13.5 STARTUP AND POWER TEST PROGRAM

This section presents a general description of the startup testing that was planned for Browns Ferry and has been retained in the FSAR as a historical reference only. This test description is not in conformance with Regulatory Guide 1.68 and should not be used as a model for future test programs. For a description of the retest program see Section 13.10.

There are numerous minor discrepancies between this description and the startup testing actually performed. An accurate description of the startup testing performed and the results of the testing are contained in three reports entitled "Summary Report of Startup Tests." The submittal date of each of these reports is given below.

Unit 1	September 27, 1974
Unit 2	May 23, 1975
Unit 3	May 9, 1977

This test description is retained in the FSAR only to serve as a readily available general description of the test program.

13.5.1 Program Description and Objectives

13.5.1.1 General

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

Phase I	Preoperational tests (Subsection 13.4)
Phase II	Fuel loading and shutdown power level tests
Phase III	Initial heatup to rated temperature and pressure
Phase IV	Power testing from 25 to 100 percent of rated output
Phase V	Warranty demonstrations

The tests performed can be broadly classified as Major Plant Transients (Table 13.5-2), Stability Tests (Table 13.5 3), and a residue of tests directed toward demonstrating correct performance of the numerous auxiliary plant systems; clearly, certain tests may be identified with more than one class. Each test is discussed later, but at this juncture the following comments are given by way of outlining the startup and power test program. The various test points as functions of core thermal power and flow are shown for Unit 1 in Table 13.5-4, Unit 2 in Table 13.5-4 and Unit 3 in Table 13.5-6.

13.5.1.2 Fuel Loading and Shutdown Power Level Tests

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

At this point in the program, a number of tests are conducted which are best described as initial shutdown power level tests (Phase II). Chemical and radiochemical tests are made to check the quality of the reactor water before fuel is loaded and to establish base and background levels which will be required to facilitate later analysis and instrument calibrations. Plant and site radiation surveys are made at specific locations for later comparison with the values obtained at the subsequent operating power levels. Shutdown margin checks are repeated for the fully loaded core, and, in turn, criticality is achieved with a prescribed rod sequence, data are recorded for each rod withdrawn. The reactor is made critical by means of a prescribed control rod sequence, using the normal Source Range Monitors (SRM) in conjunction with the operational sources to show that adequate response exists for normal operation. Each control rod drive is subjected to scram and friction testing at ambient conditions. Initially the Intermediate Range Monitors (IRM) are set to maximum gain and are verified for overlap with the SRMs. The process computer is checked to see that it is receiving correct values for the available process variables. Vibration characteristics are determined for selected reactor internal components over a range of cold recirculation flows. The water level profile is determined for a range of water levels and core flows.

13.5.1.3 Initial Heatup to Rated Temperature and Pressure

Heatup follows satisfactory completion of the Phase II tests, and further checks are made of coolant chemistry together with radiation surveys at the selected plant locations. All control rod drives are scram timed at rated temperature and pressure; selected drives are timed at intermediate reactor pressures and for different accumulator pressures. The control rod sequence is further investigated to obtain rod pattern versus temperature relationships. The process computer checkout continues as more process variables become available for input. The RCIC and HPCI systems will undergo controlled starts at low rector pressure and at rated conditions; the RCIC system is tested in the quick start mode at 1,020 psig. Correlations are obtained between reactor vessel temperatures at several locations and the values of other process variables as heatup continues. The movements of drywell piping systems as a function mainly of expansion are recorded for comparison with design data. The steam separator-dryer test equipment is checked out and initial samples taken. An intermediate APRM calibration is made using coolant temperature rise data during nuclear heatup.

13.5.1.4 Power Testing from 25 to 100 percent of Rated Output

The power test phase comprises the following tests, many of which are repeated several times at the different test levels; consequently, reference should be made to Figures 13.5-1 sheets 1 and 2 (Unit 1) and Figures 13.5-2 sheets 1, 2, and 3 (Units 2 and 3) for the probable order of execution for the full series. It must be appreciated that, while a certain basic order of testing is maintained relative to power ascension, there is, nevertheless, considerable flexibility in the test sequence at a particular power level which may be used whenever it becomes operationally expedient.

Coolant chemistry tests and radiation surveys are made at each principal test level to preserve a safe and efficient power increase. Selected control rod drives are scram timed at various power levels to provide correlation with the initial data. The effect of control rod movement on other parameters (e.g., electrical output, steam flow and neutron flux level) is examined for different power conditions. Following the first reasonably accurate heat balance (25 percent power) the IRMs are reset. At each major power level (25, 50, 75, and 100 percent) the LPRMs are calibrated; the APRMs are calibrated at each new power level initially and following LPRM calibration. Completion of the process computer checkout is made for all variables, and the various options are compared with backup calculations as soon as significant power levels are available. Further tests of the RCIC and HPCI systems are made with and without injection into the reactor pressure vessel. Collection of data from the system expansion tests is completed for those piping systems which had not previously reached full operating temperatures. The axial and radial power profiles are explored fully by means of Traversing Incore Probe (TIP) system at

representative power levels (25, 50, 75, and 100 percent) during the power ascension. Core performance evaluations are made at all test points above the 10 percent power level and for selected flow transient conditions; the work involves the determination of core thermal power, maximum fuel rod surface heat flux, and the minimum critical heat flux ratio (MCHFR).

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

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

For the first of these tests a centrally located control rod is moved and the flux response noted on a selected LPRM chamber. The next two tests require that the changes made should approximate, as closely as possible, a step change in demand, while for the remaining test the bypass valve is opened as quickly as possible. For all of these tests the plant performance is monitored by recording the transient behavior of numerous process variables, the principal one of interest being neutron flux. Other imposed transients are produced by step changes in demanded core flow, dropping a feedwater heater, and failing the operating pressure regulator to permit takeover by the backup regulator. Table 13.5-2 indicates the power and flow levels at which all these stability tests are performed.

The category of major plant transients includes full closure of all the main steam isolation valves, fast closure of the turbine generator control valves, fast closure of turbine generator stop valves, loss of the main generator and offsite power, tripping a feedwater pump, and several trips of the recirculation pumps.

The plant transient behavior is recorded for each test and the results may be compared with the predicted design performance. Table 13.5-2 shows the operating test conditions for all the proposed major transients.

A test is made of the relief valves in which the capacity, leak tightness, and general operability is demonstrated. At all major power levels the jet pump flow instrumentation is calibrated and carryover/carryunder measurements are made to facilitate assessment of the steam separator-dryer performance. The as-built characteristics of the recirculation pump drives are investigated as soon as operating conditions permit full core flow. The local control loop performance (based on the drive motor, fluid coupler, generator, drive pump, jet pumps, and control equipment is checked. The vibration is testing conducted at the cold flow

condition is extended to measurements at several power conditions as the operating power level is raised.

13.5.1.5 Warranty Demonstrations

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

13.5.2 Discussion of Startup Tests

13.5.2.1 General

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

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

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

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

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

13.5.2.2 Test Purpose, Description and Acceptance Criteria--Unit 1

TEST NUMBER 1 -- CHEMICAL AND RADIOCHEMICAL

Purpose

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

sampling equipment, procedures, and analytic techniques are adequate to supply the data required to demonstrate that the coolant chemistry meets water quality specifications and process requirements.

Secondary objectives of the test program include data to evaluate the performance of the fuel, operation of the demineralizers and filters, operation of the offgas system and calibration of certain process instruments.

Description

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

Criteria

Level 1

Water quality must be known and must conform to the water quality specifications at all times.

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

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

TEST NUMBER 2 -- RADIATION MEASUREMENTS

Purpose

The purpose of this test is to determine the background radiation levels in the plant environs prior to operation for base data on activity buildup and to monitor radiation at selected power levels to assure the protection of personnel during plant operation.

Description

A survey of natural background radiation throughout the plant site will be made before fuel loading. Subsequent to fuel loading, during reactor heatup and at power

levels of 25, 50, and 100 percent of rated power, gamma radiation level measurements and, where appropriate, thermal and fast neutron dose rate measurements will be made at significant locations throughout the plant. All potentially high radiation areas will be surveyed.

Criteria

Level 1

The radiation doses of plant origin and occupancy times shall be controlled consistent with the guidelines of the standards for protection against radiation outlined in TVA Radiological Control Instructions.

TEST NUMBER 3 -- FUEL LOADING

Purpose

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

Description

Before fuel loading, control rods will be installed and tested. A neutron source of approximately 10 neutrons per sec will be installed near the center of the core. At least three neutron detectors calibrated and connected to high flux scram trips will be located to produce acceptable signals during loading.

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

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

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

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

4. The Subcriticality Check will be repeated.

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

Shutdown margin demonstrations will be performed periodically during fuel loading.

Criteria

Level 1

The criteria for successful completion of this test are: (a) the partically-loaded core must be subcritical by at least 0.38 percent k/k with the geometrically strongest rod fully withdrawn, (b) the core is fully loaded, and (c) the full core shutdown margin demonstration has been completed.

TEST NUMBER 4 -- FULL CORE SHUTDOWN MARGIN

Purpose

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

Description

This test will be performed in the fully loaded core at ambient temperature in the xenon-free condition. Subcriticality will be demonstrated with the strongest rod fully withdrawn and a series of calibrated rods pulled to a position calculated to be equal to a shutdown margin specified to account for expected reactivity changes during core lifetime.

Criteria

Level 1

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

TEST NUMBER 5 -- CONTROL ROD DRIVE

Purpose

The purposes of the Control Rod Drive System test are: (a) to demonstrate that the Control Rod Drive (CRD) system operates properly over the full range of primary coolant temperatures and pressures from ambient to operating, and particularly that thermal expansion of core components does not bind or significantly slow control rod movements, and (b) to determine the initial operating characteristics of the entire CRD system.

Description

The CRD tests performed during Phases II through IV of the startup test program are designed as an extension of the tests performed during the preoperational CRD system tests. Thus, after it is verified that all control rod drives operate properly when installed, they are tested periodically during heatup to assure that there is no significant binding caused by thermal expansion of the core components. A list of all CRD tests to be performed during startup testing is given in Table 13.5-1.

Criteria

Level 1

(a) Each drive speed in either direction (insert or withdraw) must be 3.0 ± 0.6 in. per sec, indicated by a full 12-foot stroke in 40 to 60 sec.

