a. Accomplish a controlled, orderly, and safe initial core loading

b. Accomplish a controlled, orderly, and safe initial criticality and heatup

c. Conduct low power testing sufficient to ensure that design parameters are satisfied and safety analysis assumptions are correct or conservative

d. Perform a controlled, orderly, and safe power ascension

14.2.2 ORGANIZATION AND STAFFING

The Superintendent of Plant - Susquehanna, has overall responsibility for the Initial Test Program. The Plant Staff and Integrated Startup Group (ISG) conduct the different phases of the test program. Responsibility for the ISG may be delegated to the Assistant Superintendent of Plant-Outages. In addition to these basic organizational units the Superintendent of Plant - Susquehanna is assisted by two review organizations, the Plant Operations Review Committee (PORC) and the Test Review Board (TRB). The organization, authority, responsibility, and degree of participation of each of these organizational units during the Initial Test Program are described in the following sections.

14.2.2.1 Plant Staff

The Plant Staff consists of the permanent onsite PP&L personnel responsible for the safe operation and proper maintenance of the plant. Chapter 13 describes the Plant Staff organization. This section also establishes responsibilities, reporting relationships, and minimum qualification requirements for principal Plant Staff supervisory personnel.

The Plant Staff also includes the Startup Test Group which is a temporary group established to prepare for and implement the Startup Test Program. The Startup Test Group Supervisor reports to the Technical Supervisor and supervises the activities of the Startup Test Group. Activities include: preparation and implementation of startup tests; review and analysis of startup test results; preparation of startup test reports; and participation in test planning meetings.

The Plant Staff is utilized, to the fullest extent practicable, during the Initial Test Program. Specific responsibilities of the Plant Staff during the Initial Test Program are:

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14.2.11 TEST PROGRAM SCHEDULE

The Preoperational Test Program is scheduled for 15 months duration on the Unit 1 and Common components and for 12 months duration on the remaining Unit 2 components (see Figure 14.2-4a and 14.2-4b). The subsequent Startup Test Programs are scheduled for six months on each unit.

The Preoperational Test Program sequential test schedules presented on Figures 14.2-4a and 14.2-4b offer one possible plan for an orderly and efficient progression of the program. While these sequences may be preferred, numerous alternatives exist. The schedule will be updated periodically at the jobsite to reflect construction status, manpower availability, and the required test prerequisites.

The safety-related structures, systems, and components will be preoperationally tested. The Preoperational Test Procedures are scheduled to be developed from September 1977 to January 1979 for Unit 1 and from July 1982 to July 1983 for Unit 2. Where electrical, mechanical, physical or administrative communication exists between Unit 2 and the operating Unit 1, the Unit 2 Preoperational or Acceptance Test will be divided into 2 or more procedures to facilitate proper administrative control and scheduling. Any test procedure which involves an interplant communication will contain the suffix B on the procedure number.

The schedule of Unit 1 and Unit 2 Startup Tests is presented in Figure 14.2-5. This schedule establishes the required testing as a function of test condition. The test conditions are described on Figure 14.2-6. All testing is assigned to a specific test condition for convenience even though some testing, as identified in figure 14.2-5, is performed outside the bounds of the assigned test condition. Not all subtests of a Startup Test are performed at each assigned test condition. Startup testing will be divided into three Major Test Phases, and, within the Power Ascension Test Phase, into distinct test plateaus. The testing included in each Major Test Phase and test plateau is described in Table 14.2-4. Even though this basic order of testing is required, there is still considerable flexibility in sequencing the startup testing specified to be conducted at each plateau. Detailed startup testing schedules, commensurate with the requirements of this schedule, will be developed at the job site.

14.2.12 INDIVIDUAL TEST DESCRIPTIONS

The individual preoperational tests to be conducted on safety-related structures, systems, and components are listed in Table 14.2-1 for Unit 1 and Table 14.2-6 for Unit 2. The abstracts of

these preoperational tests are contained in Subsection 14.2.12.1 in numerical order. The Startup Test Program procedures are listed in Table 14.2-3. The abstracts of Startup Test procedures are contained in Subsections 14.2.12.2 and 14.2.12.6 for Unit 1 and Unit 2, respectively in numerical order. The abstracts identify each test by title and number, describe the test objectives, specify the test prerequisites, provide a summary description of the test method, and establish the test acceptance criteria.

Unit 2 preoperational program will be scheduled and performed in a manner that will not affect the safe operation of Unit 1. Several of the Preoperational Acceptance Tests will be subdivided into A and B tests. The A portion of the test will not affect the safe operation of Unit 1, the B portion of the Preoperational Test is dependent upon an interface with Unit 1 and may require an outage on Unit 1 to perform the test. In addition to Test Review Board approval of the Preoperational Test, B designated tests will require a written Safety Evaluation submitted and test approval by the Plant Operations Review Committee. All permanent interface connections between Unit 1 and Unit 2 will be accomplished in accordance with SSES Plant Modification Procedure. Prior to performing the B designated Preoperational Test, the Work Activity Review Committee will be briefed on the impact and requirements of the test.

14.2.12.1 Unit 1 Preoperational Test Procedure Abstracts

(P2.1) 125 Volt DC System Preoperational Test

Test Objective - To demonstrate the ability of the 125 Volt dc system to perform the following:

A. The batteries can endure a complete discharge, based on their ampere hour rating, without exceeding the battery bank minimum voltage limit. (Performance Test)

B. The batteries can provide reliable stored energy to selected loads, indicated in Table 8.3-6, in the event of a design base accident. (Service Test)

C. The battery chargers can deliver their rated output.

D. The battery chargers can fully charge their associated batteries from design minimum charged state (i.e., after the service test) simultaneously providing power to the distribution panels for normal station loads.

(P100.1) - Cold Functional Test

Test Objective - To demonstrate that the plant systems are capable of operating on an integrated basis in normal and emergency modes, to demonstrate that adequate power supplies for the class IE equipment will exist, and to assure that optimum tap settings have been selected for transformers supplying power from offsite sources to class IE busses.

Prerequisites - Required system preoperational tests have been completed and plant systems are ready for operation on an integrated basis.

Test Method - Emergency Core Cooling Systems (RHR & Core Spray) are lined up in their normal standby mode. The plant electrical system is lined up per normal electrical system lineup (For Unit 1 this lineup may be different than the lineup for two unit operation). Loss of coolant accident signals are initiated with and without a loss of offsite power. Voltages and loads are adjusted, as practical, to simulate the anticipated ranges of variations. Proper response of the electrical distribution system, diesel generators, and ECCS pumps will be verified.

Acceptance Criteria - Systems performance parameters are in accordance with the applicable design documents.

14.2.12.2 - Unit 1 Startup Test Program Procedure Abstracts

All those tests comprising the Unit 1 Startup Test Program (Table 14.2-3) are discussed in this section. For each test a description is provided for test purpose, test prerequisites, test description and statement of test acceptance criteria, where applicable. Additions, deletions, and changes to these discussions are expected to occur as the test program progresses. Such modification to these discussions will be reflected in amendments to the FSAR.

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

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

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

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

(ST-1) Chemical and Radiochemical

Test Objectives - 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 documentation of radwaste liquid discharge, documentation of baseline piping radiation levels, determination of steam quality, evaluation of the Condensate Polishing system, and evaluation of the Reactor Water Cleanup system. Data for these purposes is secured from a variety of sources: plant operating records, regular routine coolant analysis, radiochemical measurements of specific nuclides, and special chemical tests.

Prerequisites - The required preoperational tests have been completed. Instrumentation has been checked or calibrated as appropriate.

Test Method - Prior to fuel loading, chemical samples are taken to ensure that reactor coolant and Fuel Pool Cooling and Cleanup System sample stations are functioning properly and to determine initial concentrations. Additionally, subsequent to fuel loading, during reactor heatup, and at each major power level change, a complete set of samples are taken to verify that all plant sample stations are functioning properly and to determine the chemical and radiochemical quality of reactor water and reactor feedwater, and performance of filters and demineralizers.

Acceptance Criteria - Level 1 - Chemical factors defined in the Technical Specifications and Fuel Warranty must be maintained within the limits specified. The activity of liquid effluents must conform to license limitations. Water quality must be known at all times and should remain within the guidelines of the Water Quality Specifications.

Test Method – Before the first fuel assembly is taken from the fuel pool and inserted into the reactor, core components (fuel support castings, blade guides, control rod drives, etc.) will be installed, tested and/or verified. This procedure begins with the steps required to assemble and load neutron sources, includes the activities necessary to monitor neutron population using specially constructed fuel loading chambers (FLCs), and culminates with the insertion of fuel assemblies into the reactor core. Fuel loading continues until the core is fully loaded, verified and ready to perform subsequent Startup Tests.

Control rod functional tests, subcriticality checks, and shutdown margin demonstrations will be performed periodically during the loading.

Acceptance Criteria – Level 1 – The partially loaded core must be subcritical by at least 0.38% delta k/k with the analytically determined, highest worth rod fully withdrawn.

(ST-4) Full Core Shutdown Margin

Test Objective – 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.

Prerequisites – The following prerequisites will be complete prior to performing the full core shutdown margin test:

a) The predicted critical rod position is available

b) The Standby Liquid Control System is available

c) Nuclear instrumentation is available with neutron count rate of at least three counts per second and signal to noise ratio greater than two to one

d) High-flux scram trips are set conservatively low

e) Instrumentation has been checked or calibrated as appropriate

Test Method – This test will be performed in the fully loaded core in the xenon-free condition. The shutdown margin test will be performed by withdrawing the control rods from the all-rods-in configuration until criticality is reached. If the highest worth rod will not be withdrawn in sequence, other rods may be withdrawn providing that the reactivity worth is equivalent. The difference between the measured Keff and the calculated Keff or the in-sequence critical will be applied to the calculated value to obtain the true shutdown margin.

Acceptance Criteria - Level 1 - The shutdown margin of the fully loaded, cold (68°F), xenon-free core occuring at the most reactive time during the cycle must be at least 0.38% delta k/k with the analytically strongest rod (or its reactivity equivalent) withdrawn. If the shutdown margin is measured at some time during the cycle other than the most reactive time, compliance with the above criterion is shown by demonstrating that the shutdown margin is 0.38% delta k/k plus an exposure dependent correction factor which corrects the shutdown margin at that time to the minimum shutdown margin.

Level 2 - Criticality should occur within ±1.0% delta k/k of the predicted critical.

(ST-5) Control Rod Drive System

Test Objective - The objectives 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.

Prerequisites - The required preoperational tests have been completed.

Test Method - The CRD tests performed during 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 in Table 14.2-5.

Acceptance Criteria - Level 1 - Each CRD must have a normal withdraw time greater than or equal to 40 seconds.

The mean scram time of all operable CRDs must not exceed the values specified in the plant technical specifications. (Scram time is measured from the time the pilot scram valve solenoids are deenergized.)

The mean scram time of the three fastest CRDs in a two by two array must not exceed the values specified in the plant technical specifications. (Scram time is measured from the time the pilot scram solenoids are deenergized)

Level 2 - Each CRD must have a normal insert speed of 3.0 ± 0.6 inches per second, 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 for a

Unfortunately, the decay heat load is insignificant during the startup test period. Use of this mode with low core exposure results in exceeding the 100°F/hr cooldown rate of the vessel.) The shutdown cooling mode will be demonstrated after a trip or a cooldown from Test Condition 6.

The RHR system steam condensing mode is used to condense steam while the reactor is isolated from the main condenser and reactor vessel water level is being maintained by RCIC. This test will demonstrate system operability and stability.

Acceptance Criteria - Level 1 - The transient response of any system-related variable to any test input must not diverge.

Level 2 - The RHR system shall be capable of operating in the steam condensing, suppression pool cooling and shutdown cooling modes at the heat exchanger capacities indicated on the process diagrams. Both simultaneous operation of RHR loops and single loop operation shall be tested in the steam condensing and shutdown cooling modes. Each RHR loop shall be tested independently in the suppression pool cooling mode. System-related variables may contain oscillatory modes of response. In these cases, the decay ratio for each controlled mode of response must be less than or equal to 0.25.

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

(ST-9) Water Level Measurement

Test Objectives - The objectives of this test are to determine actual reference leg temperature and recalibrate instruments if necessary and to verify consistent response of the upset range, narrow range and wide range level instrumentation.

Prerequisites - The required preoperational tests have been completed. All system instrumentation is installed and calibrated.

Test Method - At rated temperature and pressure under steady state conditions, the reference leg temperature will be measured and compared to the value assumed during initial calibration. If the difference of the two temperatures exceed the Acceptance Criteria, then the instruments will be recalibrated using the measured value. Data will be recorded at rated temperature and pressure and at steady state conditions to verify consistency and proper calibration of reactor vessel level instrumentation.

Acceptance Criteria - Level 1 - Not applicable.

Level 2 - The difference between the actual reference leg temperature(s) and the value(s) assumed during calibration shall be less than that amount which will result in a scale end point error of 1% of the instrument span for each range.

The Narrow Range Level indicators should agree within ±1.5 inches of their average reading.

The Wide and Upset Range Level indicators should agree within ±6 inches of their average reading.

(ST-10) IRM Performance

Test Objectives - The objective of this test is to adjust the Intermediate Range Monitor System to obtain the desired overlap with the SRM and APRM systems.

Prerequisites - The required preoperational tests have been completed.

Test Method - Initially the IRM system is set during the Preoperational Test Program. SRM-IRM and IRM-APRM overlap is verified the first time sufficient neutron flux conditions arise. After the APRM calibration, the IRM gains will be adjusted as necessary to optimize the IRM overlap with the SRMs and APRMs.

Acceptance Criteria - Level 1 - Each IRM channel must be adjusted so that overlap with the SRMs and APRMs is assured.

(ST-11) LPRM Calibration

Test Objectives - The objective of this test is to calibrate the Local Power Range Monitoring System.

Prerequisites - The required preoperational tests have been completed. Instrumentation for calibration has been checked.

Test Method - The LPRM channels will be calibrated to make the LPRM readings proportional to the neutron flux in the water gap at the chamber elevation. Prior to this calibration, LPRM response to control rod movement is verified. 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.

Acceptance Criteria - Level 1 - Not applicable.

Level 2 - Each LPRM will be within 10% of its calculated value.

(ST-12) APRM Calibration

pumps to avoid coolant temperature stratification in the reactor pressure vessel bottom head region.

