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Subject: **Submittal of ABWR Licensing Topical Report (LTR) NEDO-33310 "Advanced Boiling Water Reactor (ABWR) Startup Test Specification"**

Reference: Letter MFN 017-97, J. Quirk to NRC, *ABWR Design Control Document, Revision 4*, dated March 28, 1997, Docket No. 52-001

The enclosed Licensing Topical Report (LTR) is submitted for NRC generic review and approval as a Combined License (COL) license information item as required by the current ABWR certified design (referenced), Docket No. 52-001. The regulatory basis for this submittal is discussed below.

This is the sixth of a number of ABWR-related LTRs GE plans to submit and which have been discussed in South Texas Project 3&4 project meetings with the NRC. In support of the ABWR Design Centered Working Group (DCWG) plans, GE requests a generic review and approval of the subject LTR in advance of any future combined license applications (COLA) submittals. Note that the submittal is the result of design detailing performed for ABWRs in the US and in Asia and provides for the generic resolution of a COL license information item, thereby contributing to standardization.

This LTR is submitted in response to DCD Tier 2, Section 14.2.13.2, COL license information item – 14.2. The information contained in this LTR is typical for an ABWR Startup Test Specification. The purpose of this LTR is to provide test requirements for the Startup Testing Phase of the ABWR Initial Test Program. This specification establishes test objectives and acceptance criteria, defines operational conditions at which the tests are to be conducted, provides testing methodologies to be utilized,

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identifies specific data to be collected, and provides acceptable data reduction techniques.

The enclosure contains no information that GE considers proprietary although full copyright protection applies.

If you have any questions about the information provided here, please contact me at 910-602-1885.

Sincerely,

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Enclosure: NEDO-33310 *"Advanced Boiling Water Reactor (ABWR) Startup Test Specification"* April 2007 - Non-Proprietary

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LICENSING TOPICAL REPORT

Advanced Boiling Water Reactor (ABWR) Startup Test Specification

INFORMATION NOTICE

This document NEDO-33310, Revision 0, contains no proprietary information.

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

PLEASE READ CAREFULLY

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1.0 Purpose and Scope

2.0 Purpose

The purpose of this document is to provide test requirements for the Startup Testing Phase of the ABWR Initial Test Program. This specification establishes test objectives and acceptance criteria, defines operational conditions at which the tests are to be conducted, provides testing methodologies to be utilized, identifies specific data to be collected, and provides acceptable data reduction techniques. This Licensing Topical Report (LTR) provides the ABWR Startup Test Specification as required by the Combined Operating License (COL) License Information item described in subsection 14.2.13.2 of the ABWR Design Control Document (Tier 2).

2.1 Scope

The Initial Test Program consists of a series of tests categorized as Construction, Preoperational, and Startup. The Construction and Preoperational test phases consist of those tests that are performed prior to the fuel loading and are described elsewhere. This specification deals with the Startup Test Phase which is generally subdivided into three parts: 1) initial fuel loading and open vessel testing; 2) testing during nuclear heatup to rated temperature and pressure (approximately 5% of rated power); and 3) power ascension testing from 5 to100 % of rated power.

The enclosed requirements represent the minimum startup testing required to be performed during the Startup Test Program. The Site GE Operations Manager will review additions to the program for safety considerations. Deletions will require approval from the ABWR Project Office, and be consistent with applicable regulatory requirements.

All required tests and their operating conditions are defined in Table 2. The individual test requirements including the test purpose, a brief discussion of the test methods and philosophy, and the test criteria for each test listed in Table 2 are presented in Section 4. A list of signals that are to be available, but not necessarily recorded, during each specified test is contained in Section 6. Detailed test procedures will be prepared to implement this program.

3.0 Applicable Documents

3.1 Supporting and Supplemental Documents

3.1.1 Supporting Documents

Supporting documents in conjunction with those documents listed in Sections 3.2 and 3.3 provide the design and licensing requirements for the Startup Test Program.

1. "Project Design Manual (PDM)"

2. ABWR Codes and Standards Database

3.1.2 Supplemental Documents

Supplemental documents are those documents that are used in conjunction with this document (some of which may not have been issued at the time of issuance of this Startup Test Specification). The specific document and revision for a given project will be identified in the Plant FSAR reference documents. These references are generic for ABWR and the titles of the document used for a specific project may differ from the titles in this section (e.g., some projects may use the term Nuclear Boiler vice Main Steam). Additionally, in some cases, documents listed in this section may have been combined with other documents (e.g. combined Project Design Manual vs individual System Design Documents).

1. 23A6100, ABWR Standard Safety Analysis Report (SSAR)
2. NEDO-33305, Advance Boiling Water Reactor (ABWR) Startup Administrative Manual
3. Startup Transient Analysis Report (STAR)
4. "GESTAR, General Electric Standard Application of Reactor Fuel"
5. NEDE-10958PA, "General Electric Thermal Analysis Basis (GETAB) Data, Correlation and Design Application"
6. ABWR Piping Penetration Design Requirements Specification
7. NEDO-33309, Advance Boiling Water Reactor (ABWR) Preoperational Test Specification
8. "Plant Working Fluids Requirements Document"
9. "Turbine-Generator Control Requirements"
10. "Feedwater Control System (FWC) Hardware/Software Specification"
11. "Feedwater Control System (FWC) BOP/NI Interface Requirements"
12. "Reactor Building Environmental Zone Drawings"
13. "Control Building Environmental Zone Drawings"
14. "Process Flow Diagram Reactor Building Cooling Water System (RCW)"
15. "Process Flow Diagram Residual Heat Removal System (RHR)"
16. "Process Flow Diagram Reactor Water Cleanup System (CUW)"
17. American Society of Mechanical Engineers (ASME) Steam Tables
18. " Nuclear Island Piping Systems Preoperational and Startup Testing Test Specification"

19. "Cycle Management Report"

20. Control Rod Drive System Design Description

21. "Turbine Building Piping Test Requirements Specification."

22. Reactor Water Cleanup System Design Description

3.2 Codes and Standards

The following codes and standards are applicable to the Initial Test Program for boiling water reactor power plants to the extent specified herein. The applicable date/revision of the code or standards is specified in the ABWR Codes and Standards Database.

1. ASME Boiler and Pressure Vessel (B&PV) Code, Section III, Rules for Construction of Nuclear Power Plant Components

2. American National Standards Institute (ANSI) N45.2.4, Quality Assurance Program Requirements for Nuclear Power Plants

3. ASME Performance Test Code (PTC) 6, Steam Turbines

3.3 Regulations and Regulatory Requirements

The following Regulations and Regulatory Requirements are applicable to the Initial Test Program for boiling water reactor power plants to the extent specified herein. The applicable date/revision is specified in the ABWR Codes and Standards Database for each project.

3.3.1 U.S. Code of Federal Regulations (CFR)

1. 10CFR20, "Standards for Protection Against Radiation"

2. 10CFR30, Section 30.53, "Tests"

3. 10CFR50, Section 50.34, "Contents of Applications: Technical Information"

4. 10CFR50.55a, "Codes and Standards"

5. 10CFR50.63, "Loss of All Alternating Power"

6. 10CFR50. Appendix A, "General Design Criteria for Nuclear Power Plants"

7. 10CFR50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants"

8. 10CFR52.79, "Contents of Application: Technical Information"

9. 10CFR 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants"

3.3.2 U.S. Nuclear Regulatory Commission (NRC) Regulatory Guides and NUREGs

1. Regulatory Guide 1.9, "Selection, Design, Qualification, and Testing of Emergency Diesel-Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Plants"
2. Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing"
3. Regulatory Guide 1.30, "Quality Assurance Requirements for Installation, Inspection and Testing of Instrumentation and Electrical Equipment (Safety Guide 30)
4. Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling Water Reactors"
5. Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants"
6. Regulatory Guide 1.68.1, "Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants"
7. Regulatory Guide 1.68.2, "Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants"
8. Regulatory Guide 1.108, "Periodic Testing of Diesel Generators Used as Onsite Electric Power Systems at Nuclear Power Plants"
9. Regulatory Guide 1.116, "Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems,
10. Regulatory Guide 1.139, "Guidance for Residual Heat Removal"
11. Regulatory Guide 1.152, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants"
12. Regulatory Guide 1.168, "Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants"
13. Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)"
14. NUREG 0800, Standard Review Plan, Section 14.2 Initial Plant Test Program Final Safety Analysis Report

4.0 Discussion

4.1 Power/Flow Operating Map

Figure 1 shows the Power/Flow operating map for 10-reactor internal pump (RIP) operation. The Power/Flow operating map is a plot of reactor power (in percent of rated) versus core flow (also in percent of rated) for various operating conditions. The approximately vertical lines represent the power relationship at constant Reactor Internal Pump (RIP) speed. Of special interest are: natural circulation line, minimum RIP speed line, and RIP speed which provides rated core flow at rated power. Other performance characteristics and limits also shown in the figure are: constant rod lines, typical startup/shutdown region, steam separator performance limit region, and instability-restricted region.

4.2 Startup Test Conditions

The test conditions at which a particular startup test is to be performed are included in Table 2. Special requirements and plant operating conditions (Test Plateaus) are also included in Table 2. Each test plateau is defined in Table 1 and illustrated on Figure 1. Table 3 provides a description of control system operating modes that are used in Table 2. Unless otherwise specified, Core Flow Control mode must be used for all startup tests specified in this specification.

4.3 Startup Test Sequence

The actual test sequence will be determined by the Startup Organization. The normal sequence of testing is to complete all tests to be performed at a particular test plateau prior to proceeding to the next plateau.

The recommended test sequence will be: core performance analysis tests, steady state testing, control system tuning, system transient tests, and finally, the major transient tests including plant trips. However, the actual test sequence can vary from the recommended test sequence due to equipment problems and other considerations.

4.4 Test Criteria Definitions

There are three levels of Test Criteria. All three may apply to a specific Startup Test. The following sections define each level and the actions to be taken if an individual test criterion is not satisfied.

4.4.1 Level 1

Level 1 criterion relate to the values of process variables assigned in the design or analysis of the plant and component systems or associated equipment. Violation of these Level 1 criteria may have plant operational or plant safety implications. Therefore, if a Level 1 test criterion is not satisfied, the plant must be placed in a suitable hold condition that is judged to be satisfactory to safety based on the results of prior testing. Plant operating or test procedures or the Technical Specifications may guide the decision on the direction to be taken. Startup tests compatible with this hold condition may be continued.

Resolution of the problem must be documented and pursued by appropriate equipment adjustments or through engineering support by offsite personnel, if needed. Following resolution, the applicable test portion must be repeated to verify that the Level 1 requirement is ultimately satisfied. A description of the problem resolution shall be included in the report documenting the successful test.

4.4.2 Level 2

Level 2 criteria are specified either a) as key plant, system or equipment performance requirements that are consistent with the plant specification, individual system or equipment design specification values or requirements for the measured response, or b) as the expected plant response predicted by best estimate computer code and the desired trip avoidance margins as applicable to plant malfunction testing.

If all Level 2 criteria requirements in a test are ultimately satisfied, there is no need to document a temporary failure (e.g. during tuning and system adjustment) in the test report; unless there is an educational benefit involved. Following resolution, the applicable test portion must be repeated to verify that the Level 2 criterion requirement is satisfied.

If a Level 2 criterion requirement is not satisfied after a reasonable effort, then the cognizant design and engineering organization shall document the results in the Corrective Action Program with a full explanation of their recommendations. In order for the system as a whole to be acceptable, all Level 2 requirements do not necessarily have to be satisfied provided that the overall system performance is evaluated to be acceptable based on engineering's recommendations.

The specific action(s) required in dealing with criteria violations and other test exceptions or anomalies shall be as described in the site based upon the Startup Administrative Manual.

4.4.3 Level 3

Level 3 criteria are associated with specifications on the expected or desired performance of individual components or inner control loop transient performance. Meeting Level 3 criteria helps assure that overall system and plant response requirements are satisfied. Therefore, Level 3 criteria are to be viewed as highly desirable rather than required to be satisfied. Good engineering judgment is appropriate in the application of these rules.

Since overall system performance is a mathematical function of its individual components, one component whose performance is slightly worse than specified can be accepted provided that a system adjustment elsewhere will positively overcome this small deficiency. Large deviations from Level 3 performance requirements are not allowable.

If a Level 3 criterion requirement is not satisfied, the subject component or inner loop shall be analyzed closely. However, if all Level 1 and Level 2 criteria are satisfied, then it is not required to repeat the transient test to satisfy the Level 3 performance requirements. The occurrence of this Level 3 criterion failure shall be documented in the test report

4.5 Test Prerequisites

The following general prerequisites apply to all testing identified in this specification. Additional requirements unique to a specific test or plateau will be identified in the specific test procedures.

1. The preoperational tests have been completed and plant management has reviewed the test procedure(s) and approved the initiation of Startup Testing.

2. For each scheduled testing iteration, the plant shall be in the appropriate operational configuration, as described in Table 2.

3. Control Systems tuning will be performed following verification of steady state operation.

4. Transient testing will be performed after the principal control systems (Feedwater Control System, Pressure Control System and Recirculation Flow Control System) have been successfully tested.

5. Applicable instrumentation shall have been installed, checked and/or calibrated.

5.0 Individual Test Requirements

The following section provides the individual test requirements for the Startup Test Program. These requirements are tabulated in Table 2.

5.1 Test Number 1 – Chemical and Radiochemical Measurements

5.1.1 Purpose

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

5.1.2 Description

Prior to fuel loading, if not demonstrated during preoperational testing, a complete set of chemical and radiochemical samples will be taken both to ensure that all sample stations function properly and to determine initial concentrations.

Subsequent to fuel loading, during heatup, and at major power levels of all test plateaus, samples will be taken and measurements will be made to determine the chemical and radiochemical qualities of reactor water and incoming feedwater, amount of radiolytic gas in the main steam, gaseous activities leaving the steam jet air ejectors, liquid effluent activities from the radwaste system discharge, decay times in the offgas lines, and performance of the Reactor Water Cleanup System (CUW) filter demineralizers (F/Ds).

Radiation monitors in effluent release paths, waste handling systems, and process lines will be calibrated. Proper functioning of such monitors will be verified, including comparison with independent laboratory or other analyses. Proper operation of the main steamline and offgas pretreatment process radiation monitors will be verified and sufficient data will be taken to assure proper setting of, or if necessary, to make needed adjustment to, the alarm and trip settings of the applicable instrumentation. Effluent releases will be monitored to confirm they are within the limits of the operating license.

Other testing conducted throughout the test program will include evaluation of fuel performance, evaluations of CUW filter/demineralizer operations (i.e., no cleanup test), measurements of condensate filter and polisher performance, confirmation of condenser integrity, demonstration of proper steam separator- dryer performance, measurement and calibration of the Offgas System, and the evaluation and calibration of certain process instrumentation (including that used to monitor condensate and reactor water conductivity as required by Regulatory Guide 1.56). The demonstration of the proper functioning, and adjustment if necessary, of the Hydrogen Water Chemistry System, the Oxygen Injection System, and the Zinc Injection Passivation System will also be performed, as appropriate, to the extent that proper functioning could not be adequately demonstrated during the pre-operational phase of testing.

Data for these purposes will be obtained from a variety of sources such as plant operating records, routine coolant analysis, radiochemical measurements of specific nuclides, and special chemical tests.

5.1.3 Criteria

5.1.3.1 Level 1

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

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

Water quality shall remain within the guidelines of the water quality specifications. (Ref. 2.1.2.8)

5.1.3.2 Level 2

Not Applicable

5.2 Test Number 2 – Radiation Measurements

5.2.1 Purpose

The principal objectives of this test are: a) to obtain baseline data on the background radiation levels in the plant environs prior to operation; and b) to monitor radiation at selected power levels to assure protection of personnel during normal plant operation.

5.2.2 Description

Two categories of radiation level measurements (i.e., Complete Standard Surveys and Limited Standard Surveys) will be made. The Complete Standard Surveys provide gamma, fast neutron, and thermal neutron data useful in verifying radiation levels and specifying occupancy times for protection of personnel. The Limited Standard Surveys are made specifically in the long-residence-time areas and are wide area surveys intended to verify that there are no unexpected radiation sources. All potentially high radiation areas throughout the plant shall be surveyed. Specifically, the following locations should be included:

1. Drywell to containment building penetrations.

2. All accessible areas where intermittent activities have the potential to produce transient high radiation conditions before, during and after such operations (example – transfer of discharged resins from CUW system backwash receiver tank).

3. All accessible floor areas within the containment building prior to fuel loading, at intermediate powers and at rated power.

The survey locations shall be clearly identified and marked prior to the time of the background measurements. It is expected that other locations will be added and similarly marked as the test program progresses. The surveyor should continuously monitor radiation levels as he moves from one test location to another. Any locations between those designated, at which higher radiation levels are observed in the course of surveying, should be added. The surveyor should also look for and monitor potentially hazardous areas, such as points where doors or pipes for an area of high radiation enter a high occupancy area.

A survey of background radiation levels throughout the plant is made prior to fuel loading. Subsequent surveys will be conducted at the following times:

- After fuel loading but prior to initial criticality,

- during heatup,

- and at several major power levels at test plateaus up to and including rated condition.

These surveys will be performed at specific locations throughout the plant and will include gamma dose rate measurements and, where appropriate, neutron dose rate measurements. This is done to verify that acceptable radiation levels exist in normally accessible locations in and around the plant.

At selected reactor conditions, radiation levels at all pre-selected locations are measured and the meter readings are recorded. A note indicating the operating status of the waste equipment should accompany radiation measurements taken in the vicinity of waste equipment shielding walls. Detectors should be held 3 feet above the designated locations. A search for radiation streaming in the vicinity of piping and other penetrations through shielding walls and near shield discontinuities will be performed. Caution should be exercised by the surveyors to minimize their exposures.

5.2.3 Criteria

5.2.3.1 Level 1

The radiation doses of plant origin and the occupancy times of personnel in radiation zones shall be controlled consistent with the guidelines outlined in 10CFR20 "Standards for Protection Against Radiation".