(b) The average scram insertion time of all operable control rods, based on the deenergization of the scram pilot valve solenoids as time zero, shall be no greater than:

Percent Inserted from Fully Withdrawn	Average Scram Insertion Times (sec)
5	0.375
20	0.90
50	2.0
90	5.0

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

Percent Inserted from Fully Withdrawn	Average Scram Insertion Times (sec)
5	0.398
20	0.954
50	2.120
90	5.3

(d) The maximum scram insertion time for 90 percent insertion of any operable control rod shall not exceed 7.00 seconds.

Level 2

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

(b) With the charging valve closed, the 90 percent scram time should be greater than 1.50 second at ambient (0 psig) reactor pressure.

(c) Scram times with normal accumulator charge should fall within prescribed time limits.

TEST NUMBER 6 -- SRM PERFORMANCE AND CONTROL ROD SEQUENCE

Purpose

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

Description

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

A control rod withdrawal sequence has been calculated which completely specifies control rod withdrawals from the all-rods-in condition to the rated power configuration. Rod patterns will be recorded periodically as the reactor is heated to rated temperature. As each rod group is completed during the power ascension, the electrical power, steam flow, and APRM response will be recorded.

Movement of rods in a prescribed sequence is monitored by the Rod Worth Minimizer and Rod Sequence Control System which will prevent acceptable out-of-sequence control rod movements during startup or shutdown.

Criteria

Satisfaction of the following criteria constitutes adequate source and SRM relationships.

Level 1

There must be a neutron signal count-to-noise count ratio of at least 2 to 1 on the required operable SRMs or Fuel Loading Chambers.

There must be a minimum count rate of 3 counts/sec. on the required operable SRMs or Fuel Loading Chambers.

TEST NUMBER 10 -- IRM CALIBRATION

Purpose

The purpose of this test is to adjust the Intermediate Range Monitor system to obtain an optimum overlap with the SRM and APRM systems.

Description

Initially, the IRM system is set to maximum gain. After the APRM heatup calibration and after the first heat balance calibration of the APRM's, the IRM-APR, overlap will be checked and the IRM gains adjusted if necessary to improve the IRM system overlap between the SRM's and IRM's.

Criteria

Level 1

(a) Each IRM channel must be adjusted so that overlap with the SRM's and APRM's is assured.

(b) The IRM's must produce a scram at 120/125 of full scale.

(c) The IRM reading 120/125 of full scale on range 10 will be set equal to or less than 30 percent of rated power.

TEST NUMBER 11 -- LPRM CALIBRATION

Purpose

The purpose of this test is to calibrate the Local Power Range Monitoring system.

Description

The LPRM channels will be calibrated to make the LPRM readings proportional to the average heat flux in the four corner fuel rods surrounding each chamber at the chamber elevation. The initial calibration factors are obtained from measurements of axial power distribution, precalculated local power distributions, and precalculated radial power distributions.

Criteria

Level 1

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

TEST NUMBER 12 -- APRM CALIBRATION

Purpose

The purpose of this test is to present the methods for calibrating the Average Power Range Monitor Channels.

Description

The APRM's will be initially adjusted to maximum amplifier gain. The LPRM's and APRM's should begin to respond in the region of 1 to 10 percent of rated power when a low power calibration will be performed based on heat balance calculations during reactor heatup. As soon as the reactor power is high enough to obtain more accurate steady state heat balances to determine actual core thermal power, the APRM channels will be calibrated to read percent of core thermal power.

Criteria

Level 1

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

(b) Technical Specifications and Fuel Warranty Limits stated in Section B shall not be exceeded.

(c) In the startup mode, all APRM channels must produce a scram at less than or equal to 15 percent of rated thermal power.

(d) Recalibration of the APRM system will not be necessary for safety considerations if at least two APRM channels per reactor protection system (RPS) trip circuit have readings greater than or equal to core power. Channels will be considered to be reading accurately if they agree with the heat balance to within plus or minus 7 percent of rated power.

TEST NUMBER 13 -- PROCESS COMPUTER

Purpose

The purpose of this test is to verify the performance of the process computer under operating conditions.

Description

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

Criteria

Level 2

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

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

TEST NUMBER 14 -- RCIC SYSTEM

Purpose

The purpose of this test is to verify the operation of the Reactor Core Isolation Coolant (RCIC) system at operating reactor pressure conditions.

Description

Flow tests of the RCIC System will be performed at reactor pressures between 150 and 1,020 psig. Except for a final demonstration without warmup, the RCIC turbine will be warmed up before a fast start at each test point. This test is designed to verify proper operation of the RCIC system, determine time to reach rated flow and adjust flow controller in RCIC system for proper flow rate. These tests will first be performed with the system in the test mode so that discharge flow will not be routed to the reactor pressure vessel. The final demonstration will be made so that the discharge flow will be routed to the reactor pressure vessel while the reactor is at partial power.

Criteria

Level 1

The reactor will be allowed to operate at all conditions, including 100 percent power, if the RCIC can deliver rated flow, 600 GPM, in less than or equal to the rated actuation time, 30 sec, against any reactor pressure between 150 and 1,020 psig.

TEST NUMBER 15 -- HPCI

Purpose

The purpose of this test is to verify the proper operation of the High Pressure Coolant Injection (HPCI) system throughout the range of reactor pressure conditions.

Description

Flow tests of the HPCI System will be performed at reactor pressures between 150 and 1,020 psig. Except for a final demonstration without warmup, the HPCI turbine will be warmed up before a fast start at each test point. The purpose of this test is to verify proper operation of the HPCI system, determine time to reach rated flow, and adjust the flow controller in HPCI system for proper flow rate. These tests will be performed with the system in the Test Mode so that discharge flow will not be routed to the reactor pressure vessel. The final demonstration will be made so that discharge flow will be routed to the reactor pressure vessel while the reactor is at partial power or following a reactor scram from low power.

Criteria

Level 1

The time from actuating signal to required flow must be less than 25 sec at any reactor pressure between 150 and 1,020 psig. With pump discharge at any pressure between 1,220 and 150 psig, the flow should be at least 5,000 gpm. The HPCI turbine must not trip off during startup.

TEST NUMBER 16 -- SELECTED PROCESS TEMPERATURES

Purpose

The purposes of this procedure are to establish the minimum reactor recirculation pump speed which will maintain water temperature in bottom head of the reactor vessel within 145°F (63°C) of reactor coolant saturation temperature as determined from reactor pressure and to provide assurance that the measured bottom head drain temperature corresponds to bottom head coolant temperature during normal operations.

Description

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

Criteria

Level 1

1. The reactor recirculation pump shall not be operated unless the coolant temperatures between the dome and the bottom head drain are within 145°F (63°C) of each other.

2. The pump in an idle recirculation loop shall not be started unless the temperature of the coolant within the loop is within 50°F (10°C) of the active loop temperature.

TEST NUMBER 17 -- SYSTEM EXPANSION

Purpose

The purpose of this test is to verify that the reactor drywell piping system is free and unrestrained in regard to thermal expansion and that suspension components are functioning in the specified manner.

Description

Observe and record the horizontal and vertical positions of major equipment and piping in the Nuclear Steam Supply System and auxiliary systems to assure components are free to move as designed. Adjust as necessary for freedom of movement.

Criteria

Level 1

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

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

Level 2

(a) Displacements of instrumented points with special recording devices shall not vary from the calculated values by more than 50 percent or 0.5 (1.27) cm) inches, whichever is smaller.

Displacements of less than 0.25 (0.64 cm) inch can be neglected, since 50 percent of this value is bordering on the accuracy of measurement. If measured displacements do not meet these criteria, the system designer must be contacted to analyze the data with regard to design stresses; (b) The trace of the instrumented points during the heatup cycle shall fall within a range of 150 percent of the calculated value from the initial cold position in the direction of the calculated value, and 50 percent of the calculated value from the initial position in the opposite direction of the calculated value; (c) Hangers shall be in their operating range (between the hot and cold settings).

TEST NUMBER 18 -- CORE POWER DISTRIBUTION

Purpose

The purposes of this test are to: (1) confirm the reproducibility of the TIP system readings, (2) determine the core power distribution in three dimensions, and (3) determine core power symmetry.

Description

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

Core Power distribution, including power symmetry, will be obtained during the power ascension program. Axial power traces will be obtained at each of the TIP locations. Several TIP systems have been provided to obtain these traces--a common location can be traversed by each TIP chamber to permit intercalibration.

The results of the complete set of TIP traces will be evaluated to determine core power symmetry.

Criteria

Level 2

In the TIP reproducibility test, the TIP traces should be reproducible within 3.5 percent relative error or 0.15 in. absolute error at each axial position, whichever is greater.

TEST NUMBER 19 -- CORE PERFORMANCE

Purpose

The purpose of this test is to evaluate the core performance parameters of core flow rate, core thermal power level, maximum fuel rod surface heat flux and core minimum critical heat flux ration (MCHFR).

Description

Core power level, maximum heat flux, recirculation flow rate, hot channel coolant flow, MCHFR, fuel assembly power, and steam qualities will be determined at

existing power levels and assumed over power conditions. Plant and in-core instrumentation, conventional heat balance techniques, and core performance worksheets and nomograms will be used. This will be performed above 10 percent power and at various pumping conditions and can be done independent of the process computer functions.

Criteria

Level 1

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

(1) Maximum fuel rod surface heat flux shall not exceed 134 W/cm (425,000 Btu/h-ft) during steady-state conditions when evaluated at the operating power level.

(2) Minimum CHF ration shall not be less than 1.9 when evaluated at the operating power level. The basis for evaluation of MCHFR shall be "Design Basis for Critical Heat Flux Condition in BWRs," APED-5286, September 1966.

(3) Steady-state reactor power shall be limited to values on or below the licensed flow control line (maximum power of 3293 MWt with flow of at least 102.5x10 lb/h).

TEST NUMBER 20 -- ELECTRICAL OUTPUT AND PRELIMINARY HEAT RATE TEST

Purpose

The purpose of this test is to demonstrate that the guaranteed gross electrical output requirements are satisfied without exceeding the reactor power level warranty and to determine a preliminary net plant heat rate value.

Description

The plant gross electrical output will be measured during sustained operation at a load of at least 1,098.4 MWe. The gross electrical power will be measured at the

generator terminals and corrected to rated conditions of 0.90 power factor, turbine exhaust pressure of 2 inches mercury absolute, and rated generator hydrogen pressure. Data may be taken during this period for an ASME turbine cycle heat rate test as defined in PTC-6 "Interim Test Code for Steam Turbines Operating Predominately Within the Moisture Region with Nuclear Steam Supply."