Prerequisites - The required preoperational tests have been completed. System instrumentation has been calibrated.

Test Method - During initial heatup while at hot standby conditions, the bottom drain line temperature, recirculation loop suction temperature and applicable reactor parameters are monitored as the recirculation flow is slowly lowered to minimum stable flow. Utilizing this data it can be determined whether coolant temperature stratification occurs when the recirculation pumps are on and if so, what minimum recirculation flow will prevent it.

Monitoring the preceeding information during planned pump trips will determine if temperature stratification occurs in the idle recirculation loops or in the lower plenum when one or more loops are inactive.

Acceptance Criteria - Level 1 - The reactor recirculation pumps shall not be started nor flow increased unless the coolant temperatures between the steam dome and bottom head drain are within 145°F.

The recirculation pump in an idle loop must not be started unless the loop suction temperature is within 50°F of the active loop.

Level 2 - Not Applicable.

(ST-17) System Expansion

Test Objectives - The purposes of this test are to demonstrate that reactor recirculation, main steam inside containment, and those piping systems identified in Table 3.9-33 respond to thermal expansion consistent with stress analysis results. (Note that this test now includes piping previously contained in ST-38.)

Prerequisities - Instrumentation has been installed and calibrated.

Test Method - Hanger positions and locations of piping in the Nuclear Steam Supply System and piping systems identified in Table 3.9-33 inside and outside the reactor drywell are recorded prior to initial heatup and after a planned cold shutdown. During intitial heatup, a visual inspection is made at an intermediate reactor water temperature to assure components are free to move as designed. Adjustments are made as necessary. Devices for measuring continuous pipe deflections are mounted on main steam, recirculation and other selected lines. Motion during heatup is compared with calculated values.

Acceptance Criteria - Level 1 - There shall be no obstructions which will interfere with the thermal expansion of the main steam and recirculation piping systems. Piping systems identified in Table 3.9-33 will not be restrained against thermal expansion except by design intent.

Hangers on piping systems identified on Table 3.9-33 shall not be bottomed out or have the spring fully stretched. Snubbers on piping systems identified in Table 3.9-33 shall not become extended or compressed to the limits of their total travel.

The measured displacements at the established transducer locations on the main steam and recirculation systems shall not exceed the allowable values calculated for the specific points.

Level 2 - The measured displacements at the established transducer locations on the main steam and recirculation systems shall not exceed the expected values calculated for the specific points. The measured displacements at the established transducer locations on the piping systems identified in Table 3.9-33 shall be within the acceptable range calculated for the specific points.

Hangers on piping systems identified in Table 3.9-33 shall be in their operating range.

(ST-18) - TIP Uncertainty

Test Objectives - The objective of this test is to determine the uncertainty of the TIP system readings.

Prerequisites - System installation is completed and required preoperational tests are completed and verified. Instrumentation has been calibrated and installed.

Test Method - The TIP uncertainty consists of a random noise component and a geometric component, the geometric component being due to variation in the water gap geometry and TIP tube orientation from TIP location to location. Measurement of these components is obtained by taking repetitive TIP readings at a single TIP location, and by analyzing pairs of TIP readings taken at TIP locations which are symmetrical about the core diagonal of fuel loading and control rod symmetry.

The random noise uncertainty is determined from successive TIP runs made at the common location (32-33) with each of the TIP machines making six runs at index position 10. The TIP data will be obtained by simultaneous operation of the Process computer OD-2 program which provides 24 nodal TIP values for each TIP traverse. The standard deviation of the random noise is derived by taking the square root of the average of the variances at nodal levels 5 through 22, where the nodal variance is obtained

MLHGR

MCPR

MAPLHGR

Prior to the verification of the Process Computer in ST-13, an independent method will be used to calculate these parameters. After the successful completion of ST-13, the process computer will be used.

Acceptance Criteria - Level 1 - The Maximum Linear Heat Generation Rate (MLHGR) of any rod during steady-state conditions shall not exceed the limit specified by the Plant Technical Specifications.

The steady-state Minimum Critical Power Ratio (MCPR) shall not exceed the limits specified by the Plant Technical Specifications.

The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) shall not exceed the limits specified by the Plant Technical Specifications.

Steady-state reactor power shall be limited to the rated MWT and values on or below the licensed analytically determined power-flow line.

Level 2 - Not applicable.

(ST-20) - Steam Production Verification

Test Objective - The objective of this test is to demonstrate that the NSSS is providing steam sufficient to satisfy all appropriate warranties.

Prerequisites - Required preoperational tests have been completed. All required instrumentation is installed and calibrated.

Test Method - A NSSS steam output performance test of 100 hours of continuous operation at the warranted steam output will be performed.

Acceptance Criteria - Level 1 - The average reactor core thermal power (CTP) shall not exceed 3293 MWt.

The Maximum Average Planar Ratio (MAPRAT) shall be less than or equal to 1.0.

The Maximum Fraction of Limiting Critical Power Ratio (MFLCPR) shall be less than or equal to 1.0.

The Maximum Fraction of Limiting Power Density (MFLPD) shall be less than or equal to 1.0.

Level 2 - The NSSS shall be capable of supplying 13,483,000 pounds per hour of steam of not less than 99.7% quality at a pressure of 985 psia at the outlet of the second main steam line isolation valve, as based upon a final feedwater temperature of 383°F measured as near the reactor pressure vessel as practicable, and a control rod drive feed flow of 32,000 pounds per hour at 80°F.

(ST-21) - Core Power-Void Mode Response

Test Objectives - The objective of this test is to verify the stability of the core power-void dynamic response.

Prerequisites - The required preoperational tests have been completed. Instrumentation has been calibrated.

Test Method - The core power void loop mode that results from a combination of the neutron kinetics and core thermal hydraulic dynamics is least stable near the natural circulation end of the rated 100 percent power rod line. A fast change in the reactivity balance is obtained by moving a very high worth rod only 1 or 2 notches and by simulating a failure of the pressure regulator.

Acceptance Criteria - Level 1 - The transient response of any system related variable to any test input must not diverge.

Level 2 - Not applicable.

(ST-22) - Pressure Regulator

Test Objectives - The objectives of this test are to demonstrate the takeover capability of the backup pressure regulator upon failure of the controlling pressure regulator and to demonstrate smooth pressure control transition between the control valves and bypass valves when reactor steam generation exceeds steam flow used by the turbine.

Prerequisites - The required preoperational tests have been completed. Instrumentation has been checked or calibrated as appropriate.

Test Method - The pressure set point will be decreased rapidly and later increased rapidly by about 10 psi and the response of the system will be measured in each case. It is desirable to accomplish the set point change in less than 1 second. At specified test conditions the load limit setpoint 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

level setpoint will be made to demonstrate proper response and operability of the feedwater system at low reactor power.

At Test Conditions 2, 3 and 6, with one feedwater pump in manual and the others in auto, a ±5% change in the manually controlled feed pump will be made. The response of the feedwater system to these steps will be analyzed and compared to the applicable acceptance criteria. The recirculation system will be in manual for these tests. At Test Conditions 1, 2, 3, 4, 5 & 6, with the recirculation system in manual, ±5 inch changes in the water level setpoint will be made to demonstrate proper response and stability of the feedwater system.

At approximately 80% to 90% power, with core flow near 100% of rated, failure of extraction steam valves to one of the feedwater heater trains is accomplished by closing the heater train steam inlet isolation valves which will isolate extraction steam to the last three stages of that train. Recordings of the transient will be analyzed and compared to the predicted response and acceptance criteria.

At Test Condition 6, one feedwater pump will be tripped to demonstrate the capability to avoid a scram and prevent a low reactor water level trip due to the loss of one feedwater pump.

A maximum feedwater runout capability test will be done to demonstrate that the actual capability is compatible with licensing assumptions.

<u>Acceptance Criteria - Level 1</u> - The transient response of any level control system-related variable to any test input must not diverge.

For the feedwater heater loss test, the maximum feedwater temperature decrease due to a single failure case must be less than or equal to 100°F. The resultant MCPR must be greater than the fuel thermal safety limit.

The increase in heat flux cannot exceed the predicted Level 2 value by more than 2%. The predicted value will be based on the actual test values of feedwater temperature change and power level.

The feedwater flow runout capability must not exceed the assumed value in the FSAR.

<u>Level 2</u> - Level control system-related variables may contain oscillatory modes of response. In these cases, the decay ratio for each controlled mode of response must be less than or equal to 0.25.

Closure time for any MSIV shall not be less than 3.0 seconds.

Feedwater control settings must prevent flooding the main steam lines during the full isolation test.

The time delay between the close initiation signal and the extrapolated initial valve movement from 100% open for any MSIV shall be less than or equal to 0.5 seconds.

Level 2 - The positive change in vessel dome pressure occurring within the first 30 seconds after the closure of all MSIVs must not exceed the predicted values. Predicted values will be referenced to actual test conditions of initial power level, scram timing and dome pressure and will use beginning of life nuclear data.

The positive change in heat flux occurring within the first 30 seconds after the closure of all MSIVs must not exceed the predicted values. Predicted values will be referenced to actual test conditions of initial power level, and dome pressure and will use beginning of life nuclear data.

If water level reaches Level 2 setpoint during the MSIV full closure test, RCIC shall automatically initiate and reach rated flow.

During the MSIV full closure test, the relief valves must reclose properly (without any detectable leakage) following the pressure transient.

During full closure of individual MSIVs, peak vessel dome pressure must remain at least 10 psi below the scram setpoint.

During full closure of individual MSIVs, peak neutron flux must remain at least 7.5% below its scram setpoint.

During full closure of individual MSIVs, steam flow in individual lines must remain at least 10% below the high flow isolation trip setpoint.

During full closure of individual MSIVs, the simulated heat flux must remain at least 5% less than its flow biased scram setpoint.

(ST-26) - Relief Valves

Test Objectives - The objectives of this test are to verify that the relief valves function properly, reseat properly after operation and contain no major blockages in the relief valve discharge piping.

Prerequisites - The required preoperational tests have been completed. Instrumentation has been checked or calibrated as

appropriate. Factory test results on SRV flow and operating times have been reviewed.

Test Method - Testing done at low reactor pressure, in conjunction with plant surveillance testing, consists of cycling each relief valve to verify proper operation. The transient monitoring system will be used to record the results of this test. The data collected will compare the operation of individual relief valves against the operation of all relief valves. During relief valve operation, core power - and therefore steam generation rate - is maintained constant. The pressure control system will close the bypass valves an amount proportional to the relief valve steam flow to maintain constant reactor pressure. This bypass valve motion will be monitored and a comparison of the response for each relief valve operation will be made. If differences exist, it could suggest a partial obstruction of the relief valve or its tailpipe. Tailpipe temperature will be recorded to verify the relief valve has properly reseated. Reactor variables will also be recorded to verify system stability during opening and closing each relief valve.

Testing done at rated reactor pressure consists of manually operating each relief valve at rated reactor pressure. The decrease in Main Generator output will be monitored during the operation of each relief valve to provide an indication of relief valve flow. By comparison of the generator output response for each relief valve operation, any flow obstruction in the valve or its tailpipe can be identified. Each valve will be opened for approximately 10 seconds to allow for variables to stabilize. Reactor variables will also be recorded to verify system stability during opening and closing each relief valve.

Acceptance Criteria - Level 1

There should be a positive indication of steam discharge during the manual actuation of each valve.

Level 2 - Pressure control system-related variables may contain oscillatory modes of response. In these cases, the decay ratio for each controlled mode of response must be less than or equal to 0.25.

The temperature measured by thermocouples on the discharge side of the valves shall return to within 10°F of the temperature recorded before the valve was opened.

During the low pressure functional tests, the change in bypass valve position for each SRV opening shall be greater than or equal to a value corresponding to the average change minus 10% of one bypass valve.

d) The positive change in simulated heat flux shall not exceed the Level 2 criteria by more than 2% of rated value.

The two pump drive flow coastdown transient, during the first three seconds of an RPT trip, must fall within the specified limits.

Level 2

a) There shall be no MSIV closure in the first 3 minutes of the transient and operator action shall not be required in that period to avoid the MSIV trip.

b) The positive change in vessel dome pressure and in simulated heat flux which occur within the first 30 seconds after the initiation of either generator or turbine trip must not exceed the predicted values.

(Predicted values will be referenced to actual test conditions of initial power level, dome pressure, scram timing, and the time from the start of stop/control valve motion to start of control rod motion, and will use beginning of life nuclear data.)

c) For the Generator trip within the bypass valves capacity (initial thermal power values less than or equal to 25 percent of rated) the reactor shall not scram.

The Total Delay from the initiation of a Turbine Stop Valve Closure or Turbine Control Valve Fast Closure to complete suppression of the Electric Arc between the fully open contacts of the Recirculation Pump Trip (RPT) Breaker shall be less than 175 milliseconds.

Recirculation pump trip, HPCI and RCIC starts shall not be initiated from a low reactor water level.

Feedwater level control shall avoid the loss of feedwater flow due to a high level (L8) trip.

(ST-28) - Shutdown from Outside the Main Control Room

Test Objective - The objective of this test is to demonstrate that the reactor can be shutdown, maintained in a hot shutdown condition, and cooled down from outside the main control room. Also, the adequacy of the Emergency Operating Procedures will be verified.

Prerequisites - The required preoperational tests have been completed. Instrumentation has been checked or calibrated as appropriate.

Test Method - While operating at approximately 20% power synchronized to the grid with normal electrical system alignment, the reactor will be scrammed and the MSIV's will be closed from inside the main control room. The control room will then be evacuated, and reactor level and pressure will be controlled from outside the main control room. The Shutdown Cooling mode of RHR will be placed into service with cooling water supplied from the ultimate heat sink. During this demonstration, some supervisory and operating personnel will remain in the control room to protect non-safety-related equipment from unnecessary damage if conditions arise and to assume control of the plant if conditions warrant. A test will be run to demonstrate that the reactor can be scrammed and isolated from outside the control room.

Acceptance Criteria - Level 1 - Not applicable.

Level 2 - During a simulated control room evacuation, the reactor must be brought to the point where cooldown is initiated and under control, and the reactor vessel pressure and water level are controlled using equipment and controls outside the control room. The test is deemed successful when reactor pressure is less than 98 psig (permissive setpoint) and the RHR shutdown cooling mode has been put in operation.

