED"

14.4.35 Piping System Vibration Test

<u>*Abstract*</u>

<u>*Purpose*</u>

To verify that piping layout and support/restraints are adequate to withstand normal transients without damage in the piping systems identified in FSAR Table 3-85.

To demonstrate that flow induced vibration is sufficiently small to cause no fatigue or stress failures in the piping systems identified in FSAR Table 3-84.

<u>*Prerequisites*</u>

1. *System components and piping supports have been installed in accordance with design drawings for system to be tested.*
2. *System piping has been installed in accordance with design drawings for system to be tested.*
3. *Hot Functional Testing and/or Pre-critical Heat-up for power escalation is underway.*
4. *System piping has been filled for normal operation.*

<u>*Test Method*</u>

1. *Steady State Vibration Testing*

   The system is placed in normal operating mode. A visual inspection is performed during a walk-through of the piping system. Points on piping systems with greatest observed displacements will be selected for measurement of piping velocity.

   This velocity reading is then compared with acceptance criteria based on the piping material (carbon or stainless steel). If the unfiltered velocity reading exceeds the acceptance criteria, the location is noted along with the thermal and hydraulic conditions of the system at the time the measurement was taken. A copy of the test data sheet, the flow diagram and the piping isometric drawing with the unacceptable piping outlined is forwarded to the Operations Analysis Group of Design Engineering for evaluation and recommendations.

2. *Transient Vibration Testing*

   Inspections of piping systems will be performed to verify that acceptance criteria is met for all systems and transients. In addition, vibration measurements will be performed at points of highest expected vibration induced stress to identify any unacceptable vibration in Reactor Coolant System during reactor coolant pump starts and trips.

<u>*Acceptance Criteria*</u>

1. *Steady State Vibration Testing*

   Acceptance criteria are based on conservatively estimated stresses which are derived from measured velocities and conservatively assumed mode shapes.

2. *Transient Vibration Testing*

   a. *No permanent deformation or damage in any system, structure, or component important to nuclear safety is observed.*
   b. *All suppressors and restraints respond within their allowable ranges, between stops or with indicators on scale.*

c. *The measured piping vibration for Reactor Coolant System during reactor coolant pump starts and trips do not exceed the values specified by Duke Power Company Design Engineering Department.*

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14.4.36 Nuclear Service Water Structure Ventilation System Functional Test

Abstract

Purpose

To verify that the Nuclear Service Water Structure Ventilation System can maintain the space temperature between 55°F and 104°F at design conditions.

Prerequisites

The structure and system must be complete to the extent necessary to perform the test. For the summer heat load test, the Nuclear Service Water pumps must be operable.

Test Method

The ventilation system will be operated at times when the external conditions are expected to approach the two (2) external design day conditions, 10°F and 95°F. Data will be recorded to verify that the internal environment is maintained within it's acceptable range. If the external design day conditions are not reached, the internal versus external temperature data taken during the test will be used to extrapolate to find the internal temperature which would have been reached at the design external conditions.

Design Hot Day testing will not be done to Unit Two "A" and "B" Train, and design Cold Day testing will not be done to Unit One "A" Train and Unit Two "A" and "B" Train. Instead, fan and unit heater performance data will be taken and compared with acceptable performance on either train of Unit One for Hot Day capabilities and with Unit One "B" Train for Cold Day capabilities.

Acceptance Criteria

The nuclear service water pump structure internal temperature remains between 55°F and 104°F at both the external design day conditions of 10°F and 95°F.

For those trains in which design day testing is not being done, fan and heater performance data is > -10% of the tested train.

"HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED"

14.4.37 Electrical Load Capacity Test

Abstract

Purpose

To verify the load carrying capability of the unit auxiliary transformers and 6900 volt switchgear, the 13.8 KV switchgear, 13.8 KV/600V transformers, 6900V/600V transformers, 20.9 KV/13.8 KV transformers, 600 VAC motor control and load centers.

Prerequisites

Systems are completed to the extent necessary to allow the necessary loads to be placed on the Auxiliary Power System.

Test Method

One 70 MVA, 20.9/6.9 KV auxiliary transformer will be connected to two 6900 volt switchgear assemblies, with the other sources to these assemblies disconnected and the tie-breaker closed. The loads on the two 6900 volt switchgear assemblies will be established to approximate maximum load conditions, by loading the 600 volt systems and switchgear and load centers back through the 13.8 KV, 6900 volt switchgear through the 20.0 KV/6.9 KV and 20.9 KV/13.8 KV transformers. Current measurements will then be taken to verify that primary currents on the transformers and incoming breaker currents at motor control centers do not exceed nameplate ratings and design ratings, respectively.

Acceptance Criteria

The transformer primary current readings taken do not exceed manufacturer's nameplate ratings. Incoming breaker currents for load centers and motor control centers tested do not exceed design ratings as specified in Duke Power Company Design Engineering Department System Descriptions.

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14.4.38 6900 Volt Auxiliary Power System Preoperational Test

Abstract

Purpose

To verify the proper operation of the manual closing and tripping functions of both incoming breakers, the tie breaker and the transformer feeder breaker in response to loss of one power source to the 6900 volt switchgear assemblies.

Prerequisites

The 6900 volt normal auxiliary power system is complete.

Test Method

Each 6900 volt switchgear assembly will be tested by manually closing and tripping normal incoming breakers and tie-breakers to verify proper operation and ability to perform a "hot-bus" transfer. The automatic transfer capability will be tested by simulating an undervoltage signal and verifying that the normal breaker opens and the tie-breaker closes. The test will also verify the actuation of the normal supply bus undervoltage alarm upon the initiation of the simulated undervoltage.

Acceptance Criteria

Manual closing and tripping operations and "hot-bus" transfers are accomplished without causing unintended breaker trips. Automatic transfer occurs and the normal supply bus undervoltage alarm actuates upon receipt of the simulated bus undervoltage signal.

"HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED"

14.4.39 Electrical Penetration O-Ring Seal Leak Rate Test

Abstract

Purpose

To verify the integrity of the electrical penetration o-ring seals, and to verify that a summation of the type B leak rate test results does not exceed the limits of 10CFR 50 App. J.

Prerequisites

Electrical penetrations must be complete with no identified exceptions or discrepancies which would affect the test.

Test Method

The volume between the o-ring seals is pressurized with clean dry air to containment post accident pressure. The pressure is allowed to stabilize and then the make up flow necessary to maintain test pressure is measured to determine the leak rate.

Acceptance Criteria

The sum of all type B and C leak rate tests corrected for instrument error must be less than or equal to the value specified in Technical Specification 3.6.1.2.b.

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14.4.40 600 VAC Power System Preoperational Test

Abstract

Purpose

To verify proper operation of breakers, interlocks, and alarms associated with the 600 VAC normal auxiliary power system and 600VA station normal auxiliary power system. To verify functioning and operability of transfers from normal to standby sources.