Criteria

Level 1

The guaranteed performance calls for a gross output of 1,098,420.0 kWe at a reactor thermal power of 3293.0 MW and a net plant heat rate of 10,459 Btu/kWh. The power factor shall be 0.9 and the generator hydrogen pressure shall be 75.0 psig.

TEST NUMBER 21 -- FLUX RESPONSE TO RODS

Purpose

The purpose of this test is to demonstrate stability in the power-reactivity feedback loop with increasing reactor power and determine the effect of control rod movement on reactor stability.

Description

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

Criteria

Level 1

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

Level 2

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

TEST NUMBER 22 -- PRESSURE REGULATOR

Purpose

The purposes of this test are to: (a) determine the reactor and pressure control system responses to pressure regulator setpoint changes, (b) demonstrate the stability of the reactivity-void feedback loop to pressure perturbations, (c) demonstrate the control characteristics of the bypass and control valves, (d) demonstrate the takeover capabilities of the backup pressure regulator, and (e) to optimize the pressure regulator settings to give the best combination of fast response and small overshoot.

Description

The pressure setpoint will be decreased rapidly and then increased rapidly in 0.5 psi steps and the response of the system will be measured. The backup regulator will be tested by increasing the operating pressure regulator setpoint rapidly until the backup regulator takes over control. The load reference setpoint will be reduced, and the test repeated with the bypass valve in control. The response of the system will be measured and evaluated and regulator settings will be optimized.

Criteria

Level 1

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

Level 2

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

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

TEST NUMBER 23 -- FEEDWATER SYSTEM

Purpose

The purposes of this test are to: (a) demonstrate acceptable reactor water level control, (b) evaluate and adjust feedwater controls, (c) demonstrate general reactor response to inlet subcooling changes on reactor power and pressure, (d) demonstrate individual feedwater pump response.

Description

Reactor water level setpoint changes of approximately 6 in. will be used to evaluate and acceptably adjust the feedwater control system settings for all power and feedwater control system settings for all power and feedwater pump modes.

Operate each pump through its flow range to verify acceptable feedwater pump linearity. Response time on each feedwater pump will be verified by changing the flow by 10 percent and measuring the turbine speed and flow response times.

Criteria

Level 1

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

Level 2

The decay ratio is expected to be less than or equal to 0.25 for each process variable that exhibits oscillatory response to feedwater system setpoint changes when the plant is operating above the lower limit of the Master Flow Controller. System response for large transients should not be unexplainably worse than preanalysis.

TEST NUMBER 24 -- BYPASS VALVES

Purpose

The purposes of this test are to: (a) demonstrate the ability of the pressure regulator to minimize the reactor pressure disturbance during an abrupt change in reactor steam flow, and (b) demonstrate that a bypass valve can be tested for proper functioning at rated power without causing a high flux scram.

Description

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

Criteria

Level 1

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

Level 2

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

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

TEST NUMBER 25 -- MAIN STEAM ISOLATION VALVES

Purpose

The purposes of this test are to: (a) functionally check the main steamline isolation valves (MSIV) for proper operation at selected power levels, (b) determine reactor transient behavior during and following simultaneous full closure of all MSIV and following closure of one valve, and (c) determine isolation valve closure time.

Description

Fast full closure of each MSIV will be performed at hot standby and selected power levels to determine the maximum power conditions at which individual valve full closure tests can be performed without a reactor scram. Functional checks (10 percent closure) of each isolation valve will be performed at selected power levels above the maximum power condition for individual MSIV full closure determined above. A test of simultaneous full closure of all MSIV's will be performed at about 100 percent of rated thermal power and proper operation of the relief valves and the RCIC will be shown. Reactor process variable will be monitored to determine the transient behavior of the system during and following each isolation test. MSIV delay and movement times will be determined. Proper seating of the MSIV's will be demonstrated.

Criteria

Level 1

MSIV stroke time will be between 3 and 5 sec, exclusive of electrical delay time. Reactor pressure shall be maintained below 1230 psig (the setpoint of the first safety valve) during the transient following closure of all valves. During full closure of individual valves, scram should not occur.

Level 2

The maximum reactor pressure should be about 1200 psig, 30 psi below the first safety valve setpoint following closure of all valves. This is a margin of safety for safety valve weeping. During full closure of individual valves, pressure must be 20 psi below scram, neutron flux must be 10 percent below scram, and steam flow in individual lines must be below the trip point.

TEST NUMBER 26 -- RELIEF VALVES

Purpose

The purposes of this test are to: (a) verify the proper operation of the dual purpose relief safety valves, (b) determine their capacity, and (c) verify proper reseating following operation.

Description

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

Criteria

Level 1

The combined capacity of the relief valves will be demonstrated to be at least 61 percent (FSAR, Sections 14.5.1.2, 14.1.5.3, 14.1.5.4, and 14.1.5.6) of their design

capacity (11 x 800,000=8.8 x 10 lb/hr at 1,100 psig) considering the relief valve with the largest measured capacity to be inoperable.

Level 2

Each relief valve is expected to have a capacity of at least 800,000 lb/h at a pressure setting of 1100 psig. Relief valve leakage must be low enough that the temperature measured by the thermocouples in the discharge side of the valves falls to within 10°F of the temperature recorded before the valve was opened.

TEST NUMBER 27 -- TURBINE STOP AND CONTROL VALVE TRIPS

Purpose

The purposes of this test are to (a) determine the response of the reactor system to a turbine stop or control valve trip and (b) evaluate the response at the bypass, relief valve and reactor protection systems. The parametric responses of particular interest are the peak values and the rate of change of both reactor power and reactor steam dome pressure.

Description

The stop of control valves will be tripped closed at selected reactor power levels and neutron flux, feedwater flow and temperature, vessel water level and pressure will be monitored. Responses of selected control valves, stop valves, relief valves, and bypass valves will be recorded.

Criteria

Level 1

The safety valves should not open; therefore, reactor pressure should not rise above 1230 psig (the setpoint of the first safety valve).

Reactor scram must limit the severity of the neutron flux and simulated heat flux transients within thermal limitations.

The turbine stop valves must close before the control valves for the turbine stop valve trip.

The turbine control valves must close before the stop valves for the turbine control valve fast closure.

Level 2

Reactor pressure should not rise so close to the safety valve setting that weeping or leakage occurs; therefore, the pressure should not rise above 1200 psig, which is 30 psi below the setpoint of the first safety valve.

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

Trip scram must meet Reactor Protection System specification. The pressure regulator must regain control before a low pressure reactor operation occurs.

TEST NUMBER 29 -- FLOW CONTROL

Purpose

The purposes of this test are to: (a) determine the plant response to changes in the recirculation flow, (b) adjust all control elements, and (c) demonstrate the plant load following capability in all flow control modes (local manual, master manual, and automatic).

Description

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

Criteria

Level 1

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

Level 2

The decay ratio is expected to be less than or equal to 0.25 for each process variable that exhibits oscillatory response to flow control changes when the plant is operating above the lower limit setting of the Master Flow Controller, and also at test point No. l. Scram must not occur. Automatic flow control range must be at least 80 to 100 percent power along the full power load line. Load response to a 20 percent load demand step must be at least 0.5 percent/sec.

TEST NUMBER 30 -- RECIRCULATION SYSTEM

Purpose

The purpose of this test is to investigate the performance of the recirculation system including transient responses and steady-state conditions following recirculation pump trips at selected power levels and calibration of the jet pump core flow measurement system.

Description

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

Criteria

Level 1

Not applicable

Level 2

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

Flow instrumentation has been calibrated such that the Reactor Jet Pump Total Flow Recorder provides correct flow indication.

TEST NUMBER 31 -- LOSS OF TURBINE-GENERATOR AND OFFSITE POWER

Purpose

The purpose of this test is to demonstrate proper performance of the reactor and the plant electrical equipment and systems during the loss of auxiliary power transient.

Description

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

Criteria

Level 1

All test pressure transients must have maximum pressure values below 1230 psig, which is the setpoint of the first safety valve. All safety systems, such as the Reactor Protection System, the diesel generator, RCIC and HPCI, must function properly without manual assistance.

Level 2

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

TEST NUMBER 32 -- RECIRCULATION M-G SET SPEED CONTROL

Purpose

The purposes of this test is: (a) to demonstrate that the recirculation speed control system can satisfactorily perform its function by comparing transient test results against system criteria.

Description

Make several small step changes in speed at various pump speeds and record appropriate recirculation loop transient signals. Demonstrate performance over the full speed range with small speed demand step tests.

Criteria

Level 1

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

Level 2

When the unit is operating above the lower limit setting of the master manual limiter, the decay ratio should be less than or equal to 0.25 for each process variable that exhibits oscillatory response to recirculation M-G set speed changes.

TEST NUMBER 35 -- RECIRCULATION AND JET PUMP SYSTEM CALIBRATION

Purpose

The purpose of this test is to obtain a complete integrated calibration of the installed jet pump and recirculation system instrumentation with the reactor shutdown.

Description

This test involves applying, simultaneously to an entire loop, a closely controlled pressure to obtain an integrated calibration check of the system instrumentation. Actual calibration of the jet pump flow instrumentation will be completed during hot pressurized operation by comparison of the single and double tapped pressure drops as a function of flow.

Criteria

Not applicable.

TEST NUMBER 36 -- EQUALIZER OPEN

Purpose

The purposes of this test are (a) to explore the allowable operating range and performance of the recirculation system under conditions of one pump operation with the equalizer line valves open, and (b) to develop operating procedures for one pump equalizer open operation.

Description

The reactor recirculation system consists of the reactor vessel and two piping loops. Each loop contains a recirculation pump, suction and discharge isolation valves, and ten parallel jet pumps situated in the reactor downcomer. An equalizer line with two valves connects the loops. Under normal one and two pump operation the equalizer valves have been kept closed. This test will explore one pump operation with the equalizer open.

Initial equalizer valve opening will be made at a high pump speed by rapid jogging until the inactive loop jet pumps go from reverse to forward flow. Successive valve openings and pump speed increases will be scheduled to avoid pump loop P, pump speed and pump motor current limits. When the valve is full open, or when limits are reached the available operating region will be explored and appropriate data will be obtained. The test will be concluded by rapidly closing the equalizer valve while recording the transient.