The reactor must be capable of being scrammed and isolated from outside the control room.

(ST-29) Recirculation Flow Control System

The objectives of this test are:

a) To demonstrate the flow control capability of the plant over the entire pump speed range, including individual local manual and combined Master Manual Operation.

b) To determine that all electrical compensators and controllers are set for desired system performance and stability.

Prerequisites - The required preoperational tests have been completed.

All instrumentation has been calibrated.

Test Method - At Test Conditions 2, 3, 5 and 6, the stability of the recirculation flow control system is demonstrated by performing step changes in recirculation pump speed. This

testing is done in individual local manual at Test Conditions 2 and 5 and in combined Master Manual operation at Test Conditions 3 and 6 to demonstrate operability and stability.

Acceptance Criteria - Level 1 - The transient response of any system-related variable to any test input must not diverge.

Level 2 - A scram shall not occur due to recirculation flow control maneuvers.

The APRM neutron flux trip avoidance margin shall be greater than or equal to 7.5% and the simulated heat flux trip avoidance margin shall be greater than or equal to 5% when the power maneuver effects are extrapolated to those that would occur along the 100% rated rod line.

The decay ratio of any oscillatory controlled variable must be less than or equal to 0.25.

Steady state limit cycles (if any) shall not produce turbine steam flow variations greater than ±.5% of rated steam flow.

(ST-30) Recirculation System

Test Objectives - The objectives of this test are:

a. Obtain recirculation system performance data during pump trip, flow coastdown, and pump restart.

b. Verify that the feedwater control system can satisfactorily control water level without a resulting turbine trip and associated scram.

c. Record and verify acceptable performance of the recirculation two pump circuit trip sytem.

d. Verify the adequacy of the recirculation runback to mitigate a scram.

e. Verify that no recirculation system cavitation will occur in the operable region of the power-flow map.

Prerequisites - The required preoperational tests have been completed. Instrumentation has been checked or calibrated as appropriate.

Test Method - Single recirculation pump trips will be made at Test Condition (TC) 3 and TC-6. These trips will be initiated by tripping the M-G Set Drive Motor Breaker from the control room. Reactor parameters will be recorded during the transient and analyzed to verify non-divergence of oscillatory responses, adequate margins to RPS scram set points, and capability of the

feedwater system to prevent a high level trip. The capability to restart the recirc. pump at a high power level will also be demonstrated. At TC-3, both recirculation pumps RPT breakers will be simultaneously tripped using a temporarily installed test switch. The data gathered will be used to demonstrate acceptable pump coastdown performance prior to high power turbine trips and generator load rejects.

Appropriate conditions will be simulated at TC-3 to demonstrate the proper operation of the recirculation pump runback circuits. This is done prior to an actual planned feed pump trip at rated power.

Both the jet pumps and the recirculation pumps will cavitate at conditions of high flow and low power where NPSH demands are high and little feedwater subcooling occurs. However, the recirculation flow will automatically runback upon sensing a decrease in feedwater flow. The maximum recirculation flow is limited by appropriate stops which will run back the recirculation flow from the possible cavitation region. At TC-3, it will be verified that these limits are sufficient to prevent operation where recirculation pump or jet pump cavitation occurs.

Acceptance Criteria - Level 1 - The response of any level related variables during a single pump trip must not diverge.

The two pump drive flow coastdown transient, during the first 3 seconds of an RPT trip, must fall within the specified bounds.

Level 2 - The reactor shall not scram during the one pump trip.

The APRM margin to avoid a scram shall be at least 7.5% during the one pump trip recovery.

The reactor water level margin to avoid a high level trip shall be at least 3.0 inches during the one pump trip.

Peak simulated heat flux must remain at least 5% below its flow biased scram value.

Runback logic shall have settings adequate to prevent recirculation pump operation in areas of potential cavitation.

The recirculation pumps shall runback upon a trip of the runback circuit.

(ST-31) Loss of Turbine-Generator and Offsite Power

Test Objectives - The objectives of this test are to demonstrate that the required safety systems will initiate and function properly without manual assistance, the electrical distribution and diesel generator systems will function properly, and the HPCI

and/or RCIC systems will maintain water level if necessary during a simultaneous loss of the main turbine-generator and offsite power.

Prerequisites - The required preoperational tests have been completed. Instrumentation has been checked or calibrated as appropriate.

Test Method - With the unit synchronized to the grid at approximately 30% power, the main turbine-generator will be manually tripped coincident with a manual trip of the unit's offsite power source breaker, both trips initiated from the control room. To ensure a full simulation of the loss of all offsite power to Unit 1 during Unit 1 testing, all Unit 1 and Common loads will be transferred to Unit 1 Auxiliary and Startup Busses and appropriate breakers racked out to prevent automatic transfer of the loads to Unit 2 sources.

Reactor water level and the operation of safety systems will be monitored to verify that the acceptance criteria are satisfied. The proper response of the electrical distribution system will be checked.

The loss of offsite power condition will be maintained for at least 30 minutes to demonstrate that necessary equipment, controls, and indication are available following station blackout to remove decay heat from the core using only emergency power supplies and distribution system.

Acceptance Criteria - Level 1 - All safety systems, such as the Reactor Protection System, the diesel-generator, RCIC and HPCI must function properly without manual assistance, and HPCI and/or RCIC system action, if necessary, shall keep the reactor water level above the initiation level of Core Spray, LPCI and ADS.

Level 2 - The temperature measured by the thermocouples on the discharge side of any SRV that actuated shall return to within 10°F of the temperature recorded before the valve opened. Permanent instrumentation for reactor power, reactor pressure, water level, control rod position, suppression pool temperature, high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) shall be demonstrated operable following re-energization of the 4kV busses by the diesel generators.

(ST-32) - Containment Atmosphere and Main Steam Tunnel Cooling

Test Objective - The objective of this test is to verify the ability of the drywell coolers/recirculation fans and the reactor building portion of the main steam tunnel coolers to maintain design conditions in the drywell and reactor building portion of the mainsteam tunnel, respectively, during operating conditions and post scram conditions. This test also demonstrates that

containment main steamline penetrations do not overheat adjacent concrete.

Prerequisites - The required preoperational tests have been completed. Instrumentation has been checked or calibrated as appropriate.

Test Method - During heatup, at test conditions 2 and 6, and following a planned scram from 100% power, data will be taken to ascertain that the containment atmospheric conditions are within design limits.

Acceptance Criteria - Level 1 - not applicable

Level 2 - The general drywell area is maintained at an average temperature less than or equal to 135°F, with maximum local temperature not to exceed 150°F.

The area beneath the reactor pressure vessel is maintained at an average temperature less than or equal to 135°F, maximum local temperature not to exceed 165°F, with minimum local temperature above 100°F.

The area around the recirculation pump motors is maintained at an average temperature less than or equal to 128°F, with maximum local temperature not to exceed 135°F.

The inside base of the shield wall in the RPV skirt area is maintained at temperatures greater than 100°F.

The reactor building portion of the mainsteam pipeway is maintained at or below 120°F.

The concrete temperature surrounding the main steamlime penetrations is maintained at less than 200°F.

(ST-33) Piping Steady State Vibration

Test Objectives - The objectives of this test is to demonstrate that steady state vibration levels on reactor recirculation, main steam inside containment, and those piping systems identified in Table 3.9-33 are within acceptable limits. (Note that this test now includes piping previously contained in ST-40. Also note that dynamic transient vibration testing previously contained in this test have been merged into ST-39.)

Prepequisites - Instrumentation has been installed and calibrated.

Test Method - Devices for measuring continuous vibration are mounted on main steam lines, recirculation lines and lines of

systems identified in Table 3.9-33 as applicable, and vibration during steady state operation is compared with calculated values.

Acceptance Criteria - Level 1 - The measured amplitude (peak to peak) of each remotely monitored point on the main steam inside containment and reactor recirculation lines shall not exceed the allowable value for that point.

Level 2 - The measured amplitude (peak to peak) of each remotely monitored point on the main steam inside containment and reactor recirculation lines shall not exceed the expected value for that point.

The vibratory response of non-remotely monitored systems or portions of systems indentified in Table 3.9-33 shall be judged to be within acceptable limits by a qualified test engineer.

The maximum measured amplitude of the piping response for each remotely monitored point on systems identified in Table 3.9-33 shall not exceed the acceptable value for that point.

(ST-34) Control Rod Sequence Exchange

(This test number was previously assigned to the RPV Internals Vibration test which is now performed during the Preoperational Test Program. The test description for the RPV Internals Vibration test is now in TP2.16 which follows the abstract for P64.1.)

Test Objective - The objective of this test is to perform a representative sequence exchange of control rod patterns at the power level at which such exchanges will be done during plant operation and demonstrate that core limits and PCIOMR threshold limits will not be exceeded.

Prerequisites - Instrumentation has been checked or calibrated as appropriate.

Test Method - The control rod sequence exchange begins on the design flow control line with core flow near minimum. Control rods will be inserted as necessary to increase the margin to local core thermal limits. Core power is maintained above the low power setpoint of the Rod Worth Minimizer and Rod Sequence Control System and below the power which will keep fuel assembly nodal power at the PCIOMR threshhold. The exchange is performed in accordance with the plant operating procedure RE-TP-009. Data taken during the exchange will be reviewed to verify that the Acceptance Criteria were satisfied.

Acceptance Criteria - Level 1 - Completion of the exchange of one rod pattern for the complimentary pattern with continual

Acceptance Criteria - Components perform in accordance with applicable design documents.

14.2.12.6 Unit 2 Startup Test Program Procedure Abstracts

All those tests comprising the Unit 2 Startup Test Program (Table 14.2-3) are discussed in this section. For each test a description is provided for test purpose, test prerequisites, test description and statement of test acceptance criteria, where applicable. Additions, deletions, and changes to these discussions are expected to occur as the test program progresses. Such modification to these discussions will be reflected in amendments to the FSAR.

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

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

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

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

(ST-1) Chemical and Radiochemical

Test Objectives - The principal objective of this test is 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 documentation of radwaste liquid discharge, evaluation of the Condensate Polishing system, and evaluation of the Reactor Water Cleanup system. Data for these purposes is secured from a variety of sources: plant

operating records, regular routine coolant analysis, radiochemical measurements of specific nuclides, and special chemical tests.

Prerequisites - The required preoperational tests have been completed. Instrumentation has been checked or calibrated as appropriate.

Test Method - Prior to fuel loading, chemical samples are taken to ensure that reactor coolant and Fuel Pool Cooling and Cleanup System sample stations are functioning properly and to determine initial concentrations. Additionally, subsequent to fuel loading, during reactor heatup, and at each major power level change, samples are taken to determine the chemical and radiochemical quality of reactor water and reactor feedwater.

Acceptance Criteria - Level 1 - Chemical factors defined in the Technical Specifications and Fuel Warranty must be maintained within the limits specified. The activity of liquid effluents must conform to license limitations. Water quality must be known at all times and should remain within the guidelines of the Water Quality Specifications.

Level 2 - Not applicable.

(ST-2) - Radiation Measurements

Test Objectives - The objectives 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.

Prerequisites - The required preoperational tests have been completed.

Test Method - A survey of natural background radiation at selected locations throughout the plant will be made prior to fuel loading. Subsequent to fuel loading, during reactor heatup and at power levels of approximately 25%, 60% and 100% of rated power, gamma radiation level measurements and, where appropriate, thermal and fast neutron measurements will be made at selected locations throughout the plant.

Acceptance 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 10CFR20.

Level 2 - The radiation doses of plant origin shall meet the following limits depending upon which Radiations Zone the radiation base survey point is located:

Radiation Zone	Limit
I	0.5 mRem/hr.
II	2.5 mRem/hr.
III	15 mRem/hr.
IV	100 mRem/hr.

Note: All areas designated Radiation Zone V have potential radiation doses of 100 mRem/hr. Readings taken in Zone V during the Startup Test Program may be less than 100 mRem/hr; however, since Zone V is defined in terms of potential levels, there are no Acceptance Criteria for Zone V base survey points.

(ST-3) - Fuel Loading

Test Objective - The objective of this test is to achieve the full and proper core complement of nuclear fuel assemblies through a safe and efficient fuel loading evolution.

Prerequisites - The required Preoperational Tests have been completed. In addition, prior to starting this test procedure, the following prerequisites will be met:

a. Fuel and Control Rod inspections will be complete.

b. Control Rods will be installed and tested.

c. Reactor vessel water level will be established and minimum level prescribed.

d. The standby liquid control system will be operable and in readiness.

e. Fuel handling equipment will have been checked and dry runs completed.

f. The status of protection systems, interlocks, mode switches, alarms, and radiation protection equipment will be prescribed and verified.

g. Water quality must meet required specifications.

The following prerequisites will be met prior to commencing actual fuel loading to assure that this operation is performed in a safe manner:

a. The status of all systems required for fuel loading will be specified and will be in the status required.

b. At least two movable neutron detectors will be calibrated and operable. At least two neutron detectors will be connected to the high flux scram trips. They will be

located so as to provide acceptable signals during fuel loading.

c. Source range monitoring Nuclear instruments will be checked with a neutron source prior to fuel loading or resumption of fuel loading if sufficient delays are incurred.

d. The status of secondary containment will be specified and established.

e. Reactor vessel status will be specified relative to internal component placement and this placement established to make the vessel ready to receive fuel.

f. The high flux trip points will be set for a relatively low power level.

g. Neutron sources will be installed near the center of the core and at other specified locations.

Test Method - Before the first fuel assembly is taken from the fuel pool and inserted into the reactor, core components (fuel support castings, blade guides, control rod drives, etc.) will be installed, tested and/or verified. This procedure begins with the steps required to load neutron sources, includes the activities necessary to monitor neutron population using specially constructed fuel loading chambers (FLCs), and culminates with the insertion of fuel assemblies into the reactor core. Fuel loading continues until the core is fully loaded, verified and ready to perform subsequent Startup Tests.

Control rod functional tests, subcriticality checks, and a shutdown margin demonstration will be performed during the loading.

Acceptance Criteria - Level 1 - The partially loaded core must be subcritical by at least 0.38% delta k/k with the analytically determined, highest worth rod fully withdrawn.

(ST-4) - Full Core Shutdown Margin

Test Objective - 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.