Prerequisites

Normal 600 VAC power systems are completed with no deficiencies or deviations which would affect the performance of the test.

Test Method

Manual operations and transfers are tested to verify operability. Automatic transfers are tested by simulating initiating conditions or signals. Associated interlocks and alarms are tested to verify proper operation.

Acceptance Criteria

Manual operation of devices and manual transfers are accomplished. Automatic transfers operate in accordance with Duke Power Company Design Engineering Department System Descriptions. All interlocks and alarms function properly as specified in the appropriate Duke Power Company Design Engineering Department System Descriptions.

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14.4.41 Emergency AC Power Systems Preoperational Test

Abstract

Purpose

To demonstrate the proper operation of the essential 4160 volt, 600 volt and 125 volt A.C. power systems. To demonstrate proper operation of feeder breakers, interlocks, and alarms. To verify proper voltages at load centers during operation.

Prerequisites

The systems to be tested are completed with no outstanding deficiencies or discrepancies which could affect the test.

Test Method

For each system, the feeder breakers are operated manually, associated interlocks and alarms are verified to operate when appropriate conditions are simulated or reached during the test, voltages at load centers are measured to assure proper operation within the design range.

Acceptance Criteria

Feeder breakers, interlocks and alarms which perform a safety-related function operate in accordance with Duke Power Company Design Engineering Department System Descriptions for the appropriate systems. Voltages measured at load centers or panels are within the limits specified by Duke Power Company Design Engineering Department for the appropriate system.

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14.4.42 250/125 VDC Auxiliary Power System Preoperational Test

Abstract

Purpose

To verify the proper operation of breakers interlocks and alarms on the 250 VDC and 125 VDC auxiliary power systems. To demonstrate proper operation of the battery chargers.

Prerequisites

Systems are complete, and sufficient DC loads are available to allow performance of the test.

Test Method

Feeder breakers are operated. Alarms and interlocks are verified to operate when appropriate conditions are reached or simulated during the test. Bus voltages are measured to assure proper voltage. Voltage from chargers is verified. Ground detection systems are tested by the use of test circuits.

Acceptance Criteria

Protective devices are verified to be operable. Alarms and interlocks operate properly as described in Duke Power Company Design Engineering Department System Descriptions. Voltages measured on DC buses and battery charger outputs are within design limits as specified by Duke Power Company Design Engineering Department.

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14.4.43 Heat Tracing Systems Test

Abstract

Purpose

To demonstrate the ability of the heat tracing system to maintain proper temperature control in the various piping systems (liquid and solid wastes, chemical volume control and boron recycle).

Prerequisites

Heat tracing system installation and component checks completed. Associated systems completed to the extent necessary to allow conduct of this test.

Test Method

Energize heat tracing system.

Monitor temperatures maintained by each heat tracing circuit with the system in a static condition.

Induce negative temperature transients.

Verify primary and backup controllers energize and de-energize at setpoint temperatures.

Acceptance Criteria

Primary and backup circuits maintain temperature range of 150° to 200°F for 12% boric acid.

Primary and backup circuits maintain temperature range of 70° to 175°F for 4% boric acid.

Primary temperature controllers energize at 175 ± 5°F from decreasing temperatures and de-energize at 175 ± 5°F from increasing temperatures for 12% boric acid.

Primary temperature controllers energize at 85 ± 5°F from decreasing temperatures and de-energize at 85 ± 5°F from increasing temperatures for 4% boric acid.

Backup temperature controllers energize at 160 ± 5°F from decreasing temperatures and de-energize at 160 ± 5°F from increasing temperatures for 12% boric acid.

Backup temperature controllers energize at 70 ± 5°F from decreasing temperatures and de-energize at 70 ± 5°F from increasing temperatures for 4% boric acid.

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14.4.44 Containment Ventilation and Purge Functional Test

Abstract

Purpose

To demonstrate the capability of the Containment Ventilation System to provide containment air recirculation, control rod drive mechanism cooling and containment purging.

Prerequisites

A cooling water supply is available for the fan-cooling units of the system. For testing portions of the system as applicable, the control rod drive mechanisms are capable of being energized, and plant conditions are established as required.

Test Method

Actual expected building heat loads are simulated during Reactor Coolant System Hot Functional Testing and data is taken to demonstrate the capability of the Containment Ventilation System to provide for containment recirculation and heat removal, by testing operation of the fans, centrifugal water chillers and the cooling coils, and by ensuring adequate flow is delivered to components and areas inside Containment as required.

Data will also be taken to verify that the control rod drive mechanisms shroud ventilation units are capable of maintaining temperatures within the shroud within design limits.

The capability of the containment purge exhaust filtration units to provide filtration is verified by testing of the filtration units.

Proper operation of the containment purge supply and purge exhaust equipment is demonstrated in normal and refueling modes.

Proper operation of Containment Ventilation and Purge System instrumentation, interlocks, and alarms which perform a safety-related function is verified.

Acceptance Criteria

1. *The Containment Ventilation System components function in accordance with Duke Power Company Design Engineering Department System Descriptions. Adequate ventilation flow is provided to containment areas to maintain or limit temperatures to design valves. System interlocks, instrumentation and alarms operate as described in Duke Power Company Design Engineering Department System Descriptions.*
2. *HEPA filter banks demonstrated an efficiency of greater than or equal to 99.0% when they are tested in-place in accordance with ANSI N5101980 while operating the system at 12,500 cfm ± 10% per train.*
3. *Charcoal absorbers remove greater than or equal to 99.0% of a halogenated hydrocarbon refrigerant test gas when they are tested in accordance with ANSI N510-1980 while operating at 12,500 cfm ± 10% per train.*

"HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED"

14.4.45 Containment Air Release and Addition System Functional Test

Abstract

Purpose

To verify the proper operation of the fans and air addition and discharge valves. To verify the operation of interlocks and alarms associated with the system. To verify capability of the filtration units.

Prerequisites

System is complete with no discrepancies which would affect the test. For the portion of the test indicated in the test method, unit Reactor Coolant System Hot Functional Test and/or Power Escalation Testing is in progress.

Test Method

Containment Air Release Fans are verified as being able to provide design flow. In order to simulate the pressure differential caused by a normal filter train (since at the time of this portion of the test, filters will not be installed) an obstruction will be placed in the filter train.

Flow will be established from containment to the unit vent. Maximum ΔP from lower to upper containment created by system operation is designed to be less than the amount required to open the ice condenser doors. This will be verified by the absence of an annunciator alarm indicating that an ice condenser door is open. This verification will be performed during power escalation and/or hot functional testing.

During Hot Functional Testing Heat-up and Cooldown (Unit 7 only), the high and low pressure annunciator alarms will be verified and proper opening and closing of unit vent and air addition valves verified (This verification will be performed prior to HFT for Unit 2). The capability of filtration units will be tested in accordance with ANSI N510-1980.

During Hot Functional Testing and/or power escalation, the ability of the system to control containment pressure will be verified.