5.2.3.2 Level 2

Not Applicable

5.3 Test Number 3 – Fuel Loading

5.3.1 Purpose

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

5.3.2 Description

Prior to fuel loading, fuel and control rod inspections shall be completed and control rods fully inserted and functionally scram tested. Neutron sources shall be installed and the required number of Startup Range Neutron Monitor (SRNM) channels shall be calibrated, operable and placed in the non-coincidence mode of operation. Reactor vessel water level shall be established above the minimum prescribed level. In addition, systems required to support fuel loading shall be operable as defined by the Technical Specifications and demonstrated by the applicable surveillance tests. Appendix B provides a recommended Initial Fuel Loading Master Checklist to assist in ensuring the site is ready to commence initial fuel load. This checklist should be included and expanded based on site requirements in the fuel load procedure. Appendix C provides a list of surveillance procedures applicable to the fuel load process.

Fuel loading will commence and proceed in a predetermined sequence in accordance with a detailed written procedure. Fuel loading will be performed in a spiral pattern around a selected fuel location until the core is fully loaded in a safe, correct and efficient manner.

The neutron count rates will be monitored as the core loading progresses to ensure sub-criticality. A partial core shutdown margin demonstration will also be performed at a specified loading interval.

5.3.3 Criteria

5.3.3.1 Level 1

The partially loaded core must be subcritical by at least the amount (in terms of reactivity) specified on the Cycle Management Report or Vendor's prediction with the analytically determined highest worth rod pair fully withdrawn (a rod pair is defined as having a shared scram accumulator).

5.3.3.2 Level 2

Not Applicable

5.4 Test Number 4 – Full Core Shutdown Margin Demonstration

5.4.1 Purpose

The purpose of this test is to demonstrate that the reactor will remain sub-critical throughout the first fuel cycle with the highest worth control rod or highest worth control rod pair fully withdrawn and all other control rods fully inserted. A control rod pair consists of two control rods that are connected to the same Hydraulic Control Unit (HCU) scram accumulator.

5.4.2 Description

Prior to performing this test, the control rod position needed to attain criticality is analytically determined and documented. The Standby Liquid Control System (SLC) shall be operable as defined by the Technical Specifications and demonstrated by the applicable surveillance tests. In addition, the SRNM instrumentation shall also be operable in the non-coincident trip mode with the minimum neutron count rate and signal-to-noise ratio as specified in the Project Design Manual or plant Technical Specifications. The SRNM high flux scram and rod block trips shall be set conservatively low.

The full core shutdown margin test will be performed with the core fully loaded and in the xenon-free condition. This test will be performed by withdrawing control rods in sequence, from the all-rods-in configuration, until criticality is achieved. If the highest worth control rod or control rod pair is not withdrawn for this sequence, other control rods or control rod pairs may be withdrawn providing that the reactivity worth is equivalent. The difference between the measured K_{eff} and the calculated K_{eff} for the in-sequence critical will be determined, and then the predicted shutdown margin corrected to obtain the measured shutdown margin.

5.4.3 Criteria

5.4.3.1 Level 1

The shutdown margin of the fully loaded, cold (20°C), xenon-free core occurring at the most reactive time during the cycle must be at least that amount required by Technical Specifications with the analytically strongest rod pair (or the reactivity equivalent) fully withdrawn. If the shutdown margin is determined 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 the specified amount plus an exposure dependent correction factor which adjusts for the difference in core reactivity between the most reactive time and the time at which the shutdown margin is demonstrated.

5.4.3.2 Level 2

Criticality shall occur within the specified tolerance of the predicted critical.

5.5 Test Number 5 – Control Rod Drive System Performance

5.5.1 Purpose

The purpose of this test is to demonstrate that the control rods operate properly over the full range of primary coolant temperatures and pressures from ambient to operating, in both the scram and fine motion control modes, in conjunction with the Rod Control and Information System (RCIS).

5.5.2 Description

While in step driving, notch driving and continuous driving modes of operation, each control rod will be tested to demonstrate that it can satisfy its performance requirements for rod movement both individually and as a gang group in response to rod movement commands from the Rod Control and Information System (RCIS). For each of these modes, the drive position at the completion of movement will be verified. Additionally, the rated drive speed during continuous drive movement will be verified by the timing of each control rod from fully inserted to the fully withdrawn position.

Each drive will be withdrawn from full-out to the over-travel position (i.e., uncoupling check position) using Coupling Check Test Mode to: 1) demonstrate actuation of the separation switches, i.e., change from closed ("ON" status) to open ("OFF" status), 2) confirm synchro position indication as the drive is withdrawn to the over-travel position, and 3) confirm the integrity of the coupling between the control blade and the hollow piston.

Gang rod operation in response to commands from the RCIS (and the Automatic Power Regulator (APR) during automatic rod movement) will be demonstrated during reactor heatup and at reactor rated temperature and pressure conditions. This test will also verify that a rod withdrawal block is activated as a result of rod gang misalignment during normal gang rod movement.

Each paired (two-CRD per HCU) and unpaired (one-CRD per HCU) CRD scram performance test will be performed at both atmospheric and rated reactor pressure conditions by using the Scram Test Switches on the test panel. The accumulator charging line valve in the associated hydraulic control unit (HCU) should be closed so that the CRDs do not ride the CRD pump head during the paired and unpaired CRD scram performance test. Four CRDs will be selected based on slow scram times as determined from preoperational or atmospheric scram performance testing, or based on unusual operating characteristics. During reactor heatup with the reactor pressure at 4.14 and 5.51 ± 0.34 MPaG, scram performance tests of these four selected CRDs with the CRD accumulators normally charged will be conducted for continuous monitoring purposes.

Additionally, the full core scram test results will be recorded and evaluated in conjunction with the various planned transient scram tests. The scram insertion time of each fully withdrawn rod will be measured during each scram test. The scram follow function will be confirmed to actuate automatically and go to completion. The intermediate position reed switches will be verified to actuate momentarily as the hollow piston is inserted. The full-in switches will be verified to actuate at the end of the scram stroke. The separation switches will be verified to actuate at the start of scram and return to their normal state at the completion of the scram follow function.

The continuous-insert friction test will be performed for each drive with the reactor at cold, atmospheric pressure conditions. The underside of each Fine Motion Control Rod Drive (FMCRD) hollow piston will be pressurized using a portable friction test cart connected to the hydraulic control unit. The pressure acting on the bottom surface of the FMCRD hollow piston will be measured by the friction test device as the hollow piston and control rod drift in. The variation in water pressure under the hollow piston is then compared against the acceptable limit for indication of abnormal drive line resistance that would adversely affect drive operation.

5.5.2.1 Criteria

5.5.2.2 Level 1

Each CRD must have a maximum withdraw speed in the continuous driving mode no greater than the value specified in the CRD System Design Description.

For vessel pressure between 6.55 PaG and 7.24 MPaG, the maximum scram insertion time for individual, fully withdrawn control rods to each specified position, based on de-energization of scram pilot valve solenoids as time zero, shall be less than or equal to the limits specified in the plant Technical Specifications.

5.5.2.3 Level 2

Each CRD must have a continuous rod motion speed while inserting or withdrawing, as specified in the applicable CRD System Design Description.

For continuous insert friction tests, the measured drive-line friction shall be consistent with the baseline data obtained during pre-operational testing or the limits specified in the applicable CRD System Design Description.

For scram tests at cold conditions with ambient pressure, the 60% scram insertion time for each CRD from the fully withdrawn position, based on de-energization of scram pilot valve solenoids as time zero, must be less than or equal to the limit specified in the plant Technical Specifications.

For continuous ganged rod motion, the rods shall move together such that the positions of all rods agree with all other rods in each ganged group within the tolerance specified by the CRD System Design Description. A rod block shall be initiated if the rod positions within the rod ganged group disagree by more than this specified tolerance.

5.6 Test Number 6 - Neutron Monitoring System Performance Tests

5.6.1 Test Number 6A – Source Range Neutron Monitoring (SRNM) Performance and Control Rod Sequence

5.6.1.1 Purpose

The purposes of this test are: a) to verify response, calibration and operation of the SRNM instrumentation during fuel loading, control rod withdrawal to achieve criticality, reactor heatup, and power ascension, and b) to confirm the adequacy of the control rod withdrawal sequence.

5.6.1.2 Description

Functional testing of the SRNM will be performed during fuel loading and prior to control rod withdrawal for the initial criticality. The SRNM count rate data will be collected and compared with the minimum count rate and signal-to-noise ratio requirements. As rods are withdrawn, the approach to critical will be monitored using the signals from the SRNMs. During reactor heatup from ambient to rated temperature, the SRNM power level, rod pattern, moderator temperature, bypass valve position, and approximate rate of change of moderator temperature at the time of rod withdrawal will also be recorded each time a group of rod movements is completed.

Each SRNM channel shall be adjusted to ensure the continuity of SRNM signals between the counting flux and Mean Square Voltage (MSV) flux regions while operating in the transition region. During initial reactor heatup, a preliminary calibration will be made by adjusting the SRNM amplifier gains so that each SRNM channel reads equal to or greater than the thermal power value determined using the methodology of bypass valve power calculation. After the first Average Power Range Monitor (APRM) calibration, each SRNM channel gain will again be readjusted, if required, to optimize the SRNM overlap with the APRM instruments.

During control rod manipulation the Rod Worth Minimizer (RWM) will be monitored to ensure the constraints programmed into the RWM to prevent out of sequence withdrawal.

5.6.1.3 Criteria

5.6.1.3.1 Level 1

Each required operable SRNM shall have a count-rate signal with a signal-to-noise ratio of at least 3:1 and a signal count rate of at least 3 counts per second.

Each required operable SRNM shall be adjusted so that a bumpless transfer is achieved when transferring between the counting and MSV flux ranges.

5.6.1.3.2 Level 2

Each SRNM channel shall overlap with the APRMs consistent with the requirements of applicable Neutron Monitoring System design document or the Plant Technical Specifications.

5.6.2 Test Number 6B – LPRM Calibration

5.6.2.1 Purpose

The purpose of this test is to calibrate and verify proper operation of the Local Power Range Monitor (LPRM) Subsystem.

5.6.2.2 Description

Initially, the gain of each LPRM amplifier is adjusted such that a calibration current, as specified by design, will produce an indication of 100. The LPRM channels will then be checked for correct connections during initial reactor heatup.

During all subsequent calibrations, the LPRM channel will be calibrated to make the LPRM readings proportional to the neutron flux in the LPRM 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.

5.6.2.3 Criteria

5.6.2.3.1 Level 1

Not Applicable

5.6.2.3.2 Level 2

Each LRPM reading shall agree with its calibrated value to within ± 10% of it's calibrated value.

5.6.3 Test Number 6C – APRM Calibration

5.6.3.1 Purpose

The purpose of this test is to calibrate and verify operation of the Average Power Range Monitor (APRM) Subsystem of the neutron monitoring system during the Startup Test Program.

5.6.3.2 Description

Initially, the gain of each APRM amplifier is adjusted to maximum. For all subsequent calibrations, each APRM channel reading will be adjusted to be consistent with the results of the core thermal power (CTP) calculation. During normal plant operation, CTP is calculated using a combination of different methodologies (e.g., SRNM Power, Bypass Valve Power and Normal Heat Balance) depending on the reactor mode of operation.

The "SRNM Power" calculation is performed by the SRNM electronics and indicates percent of rated power. Each SRNM channel provides a power indication that can be used for reactor power calculations.

The "Bypass Valve Power" calculation takes advantage of the fact that the reactor steaming rate is linear with reactor thermal power. In turn, given a known reactor pressure, the reactor steaming rate is derived from bypass valve position using a three part curve fit to the "standard" curve from the BWR control system design reports. During heatup, a preliminary calibration will be made by adjusting the APRM gains so that the APRM readings agree with the results of the thermal power value as determined by the average of the bypass valve power and the SRNM power calculations.

The Normal Heat Balance method uses the principle of conservation of mass and energy as a basis for the calculation. The APRMs should be re-calibrated in the power range by a heat balance as soon as feedwater indication is available from low flow or calibrated venturi instrumentation. The APRM gains shall be adjusted so that the APRM readings agree with the results of the heat balance calculation.

5.6.3.3 Criteria

5.6.3.3.1 Level 1

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

5.6.3.3.2 Level 2

At reactor power ≥ 25% of rated, the APRM channels will be considered to be reading accurately if they agree with the heat balance to within ±2% of rated power as specified by the plant Technical Specifications.

5.6.4 Test Number 6D – OPRM Surveillance

5.6.4.1 Purpose

The purposes of this test are: a) to verify operation of the Oscillation Power Range Monitor (OPRM) Subsystem, b) to monitor the neutron flux noise levels in the reactor and to verify that the behavior is within expected limits, and c) to determine if the natural frequency of the reactor is consistent with the T_{min} and T_{max} values specified for the OPRM algorithm.

5.6.4.2 Description

The reactor is designed so that neutron flux oscillations can be reliably detected and suppressed. The OPRM system is designed to provide this reliable detection and provide signals for suppression. Neutron flux oscillations that result from coupled neutronic/thermal hydraulic reactor instability are most likely to occur near the instability region as defined on the Power/Flow operating map with low core flow and relatively high power. The OPRM can initiate a reactor scram when plant operation is within the "OPRM Armed Region" (i.e., the aforementioned instability region).

The OPRM is composed of multiple OPRM cells in each of 4 OPRM channels. Each OPRM cell signal is the average of the signals from 4 closely spaced LPRMs. The OPRM tests the OPRM cell signals by three algorithms to determine if a reactor scram is required. The three algorithms are based on period, amplitude, and growth rate calculations.

The Period Based Detection Algorithm (PBDA) monitors the OPRM signal for periodicity and size. The PBDA produces a Confirmation Count (CC) when the OPRM signal becomes periodic. After establishing a base period, the CC is incremented by one count each half cycle as the periodic signal continues. When the periodic signal is interrupted, the CC is reset to zero. No tuning is required to be performed for the PBDA.

The PBDA produces a reactor scram signal when the consecutive period confirmation (CPC) count reaches or exceeds a user-adjustable count setpoint and the peak of the normalized signal amplitude reaches or exceeds the user-adjustable count setpoint. The amplitude based algorithm initiates a reactor scram based on the peak of the normalized signal. The growth rate algorithm initiates a reactor scram based on the growth rate of the normalized signal. The Level 1 criteria are based on exceeding any of these typical scram setpoints.

Operational data is taken at three different power/flow conditions for this test program to provide an indication of the algorithm performance. The three test points are (1) below 60% of rated core flow and above the 70% rod line, (2) near 80% of rated core flow and near the rated rod line, and (3) near rated flow and near the rated rod line.

The OPRM has the capability to record CC, relative signal, oscillation period, and growth rate data at a 25 msec sample rate for output to an off-line analysis. Approximately 10 minutes of data should be recorded for each test point. Both the LPRM raw data and the OPRM cell data should be collected for each test point for an off-line analysis. The off-line analysis will determine the distribution of oscillation periods at the three test points and confirm that the actual cycle-specific OPRM amplitude and CPC settings are adequate to prevent spurious reactor scram.

5.6.4.3 Criteria

5.6.4.3.1 Level 1

The Confirmation Count shall not exceed 10 and normalized amplitude shall not exceed 1.10 concurrently for any OPRM cell.

A growth rate of greater than 1.30 shall not be detected by the Growth Rate Detection Algorithm (GRA) when core flow is less than 60% of rated.

A normalized amplitude of greater than 1.30 shall not be detected by the Amplitude Based Algorithm (ABA) when core flow is less than 60% of rated.

5.6.4.3.2 Level 2

The Confirmation Count shall not exceed 10 for any OPRM cell.

5.6.5 Test Number 6E – ATIP Performance and Core Power Distribution

5.6.5.1 Purpose

The purposes of this test are: a) to demonstrate proper operation of the Automated Traversing Incore Probe (ATIP) Subsystem, b) to determine the reproducibility of the ATIP subsystem readings, and c) to evaluate symmetry of the core power distribution.

5.6.5.2 Description

The core top and bottom limits for each ATIP channel to start and stop data acquisitions are initially established at cold conditions. During the reactor heatup test plateau, the established core top and bottom limits of each ATIP channel will be adjusted as necessary based on the comparison with fuel channel spacer dip locations after the plant reaches near rated temperature conditions.

At intermediate and higher power levels, several sets of ATIP data will be obtained to determine the total ATIP uncertainty and will be compared with the criterion to assure proper ATIP alignment. ATIP data will be obtained with the reactor operating in an octant symmetric control rod pattern and at steady state conditions.

The total ATIP uncertainty is obtained by analyzing pairs of ATIP readings taken at ATIP locations which are symmetrical about the core diagonal of fuel loading symmetry. The total ATIP uncertainty for the test will be calculated by averaging the total ATIP uncertainty determined from each set of ATIP data taken. The total ATIP uncertainty is made up of random noise and geometric uncertainty components.

The random noise component of the total ATIP uncertainty is obtained by taking repetitive ATIP readings in the common channel, with each of the ATIP machines making a minimum of two and up to six runs. The geometric uncertainty component is obtained by statistically subtracting the random noise component from the total ATIP uncertainty.

Core power distribution data will be obtained during the power ascension phase of the startup test program by using complete sets of axial power traces taken with the ATIP system. Core power symmetry will also be calculated using the ATIP data. Any asymmetry, as determined from this analysis, will be accounted for in the calculation for Minimum Critical Power Ratio (MCPR).

5.6.5.3 Criteria

5.6.5.3.1 Level 1

Not Applicable

5.6.5.3.2 Level 2

The total ATIP uncertainty (including the random noise and geometric components) obtained by averaging the uncertainties for all data sets shall be less than [6.0%].

5.6.5.3.3 Level 3

The ATIP detector hand probed full incore position readings at the cold and hot plant conditions should not differ by more than 2.54 cm (1 inch).

5.7 Test Number 7 – Plant Computer System Operation

5.7.1 Purpose

The purpose of this test is to verify the performance of the plant Performance Monitoring and Control Subsystem (PMCS) under plant operating conditions.