Prerequisites - The following prerequisites will be complete prior to performing the full core shutdown margin test:

a) The predicted critical rod position is available

b) The Standby Liquid Control System is available

c) Nuclear instrumentation is available with neutron count rate of at least three counts per second and signal to noise ratio greater than two to one

d) High-flux scram trips are set conservatively low

e) Instrumentation has been checked or calibrated as appropriate

Test Method - This test will be performed in the fully loaded core in the xenon-free condition. The shutdown margin test will be performed by withdrawing the control rods from the all-rods-in configuration until criticality is reached. If the highest worth rod will not be withdrawn in sequence, other rods may be withdrawn providing that the reactivity worth is equivalent. The difference between the measured Keff and the calculated Keff for the in-sequence critical will be applied to the calculated value to obtain the true shutdown margin.

Acceptance Criteria - Level 1 - The shutdown margin of the fully loaded, cold (68°F), xenon-free core occuring at the most reactive time during the cycle must be at least 0.38% delta k/k with the analytically strongest rod (or its reactivity equivalent) withdrawn. If the shutdown margin is measured at some time during the cycle other than the most reactive time, compliance with the above criterion is shown by demonstrating that the shutdown margin is 0.38% delta k/k plus an exposure dependent correction factor which corrects the shutdown margin at that time to the minimum shutdown margin.

Level 2 - Criticality should occur within ±1.0% delta k/k of the predicted critical.

(ST-5) Control Rod Drive System

Test Objective - The objectives 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.

Prerequisites - The required preoperational tests have been completed.

Test Method - The CRD tests performed during 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 in Table 14.2-5.

Acceptance Criteria - Level 1 - Each CRD must have a normal withdraw time greater than or equal to 40 seconds.

The mean scram time of all operable CRDs must not exceed the values specified in the plant technical specifications. (Scram time is measured from the time the pilot scram valve solenoids are deenergized.)

The mean scram time of the three fastest CRDs in a two by two array must not exceed the values specified in the plant technical specifications. (Scram time is measured from the time the pilot scram solenoids are deenergized)

Level 2 - Each CRD must have a normal insert speed of 3.0 ± 0.6 inches per second, 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 for a continuous drive in, a settling test must be performed, in which case, the differential settling pressure should not be less than 30 psid nor should it vary by more than 10 psid over a full stroke.

(ST-6) - SRM Performance and Control Rod Sequence

The testing previously contained in this test has been merged into ST-10.

(ST-7) Reactor Water Cleanup System

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

Prerequisites - The required preoperational tests have been completed. Instrumentation has been checked or calibrated as appropriate.

Test Method - With the reactor at rated temperature and pressure, process variables will be recorded during steady state operation in three modes as defined by the System Process Diagram: Blowdown, Hot Standby, and Normal. Additional system configurations will also be aligned to verify proper performance of the bottom head flow and temperature indicators.

Acceptance Criteria - Level 1 - Not applicable.

Level 2 - The temperature at the tube side outlet of the non-regenerative heat exchangers (NRHX) shall not exceed 130°F in the blowdown mode and 120°F in the normal mode.

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

The cooling water flow to the NRHX's shall be limited to 6% above the flow corresponding to the heat exchanger capacity (as determined from the process diagram) and the existing temperature differential across the heat exchangers. The cooling water outlet temperature shall not exceed 180°F.

During two pump operations at rated core flow, the bottom head temperature as measured by the bottom drain line thermocouple should be within 30°F of the recirculation loop temperatures.

Bottom head flow indicator FI-2R610 shall indicate within 25 gpm of RWCU flow indicator FI-2R609 when total system flow is thru the bottom head drain.

(ST-8) - Residual Heat Removal System

Test Objectives - The objectives of this test are to demonstrate the ability of the Residual Heat Removal (RHR) System to: 1) remove heat from the reactor pressure vessel and the suppression pool and 2) operate in the suppression pool cooling mode, steam condensing mode and shutdown cooling mode.

Prerequisites - The required preoperational tests have been completed. Instrumentation has been checked or calibrated as appropriate.

Test Method - The suppression pool cooling mode and steam condensing mode will be used to measure the RHR heat exchanger capacity. Data will be obtained to determine the heat transfer rate with rated flow on both sides of the heat exchanger. For the suppression pool cooling mode test, attempts will be made to establish a large temperature differential between the service and suppression pool water by extended RCIC or relief valve operations. Heat exchanger capacity in the steam condensing mode will be measured with the reactor in power operation, supplying a steam source to the RHR heat exchangers. Due to the insufficient decay heat load during the startup test period, full heat exchanger heat capacity in the shutdown cooling mode cannot be measured without the risk of exceeding the 100°F/hr cooldown rate limit of the reactor pressure vessel. Shutdown cooling mode operability will be demonstrated after scheduled trips and cooldowns during the Startup Test Program.

Steam condensing mode control system stability will be demonstrated with the reactor in power operation, supplying a steam source to the RHR heat exchangers.

Acceptance Criteria - Level 1 - The transient response of any system-related variable to any test input must not diverge.

Level 2 - The RHR system shall be capable of operating in the steam condensing, suppression pool cooling and shutdown cooling modes at the heat exchanger capacities indicated on the process diagrams. Both simultaneous operation of RHR loops and single loop operation shall be tested in the steam condensing and shutdown cooling modes. Each RHR loop shall be tested independently in the suppression pool cooling mode. System-related variables may contain oscillatory modes of response. In these cases, the decay ratio for each controlled mode of response must be less than or equal to 0.25.

(ST-9) Water Level Measurement

Test Objectives - The objective of this test is to determine actual reference leg temperature and recalibrate instruments if necessary.

Prerequisites - The required preoperational tests have been completed. All system instrumentation is installed and calibrated.

Test Method - At rated temperature and pressure under steady state conditions, the reference leg temperature will be measured and compared to the value assumed during initial calibration. If the difference of the two temperatures exceed the Acceptance Criteria, then the instruments will be recalibrated using the measured value.

Acceptance Criteria - Level 1 - Not applicable.

Level 2 - The difference between the actual reference leg temperature(s) and the value(s) assumed during calibration shall be less than that amount which will result in a scale end point error of 1% of the instrument span for each range.

(ST-10) SRM and IRM Performance and Control Rod Sequence

Test Objectives - The objectives of this test are: (a) to demonstrate that the operational sources, SRM and IRM instrumentation and rod withdrawal sequences provide adequate information to achieve criticality and increase power in a safe and efficient manner for each of the specified rod withdrawal sequences and (b) to adjust the Intermediate Range Monitor System as necessary to obtain the desired overlap with the SRM and APRM

systems. (Note that this test now includes testing previously contained in ST-6).

Prerequisites - The required preoperational tests have been completed.

Test Method - Source range monitor count-rate data will be taken and compared with stated criteria.

A withdrawal sequence has been calculated which completely specifies control rod withdrawals from the all-rods-in condition to the rated power configuration. Each sequence will be used to attain cold criticality.

Movement of rods in a prescribed sequence is monitored by the Rod Worth Minimizer and rod sequence control system, which will prevent out of sequence withdrawal.

Initially the IRM system is set during the Preoperational Test Program. SRM-IRM and IRM-APRM overlap is verified the first time sufficient neutron flux conditions arise. After the APRM calibration, the IRM gains will be adjusted as necessary to optimize the IRM overlap with the SRMs and APRMs.

Acceptance Criteria - Level 1 - There must be a neutron signal count-to-noise count ratio of at least 2 to 1 on the required operable SRMs. There must be a minimum count rate of 3 counts/second on the required operable SRMs.

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

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

(ST-11) LPRM Calibration

Test Objectives - The objective of this test is to calibrate the Local Power Range Monitoring System.

Prerequisites - The required preoperational tests have been completed. Instrumentation for calibration has been checked.

Test Method - The LPRM channels will be calibrated to make the LPRM readings proportional to the neutron flux in the 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.

Acceptance Criteria - Level 1 - Not applicable.

Level 2 - Each LPRM will be within 10% of its calculated value.

(ST-12) - APRM Calibration

Test Objective - The objective of this test is to calibrate the Average Power Range Monitoring (APRM) system.

Prerequisites - The required preoperational tests have been completed. Instrumentation for calibration has been checked.

Test Method - A heat balance will be made after initially achieving power level associated with each test plateau. Each APRM channel reading will be adjusted to be consistent with the core thermal power as determined from the heat balance. During heatup a preliminary calibration will be made by adjusting the APRM amplifier gains so that the APRM readings agree with the results of a constant heatup rate heat balance. The APRMs should be recalibrated in the power range by a heat balance as soon as adequate feedwater indication is available.

Acceptance Criteria - Level 1 - The APRM channels must be calibrated to read equal to or greater than the actual core thermal power.

Level 2 - Not applicable.

(ST-13) - NSSS Process Computer

Test Objective - The objective of this test is to verify the NSSS performance of the process computer under plant operating conditions.

Prerequisites - The required preoperational tests have been completed.

Test Method - The Dynamic System Test Case will be run to verify that the results of NSSS performance calculations are correct.

Acceptance Criteria - Level 1 - Not applicable.

Level 2 -

(1) The MCPR 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 2% or,

b. For the case in which the MCPR calculated by the process computer is in a different assembly than that calculated by the independent method, for both assemblies, the MCPR and CPR calculated by

the two methods shall agree within 2% for the same assembly.

(2) The maximum LHGR calculated by the independent method and the process computer either:

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

b. For the case in which the maximum LHGR calculated by the process computer is in a different assembly than that calculated by the independent method, for both assemblies, the maximum LHGR and LHGR calculated by the two methods shall agree within 2% for the same assembly.

(3) The MAPLHGR calculated by the independent method and the process computer either:

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

b. For the case in which the MAPLHGR calculated by the process computer is in a different assembly than that calculated by the independent method for both assemblies, the MAPLHGR and APLHGR calculated by the two methods shall agree within 2% for the same assembly.

(4) The LPRM calibration factors calculated by the independent method and the process computer agree to within 2%.

<u>(ST-14) RCIC System</u>

<u>Test Objective</u> - The objectives of this test are to verify the proper operation of the Reactor Core Isolation Cooling (RCIC) system at the minimum and rated operating pressures and flow ranges, and to demonstrate reliability in automatic mode starting from cold standby when the reactor is at power conditions.

<u>Prerequisites</u> - The required preoperational tests have been completed. Initial turbine operation (uncoupled) must have been performed to verify satisfactory operation and over-speed trip. Instrumentation has been installed and calibrated.

<u>Test Method</u> - The RCIC System is designed to be tested in two ways: (1) by flow injection into a test line leading to the Condensate Storage Tank (CST), and (2) by flow injection directly into the reactor vessel.

The earlier set of CST injection tests consist of manual and automatic mode starts at approximately 150 psig and near rated reactor pressure conditions. The pump discharge pressure during these tests is throttled to be approximately 100 psi above the reactor pressure to simulate the largest expected pipeline pressure drop. This CST testing is done to demonstrate general system operability and stability.

Reactor vessel injection tests are also done which consist of manual and automatic mode starts near rated reactor pressure and automatic mode start at approximately 150 psig reactor pressure conditions to demonstrate operability and stability.

After all final controller and system adjustments have been determined, a defined set of demonstration tests must be performed with that one set of adjustments. Two consecutive reactor vessel injections starting from cold conditions in the automatic mode must satisfactorily be performed to demonstrate system reliability. ("Cold" is defined as a minimum three days without any kind of RCIC operation.)

After the manual start portion of certain of the above tests is completed, and while the system is still operating, small step disturbances in speed and flow command are input (in manual and automatic mode respectively) in order to demonstrate satisfactory stability. This is to be done at both low (above minimum turbine speed) and near rated flow initial conditions to span the RCIC operating range. During testing at 150 psig, this is done only near rated flow initial conditions.

A demonstration of extended operation of up to 2 hours (or until pump and turbine oil temperature is stabilized) of continuous running at rated flow conditions is to be scheduled at a convenient time during the Startup Test Program.

<u>Acceptance Criteria - Level 1</u> - The average pump discharge flow must be equal to or greater than the 100% rated value after 30 seconds have elapsed from automatic initiation at any reactor pressure between 150 psig (+15, -0) (10.5 kg/cm^2) and rated.

The RCIC turbine shall not trip or isolate during auto or manual start tests.

Note: If any Level 1 criteria are not met, the reactor will only be allowed to operate up to a restricted power level defined by Figure 14.2-7 until the problem is resolved. Also consult the plant Technical Specifications for actions to be taken.

<u>Level 2</u> - In order to provide an overspeed and isolation trip avoidance margin, the transient start first and subsequent speed peaks shall not exceed 5% above the rated RCIC turbine speed.

The speed and flow control loops shall be adjusted so that the decay ratio of any RCIC system related variable is not greater than 0.25.

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 calibrated to a differential pressure corresponding to less than or equal to 300% of the maximum required steady state flow, with the reactor assumed to be near the pressure for main relief valve actuation.

(ST-15) HPCI System

Test Objective - The objective of this test is to verify the proper operation of the High Pressure Coolant Injection (HPCI) system at the minimum and rated operating pressures and flow ranges, and to demonstrate reliability in automatic mode starting from cold standby when the reactor is at rated pressure conditions.

Prerequisites - The required preoperational tests have been completed. Initial turbine operation (uncoupled) must have been performed to verify satisfactory operation and over-speed trip. Instrumentation has been installed and calibrated.

Test Method - The HPCI system is designed to be tested in two ways: (1) by flow injection into a test line leading to the Condensate Storage Tank (CST), and (2) by flow injection directly into the reactor vessel.

The earlier set of CST injection tests consist of manual and automatic mode starts at approximately 150 psig and near rated reactor pressure conditions. The pump discharge pressure during these tests is throttled to be approximately 100 psi above the reactor pressure to simulate the largest expected pipeline pressure drop. This CST testing is done to demonstrate general system operability and stability.

Reactor vessel injection tests are also done which consist of manual and automatic mode start near rated reactor pressure to demonstrate operability and stability.

After all final controller and system adjustments have been determined, a defined set of demonstration tests must be performed with that one set of adjustments. Two consecutive reactor vessel injections starting from cold condidtions in the automatic mode must satisfactorily be performed to demonstrate system reliability. ("Cold" is defined to a minimum three days without any kind of HPCI operation.)