Acceptance Criteria

Fan capacity is 200 SCFM ± 20% with maximum normal filter pressure drop. All ice condenser doors remain closed with either containment air release train operating in the air release mode. Containment Pressure alarm is received at the high and low containment pressure setpoint ±

0.05 psig. Containment pressure air release valve and containment air addition valve close at the correct containment pressure ± 0.05 psig.

The filtration units demonstrate an efficiency of 99.0% or greater, when tested in accordance with ANSI N510-1980.

The system functions to maintain containment pressure within Technical Specifications limits.

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14.4.46 Seal Water Injection System Functional Test

Abstract

Purpose

To verify proper operation of the isolation valve seal water injection system, including interlocks and alarms. To measure the overall leakoff of the system.

Prerequisites

The system is complete with no identified discrepancies which could affect the test. Valves supplied by the system are installed and operable.

Test Method

The system alarms and interlocks which perform a safety related function are tested by operation of components or simulation of sensor signals. Overall system leakoff is determined by measuring the CIV Leakages in valve subsets and then totaling the subsets to obtain an overall average.

Acceptance Criteria

Alarms and interlocks function as specified by Duke Power Company Design Engineering Department. Total train leakoff does not exceed the following makeup capacity:

For train 1A, ≤ 1.3818 gpm with tank pressure ≥ 45.71 psig.

For train 1B, ≤ 1.3616 gpm with tank pressure ≥ 45.71 psig.

For train 2A, ≤ 1.5818 gpm with tank pressure ≥ 45.71 psig.

For train 2B, ≤ 1.4616 gpm with tank pressure ≥ 45.71 psig.

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14.4.47 Instrument Air System Functional Test

Abstract

Purpose

To demonstrate that the system can supply instrument quality air at the design capacity. The test will also verify the correct compressor start, loading/unloading and Station Air System backup pressure setpoints. The Containment Leak Rate Test desiccant air dryer discharge dewpoint temperature will be determined.

Prerequisites

The Instrument Air System is in normal operation.

Test Method

The start and loading/unloading pressure setpoints are verified with one compressor in "BASE," one in "STANDBY 1," and the third in "STANDBY 2." The system air pressure is lowered while pressures are recorded corresponding to compressor starts and loading. The system air pressure is then allowed to increase while pressures are recorded corresponding to the compressor's unloading. Setpoints are verified using this same procedure with compressors in each control combination.

Each compressor's flow capacity is determined by directing all the flow from the compressor through a calibrated flow orifice. The refrigerant air dryers and the CLRT air dryers discharge dewpoint temperature is determined with design air flow rate through the air dryers.

The Station Air System crossover valve is demonstrated to automatically open when Instrument Air System pressure is lowered to the design setpoint.

The Instrument Air System product air is verified to be of sufficient quality by testing of air samples taken off locations near the end of main supply headers, for a total of five samples. Samples are taken downstream of filter regulators supplying individual instrument groups. The samples are examined for particulate matter size and oil concentration.

Acceptance Criteria

The compressors start and load/unload in accordance with Duke Power Company Design Engineering Department at the correct pressures. Refrigerant and CLRT air dryers meet their maximum allowable discharge dewpoints with design flow rate. Station Air System crossover valve opens at the design setpoint ± 10%. The product air meets instrument air quality requirements as stated in the test procedure. The compressor performance meets or exceeds the acceptance flow rate specified by Duke Power Company Design Engineering Department.

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14.4.48 Plant Sump Test

Abstract

Purpose

To verify operability of instrumentation provided to detect high sump levels in the groundwater drainage sump, upper head injection room sump, containment sumps, incore instrument sumps and diesel generator room sumps.

Prerequisites

Systems are completed and instrumentation is installed.

Test Method

The operation of the sump level alarms will be verified by raising the water level in the sump to the appropriate level to actuate the alarm. In cases where this method is not feasible, the primary sensing device will be actuated to simulate the high level condition.

Acceptance Criteria

The level alarms actuate as specified in the Duke Power Company Design Engineering Department System Descriptions.

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14.4.49 Waste Gas System Functional Test

Abstract

Purpose

To demonstrate the operability of the Gaseous Waste Processing System including its capability to remove and process gases from specified sources including the volume control tank, boron recycle evaporator, reactor coolant drain tank, and waste evaporator.

Prerequisites

The system is complete, with no discrepancies which would affect the test. All necessary supporting equipment is operational.

Test Method

The system will be operated to verify the flow paths from the sources through the system. Alarms and interlocks which perform a safety-related function will be verified to operate properly. The ability of the hydrogen recombiner to combine hydrogen and oxygen will be verified by operation of the recombiner.

Acceptance Criteria

Flow paths are verified to be unblocked. Alarms and interlocks function as specified by Duke Power Company Design Engineering Department.

The hydrogen recombiner successfully combines hydrogen and oxygen when operated within normal limits.

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14.4.50 Auxiliary Building Filtered Exhaust and Shutdown Ventilation Test

Abstract

Purpose

To verify proper operation of alarms, interlocks and controls. To verify the capability of the filtration units to fulfill their design function.

Prerequisites

The system is complete with no outstanding discrepancies which would affect the test. Supporting systems are complete to the extent necessary to operate the system.

Test Method

The system will be operated in both normal and LOCA (Ss) modes. Flow rates will be verified during operation. Switchover on receipt of a simulated LOCA (Ss) signal will be verified. Proper operation of alarms and interlocks will be verified by simulation of the appropriate conditions or injection of simulated sensor signals. Filtration units will be tested to verify their capabilities in accordance with ANSI N510-1980.

Acceptance Criteria

System alarms and interlocks function as specified by Duke Power Company Design Engineering Department. Filtered exhaust flow rate is 30,000 cfm ± 10% per fan. System realigns to draw suction only from safety-related equipment rooms upon receipt of LOCA (Ss) signal.

HEPA filter banks demonstrated an efficiency of greater than or equal to 99.0% when they are tested in-place in accordance with ANSI N510-1980 while operating the system at a flow rate of 30,000 cfm ± 10% per fan.

Charcoal absorbers remove greater than or equal to 99.0% of a halogenated hydrocarbon refrigerant test gas when they are tested in accordance with ANSI N510-1980 while operating at a flow rate of 30,000 cfm ± 10% per fan.

For the interim flow balance for this system, the Unit 2 filtered exhaust fans will exhaust 30,000 cfm + 0% - 35%. Once the interim barrier is removed or the Unit 2 RCA is established the system balance will be changed prior to Unit 2 entering Mode 4. Exhaust flows will be 30,000 cfm ± 10% per fan.

14.4.51 Fuel Handling Equipment Test

Abstract

Purpose

To demonstrate the operability of fuel handling equipment. To demonstrate the operability of interlocks and alarms associated with the fuel handling equipment. To demonstrate the operability and leakage characteristics of the weir gate in the spent fuel pool.

Prerequisites

The equipment is complete and supporting systems are complete to the extent necessary to perform the test.