5.7.2 Description

PMCS program verifications and performance calculation program validations are accomplished through implementation of the Static System Test Cases (SSTC) and Dynamic System Test Cases (DSTC). The SSTC is performed at the computer supplier's site and following delivery to the plant site during the preoperational test phase. The DSTC is performed at steady state power conditions beginning with plant heatup and continuing through the ascension to rated power and flow conditions.

Following fuel loading, during reactor heatup and ascension to rated conditions, the nuclear steam supply system (NSSS) and balance-of-plant (BOP) system process variables sensed by the computer will become available. It will be verified that the computer correctly receives, validates, processes and displays the applicable NSSS and BOP process variables. Data manipulation and the results of performance calculations of the nuclear steam supply system and the balance-of-plant will be verified for accuracy. At steady-state power conditions, the dynamic system test case will be performed.

An independent offline computation system(s) will be used as a benchmark to verify plant computer thermal limits calculations.

The following criteria are applicable for the generic GE ABWR Fuel Design. Should a project use a fuel vendor other than GE, that vendor will provide the applicable test criteria.

5.7.3 Criteria

5.7.3.1 Level 1

Not Applicable

5.7.3.2 Level 2

The Full Core TIP Data Processing and Predictor functions will be considered operational when:

1. The Minimum Critical Power Ratio (MCPR) calculated by the independent method and the plant 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 plant computer is in a different assembly than that calculated by the independent method, for each assembly, the MCPR and Critical Power Ratio (CPR) calculated by two methods shall agree within 2%.

2. The Maximum Linear Heat Generation Rate (MLHGR) calculated by the independent method and the plant 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 MLHGR calculated by the plant computer is in a different assembly than that calculated by the independent method, for each assembly, the MLHGR and Linear Heat Generation Rate (LHGR) calculated by two methods shall agree within 2%.

3. The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) calculated by the independent method and the plant 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 plant computer is in a different assembly than that calculated by the independent method, for each assembly, the MAPLHGR and Average Planar Linear Heat Generation Rate (APLHGR) calculated by two methods shall agree within 2%.

4. The LPRM calibration factors calculated by the plant computer and the independent method agree within 2%.

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

5.7.3.3 Level 3

The computer interfaces with the ATIP system shall be considered operable when the computer is demonstrated capable of detecting ATIP positions during ATIP traverses for each channel and when all computer alarms and messages associated with the Full Core TIP Data Processing function have been demonstrated operable.

5.8 Test Number 8 – Core Performance

5.8.1 Purpose

The purpose of this test is to demonstrate that the various core performance characteristics such as power versus flow, core power distribution, and those parameters used to demonstrate compliance with core thermal limits and plant license conditions are in accordance with design limits and expectations.

5.8.2 Description

This test will collect data sufficient to demonstrate that reactor and core performance characteristics remain within design limits and expectations for all operational conditions that the plant is normally expected to encounter. The core performance evaluation is employed to determine the principal thermal and hydraulic parameters associated with core behavior.

During reactor heatup and ascension to rated conditions, the core performance parameters will be collected at various rod patterns, power, and core flow conditions. These parameters include core flow rate, core thermal power level, MLHGR, MAPLHGR, and MCPR.

Plant computer testing as described in section 4.7 must be performed prior to the plant computer being used as a primary means for obtaining core performance parameters (i.e. MLHGR, MAPLHGR and MCPR).

If both the plant computer and the independent method of calculation are not available, the manual calculation techniques available at the site can be used for the core performance evaluation.

The following Criteria are applicable for the generic GE ABWR Fuel Design. Should a project use a fuel vendor other than GE, that vendor will provide the applicable test criteria.

5.8.3 Criteria

5.8.3.1 Level 1

For any non-GE fuel only, the MLHGR of any rod during steady state conditions shall not exceed the limit specified by the plant Technical Specifications.

The MAPLHGR shall not exceed the limit specified by the plant Technical Specifications.

The steady-state MCPR shall exceed the minimum limit specified by the plant Technical Specifications.

Steady-state reactor power shall be limited to the rated core thermal power and the values on or below the minimum of either rated thermal power or the licensed maximum flow control line as shown on the Power/Flow operating map.

Core flow as indicated by operating data shall not exceed its designed rated value.

5.8.3.2 Level 2

Not Applicable

5.9 Test Number 9 - Nuclear Boiler Process Monitoring Tests

5.9.1 Test Number 9A – Selected Process Temperatures

5.9.1.1 Purpose

The purposes of this test are: 1) to verify proper operation of various nuclear boiler process instrumentation, 2) to collect pertinent data from such instrumentation at various plant operating conditions in order to validate design assumptions, and 3) to identify any operational limitations that may exist.

5.9.1.2 Description

During ascension to rated conditions, the bottom head temperature and selected reactor coolant temperatures will be monitored and recorded. Data will be used to confirm the adequacy of the bottom head drain line temperature sensors and identify potential stratification in the reactor bottom head region with all reactor internal pumps in operation. The bottom drain line must have adequate flow through it for the drain line temperature reading to be considered indicative of vessel bottom head temperature.

At steady state power operation with near or rated core flow, the difference between measured bottom head drain line temperature and lower plenum coolant temperature calculated in terms of reactor core inlet subcooling, based on a heat balance, is expected to be less than 5°C. If this difference exceeds 5°C, then an evaluation is required in order to determine if a problem exists.

Monitoring the temperature difference between the reactor bottom head drain and the saturated water temperature inferred from the reactor steam dome pressure during planned RIP trips will determine if temperature stratification occurs in the lower plenum. Following the 1 (or 3) RIP trip tests, i.e., Test 28A (or 28B), the nine (or seven) operating RIPs will be reduced to the allowable speed for restarting the tripped pump(s). Differential temperature between the reactor coolant and the reactor bottom head drain shall be trended during the reduction in pump speed and prior to restart of the tripped pump(s).

After the performance of the loss of turbine/generator and offsite power test (Section 4.30), all RIPs will be tripped and the plant operation will be stabilized at decay heat power level and natural circulation conditions. Reactor coolant and bottom head temperature in the natural circulation mode will be measured until temperature is stabilized or a trend has been established. The RIPs will be restarted before the differential temperature between the reactor coolant saturation temperature and the bottom head drain exceeds the Level 1 criteria (i.e., ≤ 80°C).

5.9.1.3 Criteria

5.9.1.3.1 Level 1

An idle RIP shall not be started unless the temperature difference between the reactor bottom head drain and the saturated water temperature inferred from the reactor steam dome pressure is less than 80°C.

5.9.1.3.2 Level 2

With all ten RIPs in operation at near rated core flow conditions, the difference between bottom head temperature as measured by the bottom drain line thermocouples and the saturated water temperature corresponding to the reactor steam dome pressure shall be less than 28°C.

5.9.1.3.3 Level 3

At steady state power operation, at or near rated core flow condition, the difference between measured RPV bottom head drain line temperature and the calculated bottom head fluid temperature based on heat balance shall be less than 5°C.

5.9.2 Test Number 9B – Water Level Reference Leg Temperature

5.9.2.1 Purpose

The purpose of this test is to measure the reference leg temperature of various water level instruments and to recalibrate the water level instruments if the measured temperature is different from the value assumed during the initial calibration.

5.9.2.2 Description

This test will record wide range level reference column temperatures and the indicator readings on the narrow and wide range level systems at rated temperature and pressure and under major steady-state conditions. The indicated levels will be used to check the calibration of the narrow and wide range level monitoring systems and to define the effects of core flow, subcooling, and carryunder on the wide range levels. The wide range level reference column temperatures will be compared against the value assumed in the initial calibration end points calculation. If the measured value differs from the assumed value, the instruments will be recalibrated using the measured value.

5.9.2.3 Criteria

5.9.2.3.1 Level 1

Not Applicable

5.9.2.3.2 Level 2

The difference between the actual reference leg temperature(s) and the value(s) assumed during initial 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 indicated readings for the Narrow Range Level instruments should agree within ± 3.81 cm (1.5 inches) of the average reading.

The indicated readings for the Wide Range Level instruments should agree within ± 15.2 cm (6 inches) of the average reading.

5.9.3 Test Number 9C – Core Flow Calibration

5.9.3.1 Purpose

The purpose of this test is to evaluate the three methods for measuring core flow: the Pump Deck Differential Pressure (PDdP) method, the Core Plate Differential Pressure (CPdP) method, and the Simplified PDdP method in each operating condition and to calibrate the core flow measurement subsystem.

5.9.3.2 Description

The General Electric Thermal Analysis Basis (GETAB) utilizes statistical analysis to establish the allowable MCPR for protecting the fuel from overheating during design basis transients. An input to the GETAB statistical analysis is the uncertainty in total core flow measurement. Accurate measurement of core flow is also important since it affects fuel reactivity and provides the data for maintaining vessel internal stresses below design bases.

There are 3 different methods for determining reactor core mass flow rate: 1) the CPdP core flow measurement, 2) the PDdP core flow measurement, and 3) the Simplified PDdP core flow measurement. The CPdP core flow is used for safety and control functions (e.g., APRM flow bias trip function and Recirculation Flow Control). PDdP core flow is used for reactor performance and fuel limit calculations by the plant computer system (PCS), and to provide monitoring functions (e.g., various displays in the main control room) for operator information. The PDdP core flow is also used as an input for calibrating the CPdP core flow. The Simplified PDdP core flow is used to provide an estimated value of the core flow when the Neutron Monitoring System (NMS) "Low-Cut" function is activated.

The calculation for core flow using the PDdP method is performed by the PCS using an intricate algorithm. This method uses several plant measurements as well as RIP performance data determined during RIP shop testing. This information is then adjusted for actual reactor conditions to determine a value for core flow.

The CPdP method utilizes a fairly simple algorithm that relates CPdP to core flow. This calculation is performed by the NMS.

The Simplified PDdP calculation is performed in the Recirculation Flow Control System (RFC).

Core flow measurement data will be recorded at several power-flow conditions, and in conjunction with the RIP trip tests (Test 28A and 28B). During the testing program at conditions that achieve rated core flow and rated power, the PDdP method will be used to determine total core flow. Then, the overall adjustment factor and fitting coefficients used in the CPdP algorithm will be determined per engineering procedure and adjusted if necessary to provide core flow indication using the collected core flow measurement data and the results of the PDdP core mass flow rate calculation.

After the CPdP method has been calibrated, the flow biased APRM and Multi-channel Rod Block Monitor (MRBM) systems will be verified and adjusted as necessary. The CPdP core flow rate value will be measured during startup and ascension to the high power plateau in order to confirm the adequacy of the reading above the NMS "Low-Cut" function activation point.

5.9.3.3 Criteria

5.9.3.3.1 Level 1

Not Applicable

5.9.3.3.2 Level 2

After final calibrations are completed, the magnitude of the difference between the CPdP core mass flow rate value and the PDdP core mass flow rate value shall be less than [0.8%] of rated core flow rate at rated power and rated flow conditions.

With 3 RIPs tripped and the remaining 7 RIPs running at minimum speed, the indicated core flow, based on core plate ΔP, shall be greater than the setpoint for the NMS "Low-Cut" function.

The post calibration bottom drain temperature as calculated by the PDdP core flow calculation algorithm shall be within [± 5°C] of the average value of measured reactor bottom drain flow temperature.

The flow-biased APRM and MRBM systems shall be adjusted to function properly at rated conditions.

The recirculation flow control system shall be adjusted to limit the maximum core flow setpoint to [111%] of rated.

5.10 Test Number 10 – System Expansion

5.10.1 Purpose

The purposes of this test are to confirm that: 1) the pipe suspension system is working as designed, 2) the piping is free of obstructions that could constrain free pipe movement caused by thermal expansion, and 3) measured and observed pipe thermal movement is consistent with analysis results and within the acceptable range.

5.10.2 Description

The piping considered to be within the boundary of this test is listed below:

- Main Steam Piping:

  Steam lines between the RPV nozzles and the outboard main steam isolation valves (MSIVs), and steam lines downstream of the outboard MSIVs.

- Relief Valve Discharge Piping:

  The piping attached to the main steam lines and bounded by the SRV discharge flange and the quencher in the wetwell.

- Feedwater Piping:

  The feedwater discharge piping up to the RPV feedwater nozzles.

- RCIC Piping:

  RCIC turbine steam supply and exhaust piping and RCIC pump suction and discharge piping.

- RHR Piping:

  The RHR pumps suction and discharge piping in the shutdown cooling mode of operation.

- CUW

- Piping:

  The CUW pumps suction and discharge piping including head spray line is within the scope of the test.

- RPV Head Vent

  The RPV Head Vent piping attached to the RPV Head and insulation will be monitored during heatup.

- RIP Motor Cooling (RMC) Piping:

  The RMC piping, including RIPs. (note: if a hot functional test has been performed, then the reactor recirculation system RMC piping system thermal expansion test is not required during the power ascension test).

- Piping Inside Drywell:

  All major piping systems inside the drywell including the HPCF and SLC discharge piping.

- Small attached piping:

  All small branch piping attached to those portions of piping within the scope of this test from the large pipe branch connection to the first downstream guide or anchor. (Note: small branch pipes that cannot be monitored due to limited access are excluded from the scope of this test).

- Turbine Building piping:

  All major piping systems inside the Turbine Building that are subject to thermal expansion including the Feedwater Heater Vent and Drain and Auxiliary Steam System piping. These systems and their criteria will be identified in the Turbine Building Piping Test Requirements Specification.

The system expansion test consists of measuring displacements and temperatures of piping systems using installed instrumentation or local measurements during various system and plant operating modes. A visual examination for evidence of obstruction or interference will be performed on the above mentioned system piping inside containment at appropriate hold points during reactor heatup to rated temperature and pressure conditions and again after three heatup and cooldown cycles. In addition, visual observation will be made by a system walkdown at accessible locations to determine acceptability of the piping outside containment under the conditions existing during each specified system test.

Thermal movement and temperature measurements shall be recorded inside the drywell and wetwell on the following piping: Main Steam, selected SRV discharge lines, RCIC steam piping, Feedwater lines, and RHR line B with CUW in service under the following conditions:

a) Ambient temperature (for baseline data);

b) 1.03 MPaG reactor pressure;

c) 4.14 MPaG reactor pressure;

d) Approximately 7.07 MPaG

e) 20-25%, 50%, 75% and 100% of rated thermal power;

f) At points (a) and (c) above during a total of three separate heatup and cooldown cycles (to measure possible shakedown effects).

Thermal movements will also be recorded at appropriate temperature increments up to the required test temperature for the feedwater, CUW and RHR systems piping as each system is placed in service during normal plant operation.

In addition, a special test will be performed to monitor the conditions and effects of temperature stratification that may exist on the feedwater discharge piping inside and outside of containment. This special test will be conducted during heatup, hot standby, post scram, RCIC injection, and reactor shutdown. During the performance of this test, thermal displacements, strains, and temperature measurements will be taken on at least one of the main feedwater headers inside and outside the containment, at selected feedwater riser piping, and at selected feedwater RPV nozzles to measure thermal cycling.

If there are discrepancies between this document and the applicable piping test requirement specification, the latest revision of the latter governs.

5.10.3 Criteria

The thermal expansion acceptance criteria are based upon the actual movements being within a prescribed tolerance of the movements predicted by analysis. Measured movements are not expected to precisely correspond with those mathematically predicted. Therefore, a tolerance is specified for differences between measured and predicted movements. The tolerances are based upon consideration of measurement accuracy, suspension free play, and piping temperature distributions.

If the measured displacement does not vary from the acceptance limit values by more than the specified tolerance, the piping system is responding in a manner consistent with the predictions, and is therefore acceptable. The piping response to test conditions shall be considered acceptable if the review of the test results indicates that the piping responds in a manner consistent with the predictions of the stress report and/or that piping stresses are within ASME Code Section III limits. Acceptable thermal expansion limits are determined after the completion of piping systems stress analysis and will be supplied to site personnel in the applicable piping test requirement specification.

The Level 1 limits are bounding criteria based on ASME Code Section III stress limits. If a Level 1 limit is exceeded, the test shall either be terminated or the plant placed in a satisfactory hold condition below the Level 1 limit. The cognizant engineering organization shall be advised and reconciliation or corrective action taken prior to moving beyond those plant conditions at which thermal expansion has been demonstrated to be acceptable. If practicable, a walkdown of the affected piping and suspension system should be made in an attempt to identify potential obstruction to free piping movement. Hangers and snubbers should be positioned within their expected cold and hot settings. All signs of damage to piping or supports shall be investigated.

Piping may not move smoothly to predicted positions during increase in operating temperature. During the first part of system expansion testing, vessel movements do not always move the pipe in a direction consistent with the stress report predictions. The pipe may advance in a stepwise fashion due to friction constraints. Level 1 criterion will discount spurious movement measurements while assuring safe limits on movement are met. A sufficient number of thermal movement transducers shall be installed on piping within the scope of this test to allow a comparison between similar instruments on the same line, such that failed instrument or spurious readings may be identified and discounted. When pipe movement is obstructed, movement discrepancies occur at multiple locations because of coupling effects. In specific cases, when only one instrument out of several indicated movements is outside the Level 1 criterion, that measurement should be examined and, if deemed spurious, discounted.

Main Steam Prerequisites for thermal limits application:

a) The reactor must be at normal operating temperature, and

b) Thermal transducer must indicate that main steam piping is at normal operating temperature, and

c) Thermal transducer must indicate that the RCIC turbine steam supply line is at normal operating temperature.

Feedwater Piping Prerequisites for thermal limits application:

a) Reactor must be at normal operating temperature.

b) A thermal transducer must indicate that the pipe is approaching operating temperature.

5.10.3.1 Level 1

Visual Inspection Level 1 Criteria:

The spring load indicator shall be within an expanded operable range for variable supports where the operating range has been increased by 5% of the design hot load on the hot side of the operating range, and by 5% of the design cold load on the cold side of the operating range, provided that this extension does not top or bottom out the spring.

Snubbers shall not be fully extended or fully retracted.

Snubber swing clearance shall be maintained.

The constant spring supports shall not be topped or bottomed out.