After the manual start portion of certain of the above tests is completed, and while the system is still operating, small step disturbances in speed and flow command are input (in manual and automatic mode respectively) in order to demonstrate satisfactory stability. This is to be done at both low (above minimum turbine speed) and near rated flow initial conditions to span the HPCI operating range. During testing at 150 psig this is done only near rated flow initial conditions.

A continuous running test is to be scheduled at a convenient time during the Startup Test Program. This demonstration of extended operation should be for up to 2 hours or until steady turbine and pump conditions are reached or until limits on plant operation are encountered.

Pump flow testing will also be verified since auxiliary boiler supply is insufficient to fully test the sytem during the Preoperational Test Program.

Acceptance Criteria - Level 1 - The average pump discharge flow must be equal to or greater than the 100% rated value after 25 seconds have elapsed from automatic initiation at any reactor pressure between 150 psig (+15, -0) (10.5 kg/cm²) and rated.

The HPCI turbine shall not trip or isolate during auto or manual start tests.

Level 2 - In order to provide an overspeed and isolation trip avoidance margin, the transient start first peak shall not come closer than 15% (of rated speed) to the overspeed trip, and subsequent speed peaks shall not be greater than 5% above rated turbine speed.

The speed and flow control loops shall be adjusted so that the decay ratio of any HPCI system related variable is not greater than 0.25.

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 calibrated to actuate at no greater than 300% of the maximum required steady state flow, with the reactor assumed to be near the pressure for main relief valve actuation.

(ST-16) Selected Process Temperatures

Test Objectives - The objectives of this procedure are a) to establish the proper setting of the low speed limiter for the recirculation pumps to avoid coolant temperature stratification in the reactor pressure vessel bottom head region, b) to identify any reactor operating modes that cause temperature

stratification, and c) to familiarize the plant personnel with the temperature differential limitations of the reactor system.

Prerequisites - The required preoperational tests have been completed. System instrumentation has been calibrated.

Test Method - During initial heatup while at hot standby conditions, the bottom drain line temperature, recirculation loop suction temperature and applicable reactor parameters are monitored as the recirculation flow is slowly lowered to minimum stable flow. Utilizing this data it can be determined whether coolant temperature stratification occurs when the recirculation pumps are on and if so, what minimum recirculation flow will prevent it.

Monitoring the preceeding information during planned pump trips will determine if temperature stratification occurs in the idle recirculation loops or in the lower plenum when one or more loops are inactive.

Acceptance Criteria - Level 1 - The reactor recirculation pumps shall not be started nor flow increased unless the coolant temperatures between the steam dome and bottom head drain are within 145°F.

The recirculation pump in an idle loop must not be started unless the loop suction temperature is within 50°F of the active loop.

The recirculation pump in an idle loop must not be started unless the operating loop flow rate is less than or equal to 50% of rated loop flow.

When both loops have been idle, an idle recirculation loop shall not be started unless the temperature differential between the reactor coolant within the idle loop to be started up and the coolant within the reactor pressure vessel is less than or equal to 50°F.

Level 2 - Not Applicable.

(ST-17) - System Expansion

Test Objectives - The purposes of this test are to demonstrate that reactor recirculation, main steam inside containment, and those piping systems identified in Table 3.9-33 respond to thermal expansion consistent with stress analysis results. (Note that this test now includes piping previously contained in ST-38.)

Prerequisities - Instrumentation has been installed and calibrated.

Test Method - Hanger positions and locations of piping in the Nuclear Steam Supply System and piping systems identified in Table 3.9-33 inside and outside the reactor drywell are recorded prior to initial heatup and after a planned cold shutdown. During initial heatup, visual inspections are made at intermediate reactor water temperatures and at rated temperature to assure components are free to move as designed. Adjustments are made as necessary. Devices for measuring continuous pipe deflections are mounted on main steam, recirculation and other selected lines. Motion during heatup is compared with calculated values.

Acceptance Criteria - Level 1 - There shall be no obstructions which will interfere with the thermal expansion of the main steam and recirculation piping systems. Piping systems identified in Table 3.9-33 will not be restrained against thermal expansion except by design intent.

Hangers on piping systems identified in Table 3.9-33 shall not be bottomed out or have the spring fully stretched. Snubbers on piping systems identified in Table 3.9-33 shall not become extended or compressed to the limits of their total travel.

The measured displacements at the established transducer locations on the main steam and recirculation systems shall not exceed the allowable values calculated for the specific points.

Level 2 - The measured displacements at the established transducer locations on the main steam and recirculation systems shall not exceed the expected values calculated for the specific points. The measured displacements at the established transducer locations on the piping systems identified in Table 3.9-33 shall be within the acceptable range calculated for the specific points.

Hangers on piping systems identified in Table 3.9-33 shall be in their operating range.

(ST-18) TIP Uncertainty

Test Objectives - The objective of this test is to determine the uncertainty of the TIP system readings.

Prerequisites - System installation is completed and required preoperational tests are completed and verified. Instrumentation has been calibrated and installed.

Test Method - The TIP uncertainty consists of a random noise component and a geometric component, the geometric component being due to variation in the water gap geometry and TIP tube orientation from TIP location to location. Measurement of these components is obtained by taking repetitive TIP readings at a

single TIP location, and by analyzing pairs of TIP readings taken at TIP locations which are symmetrical about the core diagonal of fuel loading and control rod symmetry.

The random noise uncertainty is determined from successive TIP runs made at the common location (32-33) with each of the TIP machines making six runs at index position 10. The TIP data will be obtained by simultaneous operation of the Process computer OD-2 program which provides 24 nodal TIP values for each TIP traverse. The standard deviation of the random noise is derived by taking the square root of the average of the variances at nodal levels 5 through 22, where the nodal variance is obtained from the fractional deviations of the successive TIP values about their nodal mean value.

The total TIP uncertainty is determined by performing a complete set of TIP traverses as required by Process Computer program OD-1. The total TIP uncertainty is obtained by dividing the standard deviation of the symmetric TIP pair nodal ratios by the square root of 2. The nodal TIP ratio is defined as the nodal BASE value of the TIP in the lower right half of the core divided by its symmetric counterpart in the upper left half.

The geometric component of TIP uncertainty is obtained by statistically subtracting the random noise component from the total TIP uncertainty.

The TIP data will be taken with the reactor operating with an octant symmetric rod pattern and at steady state conditions. One set of TIP data will be taken at approximately 50% power and at least one other set at 75% power or above. The acceptance criteria for this subtest uses the "average uncertainties" for all data sets. Therefore additional performance of the subtest may be scheduled and the previous values of uncertainty will be used in the averaging to determine the acceptability of the results.

<u>Acceptance Criteria - Level 1</u> - Not applicable.

<u>Level 2</u> - The total TIP uncertainty (including random noise and geometrical uncertainties) obtained by averaging the uncertainties for all data sets must be less than 6.0%.

NOTE: A minimum of two and up to six data sets may be used to meet the above criteria.

<u>(ST-19) Core Performance</u>

<u>Test Objectives</u> - The objectives 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 linear heat generation rate (MAPLHGR).

Prerequisites - The required preoperational tests have been completed.

Test Method - The core performance evaluation is employed to determine the principal thermal and hydraulic parameters associated with core behavior. These parameters are:

Core flow rate

Core thermal power level

MLHGR

MCPR

MAPLHGR

Prior to the verification of the Process Computer in ST-13, an independent method will be used to calculate these parameters. After the successful completion of ST-13, the process computer will be used.

Acceptance Criteria - Level 1 - The Maximum Linear Heat Generation Rate (MLHGR) of any rod during steady-state conditions shall not exceed the limit specified by the Plant Technical Specifications.

The steady-state Minimum Critical Power Ratio (MCPR) shall not exceed the limits specified by the Plant Technical Specifications.

The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) shall not exceed the limits specified by the Plant Technical Specifications.

Steady-state reactor power shall be limited to the rated MWT and values on or below the licensed analytically determined power-flow line.

Level 2 - Not applicable.

(ST-20) Steam Production Verification

(This test deleted from the FSAR for Unit 2).

(ST-21) Core Power-Void Mode Response

Test Objectives - The objective of this test is to verify the stability of the core power-void dynamic response.

Prerequisites - The required preoperational tests have been completed. Instrumentation has been calibrated.

Test Method - The core power void loop mode that results from a combination of the neutron kinetics and core thermal hydraulic dynamics is least stable near the natural circulation end of the rated 100 percent power rod line. A fast change in the reactivity balance is obtained by moving a very high worth rod only 1 or 2 notches, by simulating a failure of the pressure regulator and by performing pressure regulator setpoint changes.

Acceptance Criteria - Level 1 - The transient response of any system related variable to any test input must not diverge.

Level 2 - Not applicable.

(ST-22) Pressure Regulator

Test Objectives - The objectives of this test are to demonstrate the takeover capability of the backup pressure regulator upon failure of the controlling pressure regulator and to demonstrate smooth pressure control transition between the control valves and bypass valves when reactor steam generation exceeds steam flow used by the turbine.

Prerequisites - The required preoperational tests have been completed. Instrumentation has been checked or calibrated as appropriate.

Test Method - The pressure set point will be decreased rapidly and later increased rapidly by about 10 psi and the response of the system will be measured in each case. It is desirable to accomplish the set point change in less than 1 second. At specified test conditions the load limit setpoint 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.

Acceptance Criteria - Level 1 - The transient response of any pressure control system related variable to any test input must not diverge.

Level 2

a) Pressure control system related variables may contain oscillatory modes of response. In these cases, the decay ratio must be less than or equal to 0.25 when operating above the lower limit of the master manual controller.

b) When in the recirculation manual mode, the pressure response time from initiation of pressure setpoint step change to the turbine inlet pressure peak shall be ≤10 seconds.

c) Pressure control system deadband, delay, etc., shall be small enough that steady state limit cycles (if any) shall produce steam flow variations no larger than ±0.5 percent of rated steam flow.

d) The normal difference between regulator set points must be small enough that the peak neutron flux and/or peak vessel pressure remain below the scram settings by 7.5 percent and 10 psi respectively, for the Regulator Failure Test performed at Test Condition 6.

(ST-23) Feedwater System

Test Objectives - The objectives of this test are a) to demonstrate acceptable response to the feedwater control system for reactor water level control, b) to demonstrate stable reactor response to subcooling changes, i.e., loss of feedwater heating, 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, and d) to demonstrate the maximum feedpump runout capability is compatable with licensing assumptions.

Prerequisites - The required preoperational tests have been completed. Instrumentation has been checked or calibrated as appropriate.

Test Method - At Test Condition (TC) 1 with the water level being automatically controlled using the low load valve and the recirculation system in Manual, ±5 inch step changes in the water level setpoint will be made to demonstrate proper response and operability of the feedwater system at low reactor power.

At Test Conditions 2, 3 and 6, with one feedwater pump in manual and the others in auto, small and large flow changes in the manually controlled feed pump will be made. The response of the feedwater system to these steps will be analyzed and compared to the applicable acceptance criteria. The recirculation system will be in manual for these tests. At Test Conditions 1, 2, 3, 4, 5 & 6, with the recirculation system in manual, ±5 inch changes in the water level setpoint will be made to demonstrate proper response and stability of the feedwater system.

At approximately 80% to 90% power with core flow near 100% of rated, a simulated failure of the extraction steam valves to one of the feedwater heater trains is accomplished by closing the heater train steam inlet isolation valves which will isolate

extraction steam from the last three stages of that train. Recordings of the transient will be analyzed and compared to the predicted response and acceptance criteria.

At Test Condtion 6, one feedwater pump will be tripped to demonstrate the capability to avoid a scram and prevent a low reactor water level trip due to the loss of one feedwater pump.

A maximum feedwater runout capability test will be done to demonstrate that the actual capability is compatible with licensing assumptions.

Acceptance Criteria - Level 1 - The transient response of any level control system-related variable to any test input must not diverge.

For the feedwater heater loss test, the maximum feedwater temperature decrease due to a single failure case must be less than or equal to 100°F. The resultant MCPR must be greater than the fuel thermal safety limit.

For the feedwater heater loss test the increase in heat flux cannot exceed the predicted Level 2 value by more than 2%. The predicted value will be based on the actual test values of feedwater temperature change and power level.

The feedwater flow runout capability must not exceed the assumed value in the FSAR.

Level 2 - Level control system-related variables may contain oscillatory modes of response. In these cases, the decay ratio for each controlled mode of response must be less than or equal to 0.25.

The open loop dynamic flow response of each feedwater actuator (turbine or valve) to small (≤10%) step disturbances shall be:

(1)	Maximum time to 10% of a step disturbance	1.1 sec.
(2)	Maximum time from 10% to 90% of a step disturbance	1.9 sec.
(3)	Peak overshoot (% of step disturbance)	≤15%

The average rate of response of the feedwater actuator to large (≥20% of pump flow) step disturbances shall be between 10 percent and 25 percent rated feedwater flow/second. This average response rate will be assessed by determining the time required to pass linearly through the 10 percent and 90 percent response points.

For the feedwater heater loss test the increase in heat flux cannot exceed the predicted value referenced to the actual Feedwater temperature change and the initial power level.

A scram must be avoided from low water level with at least a 3 inch margin following a trip of one of the operating feedwater pumps.

With extrapolated reactor pressure equal to 1060 psig, the sum of the calculated maximum reactor feed pump flows must be greater than 15.4×10^6 lbs/hr.

With extrapolated reactor pressure equal to 1010 psig, the sum of the two smallest maximum reactor feed pump flows as calculated must be greater than 9.1×10^6 lbs/hr.

(ST-24) Turbine Valve Surveillance

Test Objectives - The objective of this test is to demonstrate acceptable procedures and maximum power levels for periodic surveillance testing of the main turbine control, stop, intercept and bypass valves without producing a reactor scram.

Prerequisites - The required preoperational tests have been completed. Instrumentation has been checked or calibrated as appropriate.

Test Method - The test of the control, main stop, intermediate stop and bypass valves are performed near the predicted highest power level to demonstrate that the Acceptance Criteria are satisfied. Rate of valve stroking and timing of the close-open sequence will be such that minimum practical disturbance is introduced and that PCIOMR limits are not exceeded.

Acceptance Criteria - Level 1 - Not applicable.

Level 2 - Peak neutron flux must remain at least 7.5% below the Neutron flux scram trip value. Peak vessel pressure must remain at least 10 psi below the high pressure scram setting. Peak steam flow in each line must remain at least 10% below the high flow isolation trip setting. Peak simulated heat flux must remain at least 5% below its scram trip point.