Test Method

New assembly handling fixtures, spent fuel handling grapples and tools for handling various types of core components are all checked by handling of dummy fuel assembly or components, or load test stand. The capability of each fuel handling tool or component handling to lift or support the full weight of the assembly or component which it would handle is verified. Fuel handling bridges are tested for operability by simulating fuel handling operations and movement within their respective access areas. Interlocks on the fuel handling bridges are verified to operate as designed. Interlocks associated with the fuel transfer system are verified. Transfer system operation in normal, interlock and bypass modes is performed. The weir gate seal is installed and the ability to maintain the transfer system area drained with the normal sump pump, while the spent fuel pool and transfer canal water levels are at normal levels is verified.

Acceptance Criteria

Interlocks function as specified in the Sterns-Rogers equipment manuals. Operability of tools and handling fixtures and support of full assembly component weight is in accordance with manufacturer's descriptions.

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14.4.52 Refueling Water System Functional Test

Abstract

Purpose

To demonstrate the operability of the refueling water storage tank heaters, in both automatic and manual modes. To demonstrate the operability of level, temperature and flow alarms.

Prerequisites

The refueling water storage tank, heaters and electrical circuits are complete with no outstanding exceptions which would affect the test.

Test Method

The operation of both sets of refueling water storage tank heaters is verified by energizing the heaters in each mode of operation. Current flow is verified to both sets of heaters. Control of the heaters in the automatic mode is verified by input of a test signal. The operation of the heaters is verified as this test signal is varied. Level, temperature and flow alarms are verified to operate in accordance with designs.

Acceptance Criteria

The heater banks are verified to energize and deenergize at the proper setpoints in each mode of operation as specified by Duke Power Company Design Engineering Department. The low recirculation flow, low recirculation line temperature, low refueling water storage tank temperature, low level, low-low level and puncture alarms all actuate as specified by Duke Power Company Design Engineering Department.

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14.4.53 Boron Recycle Functional Test

Abstract

Purpose

Verify the operability of alarms and annunciators which, without operator action, could result in an uncontrolled radioactive release or the spread of contaminated liquid to undesired areas. Verify the operability of all interlocks and controls whose failure could result in an uncontrolled radioactive release or the spread of contaminated liquid to undesired areas. Confirm the ability to transfer evaporator distillate and concentrates through their major flowpaths. Confirm the ability to obtain a sample from distillate and concentrates sample vessels. Confirm the ability of the evaporator to produce a distillate and a concentrates stream.

Prerequisites

The evaporator package and support systems are complete to the extent necessary to perform the test.

Test Method

The evaporator will be run in the normal operating configuration. Flow paths will be verified to be open. Alarms and interlocks are checked by varying conditions to actuate the appropriate alarms/interlocks or by simulation of a sensor signal.

Acceptance Criteria

All alarms and interlocks actuate as specified in the Westinghouse Limitations and Setpoints Manual or as specified by Duke Power Company Design Engineering Department, as appropriate. Flow paths are open. The controllers demonstrate the ability to return a process variable to a setpoint when in the "AUTOMATIC" mode. A machinist's stethoscope detects flow to the tank being tested. Concentrates and distillate sample vessels are disconnected and demonstrated to contain a sample. The refractometer monitor (0NBMT 5770) shows an upward change which indicates the boric acid is being concentrated. The CONDENSER LEVEL indicator shows that there is distillate in the Condenser.

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14.4.54 Radwaste Solidification System Functional Test

Abstract

Purpose

To demonstrate the operability of the solid waste subsystem for transporting evaporator concentrates and spent resins to the proper area for solidification. To demonstrate the operability of the miscellaneous solid waste compactor.

Prerequisites

Solid waste system and supporting systems are complete to the extent necessary to perform this test.

Test Method

Pumps necessary for transfer of liquids to the solidification area will be verified to provide required flow. Flowpaths to the solidification site will be verified to be open. Alarms, setpoints, and interlocks necessary to prevent the uncontrolled release or spread of radioactive materials will be verified to operate properly. The waste compactor will be run with simulated miscellaneous solid waste, to demonstrate the operability of the compactor.

Acceptance Criteria

Sufficient flow is verified by transporting simulated waste and resin to solidification site. Flow paths necessary for proper operation of the system are verified to be open. Alarms, setpoints and interlocks function as specified by the vendor, or as specified by Duke Power Company Design Engineering Department.

Solid waste compacter produces compacted waste containing no free liquids.

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14.4.55 Liquid Waste System Functional Test

Abstract

Purpose

To demonstrate the operability of the Liquid Waste System for collection, processing and recycling of liquid wastes and for preparation of liquid waste for release to the environment.

Prerequisites

Liquid waste system and supporting systems are complete to the extent necessary to perform the test.

Test Method

Pumps are run and verified to produce acceptable flow. Flowpaths between main components in the Liquid Waste System are verified to be open. Alarms, interlocks, and controls necessary to prevent the uncontrolled release or spread of radioactive material are verified to operate properly. Ability to draw samples from defined sample locations is verified.

Acceptance Criteria

Acceptable flow is verified by transporting liquid between main components. Flowpaths between major components are open. Samples are procured from defined sample locations. All alarms, setpoints, and interlocks tested operate as specified by Duke Power Company Design Engineering Department.

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14.4.56 Auxiliary Shutdown Panel Test

Purpose

To verify automatic valve realignment following transfer of control to auxiliary shutdown panel A (B). To demonstrate operability of auxiliary shutdown panel A (B) controls and isolation of control room following transfer of control to LOCAL. To demonstrate operability of control room controls and isolation of auxiliary shutdown panel A (B) following control transfer back to the control room.

To demonstrate that the unit can be operated from the auxiliary shutdown panels prior to loading fuel. During the Reactor Coolant System Hot Functional Test (HFT), from a hot standby condition, the ability to establish a heat transfer path to the ultimate heat sink using the Residual Heat Removal System and lowering the Reactor Coolant System temperature by 50°F is demonstrated. Instrumentation on the auxiliary shutdown panels is verified operable during this test.

Prerequisites

All systems interlocked or that can be controlled from auxiliary shutdown panel A (B) are available as required for this test.

For the demonstration portion during HFT, Hot Functional Testing is in progress with primary system at approximately 400°F.

Test Method

Prior to HFT, control is transferred to auxiliary shutdown panel A (B) and these controls are verified to be operable. All automatic interlocks are verified. Controls are verified by cycling valves and running Boric Acid Transfer Pump A (B). The remainder of the pumps and Pressurizer Heater Bank A (B) control circuits are verified operable with associated breakers in the "TEST" position. At the same time, main control room controls are verified to be isolated. Upon transfer back to the main control room, control is verified to be regained and auxiliary shutdown panel A (B) control is isolated. This is accomplished in the same manner as the previous section.