The thermal expansion measuring instrumentation values shall be in compliance with Level-1 acceptance criteria derived from the design basis thermal analysis.

There shall be no obstruction that is interfering with the expected thermal expansion of the piping system.

Electrical cables and flexible hoses shall accommodate the expected thermal expansion of the piping system.

Instrument and branch piping shall accommodate the expected thermal expansion of the piping system.

There shall be no observed damage to hangers or snubbers.

Piping and insulation shall not be in hard contact with its associated pipe whip restraint.

5.10.3.2 Level 2

The Level 2 limits are more stringent criteria based on the predicted movements using calculated deflections, plus a selected tolerance. If a Level 2 limit is exceeded, the cognizant engineering organization is required to be advised so that investigation of the measurements and of the criteria and calculations used to generate the pipe motion limit can be initiated. If a Level 2 limit is not satisfied, plant operating and startup test plans need not necessarily be altered.

Correlation of Test Data and Analysis:

The predicted movements are based on calculations that are dependent on assumed nozzle movements and temperature distributions. The measured temperatures and pipe or nozzle movement must be compared with those assumed in the analysis to determine which analysis condition corresponds to the test condition. If test conditions do not correspond to those assumed in the analysis, the cognizant pipe design engineer may find it necessary to calculate pipe movements based on measurements and then compare the predicted and actual movement to establish the latter's acceptability.

Visual Inspection Level 2 Criteria:

There shall be no evidence of unexpected restraint to thermal expansion.

There shall be no obvious indications of damage caused by excessive line vibration.

The spring load indicator shall be between the cold load and hot load marks on the load scale for variable supports.

The calculated cold to hot snubber travel shall be less than or equal to its design cold to hot travel, and both the installed cold position and operating hot position are at least 13 mm or 20% of total stroke (whichever is smaller) from being topped or bottomed out.

The constant spring supports shall be within the operating range, between the cold and hot positions, on the travel range.

The gap between the insulated pipe and its associated pipe whip restraint shall be at least 6.4 mm.

The gap between piping and structural components shall be at least 6.4 mm.

5.11 Test Number 11 – System Vibration

5.11.1 Purpose

The purposes of this test are: 1) to verify that the vibration of critical system piping and components is within acceptable limits during normal steady-state power operation, and 2) to verify during expected operational transient loading that pipe stresses are within code limits.

5.11.2 Description

This test is an extension of Section 4.10, System Expansion, and the preoperational vibration tests. Consult Section 4.10 of this specification for piping considered to be within the scope of this testing. System vibration testing during the startup test program shall be limited to those systems that could not be adequately tested during the preoperational phase. The predicted steady state and transient vibration displacements for each monitored location will be provided in the applicable piping test requirement specification.

The remote measurements shall be regularly checked to verify compliance with acceptance criteria. If trends indicate that criteria may be violated, the measurements should be monitored at more frequent intervals. The test shall be held or terminated as soon as criteria are violated. After the test hold or termination, appropriate investigative and corrective actions shall be taken as soon as possible. If practicable, a walkdown of the piping and suspension system should be made in an attempt to identify potential obstructions or improperly operating suspension components. Hangers and snubbers should be positioned such that they can accommodate the expected deflections without bottoming out or extending fully. All signs of damage to piping supports or anchors shall be investigated. In addition, remote instrumentation indicating criteria failures shall be checked for proper operation and calibration.

Because of limited access due to high radiation levels, no visual observation is required on the main steam, RMC, RHR suction and discharge, CUW suction and discharge, and RCIC turbine steam supply and exhaust piping inside containment during the startup phase of the testing. Remote measurement of piping vibrations shall be made during the following steady-state conditions:

a) Main steam piping downstream of outboard MSIV during plant shutdown with the main turbine stop valves closed

b) Main steam and feedwater piping at 25% ± 5% of rated steam flow

c) Main steam and feedwater piping at 50% ± 5% of rated steam flow

d) Main steam and feedwater piping at 75% ± 5% of rated steam flow

e) Main steam, feedwater, and SRV discharge piping at 100% of rated electrical power

f) RCIC turbine steam supply flow at 100% of rated

g) RHR suction and discharge piping at 100% of rated flow in shutdown cooling mode

h) CUW suction piping at 100% of rated flow and operating temperature

i) For feedwater piping only, when the plant is in hot standby condition following a reactor scram, feedwater flow at 100 % of rated and operating temperature, and during plant startup when hot CUW enters the cold feedwater line

j) Turbine Building Piping as identified in the BOP Piping Test Requirements document

With the system as near as obtainable to normal operating temperatures and flows, system walkdowns and local vibration measurements at each accessible location shall be conducted to look for excessive vibrations. Specific attention shall be given to small attached piping and instrument connections in order to ensure their movements are within acceptable limits.

Finally, remote vibration and strain measurements of appropriate piping subsystems will be taken during the following plant transient tests:

a) Main turbine control or stop valve closure at 75% and 100% of rated flow (Test 31)

b) Manual discharge of each safety relief valve at ≥ 6.55 MPaG or after synchronization within LP Test Plateau (Test 25) and at planned transient tests that result in safety relief valve discharge

c) Feedwater pump trip (Test 27)

d) RIP trip and recovery at 75% and 100% of rated flow (Tests 28A and 28B)

e) Reactor full isolation test at 100% power (Test 32)

f) RCIC pump start and trip (Test 22)

g) CUW pump start and trip (Test 21)

h) RHR pump start and trip in shutdown cooling mode (Test 19)

i) Plant at hot standby condition following planned or unplanned reactor scrams.

During the operating transient load testing, the amplitude of displacement of the main steam, safety relief valve discharge and feedwater discharge piping will be measured, and the displacements compared with acceptance criteria. The deflections are correlated with stresses to verify that the pipe stresses remain within Code limits. System walkdowns and local vibration measurements (where practical) shall be made during and after the transients described above to look for excessive vibration or signs of damage.

If there are discrepancies between this document and the Piping Test Requirement Specifications the latest revision of the latter governs.

5.11.3 Criteria

The piping response to test conditions shall be considered acceptable, if the review of test results indicates that the piping responds in a manner consistent with predictions of the stress report and/or that piping stresses are within ASME Code Section III limits. Acceptable limits are determined after the completion of piping systems stress analysis and will be supplied to site personnel in the applicable piping test requirement specification.

For steady-state and transient vibration, the pertinent acceptance criteria are usually expressed in terms of maximum allowable displacement/deflection. Visual observation should only be used to confirm the absence of significant levels of vibration and not to determine acceptability of any potentially excessive vibration.

There are typically two levels of acceptance criteria for allowable displacements and deflections. Level 1 criteria are bounding type criteria associated with safety limits, while Level 2 criteria are stricter criteria associated with system or component expectations.

5.11.3.1 Level 1

a) Steady-State Vibration:

   Level 1 limits on steady-state operational vibration displacements and strains are based upon keeping piping stresses and pipe mounted equipment accelerations within safe limits and ASME code limits. If any one of the remote sensors indicate that the prescribed limits are exceeded, the test shall be placed on hold.

b) Transient Vibration:

   Level 1 limits on transient vibration displacements are based on either the ASME Code Section III upset primary stress limits or the applicable snubber load capacity to keep the loads on piping and suspension components within safe limits. If any one of the transducers indicates that the prescribed limits are exceeded, the test shall be placed on hold.

5.11.3.2 Level 2

a) Steady-state Vibration:

The evaluation criteria of acceptable levels of steady-state vibration take two forms, i.e., limits on vibratory displacement along with limits on acceleration. The limits are set based on one half the endurance limit and consideration of analysis, operating experience and protection of pipe mounted components.

b) Transient Vibration:

Acceptable transient vibration limits are based on a given tolerance about the expected deflection value. Transducers have been placed near points of maximum anticipated movement. Where movement values have been predicted, tolerances are prescribed for differences between measurements and predictions. Tolerances are based on instrument accuracy and suspension free play. Where no movements have been prescribed, limits on displacements will be prescribed.

5.12 Test Number 12 - Recirculation Flow Control System Tests

5.12.1.1 Test Number 12A – Individual RIP Speed Control

5.12.1.1.1 Purpose

The purposes of this test are: a) to demonstrate the capability of the Recirculation Flow Control System (RFC) with all RIPs operating in the individual speed control mode and b) to demonstrate that all controllers are set for desired system performance and stability.

5.12.1.1.2 Description

The testing of the RFC follows an "inner loop" to "outer loop" testing approach while the plant is ascending from low to high power levels. Testing of the inner RIP Adjustable Speed Drive (ASD) frequency control loop is conducted first, followed by testing of the gang speed control and then core flow control loops of the RFC. Finally, the automatic load following (ALF) control by core flow control is tested.

Prior to startup testing, the step response performance of each of the individual RIP speed controllers is confirmed as part of the RIP ASD Factory Acceptance Tests and as part of the preoperational testing. At mid and high power plateaus, the response of the system to small frequency demand step input changes will be measured and evaluated.

5.12.1.2 4.12.1.3 Criteria

5.12.1.2.1 Level 1

The transient response of any recirculation system-related variable to any test input must not diverge.

5.12.1.2.2 Level 2

The decay ratio of RIP ASD output frequency response to any test inputs shall be ≤ 0.25.

Maximum time of RIP ASD output frequency response to 10% of a step (≤5% of rated frequency) disturbance shall be less than or equal to 0.15 seconds.

Settling time (to within ± 5% of step change) of RIP response to a step (≤5% of rated frequency) disturbance shall be less than or equal to 2.5 seconds

Maximum time of RIP ASD output frequency response from 10% to 90% of a step (≤5% of rated frequency) disturbance shall be less than or equal to 0.9 seconds.

Maximum allowable overshoot of RIP ASD output frequency response to a step disturbance with magnitude of 5% of rated shall be ≤ 5% of the demand step in increasing direction and ≤ 15.0% of the demand step in decreasing direction.

Maximum rate of a single RIP response to a large (≥ 10% of rated frequency) demand step shall be 5.0% of rated frequency per second in either decreasing or increasing direction.

5.12.1.2.3 Level 3

Frequency loop deadband shall be ≤ 0.1%.

Note: Performing tests near the high and low ends of the specified range is acceptable for verifying step input response.

5.12.2 Test Number 12B – Recirculation Flow Control

5.12.2.1 Purpose

The purposes of this test are: a) to demonstrate the RFC's control capability over the entire flow control range, including gang speed control, core flow control and load following modes of operation across the span of expected operational conditions, b) to determine that all controllers are set for desired system performance and stability, c) to assure that the RFC can meet its dynamic performance requirements, and d) to confirm the stability of the RFC in each of the operating modes.

5.12.2.2 Description

The gang speed controller step response performance should be tested as part of the preoperational testing. However, the RFC core flow control response and load control response cannot be confirmed until startup testing is performed. Following the initial individual speed control mode tests of Section 4.12.1, the preliminary values of core flow and load control loop adjustments will be made on the mid power line. This will be the most extensive testing of the RFC. Then, the response of the system will be measured and evaluated, and the final system controller settings will be adjusted, if necessary, as part of the high power plateau testing.

During mid and high power plateau testing, any needed adjustments to the system controller settings should be made prior to performing the formal step response startup testing. In addition, the core flow control algorithm of the neutron monitoring system should be calibrated based upon the available startup data collected prior to performing the formal core flow control response and load control response testing.

Finally, with all 10 RIPs operating in Gang Speed Control mode at rated flow and rated power, a controller switchover fault on the operating RIP ASD controller will be introduced while the other RIP ASD controller is in the standby condition. The successful switchover, restraint of the associated RIP speed drop and recovery to the initial operating conditions shall be observed and recorded during this test.

5.12.2.3 Criteria

5.12.2.3.1 Gang Speed Control Loop Criteria

5.12.2.3.1.1 Level 1

The transient response of any recirculation system-related variable to any test input must not diverge.

5.12.2.3.1.2 Level 2

The decay ratio of RIP frequency response to any test inputs shall be ≤ 0.25.

Maximum rate of a single RIP response to a small (≤5% of rated frequency) demand step shall be 5.0% of rated frequency per second in either increasing or decreasing direction.

Maximum rate of a single RIP response to a larger (≥10% of rated frequency) demand step shall be 5.0% of rated frequency per second in either increasing or decreasing direction.

5.12.2.3.2 Core Flow Control Loop Criteria

5.12.2.3.2.1 Level 1

The core flow control loop response to test inputs shall not diverge.

5.12.2.3.2.2 Level 2

The decay ratio of the core flow control loop response shall be ≤ 0.25.

Maximum time of core flow response to 10% of a step (i.e., a step size of ≤3% of rated core flow) disturbance shall be less than 0.5 seconds.

Maximum time of core flow response from 10% to 90% of a step (i.e., a step size of ≤3% of rated core flow) disturbance shall be less than 4.5 seconds.

Maximum allowable overshoot of core flow response to a step disturbance with magnitude of 3% of rated shall be less than 6% of the demand step in increasing and decreasing direction.

Settling time (to within ± 5% of step change) of core flow response to a step (i.e., a step size of ≤3% of rated core flow) disturbance shall be less than 6.0 seconds.

5.12.2.3.2.3 Level 3

Core flow control loop upper limit should be checked for proper setting.

5.12.2.3.3 Load Following Loop Criteria

5.12.2.3.3.1 Level 1

The load following loop response to test inputs shall not diverge.

5.12.2.3.3.2 Level 2

The decay ratio of the load following response shall be ≤ 0.25.

Maximum time of the measured steam flow response to 90% of a load demand change (i.e., step magnitude of 10% of rated or less) shall be less than or equal to 10 seconds in 65% to 100% power operating range on rated rod line.

The load following loop response shall produce steady state RIP frequency demand ringback and core flow variations (i.e., after the initial transient response from a step disturbance) no larger than ± 3% of rated value.

5.12.2.3.4 Scram Avoidance and General Criteria

5.12.2.3.4.1 Level 1

Not Applicable

5.12.2.3.4.2 Level 2

For any one of the above recirculation flow control system loop test or maneuvers, the peak neutron flux and simulated heat flux shall remain below the scram settings by 7.5% and 5.0%, respectively.

The switchover to the standby controller shall be successfully accomplished without resulting in reactor scram and an RIP ASD trip. The RIP ASD output frequency shall be returned to normal operating conditions that existed before the switchover without operator action.

5.13 Test Number 13 – Feedwater Control System

5.13.1 Purpose

The purpose of this test is to demonstrate that the stability and response characteristics of the Feedwater Control System (FWC) are in accordance with the design performance requirements for applicable configurations and operating conditions.

5.13.2 Description

FWC provides the logic for controlling the supply of feedwater flow to the reactor vessel in response to automatic or operator manual control signals. FWC testing will begin during plant heatup and continue up through the normal full power line up. Testing will be accomplished by manual manipulation of FWC controllers and/or by direct input of demand changes at various levels of control. Tests of the FWC algorithms which place reactor feedwater pumps in and out of service, and check interaction between the combined low flow control valve (LFCV) and CUW blowdown valve, will also be included.

Power ascension phase testing of the FWC consists of both open loop and closed loop testing at each specified test condition. During open loop testing, the Feedwater Pump Drivers, LFCV and CUW blowdown valve dynamic responses are verified and adjustment made, if necessary. During closed loop testing, the dynamic response of each system controller to small demand step disturbances is verified by comparing against the acceptable performance requirements. Small (i.e., a step size of ≤ 15cm) reactor water level setpoint changes are used to evaluate the FWC settings for all power and system operating modes. The level setpoint changes also demonstrate core stability to subcooling changes.

Additionally, steady-state gain curve data will be collected during ascension to rated power for verifying the linearity of the main actuators in controlling reactor water level. Linearization/smoothing of the curve will be done as necessary together with the master level controller optimization.

5.13.3 Criteria

5.13.3.1 Level 1

The transient responses of feedwater flow and reactor vessel level to any test input must not diverge.

5.13.3.2 Level 2

Level control system-related variables may contain oscillatory modes of response. In these cases, the decay ratio for the response of feedwater flow (and reactor vessel water level) to any test input must be less than or equal to 0.25.

The open loop dynamic response of main FW actuators, LFCV and CUW blowdown valve to a small valve position demand step (greater than 5% and less than or equal to 10%) disturbance shall be:

a) Maximum time to 10% of a step disturbance	≤ 1.1 sec
b) Maximum time from 10% to 90% of a step disturbance	≤ 1.9 sec
c) Peak overshoot (% of step disturbance)	≤ 15.0 %
d) Settling time, 100% ± 5% of step disturbance	≤ 14.0 sec

The average rate of response of the FW actuator to large (> 20% of rated pump flow) step disturbances, shall be between 10% and 25% of rated pump flow per second. Rated pump flow is equivalent to the capacity of a single feedwater pump. This average response will be assessed by determining the time required to pass linearly through the 10% and 90% response points.

The close loop dynamic response of the master level controller to small (≤ 15 cm) step disturbances shall be:

a) Maximum time to 10% of a step disturbance	≤ 3.7 sec
b) Maximum time from 10% to 90% of a step disturbance	≤ 12.7 sec
c) Peak overshoot (% of step disturbance)	≤ 12.6%
d) Settling time, 100% ± 5% of step disturbance	≤ 73 sec

5.13.3.3 Level 3

The dynamic response of each individual level and feedwater flow sensor, i.e., bandwidth, must be faster than 4 Hz. The dynamic response of each steam flow sensor must be faster than 2 Hz.*

Reactor water level, feedwater flow, and steam flow sensors must be installed with sufficiently short lines and proper damping adjustment so that no resonance exists. *

Initial setting of the function generators should be based on vendor supplied or previous operating data. The function generators must be adjusted so that the change in slope, i.e., actual fluid flow changes divided by demand change for small disturbances, shall not exceed a factor of 2 to 1 (i.e., maximum slope versus minimum slope) over the entire 20% to 100% feedwater flow range. Also, function generators should be used to minimize the differences between FW actuators (i.e., pumps and/or valves).