(ST-25) Main Steam Isolation Valves

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

Prerequisites - The required preoperational tests have been completed. Instrumentation has been checked or calibrated as appropriate.

Test-Method - The Main Steam Isolation Valves (MSIVs) are operated during this test to verify their functional performance and to determine closure times. While functionally testing the operation of the MSIVs, the time necessary for closing each individual valve will be noted. The fastest MSIV will then be tested to determine what power level an MSIV can experience fast closure without causing a scram. All MSIVs will later be used to demonstrate a full isolation subsequently leading to a scram. (The Nuclear Steam Supply Shutoff System (NSSSS) logic will be used to initiate the full isolation). The acceptability of the fast criteria (3 seconds) is determined by utilizing the full stroke time without delay extrapolated from measured stroke times between 10% closed and 90% closed. The acceptability of the slow criteria (5 seconds) is determined by utilizing the full stroke time with delay extrapolated for the final 10% of stroke.

Acceptance Criteria - Level 1

The positive change in vessel dome pressure occurring within 30 seconds after closure of all MSIVs must not exceed predicted values by more than 25 psi.

The positive change in heat flux following closure of all MSIVs shall not exceed predicted values by more than 2% of rated value.

Following the closure of all MSIV's, the reactor must scram.

Closure time for any MSIV, including delay, shall not be greater than 5.5 seconds.

Closure time for any MSIV shall not be less than 3.0 seconds nor greater than 5.0 seconds.

Feedwater control settings must prevent flooding the main steam lines during the full isolation test.

Level 2 - The positive change in vessel dome pressure occurring within the first 30 seconds after the closure of all MSIVs must not exceed the predicted values. Predicted values will be referenced to actual test conditions of initial power level, scram timing and dome pressure and will use beginning of life nuclear data.

The positive change in heat flux occurring within the first 30 seconds after the closure of all MSIVs must not exceed the predicted values. Predicted values will be referenced to actual test conditions of initial power level, and dome pressure and will use beginning of life nuclear data.

If water level reaches Level 2 setpoint during the MSIV full closure test, RCIC shall automatically initiate and reach rated flow.

During the MSIV full closure test, the relief valves must reclose properly (without any detectable leakage) following the pressure transient.

During full closure of individual MSIVs, peak vessel dome pressure must remain at least 10 psi below the scram setpoint.

During full closure of individual MSIVs, peak neutron flux must remain at least 7.5% below its scram setpoint.

During full closure of individual MSIVs, steam flow in individual lines must remain at least 10% below the high flow isolation trip setpoint.

During full closure of individual MSIVs, the simulated heat flux must remain at least 5% less than its flow biased scram setpoint.

(ST-26) Relief Valves

Test Objectives - The objectives of this test are to verify that the relief valves function properly, reseat properly after operation and contain no major blockages in the relief valve discharge piping.

Prerequisites - The required preoperational tests have been completed. Instrumentation has been checked or calibrated as appropriate. Factory test results on SRV flow and operating times have been reviewed.

Test Method - Testing done at low reactor pressure, in conjunction with plant surveillance testing, consists of cycling each relief valve to verify proper operation. The transient monitoring system will be used to record the results of this test. The data collected will compare the operation of individual relief valves against the operation of all relief valves. During relief valve operation, core power - and therefore steam generation rate - is maintained constant. The pressure control system will close the bypass valves an amount proportional to the relief valve steam flow to maintain constant reactor pressure. This bypass valve motion will be monitored and a comparison of the response for each relief valve operation will be made. If differences exist, it could suggest a partial obstruction of the relief valve or its tailpipe. Tailpipe temperature will be recorded to verify the relief valve has properly reseated. Reactor variables will also be recorded to verify system stability during opening and closing each relief valve.

Testing done at rated reactor pressure consists of manually operating each relief valve at rated reactor pressure. The decrease in Main Generator output will be monitored during the operation of each relief valve to provide an indication of relief

valve flow. By comparison of the generator output response for each relief valve operation, any flow obstruction in the valve or its tailpipe can be identified. Each valve will be opened for approximately 10 seconds to allow for variables to stabilize. Reactor variables will also be recorded to verify system stability during opening and closing each relief valve.

Acceptance Criteria - Level 1

There should be a positive indication of steam discharge during the manual actuation of each valve.

Level 2 - Pressure control system-related variables may contain oscillatory modes of response. In these cases, the decay ratio for each controlled mode of response must be less than or equal to 0.25.

The temperature measured by thermocouples on the discharge side of the valves shall return to within 10°F of the temperature recorded before the valve was opened.

During the low pressure functional tests, the change in bypass valve position for each SRV opening shall be greater than or equal to a value corresponding to the average change minus 10% of one bypass valve.

During the rated pressure tests, the change in MWe for each SRV opening shall be greater than or equal to a value corresponding to the average change minus 0.5% of rated MWe.

(ST-27) Turbine Trip and Generator Load Rejection

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

Prerequisites - The required preoperational tests have been completed. All instrumentation has been calibrated.

Test Method - At Test Condition 3, a turbine trip will be manually initiated by depressing the Turbine Trip pushbutton in the main control room. At Test Condition 6, a generator load rejection will be manually initiated by remotely opening the generator synchronizing breaker from the control room. During both transients, reactor water level, pressure, neutron flux and simulated heat flux will be recorded and compared to predicted results and acceptance criteria.

At approximately 24% power, a generator load rejection within bypass capacity will be manually initiated as described above. This will demonstrate the ability to ride through a load rejection within bypass capacity without a scram.

During all 3 transients, main turbine stop, control and bypass valve positions and reactor water level will be recorded and compared to the acceptance criteria.

Acceptance Criteria - Level 1

a) For Turbine and Generator trips there should be a delay of no more than 0.1 seconds following the beginning of control or stop valve closure before the beginning of bypass valve opening. The bypass valves should be opened to a point corresponding to greater than or equal to 80 percent of full open within 0.3 seconds from the beginning of control or stop valve closure motion.

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

c) The positive change in vessel dome pressure occurring within 30 seconds after either generator or turbine trip must not exceed the Level 2 criteria by more than 25 psi.

d) The positive change in simulated heat flux shall not exceed the Level 2 criteria by more than 2% of rated value.

e) The two pump drive flow coastdown transient, during the first three seconds of an RPT trip, must fall within the specified limits.

Level 2

a) There shall be no MSIV closure in the first 3 minutes of the transient and operator action shall not be required in that period to avoid the MSIV trip.

b) The positive change in vessel dome pressure and in simulated heat flux which occur within the first 30 seconds after the initiation of either generator or turbine trip must not exceed the predicted values.

(Predicted values will be referenced to actual test conditions of initial power level, dome pressure, scram timing, and the time from the start of stop/control valve motion to start of control rod motion, and will use beginning of life nuclear data.)

c) For the Generator trip within the bypass valves capacity (initial thermal power values less than or equal to 25 percent of rated) the reactor shall not scram.

d) The Total Delay from the initiation of a Turbine Stop Valve Closure or Turbine Control Valve Fast Closure to complete suppression of the Electric Arc between the fully open contacts of the Recirculation Pump Trip (RPT) Breaker shall be less than 175 milliseconds.

e) Feedwater level control shall avoid the loss of feedwater flow due to a high level (L8) trip.

f) Feedwater level control shall maintain water level above the L2 level trip setpoint for HPCI, RCIC and ATWS RPT.

(ST-28) - Shutdown from Outside the Main Control Room

Test Objective - The objective of this test is to demonstrate that the reactor can be shutdown, maintained in a hot shutdown condition, and cooled down from outside the main control room. Also, the adequacy of the Emergency Operating Procedures will be verified.

Prerequisites - The required preoperational tests have been completed. Instrumentation has been checked or calibrated as appropriate.

Test Method - While operating at approximately 20% power synchronized to the grid with normal electrical system alignment, the reactor will be scrammed and the MSIV's will be closed from inside the main control room. The control room will then be evacuated, and reactor level and pressure will be controlled from outside the main control room. The Shutdown Cooling mode of RHR will be placed into service with cooling water supplied from the ultimate heat sink. During this demonstration, some supervisory and operating personnel will remain in the control room to protect non-safety-related equipment from unnecessary damage if conditions arise and to assume control of the plant if conditions warrant. A test will be run to demonstrate that the reactor can be scrammed and isolated from outside the control room.

Acceptance Criteria - Level 1 - Not applicable.

Level 2 - During a simulated control room evacuation, the reactor must be brought to the point where cooldown is initiated and under control, and the reactor vessel pressure and water level are controlled using equipment and controls outside the control room. The test is deemed successful when reactor pressure is less than the permissive setpoint and the RHR shutdown cooling mode has been put in operation.

The reactor must be capable of being scrammed and isolated from outside the control room.

(ST-29) - Recirculation Flow Control System

The objectives of this test are:

a) To demonstrate the flow control capability of the plant over the entire pump speed range, including individual local manual and combined Master Manual Operation.

b) To determine that all electrical compensators and controllers are set for desired system performance and stability.

Prerequisites - The required preoperational tests have been completed.

All instrumentation has been calibrated.

Test Method - At Test Conditions 2, 3, 5 and 6, the stability of the recirculation flow control system is demonstrated by performing step changes in recirculation pump speed. This testing is done in individual local manual at Test Conditions 2 and 5 and in combined Master Manual operation at Test Conditions 3 and 6 to demonstrate operability and stability.

Testing will also be performed to verify that the Recirc M-G set high speed mechanical stops are properly set.

Acceptance Criteria - Level 1 - The transient response of any system-related variable to any test input must not diverge.

Level 2 - A scram shall not occur due to recirculation flow control maneuvers.

The APRM neutron flux trip avoidance margin shall be greater than or equal to 7.5% and the simulated heat flux trip avoidance margin shall be greater than or equal to 5% when the power maneuver effects are extrapolated to those that would occur along the 100% rated rod line.

The decay ratio of any oscillatory controlled variable must be less than or equal to 0.25.

Steady state limit cycles (if any) shall not produce turbine steam flow variations greater than ±.5% of rated steam flow.

(ST-30) - Recirculation System

Test Objectives - The objectives of this test are:

a. Verify that the feedwater control system can satisfactorily control water level without a resulting turbine trip and associated scram.

b. Record and verify acceptable performance of the recirculation two pump circuit trip system.

c. Verify the adequacy of the recirculation runback to mitigate a scram.

d. Verify that no recirculation system cavitation will occur in the operable region of the power-flow map.

Prerequisites - The required preoperational tests have been completed. Instrumentation has been checked or calibrated as appropriate.

Test Method - Single recirculation pump trips will be made at Test Condition (TC) 3 and TC-6. These trips will be initiated by tripping the M-G Set Drive Motor Breaker from the control room. Reactor parameters will be recorded during the transient and analyzed to verify non-divergence of oscillatory responses, adequate margins to RPS scram set points, and capability of the feedwater system to prevent a high level trip. The capability to restart the recirc. pump at a high power level will also be demonstrated. At TC-3, both recirculation pumps RPT breakers will be simultaneously tripped using a temporarily installed test switch. The data gathered will be used to demonstrate acceptable pump coastdown performance prior to high power turbine trips and generator load rejects.

Appropriate conditions will be simulated at TC-3 to demonstrate the proper operation of the recirculation pump runback circuits. This is done prior to an actual planned feed pump trip at rated power.

Both the jet pumps and the recirculation pumps will cavitate at conditions of high flow and low power where NPSH demands are high and little feedwater subcooling occurs. However, the recirculation flow will automatically runback upon sensing a decrease in feedwater flow. The maximum recirculation flow is limited by appropriate stops which will run back the recirculation flow from the possible cavitation region. At TC-3, it will be verified that these limits are sufficient to prevent operation where recirculation pump or jet pump cavitation occurs.

Acceptance Criteria - Level 1 - The response of any level related variables during a single pump trip must not diverge.

The two pump drive flow coastdown transient, during the first 3 seconds of an RPT trip, must fall within the specified limits.

Level 2 - The reactor shall not scram during the one pump trip.

The APRM margin to avoid a scram shall be at least 7.5% during the one pump trip recovery.

The reactor water level margin to avoid a high level trip shall be at least 3.0 inches during the one pump trip.

Peak simulated heat flux must remain at least 5% below its flow biased scram setpoint.

Runback logic shall have settings adequate to prevent recirculation pump operation in areas of potential cavitation.

The recirculation pumps shall runback upon a trip of the runback circuit.

(ST-31) - Loss of Turbine-Generator and Offsite Power

Test Objectives - The objectives of this test are to demonstrate that the required safety systems will initiate and function properly without manual assistance, the electrical distribution and diesel generator systems will function properly, and the HPCI and/or RCIC systems will maintain water level if necessary during a coincidental loss of the Unit 2 main turbine-generator and offsite power to Unit 2.

Prerequisites - The required preoperational tests have been completed. Instrumentation has been checked or calibrated as appropriate.

Test Method - With the unit synchronized to the grid at approximately 30% power, the main turbine-generator will be manually tripped coincident with a manual trip of the unit's offsite power source breaker, both trips initiated from the control room. During Unit 2 testing, to ensure a full simulation of the loss of all offsite power to Unit 2 while minimizing the impact on Unit 1 operations, all Unit 2 loads will be transferred to Unit 2 Auxiliary and Startup busses, all Unit 1 and common loads will be transferred to Unit 1 Auxiliary and Startup Busses, and appropriate breakers will be racked out to prevent automatic transfer of Unit 2 loads to Unit 1 sources.

Reactor water level and the operation of safety systems will be monitored to verify that the acceptance criteria are satisfied. The proper response of the electrical distribution system will be checked.

The loss of offsite power condition will be maintained for at least 30 minutes to demonstrate that necessary equipment, controls, and indication are available to remove decay heat from the core using only emergency power supplies and distribution system.

Acceptance Criteria - Level 1 - All safety systems, such as the Reactor Protection System, the diesel-generators, RCIC and HPCI must function properly without manual assistance, and HPCI and/or

RCIC system action, if necessary, shall keep the reactor water level above the initiation level of Core Spray, LPCI and ADS.

Level 2 - The temperature measured by the thermocouples on the discharge side of any SRV that actuated shall return to within 10°F of the temperature recorded before the valve opened.

Permanent instrumentation for reactor power, reactor pressure, water level, control rod position, suppression pool temperature, high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) shall be demonstrated operable following re-energization of the 4kV busses by the diesel generators.