During HFT, with the primary system at approximately 400°F, control is transferred to the auxiliary shutdown panels. Also, Reactor Coolant temperature and pressure is lowered sufficiently to permit operation of the Residual Heat Removal System from the auxiliary shutdown panels. While using the Residual Heat Removal System the Reactor Coolant temperature is reduced at least 50°F.

Acceptance Criteria

For the pre-HFT portion of the test, valves will automatically realign to their proper position when control is transferred to auxiliary shutdown panel A (B). All auxiliary shutdown panel A (B) controls are demonstrated operable and main control room controls isolated with transfer switch in LOCAL. All main control room control is regained and auxiliary shutdown panel A (B) control isolated when control is transferred back to the main control room.

For the portion of the test conducted during HFT, auxiliary shutdown panels A and B, as well as the auxiliary feedwater turbine panel instrumentation is demonstrated to be operational. With control transferred to the auxiliary shutdown and turbine panels, (a) Reactor Coolant temperature and pressure is lower sufficiently to permit operation of the Residual Heat Removal

System and (b) the Reactor Coolant temperature is lowered at least 50°F using the Residual Heat Removal System.

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HISTORICAL INFORMATION IN ITALICS BELOW NOT REQUIRED TO BE REVISED

14.5 Initial Startup Testing

14.5.1 Initial Fuel Loading

Abstract

Purpose

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

Prerequisites

Testing prior to initial fuel loading is completed sufficiently to demonstrate the operability of required systems and components as defined in Technical Specifications. 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 circulation with at least one Residual Heat Removal Pump and is sufficient to assure K_{eff} less than or equal to 0.95 during fuel loading. Containment integrity is established as defined in Technical Specifications.

Test Method

Fuel assemblies containing specified control components are placed into the reactor vessel in accordance with the specified 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. A minimum count rate of 2 cps on each responding source range channel is required following loading of the primary source assemblies.

Acceptance Criteria

The core is assembled in accordance with the configuration specified in the Westinghouse design core loading report.

14.5.2 Moveable Incore Detector Functional Test

Abstract

Purpose

To assure proper alignment, indexing and operation of the moveable incore detector drive system and readout equipment. To assure proper operation of the incore Instrument System. This test is considered to be non-safety related.

Prerequisites

Reactor core is loaded. 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 hot standby condition.

Test Method

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 movement is verified.

Acceptance Criteria

The moveable incore detector drive system and readout equipment perform as defined in Chapter 7 and the Technical Specifications.

14.5.3 Incore Thermocouple and RTD Cross-Calibration

Abstract

Purpose

To determine the response characteristics of each RTD and the response characteristics and isothermal correction factor for each incore thermocouple. To demonstrate the proper operation of temperature readout and compensating equipment. This test is considered to be non-safety related. This test will be performed initially during hot functional testing, and will be repeated anytime RTD's are replaced following hot functional testing.

Prerequisites

The reactor is in the cold shutdown condition, or is in hot functional testing. Incore thermocouple checkout is satisfactorily completed. All RTD's are installed and have satisfactorily undergone continuity, resistance and alignment checks. Cold junction box RTD's and controllers are satisfactorily calibrated.

Test Method

As the unit is heated up from cold shutdown, or cold ambient conditions during hot functional testing isothermal conditions are established at selected intervals. At these isothermal plateaus resistance is measured and recorded for all RTD's[1]. The temperature variation between each RTD and the average of the RTD readings is calculated and recorded. The operation of remote and local temperature instrumentation is observed. Cold junction box temperatures are recorded at each plateau.

Acceptance Criteria

Response characteristics for each RTD are consistent with vendor calibration data. Isothermal calibration factors are recorded and in each case, when applied to the applicable incore thermocouple, give thermocouple output consistent with RTD data with the unit at hot zero power.

14.5.4 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 standby conditions over its entire range of travel.

Prerequisites

The reactor is at hot standby, 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

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[1] Temperatures are calculated for each RTD.

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 and inserted to its fully inserted position in discrete increments. Indication and alarms are observed for proper operation.

Acceptance Criteria

The rod position indication system for each rod cluster control assembly indicates the correct position, within ± 4 steps, over its entire length of travel, when compared to the group step counter.

14.5.5 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 and in hot standby conditions.

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 as defined in 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. Stationary gripper coil outputs and rod position indication outputs are connected to recording devices.

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 deenergization of the stationary winding of the rod cluster control assembly drive mechanism. Rod control cluster assemblies for which the initial drop time differs from the average of all the drop times for that condition will be re-tested a minimum of three additional times. Proper operation of the dashpot is verified by analysis of the drop data.

Acceptance Criteria

The time from release of the rod cluster control assembly until it reaches the top of the dashpot is less than the limits defined in Technical Specification 3.1.3.4.

Steady deceleration is observed through the lower constricted region (dashpot region) for each rod cluster control assembly. The longest and shortest rod drop times for repeated drops of rod control cluster assemblies (whose initial times differed from the average by more than two standard deviations) do not differ by more than 0.04 seconds.

14.5.6 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 standby condition, to adjust Rod Control System bank-overlap setpoints and to demonstrate proper system control and indication.

Prerequisites

Initial core loading is complete. 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 shutdown or hot standby condition as dictated by the specific test requirements. Containment integrity is established as required in Technical Specifications. Nuclear Instrumentation and Rod Position Indication Systems are operable.

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 standby 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.

Acceptance Criteria

Cable connections and identifications are correct. Bank overlap controls function to sequence withdrawal of banks in accordance with the settings of the controls. Step counter accumulate changes in rod position as rods are moved in and out. Cold resistance values are within the limits specified in the vendor technical manual.

14.5.7 Rod Drive Mechanism Timing Test

Abstract

Purpose

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

Prerequisites

Initial core loading is complete. The reactor vessel upper internals are installed, the reactor vessel head is installed with studs tensioned, each rod is latched and the Reactor Coolant System is filled and vented. The reactor is in the cold shutdown or hot standby condition, as dictated by the test requirements. Containment integrity is established as defined in the Technical Specifications. Cold condition Rod Control System alignment has been completed. Nuclear Instrumentation System is operable.

Test Method

With the reactor in the cold shutdown condition, the timing for each slave cycler is set, measured and reset as necessary. Each 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 standby condition.

Acceptance Criteria

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

14.5.8 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 operating.

Test Method

The output voltage of each NC loop differential pressure transmitter is measured using a digital voltmeter. The output voltages are averaged and converted to equivalent differential pressure which is then converted to flow using a vendor supplied, plant specific graph. The loop flows are summed to give the total system flow. The flow transmitters are adjusted for 100 percent flow at normal operating conditions and zero output at zero flow.

Acceptance Criteria

Reactor coolant system flow (with allowances for measurement uncertainties) is greater than the value specified in the Technical Specifications and less than the mechanical design flow as stated in Chapter 5.

Following adjustment, the flow transmitters yield 100 percent flow signal at normal operating conditions and zero output at zero flow conditions, within the tolerances specified by the vendor.

14.5.9 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.