Criteria

Level 1

Operations shall be conducted so that the following conditions are met: (1) At pump flows greater than 115 percent of the design value, the pump loop shall at all times be greater than 71 psi. (2) At pump flows greater than 115 percent of the design value, operation with pump loop P's in the range of 71 to 80 psi shall be minimized, and shall not exceed 30 minutes total.

Operating time with partially open equalizer under conditions of reverse jet pump flow shall be minimized. If the equalizer valve sticks under this condition the following action will be taken: (1) In the absence of vibration monitoring, the remaining equalizer valve shall be immediately closed or the pump tripped. (2) If vibration is being monitored, the vibration engineer will decide whether the situation warrants closure of the remaining equalizer valve or tripping of the pump.

Pump motor and MG set temperatures and currents shall not be allowed to exceed vendor specifications. If these limits are reached, corrective actions include

reduction of pump speed to the minimum head limit, equalizer closure and pump trip.

During steady-state operation, reactor power shall be kept below 68 percent of rated with one pump operating and closed equalizer, and below 80 percent with open equalizer.

Level 2

Operation should be free of repeated running into limits. If control system instabilities preclude acceptable operation, the test should be terminated by closure of the equalizer or tripping of the pump.

TEST NUMBER 39 -- WATER LEVEL VERIFICATION IN REACTOR VESSEL

Purpose

The purpose of this test is to verify the calibration of YARWAY and GEMAC level instrumentation under varying conditions.

Description

Reactor water level is monitored by four level instrument systems: YARWAY wide and narrow range, and, GEMAC wide and narrow range instruments.

This test is divided into two parts. The first part involves measuring the YARWAY reference by temperature to verify it agrees with the temperature correction factor used in calibration.

The second portion of the test verified the ability of the feedwater control system to regulate reactor water level at two power levels: 50 percent flow/50 percent power and 100 percent flow/100 percent power.

Criteria

Not applicable.

TEST NUMBER 70

Purpose

Demonstrate the operability of the reactor water cleanup system under actual reactor operating temperature and pressure.

Description

This system provides a continuous purifying treatment of the reactor water by removing various impurities to maintain reactor water at specification quality. The system contains two recirculation pumps, regenerative heat exchangers, nonregenerative heat exchangers, and a cleanup demineralizer system. The modes of operation are normal, startup, blowdown, refueling, and hot standby.

Criteria

Level 1

Reactor water quality must be maintained according to specifications in fuel warranty documents.

Level 2

The temperature at the tube outlet of the nonregenerative heat exchangers shall not reach 140°F (60°C) in any cleanup system operating mode.

TEST NUMBER 71 -- RESIDUAL HEAT REMOVAL SYSTEM

Purpose

The purpose of this test is to demonstrate the ability of the Residual Heat Removal (RHR) system to: (a) remove residual and decay heat from the nuclear system so that refueling and nuclear system servicing can be performed, and (b) remove heat from the pressure suppression pool water.

Description

The RHR system is a closed loop system of piping, water pumps, and heat exchangers, the purpose of which is to remove post-power-operation energy from the reactor under both operational and accident conditions.

During a reactor cooldown, after operation at normal temperature and pressure, the ability of the Shutdown Cooling Subsystem to remove enough of the residual heat (decay and sensible heat) from the reactor primary system will be demonstrated. The reactor water will be cooled by pumping water from one of the recirculation loops, through the RHR system heat exchanger(s) and back to the reactor vessel by way of a (the) recirculation loop(s). It will also be possible to divert part of the flow to a spray nozzle in the reactor vessel head to condense steam generated from the hot walls of the vessel while it is being flooded, thereby maintaining saturated pressure and temperature conditions.

The Suppression Pool Cooling subsystem cools the pressure suppression pool to limit the water temperature such that the temperature immediately after a blowdown does not exceed 170°F when reactor pressure is above 135 psig. During this mode of operation, water is pumped from the pressure suppression pool, through the RHR system heat exchangers and back to the pressure suppression pool. No Suppression Pool Cooling test is necessary if the heat exchanger capability is established by the Shutdown Cooling test, and the flow capability of the RHR system in the "Suppression Pool Cooling" mode has previously been established.

Criteria

Level 1

The heat removal capability of each RHR heat exchanger in the "Shutdown Cooling" mode or the "Suppression Pool Cooling" mode shall be 18.7 by 10 Btu/hr (4.69 by 10 Kcal/hr) or greater.

TEST NUMBER 72 -- DRYWELL ATMOSPHERE COOLING SYSTEM

Purpose

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

Description

The Drywell Atmosphere Cooling System will be placed in operation and its ability to maintain the following temperature in the drywell with 8 of the 10 fans in operation will be demonstrated.

During Operation:

Average temperature throughout drywell -- 135°F
Maximum around the recirculating pump motors -- 128°F
Maximum all other areas -- 150°F
Maintain a uniform circumferential temperature of the refueling bellows/bulkhead assembly within 10°F

Within 8 to 20 hours following shutdown, all areas in the drywell beneath the vessel-to-drywell bulkhead shall be within 10°F of Reactor Building Closed Cooling Water inlet temperature.

Criteria

Level 2

The heat removal capability of the drywell cooler shall be approximately 4.8 x 10 Btu/hr.

The drywell cooling system shall have a standby capability of 25 percent of above.

The drywell cooling system shall maintain temperatures in the drywell below the design valves given in the description during normal operation.

TEST NUMBER 73 -- COOLING WATER SYSTEMS

Purpose

The purpose of this test is to verify that the performance of the Reactor Building Closed Cooling Water (RBCCW) and the raw cooling water systems is adequate with the reactor at rated condition.

Description

With the reactor at 1,000 psig (70 kg/cm) following the initial heatup, data will be obtained to verify that the flow rates in the RBCCW heat exchangers are adequate and properly balanced, and that the heat exchangers outlet temperatures are balanced and within design values. Flow rate adjustments will be made as necessary to achieve satisfactory system performance. The test will be repeated at selected power levels to verify continued satisfactory performance with higher plant heat loads.

Criteria

Level 2

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

3.2.2 The RBCCW was designed to transfer a maximum heat load of 31.3 x 10 Btu/hr (7.9 x 10 Kcal/hr) in order to limit equipment inlet water temperature to 100°F (38°C) assuming a service (raw cooling) water inlet temperature of 90°F (32°C).

TEST NUMBER 90 -- VIBRATION MEASUREMENTS

Purpose

Determine the vibration characteristics of selected reactor internals and recirculation loops induced by cold recirculation flow and by hot, two-phase flows.

Description

Vibratory responses will be recorded at various recirculation flow rates at temperatures below 150°F using strain gages on in-core guide tubes, control rod stub tube, shroud support legs, and jet pump riser braces; accelerometers on the recirculation loops and displacement gages on the shroud, steam separator and jet pumps. Portable vibration sensor surveys will be made on the recirculation loops and differential pressure measurements will be made across the core plates, shroud head and shroud wall. At hot, two-phase flow conditions, similar measurements will be made on the in-core guide tubes, shroud, jet pump riser and shroud head. The results of vibration measurements made at other BWR installations will be considered in the final selection of components to be tested. Where possible, vibration measurements will be made as Preoperational Test.

Criteria

Level 1

The criteria by which the results of the vibration tests will be judged involve complex, precalculated relationships among spatial locations, vibrational amplitudes, and vibrational frequencies as related to stress and limited by ASME Code, Section III.

TEST NUMBER 92 -- STEAM SEPARATOR-DRYER

Purpose

Determine carryunder and carryover characteristics of the steam separator-dryer.

Description

Samples will be taken from the inlet and outlet of the steam dryers, and the inlet at the steamline at various power levels at chosen water levels and recirculation flow rates. The amount of carryunder will be estimated from these samples and carryover will be determined from Na-24 activities in samples taken from the outlet of the steam dryers.

Criteria

Level 2

Water carryover from the dryer shall be no greater than 0.002 weight fraction. The design value of steam carryunder to the jet pumps in 0.01 weight fraction.

13.5.2.3 Test Purpose, Description and Acceptance Criteria -- Units Two and Three

TEST NUMBER 1 -- CHEMICAL AND RADIOCHEMICAL

Purpose

The principal objectives of this test are (a) to secure information on the chemistry and radiochemistry of the reactor coolant, and (b) to determine that the sampling equipment, procedures and analytic techniques are adequate to supply the data required to demonstrate that the chemistry of all parts of the entire reactor system meet specifications and process requirements.

Specific objectives of the test program include evaluation of fuel performance, evaluations of demineralizer operations by direct and indirect methods, measurements of filter performance, confirmation of condenser integrity, demonstration of proper steam separator-dryer operation, measurement and calibration of the off-gas system, and calibration of certain process instrumentation. Data for these purposes is secured from a variety of sources: plant operating records, regular routine coolant analysis, radio-chemical measurements of specific nuclides, and special chemical tests.

Description

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

Criteria

Level 1

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

The activity of gaseous and liquid effluents must conform to license limitations.

Level 2

Water quality must be known at all time and should remain within the guidelines of the Water Quality Specifications.

TEST NUMBER 2 -- RADIATION MEASUREMENTS

Purpose

The purposes of this test are (a) to determine the background radiation levels in the plant environs prior to operation for base data on activity buildup and (b) to monitor radiation at selected power levels to assure the protection of personnel during plant operation.

Description

A survey of natural background radiation throughout the plant site will be made prior to fuel loading. Subsequent to fuel loading, during reactor heatup and at power levels of 25 percent, 50 percent, 75 percent, and 100 percent of rated power, gamma radiation level measurements and where appropriate, thermal and fast neutron dose rate measurements will be made at significant locations throughout the plant. All potentially high radiation areas will be surveyed.

Criteria

Level 1

The radiation doses of plant origin and the occupancy times of personnel in radiation zones shall be controlled consistent with the guidelines of the standards for protection against radiation outlined in 10 CFR 20 AEC General Design Criteria.

TEST NUMBER 3 -- FUEL LOADING

Purpose

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

Description

Prior to fuel loading, control rods and neutron sources and detectors will be installed and tested. Fuel loading will commence with the loading of four fuel assemblies around the central neutron source. Fuel loading will be accomplished by loading complete control coils that sequentially complete each face of an ever-increasing square core loading in a counterclockwise direction.

Control rod drive functional tests are performed during the last week before fuel loading.