(ST-32) Containment Atmosphere and Main Steam Tunnel Cooling

Test Objective - The objective of this test is to verify the ability of the drywell coolers/recirculation fans and the reactor building portion of the main steam tunnel coolers to maintain design conditions in the drywell and reactor building portion of the mainsteam tunnel, respectively, during operating conditions and post scram conditions.

Prerequisites - The required preoperational tests have been completed. Instrumentation has been checked or calibrated as appropriate.

Test Method - During heatup, at test conditions 2 and 6, and following a planned scram from 100% power, data will be taken to ascertain that the containment atmospheric conditions are within design limits.

Acceptance Criteria - Level 1 - The area under the reactor vessel in the Control Rod Drive are is maintained below 185°F.

Level 2 - The general drywell area is maintained at an average temperature less than or equal to 135°F, with maximum local temperature not to exceed 150°F.

The area beneath the reactor pressure vessel is maintained at an average temperature less than or equal to 135°F, maximum local temperature not to exceed 167°F, with minimum local temperature above 100°F.

The area around the recirculation pump motors is maintained at an average temperature less than or equal to 128°F, with maximum local temperature not to exceed 135°F.

The inside base of the shield wall in the RPV skirt area is maintained at temperatures greater than 100°F.

The reactor building portion of the mainsteam pipeway is maintained at or below 125°F.

The area surrounding the drywell head shall have an average temperature equal to or greater than 135°F, with maximum local temperature not to exceed 150°F.

The reactor pressure vessel support skirt flange shall be maintained at or below 150°F...

The temperature of the concrete surrounding the primary containment main steamline penetrations are maintained less than 200°F.

(ST-33) Piping Steady State Vibration

Test Objectives - The objectives of this test is to demonstrate that steady state vibration levels on reactor recirculation, main steam inside containment, and those piping systems identified in Table 3.9-33 are within acceptable limits. (Note that this test now includes piping previously contained in ST-40. Also note that dynamic transient vibration testing previously contained in this test have been merged into ST-39.)

Preprequisites - Instrumentation has been installed and calibrated.

Test Method - Devices for measuring continuous vibration are mounted on main steam lines, recirculation lines and lines of systems identified in Table 3.9-33 as applicable and vibration during steady state operation is compared with calculated values.

Acceptance Criteria - Level 1 - The measured amplitude (peak to peak) of each remotely monitored point on the main steam inside containment and reactor recirculation lines shall not exceed the allowable value for that point.

Level 2 - The measured amplitude (peak to peak) of each remotely monitored point on the main steam inside containment and reactor recirculation lines shall not exceed the expected value for that point.

The vibratory response of non-remotely monitored systems or portions of systems indentified in Table 3.9-33 shall be judged to be within acceptable limits by a qualified test engineer.

The maximum measured amplitude of the piping response for each remotely monitored point on systems identified in Table 3.9-33 shall not exceed the acceptable value for that point.

(ST-34) Control Rod Sequence Exchange

This test will not be performed during the Unit 2 Startup Test Program.

Test Method - The control rod sequence exchange begins on the design flow control line with core flow near minimum. Control rods will be inserted as necessary to increase the margin to local core thermal limits. Core power is maintained above the low power setpoint of the Rod Worth Minimizer and Rod Sequence Control System and below the power which will keep fuel assembly nodal power at the PCIOMR threshhold. The exchange is performed in accordance with the plant operating procedure RE-TP-009. Data taken during the exchange will be reviewed to verify that the Acceptance Criteria were satisfied.

Acceptance Criteria - Level 1 - Completion of the exchange of one rod pattern for the complimentary pattern with continual satisfaction of all licensed core limits constitutes satisfaction of the requirements of this procedure.

Level 2 - All nodal powers shall remain below their PCIOMR threshold limit during this test.

(ST-35) Recirculation System Flow Calibration

Test Objectives - The objective of this test is to perform a complete calibration of the installed recirculation system flow instrumentation.

Prerequisites - The required preoperational tests have been completed. Instrumentation has been checked or calibrated as appropriate.

Test Method - During the testing program at selected operating conditions which allow the recirculation system to be operated at speeds required for rated flow at rated power, the jet pump flow instrumentation will be adjusted to provide correct flow indication based on the jet pump flow.

After the relationship between drive flow and core flow is established, the flow biased APRM/RBM system will be adjusted to match this relationship.

Acceptance Criteria - Level 1 - Not applicable.

Level 2 - Jet pump flow instrumentation shall be adjusted such that the jet pump total flow recorder will provide a correct core flow indication at rated conditions.

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

(ST-36) Cooling Water Systems

This test will not be performed during the Unit 2 Startup Test Program.

(ST-37) Gaseous Radwaste System

Test Objectives - The objective of this test is to demonstrate that the Gaseous Radwaste System operates within the Technical Specification during a full range of plant power operation and to demonstrate the proper operation of the offgas and containment nitrogen inerting systems during plant operation.

Prerequisites - The required preoperational tests have been completed. Instrumentation has been checked or calibrated as appropriate. In addition, the 100% power trip testing shall have been completed or 120 effective full power days shall not have elapsed prior to performing the nitrogen inerting test.

Test Method - The test will consist of collecting data and performing quantitative analysis of the off gas system effluent to determine if the performance is acceptable per the Technical Specification. For the nitrogen inerting system, the proper nitrogen concentration will be verified by the as installed plant oxygen detectors/instruments in the two major volumes of the primary containment. Proper operation of the offgas system will also be verified.

Acceptance Criteria - Level 1 - The release of radioactive gaseous and particulate effluents must not exceed the limits specified in the site technical specifications.

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 performing their required function. There shall be no less than 8000 lb/hr. of dilution steam flow when the steam jet air ejectors are pumping. The containment nitrogen inerting system shall be capable of inerting the primary containment free volume within 24 hours from the start of the test and the resulting oxygen concentration shall be less than or equal to 4%.

(ST-38) BOP Piping System Expansion

(The system expansion testing previously contained in this test has been merged into ST-17.)

(ST-39) Piping Vibration During Dynamic Transients

Test Objective - The objective of this test is to demonstrate that vibration levels on main steam inside containment, reactor recirculation, and system piping identified in Table 3.9-33 meet acceptable limits during selected dynamic transients.

Prerequisites: Instrumentation has been installed and calibration.

Test Method - Devices for measuring continuous loads, displacements, accelerations and pressures are mounted on piping systems and responses during transients are compared with calculated values. Those portions of the systems which are non-safety related are visually inspected prior to, during and subsequent to the transient loading condition.

Acceptance Criteria - Level 1 - The measured vibration amplitude (peak to peak) for each remotely monitored point of main steam inside drywell and/or reactor recirculation piping shall not exceed the allowable value for each specific point.

Level 2 - The maximum measured accelerations on those systems listed in Table 3.9-33 shall not exceed the design maximum expected values at each specific point.

The vibratory response of non-remotely monitored systems identified in Table 3.9-33 shall be judged to be within acceptable limits by a qualified test engineer.

Based on visual inspection during a post transient walkdown, there shall be no signs of excessive piping response (such as damaged insulation, markings on piping, structural or hanger steel, or walls, damaged pipe supports, etc.) on systems listed in Table 3.9-33.

The measured vibration amplitude (peak to peak) for each remotely monitored point of main steam inside drywell and/or reactor recirculation piping shall not exceed the expected value for each specific point.

(ST-40) BOP Piping Steady State Vibration

(The steady state vibration testing previously contained in this test has been merged into ST-33.)

TABLE 14.2-3

STARTUP TEST PROCEDURES

Test Number	Test Definition
ST-1	Chemical and Radiochemical
ST-2	Radiation Measurements
ST-3	Fuel Loading
ST-4	Full Core Shutdown Margin
ST-5	Control Rod Drive System
ST-6	SRM Performance and Control Rod Sequence (Unit 1 only)
ST-7	Reactor Water Cleanup System
ST-8	Residual Heat Removal System
ST-9	Water Level Measurement
ST-10	IRM Performance (Unit 1) SRM and IRM Performance and Control Rod Sequence (Unit 2)
ST-11	LPRM Calibration
ST-12	APRM Calibration
ST-13	NSSS Process Computer
ST-14	RCIC System
ST-15	HPCI System
ST-16	Selected Process Temperatures
ST-17	System Expansion
ST-18	TIP Uncertainty
ST-19	Core Performance
ST-20	Steam Production Verification (Unit 1 only)
ST-21	Core Power - Void Mode Response
ST-22	Pressure Regulator
ST-23	Feedwater System
ST-24	Turbine Valve Surveillance
ST-25	Main Steam Isolation Valves

TABLE 14.2-3 (Cont.)

STARTUP TEST PROCEDURES

ST-26	Relief Valves
ST-27	Turbine Trip and Generator Load Rejection
ST-28	Shutdown From Outside the Main Control Room
ST-29	Recirculation Flow Control System
ST-30	Recirculation System
ST-31	Loss of Turbine Generator and Offsite Power
ST-32	Containment Atmosphere and Main Steam Tunnel Cooling
ST-33	Piping Steady State Vibration
ST-34	Control Rod Sequence Exchange (Unit 1 only)
ST-35	Recirculation System Flow Calibration
ST-36	Cooling Water Systems (Unit 1 only)
ST-37	Gaseous Radwaste System
ST-38	BOP Piping System Expansion
ST-39	Piping Vibration During Dynamic Transients
ST-40	BOP Piping Steady State Vibration

TABLE 14.2-4

MAJOR TEST PHASE AND TEST PLATEAU SCHEDULE
TEST CONDITION SEQUENCE

Test Phase	Test Plateau	Test Condition Sequence
III	-	Open Vessel Test Condition
IV	-	Heatup Test Condition
V	A	Test Condition 1
V	B	Testing during approach to Test Condition 2 Test Condition 2
V	C	Testing during approach to Test Condition 3 Test Condition 3
V	D*	Testing during approach to Test Condition 5 Test Condition 5 Testing during approach to Test Condition 6 Test Condition 6 Test Condition 4

* Because of the transitory nature of testing performed along the 100% rod line during Test Phase V Test Plateau D, all testing assigned to Test Condition 6 may not be completed prior to entering Test Condition 4.

TABLE 14.2-5

CONTROL ROD DRIVE SYSTEM STARTUP TESTS

Action	Accumulator Pressure	Reactor Pressure With Core Loaded psig 0	600	800	Rated
Position Indication		all			
Normal Times Insert/Withdraw		all			4*
Coupling		all			
Friction		all			all
Scram Times	Normal	all	4*	4*	all
Scram Times	Minimum	4*			
Scram Times	Zero				4*
Scram Times	Normal				4**

* Refers to 4 CRDs selected for continuous monitoring based on slow normal accumulator pressure scram times, or unusual operating characteristics, at zero reactor pressure or rated reactor pressure when this data is available. The 4 selected CRDs must be compatible with the rod worth minimizer, RSCS system, and CRD sequence requirements.

** Scram times of the four slowest CRDs (based on scram data at rated pressure) will be determined at Test Conditions 1, 3 & 6 during planned reactor scrams.

ST.0	MAIN BODY
ST.0.1	OBJECTIVES
ST.0.2	TEST DESCRIPTION
ST.0.3	ACCEPTANCE CRITERIA
ST.0.4	REFERENCES
ST.0.5	PREREQUISITES
ST.0.6	PRECAUTIONS
ST.0.7	TEST EQUIPMENT
ST.0.8	PROCEDURE
ST.0-A	GENERAL APPENDICES

ST.X	SUBTEST
ST.X.1	DISCUSSION
ST.X.2	INITIAL STATUS
ST.X.3	TEST INSTRUCTIONS
ST.X.4	ANALYSIS
ST.X-A	SPECIFIC APPENDICES

Legend: ST – Startup Test Number
X – Subtest Number
A – Appendix Designator

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STARTUP TEST PROCEDURE
STANDARD FORMAT-UNIT 1

FIGURE 14.2-2B

ST.0	MAIN BODY
ST.0.1	OBJECTIVES
ST.0.2	TEST DESCRIPTION
ST.0.3	ACCEPTANCE CRITERIA
ST.0.4	REFERENCES
ST.0.5	PROCEDURE
ST.0-A	GENERAL APPENDICES
ST.X.	SUBTEST
ST.X.1	DISCUSSION
ST.X.2	PREREQUISITES
ST.X.3	INITIAL STATUS
ST.X.4	TEST INSTRUCTIONS
ST.X.5	SUBSEQUENT ACTIONS
ST.X.6	GROUP A ANALYSIS
ST.X.7	GROUP B ANALYSIS
ST.X-A	SPECIFIC APPENDICES