Test Method

Flow coastdown will be measured for the single loop loss of flow by tripping one of four reactor coolant pumps and monitoring flow using the elbow tap differential pressure cells. Delay times for several protective functions are measured using a strip chart recorder.

Flow coastdown will also be measured for a complete loss of flow. All four pumps will be tripped simultaneously using the Reactor Coolant Pump Electrical Monitoring System. Flow will be measured using the same method as the partial loss of flow case.

Acceptance Criteria

The core flow decrease for the 4 of 4 coastdown transient is slower than that assumed in Figure 15-60. Time delays from actuation to low flow trip, undervoltage trip and underfrequency trip actuation are less than or equal to those assumed in Chapter 15.

14.5.10 RTD Bypass Flow Verification

Abstract

Purpose

To determine the flowrate necessary to achieve the required reactor coolant transport time in each RTD bypass loop (time from NC loop to last RTD well), 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

For portions of the test other than the piping measurements, the reactor is in the hot standby 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 or equal to 1.0 second transport time. The total bypass flowrate is then measured with both loops in service, and 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(s) are reached.

Acceptance Criteria

The RTD bypass loop transport time is less than or equal to 1.0 second or if greater, is noted for comparison with results from unit trip testing at 100 percent power. The low flow alarm actuates at 90 ± 2.0 percent of full bypass loop flow.

14.5.11 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 shutdown and control assembly banks are fully inserted. All source and intermediate range channels are operable. A minimum count rate of ½ cps will be verified on both source range channels. The intermediate range high flux trip setpoints are set at 1.0 x 10^{-4} amps. Reactor coolant system leak test has been performed a no-load conditions, and leakage is within Technical Specification limits.

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 100 pcm worth remaining. Reactor Coolant System boron dilution is commenced at a rate of approximately 1000 pcm per hour with the Reactor Coolant System boron concentration being sampled at fifteen minute intervals. When the inverse neutron count rate ratio is approximately 0.2, 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 300 pcm 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 a safe and orderly manner.

14.5.12 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.*

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 of at least one decade exists between the source-intermediate range instrumentation.*

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. For Unit 2, Rod Swap technique may be utilized for determination of control rod worths. 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 vesus 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. (Unit 1 only)*
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 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. (Unit 1 only)*

Acceptance Criteria

1. *Nuclear instrumentation overlap between source and intermediate ranges is at least one decade prior to the onset of nuclear heat.*
2. *The all rods out, critical boron concentration is within ±50 ppm of the predicted value given in the Westinghouse Core Design Report.*
3. *The all-rods-out moderator temperature coefficient is negative or rod withdrawal limits have been established in accordance with Technical Specifications.*
4. *Differential boron worth, over the range measured, is within ±10% of the value given in the Westinghouse Core Design Report.*
5. *Control rod worth measurements verify the assumptions in establishing the insertion limits defined in Technical Specifications.*
6. *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 described in Chapter 15.*

14.5.13 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 not less than 25 percent power. The pressurizer safety and relief valves are operable. Pressurizer level and pressure instrumentation have 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 manner, 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 the limits given in the Westinghouse Limitations and Setpoints Manual.

14.5.14 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 25 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 Tavg at the reference value within ± 1.5°F at steady-state operation. The Rod Control System returns Tavg to within ± 1.5°F of the reference value without manual intervention from both of the initial temperature conditions given in the test method.

14.5.15 Core Power Distribution Test

Abstract

Purpose

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

Prerequisites

Reactor is critical at a steady state power level as specified by procedure (approximately 5%, less than 30%, 50%, 75%, 100%). Incore instrumentation system functional tests are complete and the systems operable. Computer systems are operable as necessary for incore map data gathering.

Test Method

With the reactor power level stable, incore flux maps are obtained and processed.

Acceptance Criteria

Core peaking factors are consistent with those predicted in the core design report and less than those specified in Technical Specifications.

14.5.16 Unit Load Steady State Test

Abstract

Purpose

To measure NSSS parameters and appropriate secondary system parameters at steady state conditions at specified power levels to compare with design predictions. To demonstrate the capability of major plant control systems to maintain equipment and system limits. To verify the proper operation of major plant systems at power. This test is considered to be non-safety related.

Prerequisites

The unit is at a steady state power level 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 required. Loose parts monitoring system baseline data is recorded, except for performance of this test at 90% power.

Acceptance Criteria

The NSSS exhibits stable operation, as determined by verifying that recorded parameters fall within the limits specified by Westinghouse and Duke Power Company Design Engineering Department System Descriptions for the appropriate power level.

Baseline data for the loose parts monitoring system is obtained.

Major plant systems operate within design limits as defined by Duke Power Company Design Engineering Department Systems and Controls descriptions.

Reactor coolant flows, levels, temperatures are within Technical Specification limits and anomalous indication or operation of instruments is identified and corrected as required by Technical Specifications.

14.5.17 Radiation Shielding Survey

Abstract

Purpose

To measure neutron and gamma 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, neutron and gamma 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. High radiation areas will be posted and controlled.

14.5.18 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.*

2. The power range channel gains reflect actual power levels, within ± 2% of rated thermal power.

3. At least one decade of overlap is observed between intermediate and power range channel outputs.

14.5.19 Process and Effluent Radiation Monitor Test

Abstract

Purpose

To verify the performance of the process and effluent monitors under actual discharge conditions. This test is not required to be completed to proceed to the next testing plateau.

Prerequisites

The reactor has been operating for a time sufficient to generate representative effluent and process levels. The effluent and process 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. Grab samples are obtained and a similar analysis is performed for process monitors. Only effluent and process monitors which experience effluent and process levels in excess of the sensitivity of the monitor will be compared with the results of radiochemistry analysis.

Acceptance Criteria

The installed effluent and process monitors perform in accordance with design standards and the indicated radioactive content of the effluent or process flow correlates with analysis as recommended by ANSI Standards N13.1-1969 and N320-1979.

14.5.20 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 a stable power condition with rods in the specified maneuvering band. The instrumentation necessary for collection of data is installed, calibrated and operable.

Test Method

Initial data is taken. With the turbine and reactor controls in manual, the turbine load is decreased then increased. Data is recorded during and after the load maneuver and used to infer a measured doppler coefficient verification factor. This factor is compared to a vendor supplied predicted doppler verification factor.

Acceptance Criteria

The inferred measured doppler coefficient verification factor agrees with predicted values as specified by the vendor.

14.5.21 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 off-sets and f(ΔI) functions. To verify the ability of Rod Control System to control induced xenon transients. This test is also perform on a quarterly basis during power operation to demonstrate continued capability to control the induced Xenon transients throughout the life of the reactor core.

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 controlling bank is inserted and/or withdrawn compensating with boron dilution/addition, to maintain constant power and axial offset. The required incorexcore correlation data are obtained during the transient. The f(ΔI) 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

Based on the data obtained, the f(ΔI) function for overpower and overtemperature differential temperature setpoints is set in accordance with limits in Technical Specifications. Rod Control system exhibits capability of damping the induced Xenon transient.