Criteria

Level 1

The partially loaded core must be subcritical by at least 0.38 percent k/k with the analytically strongest rod fully withdrawn.

TEST NUMBER 4 -- FULL CORE SHUTDOWN MARGIN

Purpose

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

Description

This test will be performed in the fully loaded core at ambient temperature in the xenon-free condition. The shutdown margin will be measured by withdrawing the analytically strongest rod or the equivalent (another rod plus an added reactivity) and one or more additional rods which have been calibrated by calculation until criticality is reached.

Criteria

Level 1

The shutdown margin of the fully loaded core with the analytically strongest rod withdrawn must be at least 0.38 percent k/k (plus an additional margin for exposure to be determined later).

Level 2

Criticality should occur within 1.0 percent k/k of the predicted rod configuration.

TEST NUMBER 5 -- CONTROL ROD DRIVE SYSTEM

Purpose

The purposes of the Control Rod Drive System test are (a) to demonstrate that the Control Rod Drive (CRD) System operates properly over the full range of primary coolant temperatures and pressures from ambient to operating, and (b) to determine the initial operating characteristics of the entire CRD system.

Description

The CRD tests performed during Phases II through IV of the startup test program are designed as an extension of the tests performed during the preoperational CRD system tests. Thus, after it is verified that all control rod drives operate properly when installed, they are tested periodically during heatup to assure that there is no significant binding caused by thermal expansion of the core components. A list of all control rod drive tests to be performed during startup testing is given below.

CONTROL ROD DRIVE SYSTEM TESTS

Test Description	Accumulator Pressure	Preop Tests	Reactor Pressure with Core Loaded psig (kg/cm) 0	600 (42.2)	800 (58.2)	Rated
Position Indication		all	all			
Normal Times Insert/ Withdrawn			all	all	4*	
Coupling		all	all***			
Friction			all		4*	
Scram	Normal	all	all	4*	4*	all
Scram	Minimum		4*			
Scram	Zero				4*	
Scram (Scram Discharge Volume High Level)	Normal	4(Full core scram)			4*	
Scram	Normal				4**	

*Value refers to the four slowest CRDs as determined from the normal accumulator pressure scram test at ambient reactor pressure. Throughout the procedure, "the four slowest CRDs" implies the four slowest compatible with rod worth minimizer and CRD sequence requirements.

**Scram times of the four slowest CRDs will be determined at 25 percent and 100 percent of rated power during planned reactor scrams.

***Establish initially that this check is normal operating procedures.

NOTE: Single CRD scrams should be performed with the charging valve closed (do not ride the charging pump head).

Criteria

Level 1

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

The mean scram time of all operable CRDs must not exceed the following times: (Scram time is measured from the time the pilot scram valve solenoids are deenergized.)

Percent Inserted	Scram Time (Seconds) Vessel Dome Pressure 950 psig (66.9 kg/cm)	Scram Time (Seconds) Vessel Dome Pressure 950 psig (66.9 kg/sm)
5	0.375	0.475
20	0.90	1.100
50	2.0	2.0
90	3.5	3.5

The mean scram time of the three fastest CRD's in a two-by-two array must not exceed the following times: (Scram time is measured from the time the pilot scram valve solenoids are deenergized.)

Percent Inserted	Scram Time (Seconds) Vessel Dome Pressure 950 psig (66.9 kg/cm)	Scram Time (Seconds) Vessel Dome Pressure 950 psig (66.9 kg/sm)
5	0.398	0.504
20	0.954	1.166
50	2.120	2.120
90	3.800	3.800

Level 2

Each CRD must have a normal insert or withdrawn speed of 3.0 ± 0.6 inches per second (7.62 sma,b 1.52 cm/sec), indicated by a full 12-foot stroke in 40 to 60 seconds.

With respect to the control rod drive friction tests, if the differential pressure variation exceeds 15 psid (1 kg/cm) for a continuous drive in, a settling test must be performed, in which case, the differential setting pressure should not be less than 30 psid (2.1 kg/cm) nor should it vary by more than 10 psid (0.7 kg/cm) over a full stroke.

Scram times with normal accumulator charge should fall within the time limits indicated on Figure 5.3-1 of the Startup Test Instructions.

TEST NUMBER 6 -- SRM PERFORMANCE AND CONTROL ROD SEQUENCE

Purpose

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

Description

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

A withdrawal sequence has been calculated which completely specifies control rod withdrawals from the all-rods-in condition to the rated power configuration. Critical rod patterns will be recorded periodically as the reactor is heated to rated temperature.

Movement of rods in a prescribed sequence is monitored by the Rod Worth Minimizer and the Rod Sequence Control System, which will prevent out-of-sequence withdrawal and insertions.

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

Data will be obtained to verify the relationship between core power and first stage turbine pressure to ensure that the RSCS properly fulfills its intended function up to the required power level.

Criteria

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

There must be a minimum count rate of 3 counts/second on the required operable SRM's or Fuel Loading Chambers.

The IRM's must be on scale before the SRM's exceed the rod block set point.

The Rod Sequence Control System shall be operable as specified in the Technical Specification.

TEST NUMBER 9 -- WATER LEVEL MEASUREMENT

Purpose

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

Description

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

Criteria

Not applicable.

TEST NUMBER 10 -- IRM PERFORMANCE

Purpose

The purpose of this test is to adjust the Intermediate Range Monitor System to obtain an optimum overlap with the SRM and APRM systems.

Description

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

Criteria

Level 1

Each IRM channel must be adjusted to that overlap with the SRM's and APRM's is assured. The IRM's must produce a scram at 96 percent of full scale.

TEST NUMBER 11 -- LPRM CALIBRATION

Purpose

The purpose of this test is to calibrate the Local Power Range Monitoring System.

Description

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

Criteria

Level 1

The meter readings of each LPRM chamber will be proportional to the neutron flux in the narrow-narrow water gap at the height of the chamber.

TEST NUMBER 12 -- APRM CALIBRATION

Purpose

The purpose of this test is to calibrate the Average Power Range Monitor System.

Description

A heat balance will generally be made each shift and after each major power level change. Each APRM channel reading will be adjusted to be consistent with the core thermal power as determined from the heat balance. During heatup a preliminary

calibration will be made by adjusting the APRM amplified gains so that the APRM readings agree with the results of a constant heatup rate heat balance. The APRM's should be recalibrated in the power range by a heat balance as soon as adequate feedwater indication is available.

Criteria

Level 1

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

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

In the startup mode, all APRM channels must produce a scram at less than or equal to 15 percent of rated thermal power.

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

Level 2

If the above criteria are satisfied then the APRM channels will be considered to be reading accurately if they do not read greater than the actual core thermal power by more than 7 percent of rated power.

TEST NUMBER 13 -- PROCESS COMPUTER

Purpose

The purpose of this test is to verify the performance of the process computer under plant operating conditions.

Description

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

the balance-of-plant are correct. At steady-state power conditions the Dynamic System Test Case will be performed.

Criteria

Level 1

Not applicable.

Level 2

Programs OD-1 P1 and OD-6 will be considered operational when:

1. The MCPR calculated by BUCLE and the process computer either:

    a. Are in the same fuel assembly and do not differ in value by more than 2 percent, or

    b. For the case in which the MCPR calculated by the process computer is in a different assembly than that calculated by BUCLE, for each assembly the MCPR and CPR calculated by the two methods shall agree within 2 percent.

2. The maximum LHGR calculated by BUCLE and the process computer either:

    a. Are in the same fuel assembly and do not differ in value by more than 2 percent, or

    b. For the case in which the maximum LHGR calculated by the process computer is in a different assembly than that calculated by BUCLE, for each assembly the maximum LHGR and LHGR calculated by the two methods shall agree within 2 percent.

3. The MAPLHGR calculated by BUCLE and the process computer either:

    a. Are in the same fuel assembly and do not differ in value by more than 2 percent, or

    b. For the case in which the MAPLHGR calculated by the process computer is in a different assembly than that calculated by BUCLE, for each assembly the MAPLHGR and APLHGR calculated by the two methods shall agree within 2 percent.

TEST NUMBER 14 -- RCIC SYSTEM

Purpose

The purpose of this test is to verify the proper operation of the Reactor Core Isolation Cooling (RCIC) system over its expected operating pressure range.

Description

The RCIC system test consists of two parts: injection to the condensate storage tank and injection to the reactor vessel. The CST injections consist of controlled and quick starts at reactor pressures ranging from 150 psig (10.5 kg/cm) to rated, with corresponding pump discharge pressures throttled between 250 psig (17.6 kg/cm) and 1220 psig (85.8 kg/cm). During this part of the testing, proper operation of the system will be verified and adjustments made as required to meet this criteria. The reactor vessel injection will consist of a cold quick start of the system with all flow routed to the reactor vessel at 25 percent power.

Criteria

Level 1

The time from actuating signal to required flow must be less than 30 seconds at any reactor pressure between 150 psig (10.5 kg/cm) and rated.

With pump discharge at any pressure between 150 psig (10.5 kg/cm) and 1220 psig (85.8 kg/cm), the required flow is 600 grm. (The limit of 1220 psig includes a conservatively high value of 100 psi for line losses. The measured value may be used if available.

The RCIC turbine must not trip off during startup.

Level 2

The turbine gland seal condenser system shall be capable of preventing steam leakage to the atmosphere. The delta P switch for the RCIC steam supply line high flow isolation trip shall be adjusted to actuate at 300 percent of the maximum required steady state flow.

TEST NUMBER 15 -- HPCI SYSTEM

Purpose

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

Description

The HPCI system test consists of two parts: injection to the condensate storage tank and injection to the reactor vessel. The CST injections consist of controlled and quick starts at three reactor pressures ranging from 150 psig (10.5 kg/cm) to rated, with corresponding pump discharge pressures throttled between 250 psig (17.6 kg/cm) and 1220 psig (85.8 kg/cm). During this part of the testing, proper operation of the system will be verified and adjustments made as required to meet the criteria. The reactor vessel injection will consist of a cold quick start of the system with all flow routed to the reactor vessel at 50 percent power.

Criteria

Level 1

The time from actuating signal to required flow must be less than 25 seconds at any reactor pressure between 150 psig (10.5 kg/cm) and rated.

With pump discharge at any pressure between 150 psig (10.5 kg/cm) and 1220 psig (85.8 kg/cm) the flow should be at least 5000 gpm. (The limit of 1220 psig includes a conservatively high value of 100 psi for line losses. The measured value may be used if available.)