All auxiliary controls which have direct impact on reactor level and FW control, e.g., the minimum flow recirculation valves, should be functional, responsive and stable. The minimum flow recirculation valves should be slow acting such that they do not couple with the FWC dynamics. In the case of a modulating valve with a proportional integrated derivative (PID) controller, the time constant should be 20 seconds or greater. The minimum flow recirculation valves should be fast enough to avoid pump trips and yet slower than the combined LFCV to avoid possible reactor flux scram due to a cold water slug. [1]

5.14 Test Number 14 – Pressure Control System

5.14.1 Purpose

The purposes of this test are: a) to determine the optimum settings for the pressure control loop and to demonstrate the adequacy of the pressure control system in maintaining a stable and fast pressure response to pressure transients, b) to demonstrate smooth pressure control transition between the turbine control valves and turbine bypass valves when the reactor steam generation exceeds the steam flow used by the turbine, and c) to demonstrate that other affected parameters are within acceptable limits during pressure regulator induced transient maneuvers.

5.14.2 Description

Preliminary component testing consists of checking pressure regulator gains, valve response timing, and frequency response characteristics of the pressure control system dynamics including lead/lag network and steam line resonance compensator.

During the startup test program, pressure setpoint changes will be made in order to determine the optimum settings which provide the satisfactory stability and response at all power levels. At specified test conditions, the load limit setpoint will be adjusted so that control valves and/or bypass valves control pressure. The pressure setpoint will be decreased and then increased rapidly by steps up to 69 kPa (10 psi) and the response of the system will be measured in each case. The response of the system will be measured and evaluated and the pressure regulator settings will be optimized.

Steady-state data will also be collected during ascension to 100% power and used to verify adequate control system linearity (i.e., incremental regulation) over the entire control range. The linearity must meet the applicable requirements and, if not, the function generator curves of the turbine control valves will need to be redone.

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[1] These criteria should be verified prior to power ascension testing

5.14.3 Criteria

5.14.3.1 Level 1

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

5.14.3.2 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 pressure response time from initiation of pressure setpoint change to the pressure regulator sensed pressure peak shall be ≤ 10 seconds.

Pressure control system deadband, delay, etc., shall be small enough that steady state limit cycles (if any) shall produce reactor steam flow variations no larger than ± 1.0% of rated steam flow.

For all pressure regulator transients, the peak neutron flux and/or peak reactor pressure shall remain below the scram settings by 7.5% and 69 kPa (10 psi) respectively. Peak heat flux must remain at least 5.0% below its scram trip point.

The variation in incremental regulation (ratio of the maximum to the minimum value of the quantity, "incremental change in steam flow demand/incremental change in steam flow"), shall not exceed one of the following limits for each of the specified flow ranges:

a) For Partial Arc Machines

% Valve-Wide-Open Flow	Variation in Incremental Regulation
0 to 90	≤ 4 : 1
90 to 97	≤ 2 : 1 (overall)
90 to 99	≤ 5 : 1 (any interval)

b) For Full Arc Machines

% Valve-Wide-Open Flow	Variation in Incremental Regulation
0 to 85	≤ 4 : 1
85 to 97	≤ 2 : 1 (overall)
85 to 99	≤ 5 : 1 (any interval)

5.14.3.3 Level 3

The pressure regulation deadband from the pressure sensor to the pressure regulation demand and to the steam bypass demand inclusive shall be less than or equal to 0.69 kPa (0.1 psi). The deadband from pressure regulation demand to control valve steam flow shall be no greater than ± 0.1 percent rated steam flow. The deadband from bypass steam flow demand to bypass valve motion shall be within ± 0.1 percent rated bypass steam flow demand.

5.15 Test Number 15 – Plant Automation and Control

5.15.1 Purpose

To verify that the Power Generation Control System (PGCS) will operate properly with automatic power regulator (APR) to support automation of the normal plant startup, shutdown and power range operations.

5.15.2 Description

PGCS is designed as an operator aid for normal plant startup, power range operation, and plant shutdown. As a top level controller, the PGCS issues task demand signals to lower level controllers for initiation of specified systems modes or equipment operation. The lower level controllers accept these task demands, perform specified control actions based on allowances of system/component interlocks, and send task completion signals back to PGCS to allow it to manage the startup, power range operation, and shutdown sequences. During the performance of this testing, the system will be monitored to ensure that unexplained control functions do not occur.

During ascension to rated conditions, testing will be performed to demonstrate that the PGCS, APR and the individual systems interface correctly. The Integrated Operating Procedure that controls each step in the process will be used and evaluated during this evolution. Initial testing will be performed on an individual system basis to assure that the prescribed automated sequence of system operation is ready to support the higher level of automation. After each individual system has been successfully tested, the next level of automation will be implemented until a final demonstration in the full automatic mode is performed.

Performance parameters for the total system and its components will be measured during each stage of testing. The capability of the PGCS and APR to manage the progression of the normal plant startup, power range operation, and shutdown sequences is demonstrated by verifying that each controlled parameter is within limits specified. The software setting values used in various software control algorithms designed to perform the PGCS, APR and individual system automated functions will be optimized during the performance of this testing.

Additionally, as part of the power range operation, the load control response will be demonstrated to ensure that the electrical load changing, maneuvering, and automatic load following capabilities are in accordance with the contractual requirements.

5.15.3 Criteria

5.15.3.1 Level 1

Not Applicable

5.15.3.2 Level 2

The PGCS shall provide supervisory control commands to individual non-safety control systems to support normal plant startup operation from the cold or hot shutdown condition.

The PGCS shall provide supervisory control commands to individual non-safety control systems to support normal plant power range operation.

The PGCS shall provide supervisory control commands to individual non-safety control systems to support plant shutdown to cold or hot conditions from any power level.

The PGCS shall provide sufficient control system interface and indication to plant operators during the progression of normal plant startup, power range operation, and shutdown.

The PGCS shall provide sufficient prompts in the prescribed sequence identifying manual functions that must be performed by plant operators to support normal plant startup, power range operation, and shutdown.

The PGCS shall provide sufficient breakpoints or pauses to allow the operator to evaluate the overall plant and system status during the management of the progression of normal plant startup, power range, and shutdown operation.

The plant shall respond to an instantaneous load demand decrease of 10% of guaranteed output without the use of steam dump or bypass systems within a range of 100% to 35% of rated power.

The plant shall pick up a rapid load change of 10% of guaranteed output within an operating range of 25 to 90% of rated power in accordance with specification.

From an initial power between 50 and 100% of rated, the plant shall respond automatically to change in power demand from the system load dispatcher of ± 5% of rated electricity output at a minimum rate of 5 MWe/min as designed.

In terms of power change capability for automatic frequency control, the plant shall be capable of satisfying peak-to-peak power change demands of 10 percent of rated power at 5 percent rated power per minute while performing ramp power changes required for load following within the operating range of 65 to 100 percent power.

The plant shall be able to operate for a 24-hour load cycle with the following profile: starting at 100% of rated power, power ramps down to 50% power in two hours, power remains at 50% for two to ten hours, and then ramps up to 100% in two hours. Power remains at 100% of rated for the remainder of the 24-hour cycle.

For any one of the above load changing and maneuvering, the reactor shall not be tripped, the primary and/or secondary relief valves shall not be opened, and the reactor-turbine combination with their associated systems shall not be damaged as a result of any of the operating conditions.

Under normal conditions, no single equipment failure shall result in the disturbance of plant operation (i.e., the plant shall be operated without the PGCS functions).

No single equipment failure shall cause the inadvertent setpoint change and switch over to the system operation mode.

5.16 Test Number 16 – Reactor Recirculation System Performance

5.16.1 Purpose

The purpose of this test is to verify that reactor recirculation system steady-state performance characteristics are in accordance with the design requirements.

5.16.2 Description

The reactor recirculation system features an arrangement of ten variable-speed reactor internal pumps commonly referred to as RIPs. The RIPs are mounted vertically with their shafts penetrating the reactor vessel through pump nozzles arranged in an equally-spaced ring pattern on the bottom head. The RIPs provide forced circulation of the reactor coolant through the lower plenum of the reactor and up through the lower grid, the reactor core, steam separators, and back down the downcomer annulus.

Ganged RIP steady-state performance testing with 9 and 10 RIPs in service will be conducted at several power-flow conditions during the Startup Test Program. Pertinent recirculation system and system-related performance parameters will be recorded beginning with minimum pump speed and continuing through various pump speeds up to the attainable maximum possible speed (MPS) for 9 and 10 pump operation.

Parameters to be evaluated shall at least include RIP speeds, pump deck and core plate differential pressures, maximum core flow capability, and any number of other variables that may indicate operating conditions of the RIPs and their shafts, motors, or heat exchangers.

Data shall also be taken and evaluated during the planned plant startup transient tests such as recirculation pump trip and restart tests (Section 4.28), and during the off normal operation tests such as turbine trip and load rejection test (Section 4.31), full reactor isolation test (Section 4.32), loss of turbine/generator and offsite power test (Section 4.30), and feedwater pump trip test (Section 4.27).

5.16.3 Criteria

5.16.3.1 Level 1

Not Applicable

5.16.3.2 Level 2

At rated power and flow, the measured efficiency for each RIP shall meet or exceed the value as stated for the applicable Reactor Recirculation System Design.

At rated power and flow, each individual RIP shall be capable of providing the flow and head as stated on the Reactor Recirculation System Process Flow Diagram.

The Recirculation Flow Control System (RFCS) shall provide controls to limit each RIP pump speed to a maximum speed consistent with the reactor operating conditions and nuclear safety operational analysis requirements (Subsection 15.4.5).

With all 10 RIPs in service, the RIPs performing as a group shall provide a minimum core flow at the flow rate and system operating conditions as defined in the System Design Description.

With one RIP out-of-service, the RIPs performing as a group shall provide 100% core flow at least at the flow rate and system-operating conditions as defined in the System Design Description.

The measured core pressure drop shall not exceed the predicted value by [4.1 KPaD] using beginning-of-life values at rated power and flow.

5.17 Test Number 17 – Feedwater System Performance

5.17.1 Purpose

The purposes of this test are: 1) to verify that the overall feedwater system (FW) performance characteristics are in accordance with the design requirements, 2) to calibrate the feedwater flow, and 3) to determine if the maximum feedwater runout capability is compatible with the licensing requirements.

5.17.2 Description

Feedwater (FW) and Condensate System parameters will be recorded across the spectrum of system flow and plant operating conditions. Parameters to be recorded include temperatures, pressures, flow rates, pressure drops, pump speeds, developed heads, and general equipment status. Measured data will be compared against the expected values to ensure proper instrument calibration and compliance with design requirements. Steady-state and transient testing shall be conducted, as necessary, to assure that adequate margins exist between system variables and setpoints of instruments monitoring these variables to prevent spurious actuations or losses of system pumps and motor-operated valves. Prior to test commencement, all instrumentation used for the data collection will be verified to be within the required calibration frequency and accuracy.

This data will be used to calibrate the feedwater pump driver controller and to verify that the maximum feedwater flows do not exceed the value assumed in the transient performance and safety analysis calculations.

The feedwater pump controller calibration is done by first obtaining vendor pump performance curves. The pump performance curves are then used to determine the pump speed corresponding to the maximum allowable flow at rated reactor vessel pressure. The FW pump speed which produces no net flow to the vessel with the FW header at 6 MPaG (i.e., estimated Low Turbine Inlet Pressure MSIV isolation setpoint) will also be determined. Adjustable equipment (i.e., feed water pump control loops, mechanical limiters, feedwater control system function generator, etc.) are set to prevent the feedwater pumps from exceeding their maximum allowed output, and yet allow desirable performance.

During the data collection and verification portion of the test, pressure, flow and controller data will be collected between 60% and 100% power. The measured maximum flow will be adjusted to the transient design pressures as identified in the FSAR Chapter 15. The maximum flows stated in the FSAR are used as licensing assumptions, and therefore, should not be exceeded.

If FSAR maximum flows are exceeded the system can either be adjusted so that the licensing assumption is not exceeded or an additional penalty can be applied to the ΔCPR. The ΔCPR can be revised by applying a 0.01 adder for each five percent of rated feedwater flow difference between the determined actual maximum flow and the FSAR maximum flow.

5.17.3 Criteria

5.17.3.1 Level 1

Maximum speed attained shall not exceed the speeds which will give the following Nuclear Boiler Rated (NBR) Feedwater Flows with the normal complement of pumps operating. The change of Feedwater flow below the pressure specified shall not exceed the sensitivity value stated in the FSAR.

a) 130% at 7.35 MPaG (1065 psig)

b) 130% + 0.029 (7350 – P_{actual}) at P_{actual} kPaG

5.17.3.2 Level 2

The maximum speed must be greater than the calculated speeds required to supply the following NBR Feedwater Flows:

a) 115% at 7.35 MPaG (1065 psig) with rated complement of pumps

b) 75% at 7.35 MPaG (1065 psig) with one feedwater pump

5.18 Test Number 18 - Main Steam System Performance Tests

5.18.1 Test Number 18A – Main Steam System Performance Data

5.18.1.1 Purpose

The purpose of this test is to verify that the main steam system related performance characteristics are in accordance with design requirements.

5.18.1.2 Description

Beginning at approximately 40% core thermal power, pertinent plant data such as reactor coolant temperatures, reactor steam dome pressure, pressures downstream of the outboard MSIVs, main steamline header pressure, and main steamline flows will be taken along the 75% rod line at various power levels.

The same data collection process will be repeated along the 100% rod line. The accumulated data will be used to demonstrate that the main steam system operation is within the design requirements.

During the analysis, the pressure drops, from the reactor vessel steam space to the pressure tap downstream of the outboard MSIVs and from the pressure tap downstream of the outboard MSIVs to the pressure tap at the main steamline header, will be evaluated against acceptance criteria while reactor vessel pressure is at the design rated value. The evaluation is accomplished by using temporary instruments measuring steam delivery pressure at the pressure tap downstream of the outboard MSIVs.

5.18.1.3 Criteria

5.18.1.3.1 Level 1

Not Applicable

5.18.1.3.2 Level 2

The maximum allowable pressure drop from the reactor vessel steam space to the pressure taps downstream of the outboard MSIVs shall be less than 276 kPa (40 psi) for 100% NBR steam flow.

The maximum allowable pressure drop from the pressure taps downstream of outboard MSIVs turbine inlet shall be less than 103 kPa (15 psi) for 100% of NBR steam flow.

5.18.2 Test Number 18B – Main Steamline Flow Calibration

5.18.2.1 Purpose

The purpose of this test is to calibrate the main steamline flow devices in the main steam outlet nozzles at selected power levels over the entire core flow range.

5.18.2.2 Description

Beginning at approximately 40% core thermal power, pertinent plant data such as reactor steam dome pressure, pressures at the throat of steam outlet nozzles, CRD flow, main steam and feedwater flows will be taken along the 75% rod line at various power levels. The same data collection process will be repeated along the 100% rod line. The accumulated data will be compared against the design data and a known flow source to verify that acceptable steam flow measurements have been made.

5.18.2.3 Criteria

5.18.2.3.1 Level 1

Not Applicable

5.18.2.3.2 Level 2

The dP measured from the pressure taps at the throat of the steam outlet nozzles shall be equal to or greater than [24.82 kPa (3.6 psi)] at 100% of NBR flow.

The difference of the total steam flow measurement obtained from the main steam line flow nozzles relative to the calibrated total feedwater flow measurement shall be within [± 4] percent of rated flow at feedwater flow rates between 25 and 100 % of rated.

5.19 Test Number 19 – Residual Heat Removal System Performance

5.19.1 Purpose

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

5.19.2 Description

Startup phase testing of the RHR system is intended to demonstrate the capabilities of the system beyond what was possible during the preoperational test phase of the startup test program due to insufficient temperature and pressure conditions.

During the first suitable reactor cooldown, the Shutdown Cooling (SDC) Mode of the RHR system will be demonstrated. Pertinent system parameters will be monitored to confirm that overall system operation and heat removal capabilities are in accordance with the design requirements. Data will be collected with flow rates and temperatures as near to the process flow diagram values as possible.

Normally, the decay heat load during the startup test period is expected to be less than design conditions due to low core exposure. Use of SDC mode with low core exposure could result in exceeding the 56°C/hr cooldown rate of the vessel if 2 RHR heat exchangers are used simultaneously. Testing may need to be delayed to late in the test program (e.g. Test Condition MP or HP) after accumulating sufficient core exposure.

The RHR heat exchangers will also be tested in the suppression pool cooling mode. This test must be coordinated with a test that adds heat to the suppression pool (e.g., Test 25) to be able to obtain adequate data to perform the evaluation.

5.19.3 Criteria

5.19.3.1 Level 1

Not Applicable

5.19.3.2 Level 2[2]

The RHR system shall be capable of operating in the suppression pool cooling and shutdown-cooling modes (with one or two heat exchangers) at heat exchanger heat removal rates equivalent to or greater than the values indicated on the Process Flow Diagrams [Ref. 2.1.2.15].

5.20 Test Number 20 – Guarantee Plant Performance Warranty Run

5.20.1 Purpose

The purpose of this test is to verify that the ABWR unit satisfies the performance warranty parameters as described in the contract.

5.20.2 Description

The warranty demonstration consists of recording sufficient data under steady-state conditions to determine the reactor power level, the pressure, and quality of the steam, and the steam flow rate from the reactor.

[2] If the available decay heat will not support data collection for the RHR Heat Exchanger performance in the shutdown cooling mode, then the heat capacity may be inferred from data taken in the Suppression Pool Cooling mode; provided that the data taken was taken with the system as close as possible to the flows and temperatures indicated in the RHR Process Flow diagram.

The measurements will include: temperature and flow rate of feedwater flow entering the reactor, the calculated energy added to the reactor water by the RIPs, the flow rate through and the temperature entering and leaving the reactor water cleanup system (CUW), the flow rate and temperature of CRD and RIP purge water, the moisture content of the steam and the steam pressure outside the drywell and near the outboard MSIVs.

This test will require a one hundred (100) hour demonstration at the warranty thermal output. During the 100 hours, two 8 hours data collection runs at rated output will be performed to record a number of precise measurements and verify plant compliance with the performance warranty values.