14.5.22 Below-Bank Rod Test (Unit 1 only) 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 (RCCA) 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. The unit is at the 50 percent power level for testing.

Test Method

Initial data is obtained with the reactor at stable conditions. The lift coils of all rods in Bank D except for the most reactive RCCA are disconnected. The most reactive RCCA is diluted to the fully inserted position while taking data. When all data has been taken, the RCCA is borated back to its bank position.

Acceptance Criteria

Hot channel factors, corrected for measurement uncertainty, are less than or equal to those assumed in Chapter 15, with the RCCA fully inserted and with the RCCA partially inserted at positions between it's bank position and full insertion. Core peaking factors remain within technical specification limits with the rod control cluster assembly less than or equal to fifteen inches below its bank. Incore instrumentation is demonstrated capable of detecting the power maldistribution caused by the misaligned rod cluster control assembly.

14.5.23 Pseudo Rod Ejection Test (Unit 1 Only)

Abstract

Purpose

To determine ejected rod worth and hot channel factors. 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 30 percent reactor power level. Reactor is a 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 to maintain nominal system criticality during a continuous, controlled Reactor Coolant System boration/dilution. The flux level 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 and moveable detector data provide information to relate core power distribution to 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 (Chapter 15). Core peaking factors remain within technical specification limits with the rod cluster control assembly less than or equal to fifteen inches above its bank. Incore and/or excore instrumentation is demonstrated capable of detecting the power maldistribution caused by the misaligned rod cluster control assembly per Chapter 15.

14.5.24 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 recording devices.

Test Method

Turbine 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 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, main steam relief or safety valves lift. No operator action is required to restore conditions to steady state. Parameters affected by the load change do not incur sustained or divergent oscillations.

14.5.25 Unit Loss of Electrical Load Test

Abstract

Purpose

To demonstrate the ability of the unit to sustain a net electrical loss of load without exceeding turbine design overspeed conditions. To evaluate interaction between control systems and to evaluate system responses to the transient to determine if any control system changes are required to improve transient response. To demonstrate proper Steam Dump Control System response.

Prerequisites

The unit is at steady state full power with rods within their respective maneuvering bands. Pressurizer and main steam safety valves are operable. The following systems are in the automatic mode:

1. *Reactor Rod Control*
2. *Pressurizer Pressure Control*
3. *Pressurizer Level Control*
4. *Steam Dump Control*
5. *Feedwater Pump Speed Control*
6. *Steam Generator Level Control*

Pertinent plant parameters (such as turbine speed, feedwater and steam flows, flux, steam generator and pressurizer levels, feedwater pump speed) are connected to recording devices.

Test Method

Both main generator output breakers are manually placed in the tripped position to simulate net loss of electrical load. Pertinent plant parameters are recorded and the data evaluated to determine control system responses to the transient.

Acceptance Criteria

The turbine does not exceed the "Electrical" backup trip (111.5%). Safety injection is not initiated. Pressurizer safety valves do not lift. No reactor coolant system safety limits as given in Technical Specification Section Section 2.1 are exceeded.

14.5.26 Turbine Trip Test

Abstract

Purpose

To demonstrate the ability of the unit to sustain a trip of the main turbine generator from approximately 68% power. To evaluate interaction between control systems and to evaluate system responses to the transient to determine if any control system changes are required to improve transient response. This test is not required to be completed to escalate to the next testing plateau.

Prerequisites

The unit is at a steady state power level, just below the P-9 setpoint, with rods within their respective maneuvering bands. Pressurizer and main steam safety valves are operable. The following systems are in the automatic mode:

1. *Reactor Rod Control*
2. *Pressurizer Pressure Control*
3. *Pressurizer Level Control*
4. *Steam Dump Control*
5. *Feedwater Pump Speed Control*
6. *Steam Generator Level Control*

Pertinent plant parameters (such as turbine speed, feedwater and steam flows, flux, steam generator and pressurizer levels, feedwater pump speed) are connected to recording devices.

Test Method

The main turbine generator is manually 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 injections are not initiated. Main steam and pressurizer safety valves do not lift. No reactor coolant system safety limits given in Technical Specifications Section 2.1 are exceeded.

14.5.27 Feedwater Temperature Variation Test (Unit 1 Only)

Abstract

Purpose

To demonstrate the ability of the unit to sustain a reduction in feedwater temperature from opening a feedwater heater train bypass valve. To evaluate interaction between control systems and to evaluate system responses to the transient to determine if any control system changes are required to improve transient response. This test is not required to be completed to escalate to the next testing plateau.

Prerequisites

The unit is at steady state conditions at the specified power level. Pressurizer and main steam safety valves are operable. The following systems are in the automatic mode:

1. *Reactor Rod Control*
2. *Pressurizer Pressure Control*
3. *Pressurizer Level Control*
4. *Steam Dump Control*
5. *Feedwater Pump Speed Control*
6. *Steam Generator Level Control*

Pertinent plant parameters (such as feedwater temperature, feedwater and steam flows, flux, steam generator and pressurizer levels, feedwater pump speeds) are connected to recording devices.

Test Method

The A-B heater train bypass valve is opened. Pertinent plant parameters are recorded and the data evaluated to determine control system responses to the transient.

Acceptance Criteria

Turbine generator and reactor do not trip due to Reactor Coolant System transients. Safety injection is not initiated. Main steam and pressurizer safety valves do not lift. No sustained or divergent oscillations occur in the parameters affected by the feedwater temperature variation.

14.5.28 Loss of Control Room Test

Abstract

Purpose

To demonstrate that the unit can be brought to hot standby conditions from a moderate power level using Auxiliary Shutdown Panel controls and only the minimum shift crew required for operation. To demonstrate that hot standby conditions can be maintained from outside the control room. This test is not required to be completed to escalate to the next testing plateau.

Prerequisites

Power escalation testing is in progress with the reactor at a moderate power level (10-25%) sufficiently high that plant systems are in normal configuration with the turbine - generator in operation. All personnel in the control room area not actively participating in the test as well as those performing the test are identified and their authority and responsibility documented in the test procedure.

Test Method

The control room is evacuated of normal operating personnel following the Normal Loss of Control Room operating procedure. Additional operators, not actively participating in the test, remain in the control room to monitor unit behavior. The reactor is tripped and the unit is brought to hot standby conditions using local controls and indications and maintained at this condition for at least 30 minutes. The Reactor Coolant System temperature will then be reduced by at least 50°F. Control is then transferred back to the control room and power escalation testing continued.

Acceptance Criteria

The unit is satisfactorily brought to hot standby conditions from a moderate power level and maintained at this condition for at least 30 minutes from outside the control room. The Reactor Coolant System temperature can be reduced by at least 50°F from outside the control room. Only the minimum number of personnel required to be assigned to the unit at any one time take an active part in this demonstration.