The HPCI turbine must not trip off during startup.

Level 2

The turbine gland seal condenser system shall be capable of preventing steam leakage to the atmosphere.

The delta P switch for the HPCI steam supply line high flow isolation trip shall be adjusted to actuate at 225 percent of the maximum required steady state flow.

TEST NUMBER 16 -- SELECTED PROCESS TEMPERATURES

Purpose

The purposes of this procedure are to establish the proper setting of the low speed limiter for the recirculation pumps and to provide assurance that the measured bottom head drain temperature corresponds to bottom head coolant temperature during normal operations.

Description

During initial heatup while at hot standby conditions, the bottom drain line temperature and applicable reactor parameters are monitored as the recirculation pump speed is slowly lowered from 30 percent of maximum pump speed to either minimum stable speed or 20 percent of maximum pump speed, whichever is the greater. The parameters above are recorded during pump trips as well. Utilizing this data the first purpose as stated above is satisfied. The second purpose is satisfied by comparing recirculation loop coolant temperature with bottom drain line temperature when core flow is 100 percent.

Criteria

Level 1

The reactor recirculation pump shall not be operated unless the coolant temperatures between the upper and lower regions of the reactor vessel are within 145°F (80°C).

Level 2

The bottom head coolant temperature as measured by the bottom drain line thermocouple should be within 50°F (28°C) of reactor coolant saturation temperature.

TEST NUMBER 17 -- SYSTEM EXPANSION

Purpose

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

Description

Observe and record the horizontal and vertical positions of major equipment and piping in the Nuclear Steam Supply System and auxiliary systems to ensure components are free to move as designed. Adjust as necessary for freedom of movement.

Criteria

Level 1

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

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

The shock suppressor pistons must be within their operating range.

Electrical cables shall not be fully stretched.

Level 2

At the steady-state condition the displacements of instrumented points with displacement measuring devices shall not vary from the calculated values by more than 50 percent or 0.5 inch (1.27 cm), whichever is smaller. Displacements of less than 0.25 inch (0.64 cm) can be neglected since 50 percent of this value is bordering on the accuracy of measurement. If measured displacements do not meet these criteria, the piping design engineer must be contacted to analyze the data with regard to design stresses.

During the heatup cycle the trace of the instrumented points shall fall within a range of 150 percent of the calculated value from the initial cold position in the direction of the calculated value, and 50 percent of the calculated value from the initial position in the opposite direction of the calculated value. Hangers will be in their operation range (between the hot and cold settings).

TEST NUMBER 18 -- CORE POWER DISTRIBUTION

Purpose

The purposes of this test are to (a) confirm the reproducibility of the TIP system readings, (b) determine the core power distribution in three dimensions, and (c) determine core power symmetry.

Description

Core power distribution data will be obtained during the power ascension program. Axial power traces will be obtained at each TIP location at the 50-percent power level. At least one and possibly more sets of TIP data will be taken above the 75-percent power level. These data sets will be used to determine overall TIP uncertainty including random noise and geometrical uncertainties. TIP data taken and used in this test will be submitted to NRC with the summary test report.

TIP data will be obtained at the power levels discussed above with the reactor operating with a symmetric rod pattern and at steady-state conditions. The total TIP uncertainty for the test will be calculated by averaging the total TIP uncertainty determined from each set of TIP data taken. The total TIP uncertainty is made up of random noise and geometric components.

The random noise uncertainty is obtained from four traces from each of the five TIP machines. The standard deviation due to random noise is calculated from the individual deviations of nodal power at each nodal level 5 through 22. The total random noise deviation is the average of the standard deviations for each nodal

level. The geometric uncertainty is determined for each complete set by dividing, for each symmetric TIP pair, the nodal BASE value in the upper left half of the core by its counterpart in the lower right half. The average and standard deviations of these ratios are calculated. The geometric standard deviation is calculated by first dividing the deviation of the ratios by 2 and then statistically subtracting the random noise uncertainty. The total TIP uncertainty is calculated by statistically adding the random noise and geometrical uncertainties.

Because the TIP/LPRM instrument tubes are arranged symmetrically about the core diagonal, only half-core symmetry about this diagonal can be measured. A meaningful level criteria cannot be applied to this measurement. Any assymetry, as measured by the TIP system, will be accounted for on the calculation of MCPR using the GETAB method (or later the GEXL as it is incorporated). The thermal limits of the four assemblies surrounding an LPRM string are calculated based on the LPRM/TIP data from that string, independent of the data from its symmetric counterpart. This TIP/LPRM data is also reflected to its three pseudo locations with the appropriate corrections and uncertainties applied for the differences in bundle configurations and fuel types.

Criteria

Level 1

The total TIP uncertainty (including random noise and geometrical uncertainties) shall be less than 7.8 percent. This total TIP uncertainty will be obtained by averaging the total uncertainty for all data sets obtained. A minimum of two data is sufficient for the determination of total TIP uncertainty. However, if the first two data sets do not meet the above criteria, testing may be continued and up to six data sets obtained and compared with the criteria. If the 7.8 percent total TIP uncertainty criteria has not been met by the six sets of data, testing may continue and additional data sets may be obtained provided (a) the MCPR limit is adjusted to reflect the TIP uncertainty determined by the six data sets, (b) the NRC is informed of the adjusted MCPR limit, (c) the data generated from the six sets of data is transmitted to the NRC, and (d) TVA's intentions for continuing to test and expand the data base are provided to NRC. If the total TIP uncertainty is reduced by taking additional sets of data to expand the data base, the MCPR limit will be adjusted accordingly until the 7.8 percent total TIP uncertainty is met. At this time, the MCPR limit will be returned to its original value.

Level 2

Not applicable

TEST NUMBER 19 -- CORE PERFORMANCE (Unit 2 Only)

Purpose

The purpose of this test is to evaluate the core performance parameters of core flow rate, core thermal power level, maximum fuel rod surface heat flux, core minimum critical heat flux ratio (MCHFR), Minimum Bundle Power Ratio (MBPR) and Maximum Average Planar Linear Heat Generation Rate (MAPLHGR).

Description

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

Criteria

Level 1

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

The Minimum Bundle Power Ratio (MBPR) shall be maintained greater than or equal to 1.0.

The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) shall not exceed the limits of the Plant Technical Specification.

MCHFR shall be maintained at or above the flow dependent Minimum Fuel Warranty MCHFR Limit (Line "B," Figure 19.3-2, of the Startup Test Instructions).

Steady-state reactor power shall be limited to 3293 mWt and values on or below the design flow control line (defined at 3440 MWt with core flow of at least 102.5 x 10 lb/hr).

TEST NUMBER 19 -- CORE PERFORMANCE (Brown Ferry Unit 3 Only)

Purpose

The purposes of this test are (a) to evaluate the core thermal power and (b) to evaluate the following core performance parameters: (1) maximum linear heat

generation rate (MLHGR), (2) minimum critical power ratio (MCPR), and (3) maximum average planar heat generation rate (MAPLHGR).

Description

The core performance evaluation is intended to determine the principal thermal and hydraulic parameters associated with core behavior. These parameters are core flow, core thermal power, MLHGR, MCPR, and MAPLHGR. These core parameters will be evaluated by manual calculations, the process computer, or the off-line computer program BUCLE. If the process computer is used as a primary means to obtain these parameters, it must be proved that it agrees with BUCLE within 2 percent on all thermal parameters (see test number 13) or the results must be corrected to do so. If the BUCLE and process computer results do not agree within 2 percent for any thermal parameter, the parameter calculated by the process computer will be corrected by a multiplication factor to bring it within the 2-percent criteria.

Criteria

Level 1

The maximum linear heat generation rate (LHGR) of any rod during steady-state conditions shall not exceed the limit specified by the technical specifications.

Steady-state reactor power shall be limited to 3,293 MWt and values on or below the design flow control line (defined as 3,440 MWt with core flow of at least 102.5 x 10 lb/hr).

The minimum critical power ratio (MCPR) shall not exceed the limits specified by the technical specifications.

The maximum average planar linear heat generation rate (MAPLHGR) shall not exceed the limits of the technical specifications.

TEST NUMBER 20 -- ELECTRICAL OUTPUT AND HEAT RATE
(Browns Ferry Unit 2 Only)

Purpose

The purpose of this test is to demonstrate that the plant net electrical output and net heat rate requirements are satisfied.

Description

The plant gross electrical output and net heat rate will be measured during sustained operation at rated conditions. The gross electrical output is defined as the gross electrical output measured at the generator terminals and must be maintained for 300 hours. The net plant heat rate is defined as the thermal output from the reactor less the thermal content in the feedwater supplied to the reactor all divided by the net electrical output. All corrections for losses and auxiliary loads will be agreed to prior to the start of the test. The 2-hour net plant heat rate test would normally be done concurrently with the net plant electrical output test although this is not necessary.

Criteria

Level 1

The guaranteed performance calls for a gross output of 1,098,420 kWe at a reactor thermal power of 3293.0 MW and a net plant heat rate of 10,359 Btu/kWh. The power factor shall be 0.9 and the generator hydrogen pressure shall be 75.0 psig.

TEST NUMBER 20 -- STEAM PRODUCTION (Browns Ferry Unit 3 Only)

Purpose

The purpose of this test is to demonstrate that the Nuclear Steam Supply System is providing steam sufficient to satisfy all appropriate warranties.

Description

Operate continuously for 300 hours at rated reactor conditions. When it is determined that all plant conditions are stabilized, the steam production rate will be measured during two 2-hour periods at conditions prescribed in the Nuclear Steam Generating System warranty.

Criteria

Level 1

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

The Nuclear Steam Supply System must produce 13,422,000 lbs/hr of steam of not less than 99.7 percent of quality and 985 psia pressure at the second isolation valve.

This output is contingent upon the feedwater flow being 13,372,000 lbs/hr at 378°F, and CRD flow being 50,000 lbs/hr at 80°F.

TEST NUMBER 21 -- FLUX RESPONSE TO RODS

Purpose

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

Description

Rod movement tests will be made at chosen power levels to prove that the transient response of the reactor to a reactivity perturbation is sufficiently stable over the full range of reactor power and flow conditions. The signal from a nearby LPRM will be recorded and evaluated to determine the local core dynamic effects of the rod movement. The control rod chosen should be strong enough to produce a 5 percent local power change and should be near the channel with most limiting thermal conditions.