Data acquisition begins after plant operation has been adjusted to the rated power conditions as defined in the contract and as given in Section 4.20.3.2 below. These operating conditions will be maintained stable during the one hundred hours period. Successful completion of this run confirms that the plant can be operated continuously in a safe manner in conformance with the applicable safety regulation and license restrictions.

During the 8-hour data collection periods, a large volume of data will be recorded. Each set of measurements shall be taken at frequent intervals such as every 5 or 10 minutes, (frequency will be determined based on expected change of parameter (e.g. Temperature of Circulating water could be 10 minutes while APRM reading should be every 5 minutes). Two 8-hour test runs shall be made during the 100 hour demonstration, one in the first 50 hours and the other in the second 50 hours. Warranty parameter values are the average of the values from the two.

5.20.3 Criteria

5.20.3.1 Level 1

The NSSS parameters such as reactor performance and associated fuel limits as determined by using normal plant operating procedures, shall be within the appropriate licensing restrictions.

5.20.3.2 Level 2

The NSSS will be capable of supplying steam of not less than 99.65 percent quality at a pressure of 6.89MPaA (1000 psia) (68.95bars, abs) at the NSSS boundary, in an amount of 7,640.3 tons per hour from the reactor, dependent on a final reactor feedwater temperature of 215.6°C and a total purge flow (CRD and RIP) of 16.42 tons per hour at a temperature of 42.6 °C for each unit. The reactor feedwater flow must equal the steam flow less the total purge flow. [Thermodynamic parameters are consistent with the 1967 ASME Steam Tables. Correction techniques for conditions that differ from the contracted conditions will be mutually agreed to prior to the performance of the test.]

5.21 Test Number 21 – Reactor Water Cleanup System Performance

5.21.1 Purpose

The purpose of this test is to verify that Reactor Water Cleanup System (CUW) performance in various modes of operation is in accordance with design requirements at rated reactor temperature and pressure conditions. (This test will complete the preoperational testing that could not be done without nuclear heating.)

5.21.2 Description

With the reactor at rated temperature and pressure, process variables (e.g., system flow rates, pressure and temperature) will be measured and recorded when the CUW system is in steady state conditions at various modes of operations, i.e., Normal Operation, Startup Operation, Hot Standby Operation, and RPV Head Spray modes.

At rated conditions during CUW normal mode of operation, the temperatures at inlet and the outlet of each heat exchanger will be measured to confirm that each CUW heat exchanger meets the established heat exchange rate performance requirements.

When the reactor water is at rated temperature during heatup, the entire planned blowdown flow rate will be discharged into either main condenser or low conductivity waste collector tank to confirm stable operation in the CUW Startup Operation mode.

During the process of reactor shutdown, CUW flow will be passed into the RPV head spray line and the temperature changes of various parts of the RPV will be measured and recorded to confirm the RPV Head Spray mode of operation.

During reactor Hot Standby mode, CUW is operated to circulate reactor water without the benefit of the filtration process to confirm stable operation in the Hot Standby Operation mode and reduce the reactor temperature gradient in the reactor vessel. The filter demineralizer units are bypassed to preclude degradation of the ion-exchange resin due to high temperature.

In addition, as a part of the water quality confirmation test (Section 4.1), the CUW system sample station will be tested at hot process conditions. By sampling water at the inlet and outlet of the filter demineralizer, it shall be confirmed that each CUW filter demineralizer meets the established performance requirements.

5.21.3 Criteria

5.21.3.1 Level 1

Not Applicable

5.21.3.2 Level 2

The temperature at the tube side outlet of the CUW non-regenerative heat exchangers shall not exceed the limits specified by the applicable CUW System Design Description while the system is in the blowdown mode and normal mode of operations.

The total dynamic head (TDH) of the CUW pump shall meet the design values specified by the applicable CUW System Design Description for all modes of system operations.

The measured heat exchange capacity of each CUW non-regenerative heat exchanger shall meet the established heat exchange rate performance requirement as stated in the applicable CUW System Design Description.

The CUW pump and motor vibration shall be less than or equal to the limits given by the Hydraulic Institute Standard and the Pump Vendor requirements during all modes of system operations.

The cooling water supplied to the CUW non-regenerative heat exchangers shall be within the flow and outlet temperature limits indicated in the CUW System Process Flow Diagram and applicable CUW System Design Description.

5.22 Test Number 22 – RCIC System Performance

5.22.1 Purpose

The purpose of this test is to verify proper operation of the Reactor Core Isolation Cooling (RCIC) System over its expected operating pressure and flow ranges and to demonstrate reliability in automatic starting from cold standby when the reactor is at power conditions.

5.22.2 Description

The RCIC System is designed to be tested in two ways: (1) by flow injection into a test line leading to the suppression pool (SP), i.e., SP Injection Tests, and (2) by flow injection directly into the reactor vessel, i.e., Vessel Injection Tests.

The earlier set of SP Injection Tests consist of manual and automatic mode starts and steady-state operation, at 1.03 MPaG and near rated reactor pressure conditions. The RCIC system will be operated in the Full Flow Test Mode in which the RCIC system is lined up to take suction from and discharge to the suppression pool. The SP Injection Test is done to demonstrate general system operability. During operability demonstration, the RCIC turbine speed adjustment will also be performed at near the rated reactor pressure condition.

During testing of the RCIC system, a pump discharge pressure 690 ± 138 kPa (100 ± 20 psi) greater than the primary system pressure is required in order to simulate the pressure drop due to line losses between the RCIC pump and the reactor vessel. If the system fails to meet the flow and time criteria during the SP injection test, the line losses should be evaluated as closely as possible and this new number, plus 69 kPa (10 psi), should be used as the pump discharge pressure.

A reactor vessel injection test will be performed at rated pressure condition to confirm system performance and to demonstrate automatic starting from a hot standby condition. Subsequently, an automatic mode start at 1.03 MPaG reactor pressure will be performed to demonstrate the system performance with the optimized RCIC turbine adjustment from either hot or "cold" standby condition. "Cold" is defined as a minimum of 72 hours without any kind of RCIC operation.

After all final system adjustments have been determined, two consecutive Vessel Injection Tests, starting from cold standby conditions in the automatic mode, must satisfactorily be performed in order to demonstrate system reliability.

Following these tests, RCIC system performance data will be collected while operating in the Full Flow Test mode in order to provide a benchmark for comparison with future surveillance testing. Demonstration of extended system operation of up to two hours (or until the pump and turbine and their auxiliaries have stabilized) of continuous operation at rated flow and near rated reactor pressure conditions will also be performed.

Sufficient operating data will be taken in order to verify proper setting of, or to adjust as necessary, the RCIC Steam Line High Flow Isolation Trip setting for the Leak Detection and Isolation System (LDI) trip logic.

5.22.3 Criteria

5.22.3.1 Level 1

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

The average pump discharge flow must be equal to or greater than the 100% rated value specified on the RCIC System Process Flow Diagram for all operating modes.

The starting time for the RCIC System from receipt of actuation signal to delivering design flow shall be within the limit specified by the applicable RCIC System Design Specification at any reactor pressure between 1.03 MPaG and rated.

5.22.3.2 Level 2

In order to provide an overspeed and isolation trip avoidance margin, the first transient start and all subsequent speed peaks shall not exceed 5% above the rated RCIC turbine speed.

The differential pressure instruments for the RCIC steam supply line high flow isolation trip shall be calibrated to actuate at the value specified in the plant Technical Specifications (i.e., about 300% of the normal steam supply flow value).

5.23 Test Number 23 – Turbine Valve Performance

5.23.1 Purpose

The purpose of this test is to demonstrate acceptable procedures for testing proper functioning of the main turbine control, stop, and bypass valves during reactor power operation. This test will also verify the maximum capacity of the bypass system.

5.23.2 Description

Main turbine control and stop valves are tested routinely during plant operation to reduce the possibility of main turbine overspeed. Periodic bypass valve surveillance testing is performed to ensure the operability of each installed turbine bypass valve and to minimize the impact on the reactor following main turbine and/or generator trip.

During demonstration of the valve operability test, response of the reactor will be observed and recorded at several test points. In addition, capacity of the installed bypass valves will be measured. Each installed bypass valve will be fully stroked to measure its capacity. After completion of the full power turbine trip and/or load rejection test (Section 4.31), the maximum flow capacity of the bypass system is again demonstrated.

Each valve will be initially tested and reset at a power level between 45 and 55% of rated. The rate of valve stroking and the timing of the close-open sequence will be such that the minimum practical disturbance is introduced. The Turbine Bypass Valves are required to be stroked to their 10% position. Bypass valve performance is also demonstrated by a full stroke test performed for the purpose of capacity measurement.

This testing will establish a maximum possible power level at which these valve tests can be performed without risk of a scram. Testing will initially be performed at 65% power, and results used to extrapolate to the next test point between 75 and 90% power. Ultimately a maximum power level will be chosen for all subsequent surveillance tests with ample margin to scram. A confirmatory test will be performed at this power level and used for baseline data collection.

5.23.3 Criteria

5.23.3.1 Level 1

The reactor shall not scram or isolate during full stroke testing of main turbine control, stop, and bypass valves at power levels up to the maximum allowable power level for conducting such tests.

5.23.3.2 Level 2

During full stroke testing of main turbine control, stop, and bypass valves, peak neutron flux must be at least 7.5% below the scram trip setting, peak vessel pressure must remain at least 69 kPa (10 psi) below the high pressure scram setting, and peak simulated fuel surface heat flux must be at least 5.0% below its scram trip point. Peak steam flow in each line must remain 10% below the high flow isolation trip setting.

The sum of measured individual bypass valve capacity shall be equal to or greater than [33%] of rated steam flow (the value used for the Nuclear Safety Operational Analysis).

5.24 Test Number 24 – MSIV Performance

5.24.1 Purpose

The purposes of this test are: a) to functionally check the MSIVs for proper operation at selected power levels, and b) to determine the closure time of each MSIV.

5.24.1.1 Description

At rated temperature and pressure during initial heatup, and then again at a power level within the low power plateau, each MSIV will be individually stroked in the fast closure mode. Valve operability and closure time will be verified and overall plant response observed. The times to be determined are: a) the time from de-energizing the solenoids until the valve is 100% closed (t_{sol}), and b) the valve stroke time (t_s). Time t_{sol} equals the interval from de-energizing the solenoids until the valve reaches 90% closed plus 1/8 times the interval from 10% to 90% closure. Time t_s equals the interval from when the valve starts to move from full open until it is 100% closed and is based on the interval from 10% to 90% closure and linear valve travel from 0% to 100% closure.

To determine the maximum power level at which periodic MSIV full individual closure surveillance testing can be performed without a scram, choose the fastest MSIV and perform a fast closure at successively higher power levels along the rated power rod line. The first actuation will be performed at 65% power and will be used to extrapolate to the next test point between 70 and 85% power, and ultimately to the maximum power test condition with ample margin to scram. The final test recorded in this series will be the demonstration of the best procedure for individual MSIV surveillance testing as required by the plant Technical Specifications.

5.24.2 Criteria

5.24.2.1 Level 1

The MSIV stroke time (t_s), exclusive of electrical delay, for any individual valve shall be 3.0 seconds $\leq t_s \leq$ 4.5 seconds. Total effective closure time for any individual MSIV shall be t_{sol} plus the maximum MSIV closure instrumentation delay time as determined in pre-operational test GE-15 and shall be ≤ [4.56] seconds. Maximum MSIV closure instrumentation delay time is defined as the elapsed time from when the trip setpoint value is exceeded at the sensor to when power is lost to the MSIV latch solenoids.

The reactor shall not scram or isolate during full closure of an individual MSIV at power levels up to the maximum allowable power level determined for conducting such tests.

5.24.2.2 Level 2

During full closure of an individual MSIV, peak neutron flux must be at least 7.5% below the scram trip setting, peak vessel pressure must remain at least 69 kPa (10 psi) below the high pressure scram setting, and peak steam flow in individual line must be 10% below the high flow isolation trip setting. The peak simulated fuel surface heat flux must be at least 5.0% below its scram trip point.

5.25 Test Number 25 – SRV Performance

5.25.1 Purpose

The purposes of this test are: a) to demonstrate that each safety/relief valve (SRV) can be opened and closed in the manual actuation mode during power operation; b) to verify that the relief valves reseat properly after operation, and c) to verify that there are no major blockages in the SRV discharge piping.

5.25.2 Description

A functional and flow demonstration test of each SRV shall be made as early as practical in the Startup Test Program when adequate steam flow is available to minimize the potential for valve seat damage upon valve closure and the subsequent relief valve leakage. This is normally the first time the plant reaches ≥ 6.55 MPaG with steam flow greater than the individual relief valve capacity.

Each SRV will be manually actuated. Turbine bypass valve (TBV) response shall be monitored if the test is performed during reactor heatup. The electrical output (MWe) response is monitored if the test is performed after T/G synchronization during the low power plateau. The test duration will be about 10 seconds to allow turbine valves and tailpipe sensors to reach a steady state.

The tailpipe sensor responses will be used to detect the opening and subsequent closure of each SRV. Additionally, the TBV and MWe responses will be analyzed, by making valve-to-valve comparisons, for anomalies indicating a restriction in an SRV tailpipe.

Vendor bench test data of the SRV opening responses will be available on-site for the comparison with design requirements.

Valve capacity will be based on certification by ASME code stamp and the applicable documentation being available in the on-site records. Note that the nameplate capacity and pressure rating assumes that the flow is sonic. A major blockage of the line may prevent sonic flow and it should be determined that no major blockage exists through the TBV or MWe response signatures.

During pressurization transients, e.g., full reactor isolation test (Section 4.32), reactor pressure and the operation of the SRV(s) will also be monitored. A comparison between the reactor pressure behavior and SRV actuations will be made to confirm the operability, open/close setpoints, and test pressure of the actuated SRVs.

5.25.3 Criteria

5.25.3.1 Level 1

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

5.25.3.2 Level 2

During opening and closing of each SRV, the 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 SRVs shall return to within 5.6 °C (10°F) of the temperature recorded before the valve was opened.

During the test, the steam flow through each relief valve shall not differ by more than 10% from the average relief valve steam flow as measured by bypass valve position; or the steam flow through each relief valve, as measured by change in MWe, shall not differ by more than 0.5% of rated MWe from the average of all the valve responses.

5.26 Test Number 26 – Loss of Feedwater Heating

5.26.1 Purpose

The purposes of this test are: a) to demonstrate adequate plant response to a loss of feedwater heating event, and b) to verify the adequacy of the modeling and associated assumptions used for this transient in the plant transient analysis.

5.26.2 Description

The plant is designed such that no single equipment failure or operator error shall cause the largest design basis feedwater temperature reduction. When the difference between actual and reference temperature exceeds a ΔT setpoint, the Feedwater Control System (FWC) sends an alarm to the main control room operator.

Upon detection of a loss of feedwater heating, the FWC will send a signal to the Recirculation Flow Control System (RFC), which will provide a signal to the Rod Control and Information System (RCIS) to initiate automatic selected control rods run-in (SCRRI). Additionally, the RFC will initiate a simultaneous runback of all operating RIPs to a designated speed limit to automatically reduce the reactor power and avoid a scram.

Prior to the test, the condensate/feedwater system design will be evaluated to determine the maximum number of feedwater heaters that can be tripped or bypassed by a credible single event. This event represents the most severe transient for analysis consideration. The event will then be initiated at a power level between 80 and 90% of rated with the recirculation flow near rated value.

Core performance and overall plant response will be observed in order to demonstrate proper integrated response and to compare actual results with those predicted. This comparison will take into account the differences between actual initial conditions and observed results and the assumptions used for the analytical predictions. Proper operation of the SCRRI function will also be verified, if the difference between actual and reference feedwater temperatures exceeds the ΔT setpoint as a result of the loss of feedwater heating transient.

5.26.3 Criteria

5.26.3.1 Level 1

The maximum feedwater temperature loss due to single operator error or equipment failure shall not exceed the value assumed in the design basis plant transient analysis (FSAR Subsection 15.1.1.1.1).

The resultant MCPR due to loss of feedwater heating must be greater than the fuel thermal safety limit.

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

5.26.3.2 Level 2

The increase in simulated fuel surface heat flux cannot exceed the predicted value referenced to the actual test values of feedwater temperature change and power level. The predicted value is provided in the plant Startup Transient Analysis Report document and will be used as the basis to which the actual transient is compared.

The SCRRI function and a simultaneous runback of all operating RIPs to minimum speed shall be automatically initiated if the differential between the actual and reference feedwater temperatures exceeds the ΔT setpoint. The reactor power shall be brought down to below 60 % rod line as shown on the power/flow-operating map.

5.27 Test Number 27 – Feedwater Pump Trip

5.27.1 Purpose

The purpose of this test is to demonstrate the capability of the plant to respond to and continue operation following the trip of one operating feedwater pump.

5.27.2 Description

From an initial reactor power level of greater than the capacity of one feedwater pump, one of the operating feedwater pumps will be tripped and it will be demonstrated that the RIP speed runback circuits are actuated during the vessel level transient to drop the reactor power to within the capacity of the remaining feedwater pump. Specifically, it shall be verified that the Feedwater Control System is sufficiently responsive, in conjunction with the automatic RIP speed runback feature to prevent a reactor trip due to the water level transient. Separate tests may be required to demonstrate features such as automatic core flow runback or auto start of a standby feedpump, if appropriate.

The ability to minimize reactor water level decrease is directly related to the rate of recirculation flow runback at which reactor power is reduced and the margin between the remaining feedwater capacity and the steam flow at the reduced power level. This test will confirm that a reactor scram does not occur due to tripping one of the normally operating feedwater pumps from rated power level, with ample margin above the low level (Level 3) scram setpoint. This test should be performed after the speed of each RFP is limited in accordance with the Feedwater System Performance test (Test 17).

5.27.3 Criteria

5.27.3.1 Level 1

Not Applicable

5.27.3.2 Level 2

The reactor shall avoid low water level scram by at least 7.62 cm margin from an initial water level midway between the high and low water level alarm setpoints.