Legend: ST – Startup Test Number
X – Subtest Number
A – Appendix Designator

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STARTUP TEST PROCEDURE
STANDARD FORMAT-UNIT 2

FIGURE 14.2-2C

Test No.	Test Name	Open Vessel	Heat Up	Test Condition$^{(1)}$ 1	2	3	4	5	6	Warranty
ST-1	Chemical & Radiochemical	X$^{(2)}$	X	X	X	X		X	X	X
ST-2	Radiation Measurements	X$^{(2)}$	X	X		X			X	
ST-3	Fuel Loading	X$^{(2)}$								
ST-4	Full Core Shutdown Margin		X$^{(6)}$							
ST-5	Control Rod Drive	X$^{(2,3)}$	X$^{(3)}$	X$^{(3)}$	X$^{(3)}$	X$^{(3)}$			X$^{(3)}$	
ST-6	SRM Perf. & Control Rod Seq.		X$^{(6)}$					X$^{(12)}$	X$^{(12)}$	
ST-7	Reactor Water Cleanup		X	X$^{(7)}$		X				
ST-8	Residual Heat Removal				X				X$^{(9,13)}$	
ST-9	Water Level Measurements		X							
ST-10	IRM Performance		X$^{(6)}$	X	X$^{(8)}$					
ST-11	LPRM Calibration			X$^{(7)}$	X	X			X	
ST-12	APRM Calibration		X	X	X	X		X	X	X
ST-13	Process Computer					X$^{(9)}$				
ST-14	RCIC		X	X$^{(7)}$					X$^{(8,9)}$	
ST-15	HPCI		X			X		X$^{(8,9)}$	X$^{(8)}$	
ST-16	Selected Process Temps		X			X	X$^{(14)}$		X$^{(14)}$	
ST-17	System Expansion		X$^{(6)}$		X$^{(8)}$				X$^{(9,13)}$	
ST-18	Tip Uncertainty					X			X	
ST-19	Core Performance			X	X	X	X	X	X	X
ST-20	Steam Production									X
ST-21	Core Power-Void Mode Response						X		X$^{(17)}$	
ST-22	Pressure Regulator			X	X	X	X	X	X	
ST-23	Feedwater			X	X	X	X	X	X$^{(8,15)}$	
ST-24	Turbine Valve Surv.					X		X	X$^{(8,16)}$	
ST-25	MSIVs		X	X		X		X	X$^{(8,16)}$	
ST-26	Relief Valves		X		X					
ST-27	Turbine Stop Valve Trip Generator Load Rejection				X$^{(10)}$	X			X	
ST-28	Shutdown From Outside Control Room			X						
ST-29	Recirculation Flow Control				X	X		X	X	
ST-30	Recirculation System					X$^{(11)}$			X	
ST-31	Loss of T-G & Offsite Power				X					
ST-32	Containment Atmosphere and Main Steam Tunnel Cooling		X		X				X$^{(13)}$	
ST-33	Piping Steady State Vibration		X		X	X		X	X$^{(8,13,9)}$	
ST-34	Rod Sequence Exchange								X$^{(17)}$	
ST-35	Recirculation System Flow Calibration					X			X	
ST-36	Cooling Water Systems			X		X			X	
ST-37	Gaseous Radwaste System		X	X		X		X	X	X
ST-38	BOP Piping System Expansion$^{(4)}$									
ST-39	Piping Vibration During Dynamic Transients		X		X	X	X		X	
ST-40	BOP Piping Steady State Vibration$^{(5)}$									

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INDIVIDUAL STARTUP TEST SEQUENCE - UNIT 1

FIGURE 14.2-5, Sheet 1

Descriptive Notes:

(1) See Figure 14.2-6 for Test Condition (TC) region map.
(2) Some Subtests required to be completed prior to fuel load and may be performed during Phase II.
(3) Refer to Table 14.2-5.
(4) Testing merged into ST-17.
(5) Testing merged into ST-33.
(6) May be done during Open Vessel Testing.
(7) May be done during Heatup.
(8) Some Subtests done during approach to Test Condition.
(9) May be done during earlier Test Condition if conditions warrant.
(10) Done within steam bypass capacity.
(11) The simultaneous trip of two Reactor recirculation pumps is done at 100% core flow on the 75% rod line.
(12) Started during approach to Test Condition 5, continued during approach to Test Condition 6.
(13) Some Subtests done after planned major trips from 100% power.
(14) Started during Test Condition 6 and continued during Test Condition 4.
(15) Loss of feedwater heating test done at 80% power.
(16) Determine maximum power Subtest can be performed without causing reactor scram.
(17) Done on 100% rod line near minimum core flow with recirc pumps on.

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INDIVIDUAL STARTUP TEST
SEQUENCE - UNIT 1

FIGURE 14.2-5, Sheet 2

FIGURE 14.2-5, Sht. 3

Test No.	Test Name	Open Vessel	Heat Up	Test Condition[1] 1	2	3	4	5	6
ST-1	Chemical & Radiochemical	X[2]	X	X	X	X		X	X
ST-2	Radiation Measurements	X[2]	X	X		X			X
ST-3	Fuel Loading	X[2]							
ST-4	Full Core Shutdown Margin		X[6]						
ST-5	Control Rod Drive	X[2,3]	X[3]	X[3]		X[3]			X[3]
ST-6	SRM Performance & Control Rod Seq.[18]								
ST-7	Reactor Water Cleanup			X[7]		X			
ST-8	Residual Heat Removal								X[9,13]
ST-9	Water Level Measurements		X						
ST-10	SRM & IRM Performance & Control Rod Seq.		X[6]	X					
ST-11	LPRM Calibration			X[7]	X	X			X
ST-12	APRM Calibration		X	X	X	X		X	X
ST-13	Process Computer					X[9]			
ST-14	RCIC		X	X[7]					X[8,9]
ST-15	HPCI		X			X			X[8,9]
ST-16	Selected Process Temps		X			X	X[14]		X[14]
ST-17	System Expansion		X[6]		X[8]				X[9,13]
ST-18	Tip Uncertainty					X			X
ST-19	Core Performance			X	X	X	X	X	X
ST-20	Steam Production[19]								
ST-21	Core Power-Void Mode Response						X		X[12]
ST-22	Pressure Regulator			X	X	X	X	X	X
ST-23	Feedwater			X	X	X	X	X	X[8,15]
ST-24	Turbine Valve Surv.								X[8,16]
ST-25	MSIVs		X	X					X[8,16]
ST-26	Relief Valves		X		X				
ST-27	Turbine Stop Valve Trip Generator Load Rejection				X[10]	X			X
ST-28	Shutdown From Outside Control Room			X					
ST-29	Recirculation Flow Control				X	X		X	X
ST-30	Recirculation System					X[11]			X
ST-31	Loss of T-G & Offsite Power				X				
ST-32	Containment Atmosphere and Main Steam Tunnel Cooling		X		X				X[13]
ST-33	Piping Steady State Vibration		X		X	X		X	X[8,9]
ST-34	Rod Sequence Exchange[17]								
ST-35	Recirculation System Flow Calibration					X			X
ST-36	Cooling Water Systems[17]								
ST-37	Gaseous Radwaste System		X	X		X		X	X
ST-38	BOP Piping System Expansion[4]								
ST-39	Piping Vibration During Dynamic Transients		X		X	X	X		X
ST-40	BOP Piping Steady State Vibration[5]								

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INDIVIDUAL STARTUP TEST SEQUENCE - UNIT 2

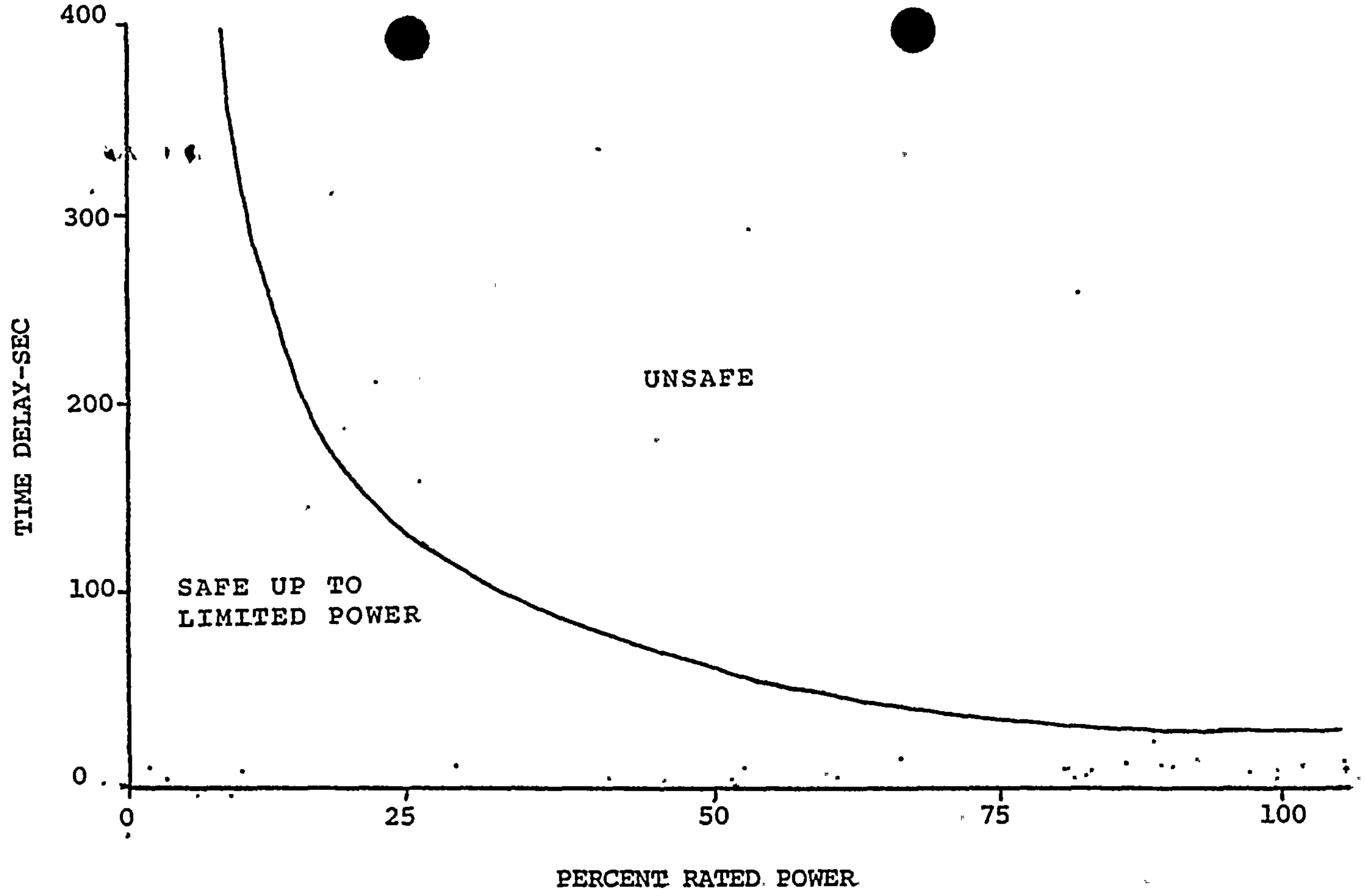
FIGURE 14.2-5, Sheet 3

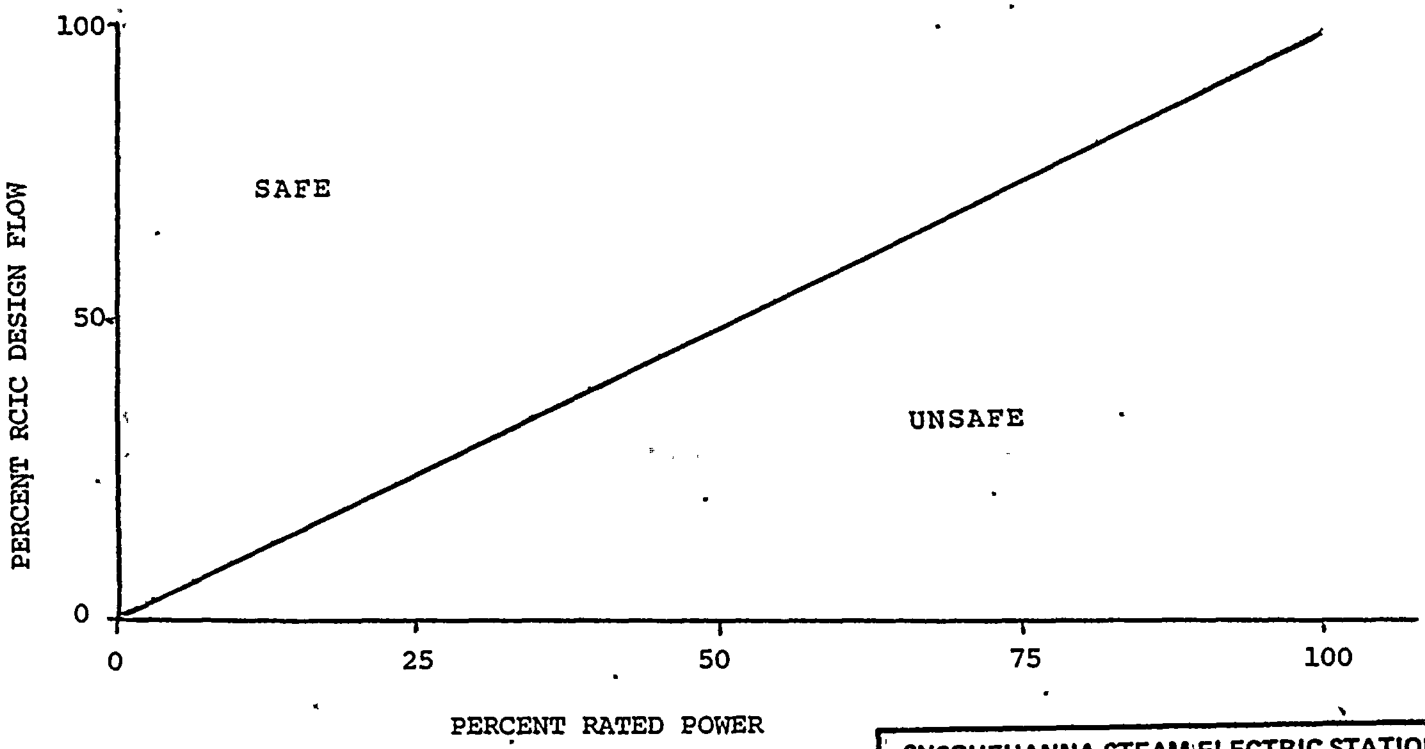
Descriptive Notes:

(1) See Figure 14.2-6 for Test Condition (TC) region map.
(2) Some Subtests required to be completed prior to fuel load and may be performed during Phase II.
(3) Refer to Table 14.2-5.
(4) Testing merged into ST-17.
(5) Testing merged into ST-33.
(6) May be done during Open Vessel Testing.
(7) May be done during Heatup.
(8) Some Subtests done during approach to Test Condition.
(9) May be done during earlier Test Condition if conditions warrant.
(10) Done within steam bypass capacity.
(11) The simultaneous trip of two Reactor recirculation pumps is done at 100% core flow on the 75% rod line.
(12) Done on 100% rod line near minimum core flow with recirc pumps at minimum speed.
(13) Some Subtests done after planned major trips from 100% power.
(14) Started during Test Condition 6 and continued during Test Condition 4.
(15) Loss of feedwater heating test done at 80% to 90% power.
(16) Determine maximum power Subtest can be performed without causing reactor scram.
(17) This test will not be performed during the Unit 2 Startup Test Program.
(18) Testing merged into ST-10.
(19) Deleted from Unit 2 FSAR.

INDIVIDUAL STARTUP TEST SEQUENCE - UNIT 2

FIGURE 14.2-5, Sheet 4





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RCIC ACCEPTANCE CRITERIA CURVES FOR CAPACITY AND ACTUATION TIME

FIGURE 14.2-7