Criteria

Level 1

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

Level 2

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

TEST NUMBER 22 -- PRESSURE REGULATOR

Purpose

The purposes of this test are to (a) determine the optimum settings for the pressure control loop by analysis of the transients induced in the reactor pressure control system by means of the pressure regulators, (b) to demonstrate the takeover capability of the backup pressure regulator upon failure of the controlling pressure regulator and to set spacing between the set points at an appropriate value, and (c)

to demonstrate smooth pressure control transition between the control valves and bypass valves when reactor steam generation exceeds steam used by the turbine.

Description

The pressure set point will be decreased rapidly and then increased rapidly by about 10 psi (0.7 kg/cm) and the response of the system will be measured in each case. It is desirable to accomplish the set point change in less than 1 second. At applicable test conditions the load reference set point will be set so that the transient is handled by control valves, bypass valves and both. The backup regulator will be tested by simulating a failure of the operating pressure regulator so that the backup regulator takes over control. The response of the system will be measured and evaluated and regulator settings will be optimized.

Criteria

Level 1 The decay ratio must be less than 1.0 for each process variable that exhibits oscillatory response to pressure regulator changes.

Level 2

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

Pressure control, deadband, delay, etc., if any shall produce variations in steam flow to the turbine no larger than the values of rated flow specified in the following table, as measured by gross generated electrical power.

Percent of Full Power	Percent of Rated Flow
90 - 100	0.5
70 - 90	1.5 to 0.5
70 and below	1.5

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

During the simulated failure of the controlling pressure regulator, if the set point of the backup pressure regulator is optimumly set, the backup regulator shall control the transient such that the reactor does not scram. Following a 10 psi (0.7 kg/cm) pressure set-point adjustment, the time between the set-point change and the occurrence of the pressure peak shall be 10 seconds or less.

TEST NUMBER 23 -- FEEDWATER SYSTEM

Purpose

The purposes of this test are (a) to adjust the feedwater control system for acceptable reactor water level control, (b) to demonstrate stable reactor response to subcooling changes, (c) to demonstrate the capability of the automatic core flow runback feature to prevent low water level scram following the trip of one feedwater pump.

Description

Reactor water level setpoint changes of approximately 3 to 5 inches (7.5 to 12.5 cm) will be used to evaluate and adjust the feedwater control system settings for all power and feedwater pump modes. The level set-point changes will also demonstrate core stability to subcooling changes.

One of the three operating feedwater pumps will be tripped and the automatic flow runback circuit will act to drop power to within the capacity of the remaining pump.

Criteria

Level 1

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

Level 2

The decay ratio is expected to be less than or equal to 0.25 for each process variable that exhibits oscillatory response to feedwater system changes when the plant is operating above the lower limit of the Master Flow Controller.

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

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

With the condensate system operating normally, the maximum turbine speed limit shall prevent pump damage due to cavitation.

TEST NUMBER 24 -- BYPASS VALVES

Purpose

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

Description

One of the turbine bypass valves will be tripped open and closed (after the opening disturbance has settled out) by a test switch; an opening time and closing time of less than 3 seconds is desirable. The pressure transient will be measured and evaluated to aid in making final adjustments to the pressure regulator.

Criteria

Level 1

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

Level 2

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

The maximum pressure decrease at the turbine inlet during valve opening shall not exceed 50 psi to avoid approaching low steam line pressure isolation. The regulator shall limit the pressure disturbance during valve reclosure so that a margin of at least 5 percent shall be maintained below flux scram.

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

TEST NUMBER 25 -- MAIN STEAM ISOLATION VALVES

Purpose

The purposes of this test are (a) to functionally check the main steam line isolation valves (MSIVs) for proper operation at selected power levels, (b) to determine reactor transient behavior during and following simultaneous full closure of all MSIV's and following full closure of one valve, (c) to determine isolation valve closure time, and (d) to determine maximum power at which a single valve closure can be made without scram.

Description

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

Criteria

MSIV closure time must be greater than 3 and less than 5 seconds. The initial transient rise in vessel dome pressure occurring within 20 seconds of the main steam isolation valve trip initiation shall not be greater than 150 psi, and the transient rise in simulated heat flux shall not exceed 10 percent.

Level 2

The initial transient peak in vessel dome pressure occurring within 20 seconds following initiation of the MSIV closure and the transient peak in simulated surface

heat flux shall not be more limiting than the predicted transients in the Transient Analysis Design Report (100 psi and no heat flux increase).

TEST NUMBER 26 -- RELIEF VALVES (Browns Ferry Unit 2 Only)

Purpose

The purposes of this test are (a) to verify the proper operation of the primary system relief valves, (b) to determine their capacity and response characteristics and (c) to verify their proper seating following operation.

Description

The main steam relief valves will each be opened using the "manual" control mode so that at any time only one is open. During heatup at 250 psig (17.5 kg/cm), each valve will be opened and closed to demonstrate proper functioning. Capacity of each relief valve will be determined at rated pressure by the amount of bypass or control valve closure required to maintain reactor pressure. Proper reseating of each relief valve will be verified by observation of temperatures in the relief valve discharge piping. At selected test conditions each valve will be manually actuated, and at least one valve will be timed. Additional timing data will be obtained in conjunction with those transient tests which result in automatic relief valve opening.

Criteria

Level 1

Each relief valve shall have a capacity of at least 800,000 lb/hr at an inlet pressure of 1112 psig.

Level 2

Relief valve leakage shall be low enough that the temperature measured by the thermocouples in the discharge side of the valves returns to within 10°F (5.6°C) of the temperature recorded before the valve was opened.

TEST NUMBER 26 -- RELIEF VALVES (Browns Ferry Unit 3 Only)

Purpose

The purposes of this test are (a) to verify the proper operation of the primary system relief valves, (b) to determine their capacity and response characteristics, and (c) to verify their proper seating following operation.

Description

The main steam relief valves will each be opened using the "manual" control mode so that at any time only one is open. During heatup at 250 psig (17.5 kg/cm), each valve will be opened and closed to demonstrate proper functioning. Capacity of each relief valve will be determined at rated pressure by the amount of bypass or control valve closure required to maintain reactor pressure. Proper reseating of each relief valve will be verified by observation of temperatures in the relief valve discharge piping. At selected test conditions each valve will be manually actuated, and at least one valve will be timed.

Criteria

Level 1

The sum total of capacities from 11 relief valves shall be equal to or greater than 8.83 x 10 lb/hr 2 percent corrected for an inlet pressure of 1112 psig.

Level 2

Relief valve leakage shall be low enough that the temperature measured by the thermocouples in the discharge side of the valves returns to within 10°F (5.6°C) of the temperature recorded before the valve was opened.

Each individual relief valve shall have a minimum capacity of 720,000 corrected for an inlet pressure of 1112 psig.

TEST NUMBER 27 -- TURBINE TRIP AND GENERATOR LOAD REJECTION

Purpose

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

Description

The turbine stop valves will be tripped at selected reactor power levels and the main generator breaker will be tripped in such a way that a load imbalance trip occurs. Several reactor and turbine operating parameters will be monitored to evaluate the response of the bypass valves, relief valves, and reactor protection system (RPS).

Additionally, the peak values and change rates of reactor steam pressure and heat flux will be determined. The effect of recirculation pump overspeed, if any, will be checked during the generator load rejection. The ability to ride through a load rejection within bypass capacity without a scram will be demonstrated.

Criteria

Level 1

The initial transient rise in vessel dome pressure occurring within 10 seconds of the turbine/generator trip initiation shall not be greater than 150 psi and the transient rise in simulated heat flux shall not exceed 10 percent.

The turbine stop valves must begin to close before the control valves for the turbine trip. The turbine control valves must begin to close before the stop valves during the generator load rejection.

Following fast closure of the turbine stop and control valves, a reactor scram shall occur if the turbine first stage pressure is greater than 154 psig.

Feedwater systems must prevent flooding of the steamline following the transients.

Level 2

The initial transient rise in vessel dome pressure occurring within 10 seconds of the turbine/generator trip initiation and the transient rise in simulated surface heat flux shall not be more limiting than the predicted transient presented in the Transient Analysis Design Report (100 psi and no heat flux increase).

The pressure regulator must prevent a low-pressure reactor isolation. The feedwater controller must prevent a low-level initiation of the HPCI and MSIV's as long as feedwater remains available.

The load rejection within bypass capacity must not cause a scram. The trip scram function for higher power levels must meet RPS specifications.

For the case of turbine trip at 75-percent power, the measured transient parameters will be compared with the predicted values. If any parameter is significantly different from the predicted values, the test will be repeated at 100-percent power.

TEST NUMBER 29 -- FLOW CONTROL (Browns Ferry Unit 2 Only)

Purpose

The purposes of this test are (a) to determine the plant response to changes in the recirculation flow, (b) to optimize the setpoints of the Master Flow Controller and Transient Pressure Set-Point Adjuster, and (c) to demonstrate the plant load following capability in Master Manual, and Automatic Flow Control modes.

Description

Various process variables will be recorded while load changes (increase and decrease) are introduced into the recirculation flow control system at chosen points on the 50 percent, 75 percent, and 100 percent load lines. The master flow controller and transient pressure set-point adjuster will be set to achieve acceptable performance over the entire auto flow control range.

Ramp changes will be made with the concurrence of the customer at rates within the range of 10 percent-30 percent power per minute. Load following capability will be demonstrated in the automatic and master manual flow control modes.

Criteria

Level 1

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

Level 2

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

Following a 10 percent step change in load demand on the 100 percent load line, the reactor must not scram and the load change must be achieved within 40 seconds.

The automatic flow control range must be at least 80 percent to 100 percent power along the 100 percent load line. The Master Flow Controller output limiters shall be set accordingly. The load change resulting from a maximum ramp increase in load reference from 80 percent to 100 percent load must be achieved within 1 minute without reactor scram.

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

TEST NUMBER 30 -- RECIRCULATION SYSTEM (Browns Ferry Unit 2 Only)

Purpose

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

Description

Single and both recirculation pumps will be tripped at various power levels. Two pump trips will be initiated by tripping the MG set drive motors. One single pump trip at 50 percent power will be initiated by opening the generator field breaker. The remaining single pump trips are to be initiated by tripping the MG set drive motor. Reactor operating parameters will be recorded during the transient and at steady-state conditions. MCHFR evaluations will be made for conditions encountered during the transient.