14.5.29 Station Blackout Test

Abstract

Purpose

To demonstrate the ability of the unit to sustain a turbine-generator trip following isolation of the offsite power distribution system. To evaluate the interaction between control systems and to evaluate system response to the transient. To verify that natural circulation can be established and used to remove core decay heat for a minimum of 30 minutes following a loss of offsite power sources and a partial loss of onsite sources. This test is not required to be completed to escalate to the next testing plateau.

Prerequisites

Unit is at stable conditions at a power level greater than 10 percent of rated generator load. Pressurizer and main steam safety valves are operable. The reactor control and protective systems, onsite emergency power and auxiliary feedwater systems are operable. Pertinent unit parameters (such as turbine speed, feedwater and steam flows, flux, steam generator and pressurizer levels, diesel generator frequency and voltage) are connected to recording devices.

Test Method

Selected portions of the onsite power distribution system will be isolated from offsite power sources concurrently with opening of the main generator headers. Reactor and turbine generator trip will occur from a minimum power level of 10% full power. The normal on site power distribution system will be sufficiently deenergized to demonstrate the following:

1. *Deenergization of the 4.16 KV Essential and Blackout busses with subsequent starting and sequencing of loads onto the emergency diesel generators.*
2. *Verification that DC power sources designed as vital or normal non-interruptable power supplies provide adequate operator indication and control to establish and maintain hot standby conditions following the transient.*
3. *Establishment of natural circulation following loss of forced flow with actual or simulated loss of normal onsite and offsite power sources.*
4. *Verification of auxiliary feedwater start and control of steam generator levels and steam dump control of steam generator pressure utilizing atmospheric steam dump capability.*
5. *Control of pressurizer level and pressure utilizing minimum heater capability available following a loss of offsite and normal onsite power sources.*

Normal station procedures will be used to the greatest extent possible in recovering from the transient.

Acceptance Criteria

The turbine generator and reactor trip; auxiliary feedwater pumps start; the unit is maintained in a safe hot standby condition for a minimum of 30 minutes utilizing natural circulation. Control rod trip into the core as a result of the loss of power to the rod control system.

14.5.30 Natural Circulation Verification Test (Unit 1 Only)

Abstract

Purpose

To demonstrate the capability of the NSSS to remove sensible heat by natural circulation flow in the primary loop. To verify that pressurizer pressure and level control systems can respond automatically to a loss of forced circulation and can maintain reactor coolant pressure within acceptable limits. To verify that steam generator level and feedwater flow can be maintained under natural circulation conditions in order to maintain effective heat transfer from the reactor coolant system. To provide operator training to satisfy NUREG 0737 requirements.

Prerequisites

The reactor is critical at a power level of approximately 3% full power with all reactor coolant pumps in operation. Rod control is in manual with Bank D positioned to maintain a slightly negative isothermal temperature coefficient. Pressurizer pressure and level control are in automatic. Steam dump control is in the pressure control mode. Steam generator level is being maintained through use of the auxiliary feedwater header.

The intermediate and power range (low setpoint) high level reactor trips have been reduced to approximately 7% rated thermal power. UHI isolation valves have been gagged. Overtemperature and overpower ΔT reactor trip signals have been blocked.

Various Technical Specifications test exemptions are required for the conduct of this test. These special test exemptions are provided in Technical Specifications. Special operator action guidelines are provided by the test procedure to compensate for the blocking of various safety injection functions and reactor trips. The test is required to be performed at core burnups which ensure that no significant core decay heat levels are present.

Test Method

The test will be initiated by tripping all operating reactor coolant pumps. The establishment of natural circulation will be verified by observing the response of wide range hot and cold leg temperatures as well as core exit thermocouples. The response of pressurizer level and pressure will be observed. Steam generator level and pressure response will be monitored. During the performance of this test on Catawba Unit 1 only, the test will be repeated for each operating shift at Catawba or suitable simulator facility, for the purpose of initial operator training. Each RO and SRO will observe or participate in the initiation, detection and maintenance of natural circulation conditions during at least one of the test runs.

Acceptance Criteria

Core exit temperatures, loop ΔTs, and loop average temperatures do not exceed values specified by the NSSS vendor. ΔTs as determined by the hot leg wide range and core exit temperature indications when compared to the wide range cold leg temperatures stabilize and do not exceed limits supplied by the NSSS vendor. Steam generator and pressurizer levels are maintained above the levels recommended by the NSSS vendor. If data taken during first performance of test demonstrates acceptable performance, data does not need to be taken during subsequent operator training.

14.5.31 Pressurizer Functional Test

Abstract

Purpose

To establish the continuous spray flow rate, determine the effectiveness of the pressurizer normal control spray and of the pressurizer heaters, and verify the response time of the pressurizer power operated relief valves.

Prerequisites

The Reactor Coolant System is at hot standby temperature and pressure. 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 125°F temperature difference between the spray line and the pressurizer steam space.

To determine pressurizer heater and spray capability, the main 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 each spray valve individually and in parallel. 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.

With the Unit at normal operating no load temperature and pressure, each PORV shall be cycled for response time testing. The 2185 psig interlock closes the valve and original conditions are re-established.

This test is performed following initial fuel loading due to the need to establish the effectiveness of actual spray flow with core pressure drop acting as the driving head. This test is a prerequisite test for initial criticality.

Acceptance Criteria

For setting of continuous spray flow, the flow through each bypass valve is established such that the temperature difference between the spray line and the pressurizer steam space is less than 125°F.

For pressurizer PORV response times, each PORV response time is $\leq$3 seconds ($\leq$4 seconds for Unit 1, cycle 1).

For spray and heater response tests, the response to induced transients is within limits specified in vendor guidelines.

14.5.32 Support Systems Verification Test

Abstract

Purpose

To verify that temperatures within rooms containing engineered safety features pumps and motors are maintained within design limits during power operation by normal operation of the cooling systems serving those areas.

Prerequisites

Unit in power operation at the power level specified in the procedure.

Test Method

Temperature readings will be taken within the rooms in the auxiliary building which contain engineered safety features pumps. These readings will be compared with the design limits for these rooms.

Acceptance Criteria

Temperature readings do not exceed the design limits specified in FSAR Chapter 9.

14.5.33 Steam Generator Water Hammer Test

<u>*Abstract*</u>

<u>*Purpose*</u>

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

<u>*Prerequisites*</u>

Reactor power at less than or equal to 30%. Feedwater supply to the steam generator is through the Feedwater System with the feedwater isolation valve in the main feedwater line shut. Provisions are made for recording pertinent process signals during the test.

<u>*Test Method*</u>

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 associated process instrumentation are recorded during the transient. A post-test walk-through will be conducted to check for damage in the steam generator enclosures.

<u>*Acceptance Criteria*</u>

No pressure transients exceeding 50 psi (peak to peak) are noted during switchover. No deformation or damage to the steam generators, supports, feedwater piping or restraints is noted following the test.

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