5.28 Test Number 28 - RIP Trip Tests

5.28.1 Test Number 28A – One RIP Trip Test

5.28.1.1 Purpose

The purposes of this test are: a) to demonstrate acceptable plant response to the one RIP trip and recovery transient; and b) to determine the maximum speed along the maximum rod line for the operating RIPs when restarting one RIP.

5.28.1.2 Description

Initially, one RIP trip will be initiated from near rated flow conditions at or below the 75% rod line within the mid power plateau. Then, the test will be repeated at near rated power conditions. Recirculation system performance data during the one RIP trip transient will be recorded and compared against the applicable criteria.

Of particular interest is the determination of the maximum speed for the operating RIPs on the maximum rod line for restarting one tripped RIP. The recommendation for the maximum speed of the operating RIPs is based on assuring adequate margin to the over-current trip function of the RIP ASD for the RIP that is restarted. If the one RIP restart is performed with the other RIPs operating at too high a RIP speed, the starting torque (e.g., to restart against a large reverse flow through the idle RIP) requirement can become so large that the RIP ASD may trip on over-current.

After the first One RIP Trip Test at the mid power plateau is performed, the RIP restart test should be performed with the other 9 RIPs operating in the Gang Speed Control mode at minimum RIP speed. The RIP ASD instantaneous over-current nominal setting is 200% of rated ASD output current. If the peak measured over-current for the RIP ASD that is restarted is less than 125% of rated ASD output current, a higher recommended speed for the 9 operating RIPs should be estimated, then testing performed to obtain more data on the over-current observed. To allow for some variability among the RIP ASDs and RIPs, the final startup test criteria for peak measured over-current is set at 150% of rated ASD output current.

After the one RIP trip test is performed in the high power plateau, a final formal test of one RIP restart along the maximum rod line with the other 9 RIPs operating at the maximum speed determined at the prior mid power plateau testing should be performed. If the measured value of over-current exceeds 150% of rated ASD output current, then the maximum recommended speed should be lowered and a retest performed to confirm that adequate avoidance margins for the over-current trip and the neutron flux scram exist.

5.28.1.3 Criteria

5.28.1.3.1 Level 1

The reactor shall not scram during one RIP trip recovery transients.

The resultant MCPR after one RIP trip must be greater than the fuel thermal limit.

5.28.1.3.2 Level 2

During one RIP trip recovery, the peak neutron flux shall remain below the scram setting by 7.5% and peak simulated heat flux must be at least 5.0% below its scram trip setpoint.

For one RIP restart with the other 9 RIPs operating in Gang Speed Control mode at or above the final recommended maximum speed, the peak over-current shall not exceed 150 % of rated RIP ASD output current.

5.28.2 Test Number 28B – Three RIPs Trip Test

5.28.2.1 Purpose

The purposes of this test are: a) to record and verify acceptable performance of the three RIPs trip transient; b) to demonstrate that the Feedwater Control System can satisfactorily control water level to prevent a main turbine and feedwater pump trip; and c) to demonstrate that SCRRI will function as designed, if flow and power setpoints are reached during the three RIPs trip.

5.28.2.2 Description

The three RIPs trip tests are performed to assure that the water level will not rise enough to threaten a high water level trip of the main turbine and the operating feedwater pumps.

The RIP speed coastdown characteristics of the tripped RIPs will also be demonstrated. This three RIPs trip test verifies that the RIP speed coastdown is satisfactory prior to the Turbine Trip and Load Rejection Tests (see Section 4.31) at high power plateau and subsequent operation. Additionally, the operability of the SCRRI function will be verified if the reactor is forced into the high power/low flow exclusion zone (Region III) due to the three RIPs trip.

At selected power levels within the mid and high power plateaus, a generator trip of an M/G set that is connected to three RIPs will be initiated. To avoid an inadvertent high water level trip, the initial water level should be set slightly above the low level alarm setting prior to the test. The level swell data at the mid power plateau should be extrapolated to 100% power prior to performing the test at the high power plateau. The simulated fuel surface heat flux should also be extrapolated to the maximum rod line in order to determine the scram trip avoidance margin at 100% power.

The margin to avoid a high water level trip should be determined by the margin between the high water level trip (Level 8) and the transient high water level which would occur if the initial water level was set at the high water level alarm setting (Level 7), i.e., margin to trip is defined as:

Margin = (High Level Trip (Level 8) Setpoint) – (Maximum water level reached during three RIP trip test) – (High Level Alarm (Level 7) Setpoint – Initial water level)

5.28.2.3 Criteria

5.28.2.3.1 Level 1

The reactor shall not scram as a result of the transient resulting from the trip of three RIPs.

The resultant MCPR following the three RIP trip must be greater than the fuel thermal limit.

The positive change in vessel dome pressure occurring within 30 seconds after the trip of three RIPs must not exceed the Level 2 criteria (i.e., 4.28.2.3.2) by more than 172.4 kPa (25 psi).

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

The tripped RIP ASD pump speed coastdown curve during the first 3 seconds must fall between the minimum safety limit curve and the maximum safety limit curve defined in the Startup Transient Analysis Report.

5.28.2.3.2 Level 2

The reactor water level margin to avoid a high level trip shall be ≥ 7.62 cm during the three RIPs trip.

The positive change in vessel dome pressure and in simulated fuel surface heat flux which occur within the first 30 seconds after the initiation of the three RIP trip must not exceed the predicted values, corrected for as tested initial power and dome pressure. The predicted analytical values are provided in the Startup Transient Analysis Report document.

The simulated heat flux margin to avoid a scram shall be ≥ 5.0 percent during transients as a result of the trips of three RIPs.

The APRM margin to avoid a scram shall be ≥ 7.5% during the transient.

The SCRRI shall automatically initiate if flow and power setpoints are reached after the three RIP trip, and its functional performance shall be as designed (i.e., the total delay time between start of RIP trip and start of pre-selected SCRRI control rods motion shall not exceed 3 seconds, the pre-selected SCRRI control rods shall be fully inserted within 145 seconds after the start of the control rod motion, and the reactor power shall be brought to below 60 % rod line as defined on the power/flow operating map).

5.29 Test Number 29 – Shutdown From Outside the Main Control Room

5.29.1 Purpose

The purpose of this test is to demonstrate that the reactor can be brought from a normal initial steady-state power level to the point where cooldown is initiated and under control with reactor water level and pressure controlled using the remote shutdown panels and equipment outside the main control room (MCR).

5.29.2 Description

The hot standby capability demonstration portion of this test shall be performed from a low initial power level that is sufficiently high such that a majority of plant systems are in their normal configurations and the Turbine Generator is in operation. The test shall be performed using the minimum shift crew complement as defined by the plant Technical Specifications. Additional qualified personnel will be available as control room observers to monitor the progress of the test and to re-establish control of the plant should an unsafe condition develop. These personnel will also perform predefined non-safety-related activities to protect plant equipment where such activities would not be required during an actual emergency situation.

This test will start with a simulated control room evacuation. The reactor will then be tripped and isolated by means outside of the Main Control Room. Achievement and maintenance of the hot standby condition is then demonstrated through control of vessel pressure and water level from outside control room using SRVs, HPCF, and RHR.

The ability to reach cold shutdown is demonstrated by placing at least one loop of the RHR system in shutdown cooling and establishing a heat rejection path to the ultimate heat sink from the Remote Shutdown Panel. The cold shutdown capability does not necessarily have to be demonstrated immediately following the shutdown and hot standby demonstration as long as the total integrated capability is adequately demonstrated. The cooldown portion of the test may be performed in conjunction with another startup test or plant event when sufficient decay heat is available and reactor cooldown is required. Also, additional personnel, over and above the minimum shift crew, may be utilized for the cold shutdown portion of the test, consistent with plant procedure and management's ability to assemble extra help at the plant site in emergency situations.

5.29.3 Criteria

5.29.3.1 Level 1

Not Applicable

5.29.3.2 Level 2

During a simulated control room evacuation, the hot standby capability demonstration portion of the test must demonstrate that the reactor can be brought down from a normal initial steady-state power level to hot standby condition with reactor vessel pressure and water level under control using minimum shift crew and equipment and controls outside the main control room. The plant shall be maintained at stable hot standby conditions for at least 30 minutes.

The cold shutdown capability demonstration portion of the test must demonstrate that the reactor coolant temperature and pressure can be lowered sufficiently to put the RHR System in the Shutdown Cooling mode of operation and under control from outside the main control room. The reactor coolant temperature shall be reduced at least 27.8°C at a rate that would not exceed the plant Technical Specifications limit using the RHR System.

5.30 Test Number 30 – Loss of Turbine/Generator and Offsite Power

5.30.1 Purpose

The purpose of this test is to demonstrate electrical equipment response and reactor system transient performance during and subsequent to a turbine/generator (T/G) trip event with a coincident loss of all offsite power sources.

5.30.2 Description

The loss of T/G and offsite power test will be performed in the 10% to 20% power range. This test will be initiated by tripping the main generator in addition to disconnecting the plant completely from the normal and alternate preferred offsite power sources.

TSV closure or TCV fast closure will not cause scram since the reactor is at low power, i.e., less than [40]% of rated. However, it is expected that the reactor will scram on low water level due to the loss of a RFP or high reactor pressure as a result of losing EHC control. The transient will be extended until SRV(s) action (if required) shows adequate pressure control, and the decreasing reactor water level rate is established to estimate when automatic initiation of the RCIC would occur. Manual intervention can be performed once this pressure and level control data is determined.

The operator may take action as he desires after the first three minutes, including taking the Reactor Mode Switch out of the RUN mode. The operator may also switch out of the RUN mode in the first three minutes, after confirming from the measured data that this action will not prevent an automatic MSIV closure trip due to low reactor water level. The plant will be maintained isolated from both offsite power sources for at least 30 minutes. During this time, appropriate reactor parameters will be recorded in order to verify proper response of plant systems and equipment. In addition, proper sequencing of emergency diesel generator loads during the transient will be confirmed.

Upon the loss of T/G and all offsite power sources, all electric pumps including four RIPs not connected to the M/G set, condensate and feedwater pumps, and circulating water pumps, are tripped. The other six RIPs powered by M/G sets will maintain their original speed initially and eventually be tripped due to RIP ASD trip logic, but will coastdown at a slower rate due to inertia from the M/G set flywheels. The condensate, feedwater and circulating water systems will be coasting down such that the reactor eventually scrams on low water level (Level 3) and/or low condenser vacuum MSIV isolation. The TBVs will remain operable in controlling reactor pressure until TBV hydraulic accumulators are exhausted, or until inhibited from opening by low condenser vacuum.

5.30.3 Criteria

5.30.3.1 Level 1

Reactor Protection System actions shall prevent violation of fuel thermal limits.

All safety systems, such as the Reactor Protection System, the diesel-generators, and the ECCS system must function properly without manual assistance, and RCIC system action, if necessary, shall keep the reactor water level above the initiation level of the HPCF system.

The turbine steam bypass valves shall open after the turbine/generator trips and remain operable until the MSIV's are closed or until the high condenser pressure signal closes the bypass valves.

5.30.3.2 Level 2

Proper instrument display to the reactor operator shall be demonstrated, including power monitors, vessel pressure, reactor water level, control rod positions, suppression pool temperature, and reactor cooling system status. Displays shall not be dependent on specially installed instrumentation. Temporary interruption of instrument display is acceptable provided that the operator has sufficient information available for long-term operation to properly access the plant status.

The CTG automatically starts on detection of a loss of power to the Plant Investment Protection buses and after reaching rated voltage and frequency, can supply power to those busses.

5.31 Test Number 31 – Turbine Trip and Load Rejection

5.31.1 Purpose

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

5.31.2 Description

From an initial power level near rated, a load rejection event will be initiated in order to verify the proper reactor and integrated plant response. The method for initiating the trip shall be chosen so that the turbine is subjected to maximum overspeed conditions. The results of this evaluation will be documented in the Startup Transient Analysis Report (STAR). Typically, this trip is initiated by opening of the generator output breakers. Reactor parameters such as vessel dome pressure and simulated fuel surface heat flux will be monitored and compared with predictions so that the adequacy and conservatism of the analytical models and assumptions used to license the plant can be verified. Proper response of systems and equipment such as the turbine stop, control, and bypass valves, main steam safety relief valves, the Reactor Protection System, the Feedwater System and Recirculation System will also be demonstrated. The core flow coastdown characteristics should be evaluated upon actuation of the recirculation pump trip logic. The ability of the Feedwater System to control vessel level after a reactor trip shall also be verified. Overspeed of the main turbine shall also be evaluated, since the generator is unloaded prior to complete shutoff of steam to the turbine.

For a direct trip of the turbine, the generator remains loaded and there is no overspeed. At high power levels, the dynamic response of the reactor is very similar between the generator and turbine trip transient. Therefore, a separate turbine trip test at high power level is not required.

A turbine trip or load rejection test shall also be performed at low power level such that nuclear boiler steam generation is just within the capacity of the bypass valves to demonstrate scram avoidance. At low power levels, sufficient reactor protection following the trip is provided, if needed, by high neutron flux and vessel high-pressure scrams. Therefore, the protective trip actuated by stop/control valve motion is automatically bypassed at low power levels. There will be no significant pressure or power transient as a result of this low power turbine trip or load rejection event, and therefore, no reactor scram should occur for this test.

5.31.3 Criteria

5.31.3.1 Level 1

The core flow coastdown transient during the first three seconds after either turbine trip or load rejection at greater than 50% of rated power must be bounded by the limiting curves defined in the Startup Transient Analysis Report (STAR).

At power level > 50% of rated, the TBV fast opening shall begin no later than 0.02 seconds after the start of TSV closure or TCV fast closure motion, and bypass flow shall be at least 80 percent of total bypass capacity within 0.17 seconds of start of TSV closure or TCV fast closure motion.

Feedwater system settings must prevent flooding of the steam lines following generator or turbine trip transients.

The positive change in vessel dome pressure occurring within 30 seconds after either turbine or generator trip must not exceed the Level 2 criteria by more than 172.6 kPa (25 psi).

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

The positive change in simulated fuel surface heat flux shall not exceed the Level 2 criteria by more than 2% as specified by the Startup Transient Analysis Report.

5.31.3.2 Level 2

An automatic MSIV isolation shall not occur during the first three minutes of the transient resulting from either turbine trip or load rejection at greater than 50% of rated power. Operator actions shall not be required during that period to avoid an MSIV closure trip. (NOTE: The operator may take action as he desires after the first three minutes, including switch out of the RUN mode. The operator may also switch out of the RUN mode in the first three minutes, if he confirms from the measured data that this action will not prevent an automatic MSIV closure trip due to low reactor water level.)

The reactor shall not scram for turbine trip or load rejection event initiated from initial thermal power values within the bypass valve capacity.

The Feedwater Control System shall be capable of avoiding a loss of feedwater pumps due to high reactor water level (Level 8) trip during the event.

Low water level RIP trips and HPCF/RCIC initiations shall not occur during the transient.

The positive change in vessel dome pressure and simulated fuel surface heat flux occurring within the first 30 seconds after the initiation of either turbine trip or load rejection must not exceed the predicted values referenced to actual test conditions of initial power level and vessel dome pressure and corrected for the measured control rod insertion speed and initiation time. The predicted values are provided in the applicable STAR document based on the beginning-of-cycle design basis analysis and shall be used as the basis to which the actual transient is compared.

If any SRVs open, the temperature measured by thermocouples on the discharge side of the SRVs shall return to within 5.6 °C of the temperature recorded before the valve was opened.

5.32 Test Number 32 – Reactor Full Isolation

5.32.1 Purpose

The purpose of this test is to verify that the dynamic response of the reactor and applicable systems and equipment is in accordance with design for a simultaneous full closure of all MSIVs.

5.32.2 Description

A simultaneous full closure of all MSIVs will be performed at 100% (+0, -5%) of rated thermal power. Reactor dynamic response, as determined by such parameters as vessel dome pressure and simulated fuel surface heat flux, will be compared with analytical predictions in order to verify the adequacy and conservatism of the models and assumptions used in the plant safety and licensing analysis.

Upon receipt of the initiation signal, all MSIV's will begin closing. A reactor scram will be initiated based upon the change in status of the position switches on the valves. No significant neutron flux peak or fuel surface heat flux peak is expected because the reactor scram is initiated before any significant steam flow interruption occurs. After the valves have closed, reactor pressure will rise quickly and SRV(s) will open.

By comparison of actual test results with predictions, the responses of the MSIVs, SRVs, reactor protection, reactor pressure and water level control systems, and the recirculation flow control system are evaluated. Reactor pressure peak and fuel surface heat flux peak values will be compared against the analytical predicted results from the STAR in order to verify the adequacy and conservatism of the models and assumptions used in the plant safety and licensing analysis.

5.32.3 Criteria

5.32.3.1 Level 1

The reactor must scram to limit the severity of the neutron flux and simulated fuel surface heat flux transients.

Feedwater system settings must prevent flooding of the main steam lines following the full reactor isolation transient event.

The recorded MSIV full closure times must meet the limits specified in the Technical Specifications.

The positive change in vessel dome pressure occurring within 30 seconds after closure of all MSIVs must not exceed the Level 2 criteria by more than 172.4 kPa (25 psi).

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

5.32.3.2 Level 2

The positive changes in vessel dome pressure and in simulated fuel surface heat flux which occur within the first 30 seconds after the initiation of full reactor isolation must not exceed the predicted values with corrections for actual measured values of initial power and dome pressure, control rod insertion speed and initiation time. For the full reactor isolation from full power predicted analytical results based on Beginning-Of-Cycle (BOC) design basis analysis, assuming no equipment failures and applying appropriate parametric corrections, shall be used as the basis to which the actual transient is compared. The following table specifies the upper limits of these criteria during the first 30 seconds following initiation of the indicated conditions.

Initial Conditions		Criteria	
Power (%)	Reactor Pressure (MPaA)	Increase In Heat Flux (%)	Increase In Reactor Press (MPaG)
100	7.17	*	*

* Defined in the Unit specific Startup Transient Analysis Report.

If any SRVs open, the temperature measured by thermocouples on the discharge side of the SRVs shall return to within 5.6 °C of the temperature recorded before the valve was opened.