With the recirculation pumps operating at the speed corresponding to rated flow at rated power, power will be reduced by inserting rods to 23 percent power where the recirculation pumps will automatically run back to 20 percent speed. A check will be made to determine if recirculation or jet pump cavitation occurs.

Criteria

Level 1

MCHFR shall be greater than 1.0 during the pump trip transient.

Level 2

For each pump trip test, the minimum transient MCHFR based on operating data divided by the corresponding minimum transient

MCHFR evaluated from design values is expected to be equal to or greater than 1.0.

TEST NUMBER 30 -- RECIRCULATION SYSTEM (Browns Ferry Unit 3 Only)

Purpose

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

Description

Single and both recirculation pumps will be tripped at various power levels. Two pump trips will be initiated by tripping the MG set drive motors. One single pump trip at 50 percent power will be initiated by opening the generator field breaker. The remaining single pump trips are to be initiated by tripping the MG set drive motor. Reactor operating parameters will be recorded during the transient and at steady-state conditions.

Criteria

Level 1

Not applicable.

Level 2

The power and flow coastdowns are expected to agree with precalculated power and flow coastdown rates. The plant shall not scram as a result of a high level turbine trip.

TEST NUMBER 31 -- LOSS OF TURBINE-GENERATOR AND OFFSITE POWER

Purpose

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

Description

The loss of auxiliary power test will be performed at 20 to 30 percent of rated power. The proper response of reactor plant equipment, automatic switching equipment,

and the proper sequencing of the diesel generator load will be checked. Appropriate reactor parameters will be recorded during the resultant transient.

Criteria

Level 1

The initial transient rise in vessel dome pressure occurring within 10 seconds of turbine/generator trip action when initiated simultaneously with loss of offsite power when performed at 25-percent power shall not exceed 150 psi and the simulated heat flux rise shall not exceed 10 percent.

All safety systems, such as the RPS, its diesel generators, and the RCIC and HPCI, must function properly without manual assistance.

Level 2

The initial transient rise in vessel dome pressure occurring within 10 seconds of turbine/generator trip shall not be greater than 75 psi, and there shall be no significant increase in simulated heat flux.

Normal reactor cooling water systems should be able to maintain adequate pressure suppression pool water temperature, adequate drywell cooling, and prevent actuation of the autodepressurization system.

TEST NUMBER 32 -- RECIRCULATION MG SET SPEED CONTROL (Browns Ferry Unit 2 Only)

Purpose

The purposes of this test are (a) to determine the speed control characteristics of the MG sets in the recirculation control system, (b) to obtain acceptable speed control system performance, and (c) to determine the maximum allowable pump speed.

Description

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

Criteria

Level 1

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

Level 2

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

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

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

TEST NUMBER 32 -- RECIRCULATION SPEED CONTROL AND LOAD FOLLOWING (Browns Ferry Unit 3 Only)

Purpose

The purposes of this test are (a) to determine correct gain for optimum performance of individual recirculation loops, (b) to determine that the recirculation loops are correctly set up for desired speed range and for acceptable variations in loop gain, (c) to demonstrate plant response to changes in recirculation flow.

Description

During the initial startup testing, data will be collected for programming the CAM's in the scoop tube positioner feedback loops to reduce the effect of abrupt nonlinearities in the coupler characteristics. The time response of the individual recirculation pump speed loops will be optimized for smooth control while staying within the PCIOMR. The response of the speed loops will then be checked by step changes in speed demand at all test conditions. The master flow controller and transient pressure setpoint adjuster will be set to achieve acceptable performance over the auto flow control range.

Criteria

Level 1

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

Level 2

The decay ratio should be less than 0.25 for any process variable that exhibits oscillatory response to 10 percent speed change inputs in local or master manual modes.

Steady-state limit cycles, if any exist, must not cause turbine steam flow to vary in excess of 0.5 percent rated flow as measured by the gross generator electrical power output.

TEST NUMBER 33 -- MAIN TURBINE STOP VALVE SURVEILLANCE TEST

Purpose

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

Description

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

Criteria

Level 1

Not applicable

Criteria

Level 1

The release of radioactive gaseous and particulate effluents must not exceed the limits specified in the site technical specifications. There shall be no loss of flow of dilution steam to the noncondensing stage when the steam jet air ejectors are pumping.

Level 2

The system flow, pressure, temperature, and relative humidity shall comply with design specifications. The catalytic recombiner, the hydrogen analyzer, the activated carbon beds, and the filters shall be working properly during operation.

13.5.3 Nuclear System Startup Test Restrictions

All operations and tests must comply with the warranty limitations specified by GE, and primarily with the safety limitations and limiting conditions for operations specified by licensing authorities. Restrictions are minimized because the prime objective of the startup program is to demonstrate that the plant is capable of operating safely and satisfactorily up to rated power. Any restrictions are detailed in the written startup procedures and in supporting instructions. An example is that during initial fuel loading operations, a special test neutron source is installed near the first fuel loading location such that sufficient neutron flux is obtained to provide positive indications of neutron multiplication. The source is enclosed in a rod-like holder and is installed in the core during initial criticality measurements. The source is removed when no longer needed for initial tests, and is replaced with the normal operational neutron sources, which provide ample neutron flux for nuclear instrument readings to meet the criteria on noise, signal-to-noise ratio and response to changes in core reactivity.

Several more-sensitive neutron detectors may be submerged in the core near the loading location and surrounding the test source to provide additional information during initial fuel loading. These detectors will be removed following completion of the open vessel reactivity testing.

During the startup program the power level will not exceed the upper power level for the test plateau (flow control line) at which testing is being performed. The upper power level values for the test plateaus are stated in Tables 13.5-4, 5 and 6. These upper power level values for testing at the 50 percent, 75 percent, and 100 percent flow control lines are 60 percent, 85 percent, and 100 percent power respectively.

TEST NUMBER 75 -- REACTOR SHUTDOWN FROM OUTSIDE THE MAIN CONTROL ROOM

Purpose

The purpose of this test is to demonstrate that the plant is designed and constructed with adequate instruments and controls to permit safe reactor shutdown from outside the main control room and maintain it in a safe condition, that the minimum number of personnel required by the technical specifications is adequate without affecting the safe, continuous operation of the other units, and that the plant emergency operating instruction is adequate.

Description

With the unit operating at greater than 10-percent generator output, the reactor will be scrammed by closing the MSIV's from the backup control station. Operators will man their backup control stations as described in the emergency operating instruction. The RCIC system will be operated from the backup controls to supply water to the reactor vessel. The suppression pool cooling system shall be placed in operation using the backup controls.

An extra licensed operator will remain in the main control room to assure that the test is terminated and control returned to normal if any unexpected conditions occur. The test will be terminated when it is assured that the reactor can be maintained in a safe hot standby condition from the backup controls.

Criteria

Level 1

Not applicable.

Level 2

Reactor scram initiated from outside the control room must occur. Reactor water level must be maintained greater than 490 inches above vessel 0 and less than the high level turbine trip point.

The RHR and RHRSW pumps and control valves shall be operable from the backup controls to initiate suppression pool cooling.

The minimum number of shift personnel as specified in the technical specifications is adequate for shutdown from outside the main control room.

Level 2

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

The pump available NPSH will be 13 feet or greater during the hot standby mode defined in the process diagrams.

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

TEST NUMBER 71 -- RESIDUAL HEAT REMOVAL SYSTEM

Purpose

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

Description

With the reactor at 100 psig (3.5 kg/cm) or less, the shutdown cooling mode of the RHR system will be demonstrated. The suppression pool cooling mode will also be demonstrated unless its functionability is shown elsewhere.

Criteria

Level 1

Not applicable.

Level 2

The heat removal capability of each RHR heat exchanger in the shutdown cooling mode shall be at least 187 x 10 Btu/hr when the inlet flows and temperatures are as indicated on the process diagrams.

TEST NUMBER 72 -- DRYWELL ATMOSPHERE COOLING SYSTEM

Purpose

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

Description

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

Criteria

Level 1

Not applicable.

Level 2

The heat removal capability of the drywell coolers shall be approximately 5.19 x 10 Btu/hr with eight fans and coils in operation.

The drywell cooling system shall have a standby capability of 25 percent of the above heat removal capability.

The drywell cooling system shall maintain temperatures in the drywell below the following design values during normal operation.

Average temperature throughout the drywell	-	150°F
Maximum around the recirculating pump motors	-	135°F
Maximum above the bulkhead	-	200°F
Maximum all other areas	-	180°F

Maintain a uniform circumferential temperature of the refueling bellows/bulkhead within 25°F point-to-point variation.

Within 10 hours following shutdown the average temperature throughout the drywell will be within 15°F of the reactor building closed cooling water inlet temperature.

TEST NUMBER 73 -- COOLING WATER SYSTEMS

Purpose

The purpose of this test is to verify that the performance of the Reactor Building Closed Cooling Water (RBCCW) and the raw cooling water systems is adequate with the reactor at rated temperature.

Description

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

Criteria

Level 1

Not applicable.

Level 2

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

The RBCCW system was designed to transfer a maximum heat load of 31.3 x 10 Btu/hr in order to limit equipment inlet water temperature to 100°F, assuming a service (raw cooling) water inlet temperature of 90°F.

TEST NUMBER 74 -- OFF GAS SYSTEM*

Purpose

The purposes of this test are to verify the proper operation of the Off Gas System over its expected operating parameters and to determine the performance of the activated carbon adsorbers.

Description

The pressure, temperature, relative humidity, system flow, and percentage of radiolytic hydrogen in the off gas are periodically monitored during startup and at

steadystate conditions. Provided that measurable and sufficient fission gases and fission gas daughter products are present in the off gas, decontamination factors across the pre- and post-filters and several charcoal beds are determined. The performance of the catalytic recombiner will be compared with the Catalytic Recombiner Guaranteed Performance Curve.

Criteria

Level 1

The release of radioactive gaseous and particulate effluents must not exceed the limits specified in the site technical specifications. There shall be no loss of flow of dilution steam to the noncondensing stage when the steam jet air ejectors are pumping.

Level 2

The system flow, pressure, temperature, and relative humidity shall comply with design specifications. The catalytic recombiner, the hydrogen analyzer, the activated carbon beds, and the filters shall be working properly during operation.

*Applies to Unit 3 and, subsequent to equipment installation, to Units 1 and 2.