Initial action of RCIC and HPCF shall be automatic if the low water level (Level 2 and 1.5, respectively) setpoints are reached. The minimum capacity and maximum delay time (including instrumentation delay) between the time vessel water level drops below the setpoint and makeup water enters the vessel shall meet the safety analysis requirements specified in the applicable plant Project Design Manual.

Recirculation Pump Trip (RPT) shall be initiated and the four RIPs not connected to M/G sets shall trip automatically if the low water level (Level 3) setpoint is reached. In addition, trip of the remaining six RIPs that are connected to M/G sets shall be initiated if the low water level (Level 2) setpoint is reached. This shall cause three out of the six specified RIPs to trip instantaneously and the remaining three RIPs to trip 6 seconds later.

5.33 Test Number 33 – Loose Parts Monitoring System Baseline Data

5.33.1 Purpose

The purpose of this test is to record a full range of initial baseline data for the Loose Parts Monitoring System (LPMS) during the Startup Test Program.

5.33.2 Description

The LPMS overall and individual sensor signal channels will be characterized under normal plant operating conditions. At several power-flow levels, the LPMS will be placed in the manual mode of operation and, using the appropriate operating procedure for guidance, the baseline set of data indicative of normal plant operation for each sensor signal channel is obtained.

LPMS sensors are accelerometers. Each accelerometer channel will exhibit its own particular signature and corresponding unique frequency spectrum. Initial baseline data will be obtained in the form of waveform plots.

The data obtained will be used to benchmark the adequacy of, or to facilitate the needed changes to, initial alert level settings for the alarm initiations and the automatic activation of data acquisition mode of system operation.

5.33.3 Criteria

5.33.3.1 Level 1

Not Applicable

5.33.3.2 Level 2

Initial baseline data for the Loose Parts Monitoring Systems has been satisfactorily established for each specified power and flow condition during steady-state operation.

5.34 Test Number 34 – Steam Separator/Dryer Performance Test

5.34.1 Purpose

The purpose of this test is to verify that the steam separator/dryer system will meet the minimum performance requirements at conditions within allowable regions of the Power/Flow operating map.

5.34.2 Description

Maximum moisture carryover is predicted to occur at low steam flow and high core flow conditions. If the separator/dryer performance limit line is reached, the RFC will perform a RIP speed runback to reduce core flow and power to an allowable region. This test will verify that the analytically established limit line is sufficient to prevent excessive moisture carryover of the steam exiting the reactor, and the established steam separator limit line is adequate to prevent plant operation outside the allowable region.

With the RIPs at maximum permissible speeds and flows, this test is initiated from a power level just above the most limiting region of the power-flow operating map. At this point, the moisture carryover in the steam exiting the reactor is determined. Injecting Na-24 into the feedwater and then measuring the concentration of the Na-24 that reaches the condenser versus that remaining in the reactor coolant can determine moisture carryover. Other methods are acceptable as long as they can provide an accuracy of 0.005%. Inserting control rods will then gradually lower reactor power. At each incremental power level, the moisture carryover is again determined. This incremental power reduction and moisture carryover determination is continued until either the separator limit line (as shown on the power-flow operating map) is reached or excessive moisture carryover (i.e., 0.01% or greater) of the exit steam is determined, whichever occurs first. If the separator performance limit line is reached first, testing may be continued at incrementally lower power levels until such time as the 0.01% limit is reached, if desired to justify lowering of the established limit. Further power reduction shall be discontinued once moisture carryover of the exit steam exceeds 0.01%. The point at which the RIP speed runback logic is automatically actuated during power reduction shall be recorded. During the performance of this test, the actual RIP speed runback may be temporarily defeated as long as the moisture content of the exit steam is determined to be acceptable. This is to simplify the test without having to recover from a recirculation runback and to allow testing at low power levels.

5.34.3 Criteria

5.34.3.1 Level 1

Steam separator/dryer exit moisture must not exceed 0.1% while operating in the analytically allowable region of the power/flow operating map, i.e., on or above the steam separator limit line.

5.34.3.2 Level 2

Further power reduction shall be discontinued once moisture carryover of the exit steam exceeds 0.01%.

The RIP speed runback logic shall be verified to be conservatively established relative to the steam separator limit line on the Power/Flow operating map in order to prevent operation in areas where excessive moisture carryover in exit steam from separator/dryer is predicted to occur.

5.35 Test Number 35 – Concrete Penetration Temperature Survey

5.35.1 Purpose

The purpose of this test is to verify the acceptability of concrete temperatures in the vicinity of selected high-energy fluid piping penetrations under normal operational conditions.

5.35.2 Description

The penetration concrete temperature survey test consists of measuring concrete temperature surrounding the penetrations of high energy fluid piping, including main steam, feedwater, CUW suction, and RHR shutdown cooling piping penetrations through the shield building. Penetrations and measurement locations selected for monitoring, as well as the test conditions at which data is to be collected, shall be sufficiently comprehensive so as to include the expected limiting thermal loading conditions on critical concrete walls and structures within the plant.

Measurements from temperature sensors on the concrete are recorded during initial heatup and at each major power level during the power ascension testing. The measured temperatures are compared, and proven to be acceptable with respect to the maximum levels specified in the Piping Penetration Design Requirement Specifications for long-term normal operating conditions.

5.35.3 Criteria

5.35.3.1 Level 1

Not Applicable

5.35.3.2 Level 2

The concrete temperature adjacent to the selected high-energy fluid piping penetrations does not exceed the level in the piping penetration design requirements [93.3 °C (200° F)] for long term normal operating conditions.

5.36 Test Number 36 – Reactor Building Cooling/Service Water Systems

5.36.1 Purpose

The purpose of this test is to verify performance of the Reactor Building Cooling Water (RCW) and the Reactor Building Service Water (RSW) systems under expected reactor power operation load conditions. (This test performed during power ascension is necessary only to the extent that fully loaded conditions could not be approached during the preoperational test phase). Data collected during normal operating conditions and during planned transients will be used to also verify that performance would be satisfactory under emergency conditions.

5.36.2 Description

The RCW and RSW systems are designed to provide a reliable supply of cooling water for the removal of heat from unit auxiliaries, such as RHR heat exchangers, standby diesel (DG) generators, and room coolers for Emergency Core Cooling System equipment required for a safe reactor shutdown following a Design Basis Accident (DBA) or transient. The RCW/RSW Systems also provide cooling to other system components, as required, during normal operation, shutdown and reactor isolation modes.

Pertinent parameters, including the RCW/RSW system flow rates and system operational data, RSW water temperature at the inlet to the RCW heat exchangers, and RSW pump head specified in the plant Technical Specifications should be recorded and verified. The performance of the RHR cooling water system during the RHR System Performance test (Section 4.19) is also monitored to ensure sufficient cooling capacity of the RHR heat exchangers to limit suppression pool temperature and for supporting the required safety-related systems during safe shutdown of the unit following major transients.

5.36.3 Criteria

5.36.3.1 Level 1

Not Applicable

5.36.3.2 Level 2

The RCW system shall be capable of removing heat loads from plant auxiliaries at the heat exchanger capacity determined by the flow rate and temperature differentials indicated on the RCW system process diagram for normal and emergency operating conditions.

The RSW system shall be capable of providing cooling water to remove heat from the RCW system heat exchangers while maintaining RCW heat exchanger outlet temperature within the design limits for both normal and emergency operating conditions.

5.37 Test Number 37 – HVAC System Performance

5.37.1 Purpose

The purpose of this test is to verify that important heating, ventilation and air-conditioning (HVAC) systems including Reactor Building HVAC System (RBHV) Control Building HVAC System (CBHV), and Turbine Building HVAC System (TBSV) maintain their service areas within design limits during reactor power operation.

5.37.2 Description

HVAC tests performed during the Startup Test Phase are necessary only to the extent that fully loaded conditions could not be achieved during the preoperational test phase.

RBHV provides reactor building space air-conditioning and also maintains the secondary containment environment at a negative pressure with respect to the outside atmosphere. RBHV includes both safety- and non-safety-related subsystems. This test will verify the adequacy of safety-related RBHV subsystems to maintain an environment in those areas of the plant within the qualification pressures, humidities, and temperatures of that affected equipment. The capability of non-safety-related subsystems to provide non-safety-related equipment rooms at a pressure, humidity and temperature to assure satisfactory function will also be verified.

CBHV is comprised of Control Room Habitability Area (CBHA) HVAC, Safety-Related Equipment Area (SREA) HVAC, and Non-Safety-Related Equipment Area (NSREA) HVAC subsystems. The areas served by the CBHV include the main control room area envelope, various safety-related equipment areas, safety and non-safety related electrical equipment rooms, and two non-safety related motor generator set rooms located in the Control Building. This test will verify that the CBHV is capable of providing a controlled environment in the respective areas that CBHV serves to ensure the continued operation of safety and non-safety-related equipment during normal plant operation.

In addition to the CBHV and the RBHV, additional tests will be performed on systems that cool areas in which fully loaded conditions could not be achieved during the preoperational test phase (e.g. Turbine Building).

After stabilizing plant conditions by maintaining power at the specified level, temperatures and relative humidity in various rooms and areas will be determined and recorded. These readings will then be compared against the specified acceptance criteria. If required, pertinent system operating parameters during various system operating modes will be collected in order to provide a final verification of proper system flow balancing and cooler performance under design conditions.

5.37.3 Criteria

5.37.3.1 Level 1

Not Applicable

5.37.3.2 Level 2

Area ventilation systems shall be capable of maintaining the temperature and relative humidity within the environmental qualification requirements for the affected equipment as stated in FSAR Appendix 3I, Tables 3I.3-1 to 3I.3-5.

6.0 Additional Tests

6.1 Turbine Building Cooling/Service Water Systems Performance

6.1.1 Purpose

To verify performance of the Turbine Building Cooling Water (TCW) System and the Turbine Building Service Water (TSW) System, under expected reactor power operation load conditions.

6.1.2 Description

The TCW and TSW systems serve to provide cooling water to various system loads, including condensate and condensate booster pump oil coolers, turbine EHC fluid coolers, and process sample coolers located in the turbine building. The adequacy of the TCW/TSW systems to maintain system parameters, such as heat exchanger outlet temperature, within the design operating limits during normal plant operation will be verified.

Pertinent parameters, including the TCW/TSW system flow rates and system operational data, TSW water temperature at the inlet to the TCW heat exchangers, and TSW pump head will be recorded and verified. The demonstration, and adjustment if necessary of the Iron Ion Injection System (if installed) will be performed.

6.1.3 Criteria

6.1.3.1 Level 1

Not Applicable

6.1.3.2 Level 2

The TCW System shall be capable of providing cooling water to the various heat exchangers identified in FSAR Table 9.2-12. It serves to maintain system parameters within the design temperature limits as specified in FSAR Subsection 9.2.14.1.2 and applicable vendor's manuals.

The TSW System shall be capable of providing cooling water to remove heat from the TCW System heat exchangers to maintain TCW heat exchanger outlet temperatures within the design limits identified in FSAR Subsection 9.2.16.1.2.

6.2 Liquid Radwaste System Performance

6.2.1 Purpose

To verify the proper operation of the various equipment and pathways which make up the Radioactive Drain Transfer System and the Liquid Radwaste System.

6.2.2 Description

Performance of the Liquid Radwaste System (e.g. low conductivity, high conductivity, and detergent waste subsystems) shall be verified throughout the Startup Test Program while at various steady state conditions. This test shall demonstrate that system operation in processing, storage and release of liquid radioactive waste is in accordance with design requirements. Additionally, all applicable system components shall be verified to function properly during various modes of plant operations.

6.2.3 Criteria

6.2.3.1 Level 1

The Liquid Radioactive Waste Management System shall be capable of collecting, processing, and controlling liquid wastes, as designed, such that releases of radioactive liquid effluents remain within the limits specified in the plant Technical Specifications or license conditions.

6.2.3.2 Level 2

The system operates as described in FSAR Subsection 9.3.8 as demonstrated by:

- Acceptable system and component flow paths and flow rates, including pump capacities and sump or tank volumes
- Proper operation of system pumps, valves, and motors under expected operating conditions
- Proper functioning of drains and sumps, including those dedicated to handling specific agents such as detergents

6.3 Gaseous Radwaste/Offgas System Performance

6.3.1 Purpose

To verify proper operation of the various components of the Gaseous Radwaste Management/Offgas System over the expected operating range of the system.

6.3.2 Description

The Gaseous Radwaste/Offgas System is designed to provide for hold up and decay of the radioactive gases in the off-gas from the Steam Jet Air Ejector System before discharge to the atmosphere. The Gaseous Radwaste/Offgas System will further minimize the release of the radioactive particulate matter into the atmosphere, and minimize the explosion potential in the off-gas system through recombination of radiolytic hydrogen and oxygen under controlled conditions.

Proper operation of the Gaseous Radwaste/Offgas System while at steady-state conditions shall be verified throughout the Startup Test Program for specific components such as catalytic recombiners and activated carbon adsorbers as well as various heaters, coolers, dryers and filters. Also to be evaluated are the piping, valving, instrumentation and control that comprise the overall system. Performance of the system shall be verified by monitoring parameters such as temperature, pressure, flow rate, humidity, hydrogen content, dewpoint, dilution steam flow, radiolytic gas production rate, and effluent radioactivity.

6.3.3 Criteria

6.3.3.1 Level 1

The release of radioactive gaseous and particulate effluents must not exceed the limits specified by the plant Technical Specifications or License conditions.

Flow of dilution steam to the non-condensing stage must be maintained at an amount no less than the low alarm setpoint when the steam jet air ejectors are operating.

6.3.3.2 Level 2

The system flow rate, temperature, humidity, and hydrogen concentration shall comply with Subsection 11.3.4 in all design modes of operations.

All applicable system components such as offgas preheater, offgas recombiner, offgas condensers, cooler condensers, refrigeration units, pre-filters, charcoal adsorbers, and offgas filters shall function properly during all design modes of operation (i.e., there shall be no gross malfunctioning of these components).

6.4 Steam and Power Conversion System Performance

6.4.1 Purpose

To demonstrate acceptable performance of the various plant steam-driven auxiliaries and power conversion systems under expected operational conditions, particularly that equipment that could not be fully tested during the preoperational phase due to inadequate steam flow conditions.

6.4.2 Description

Operation of steam-driven plant auxiliaries and power conversion systems will be monitored, and appropriate data collected, during the power ascension test phase to demonstrate that system operation is in accordance with design requirements. Systems to be monitored include the main turbine and generator and their auxiliaries, the feedwater heaters and moisture separator/reheaters, the main condenser and condenser evacuation system, and the main circulating water system. Operation and testing of power conversion systems is discussed in detail in FSAR Chapter 10. The main turbine generator and related auxiliaries are discussed in FSAR Section 10.2 and other power conversion equipment and systems are discussed in FSAR Section 10.4. Testing specific to turbine valves and the plant transient testing involving the main turbine generator are described in Subsection 4.

6.4.3 Criteria

6.4.3.1 Level 1

Not applicable.

6.4.3.2 Level 2

Each individual steam jet air ejector (SJAE) must be able to maintain the main condenser pressure within design limits during normal full load operation.

The Circulating Water (CW) System shall supply cooling water at a sufficient flow rate to condense the steam in the condenser, as required for optimum heat cycle efficiency.

The Feedwater Heater System shall heatup the reactor feedwater to a nominal temperature of 215.6°C during full load operation and to lower temperatures during part load operation.

The feedwater heater drains and vents system shall be capable of maintaining a water level in each of the LP/HP heaters and heater drain tanks within the normal operating limits during power operations.

The MSR shall maintain a balanced steam flow to the LP turbine and feedwater heaters during steady-state and transient operations in accordance with design requirements.

The main condenser shall be capable of maintaining the LP turbine exhaust conditions below the maximum allowable pressure and temperature.

The main condenser hotwell water level control system shall maintain the hotwell water level at nominal operating range during normal full load operation.

The turbine/generator shall be operated with a heat rate compatible with the design value during normal full load operations.

7.0 Startup Test Signal List

During the course of the Startup Test Program, requirements exist for recording and examining the values of many different types of parameters. The purposes of these recordings are:

a) To provide a permanent record of the performance tests,

b) To yield a recording of transients for quick comparison to the simulated predictions and to the test performance requirements, and

c) To provide helpful information for adjustments or replacement of malfunctioning components during the startup test program.

Table 4 lists the signals that are required (R) and the most useful (U) to be monitored by a Transient Recording and Analysis System (TRA) for each applicable startup test specified in this document. The TRA is a collection of functions that the test engineer will use to monitor a signal point or a set of signal points in the plant. Table 4 also identifies the startup tests that utilized the signal and provide a recommended sample plan.

The sample plan defines the rate at which the computer processes and records the sampled signal. The sampling rate necessary to evaluate a parameter must be specified in the sample plan for the associated signal. The value specified in the sample plan must be no greater than the sample rate described in the preceding discussion. The accuracy to which these signals must be measured is inherent in the specification of original equipment sensor from which the signal is obtained.

The signal list given in Table 4 is generic. The particular test requirements of each individual plant will lead to a revision of the list as it is applied to each plant unit.

8.0 Data Reduction Techniques

A Startup Transient Analysis Report (STAR) shall be prepared to predict plant response for the transients defined in the Startup Test Program. This analysis will use the beginning of cycle 1 "as loaded core" nuclear parameters, actual instrument setpoints, minimal expected limits of plant hardware required by Technical Specifications and design requirements. These results will provide a better estimate of expected plant response under the conditions at the time the testing is performed.

In general, correction factors will normally utilize a linear interpolation to correct the as tested condition to the assumed test condition for which results are being prepared.

When specific, complex calculations are necessary in order to establish the acceptable system performance (e.g. Core Flow Measurement), an engineering document will be generated (e.g. vendor manual, software application specification, etc.) to provide the equations and data reduction manipulations necessary to generate the needed parameter.

Control system tuning will require calculations to evaluate system performance and stability. Appendix A provides a description of the method used to determine Decay Ratio and other parameters related to performance.

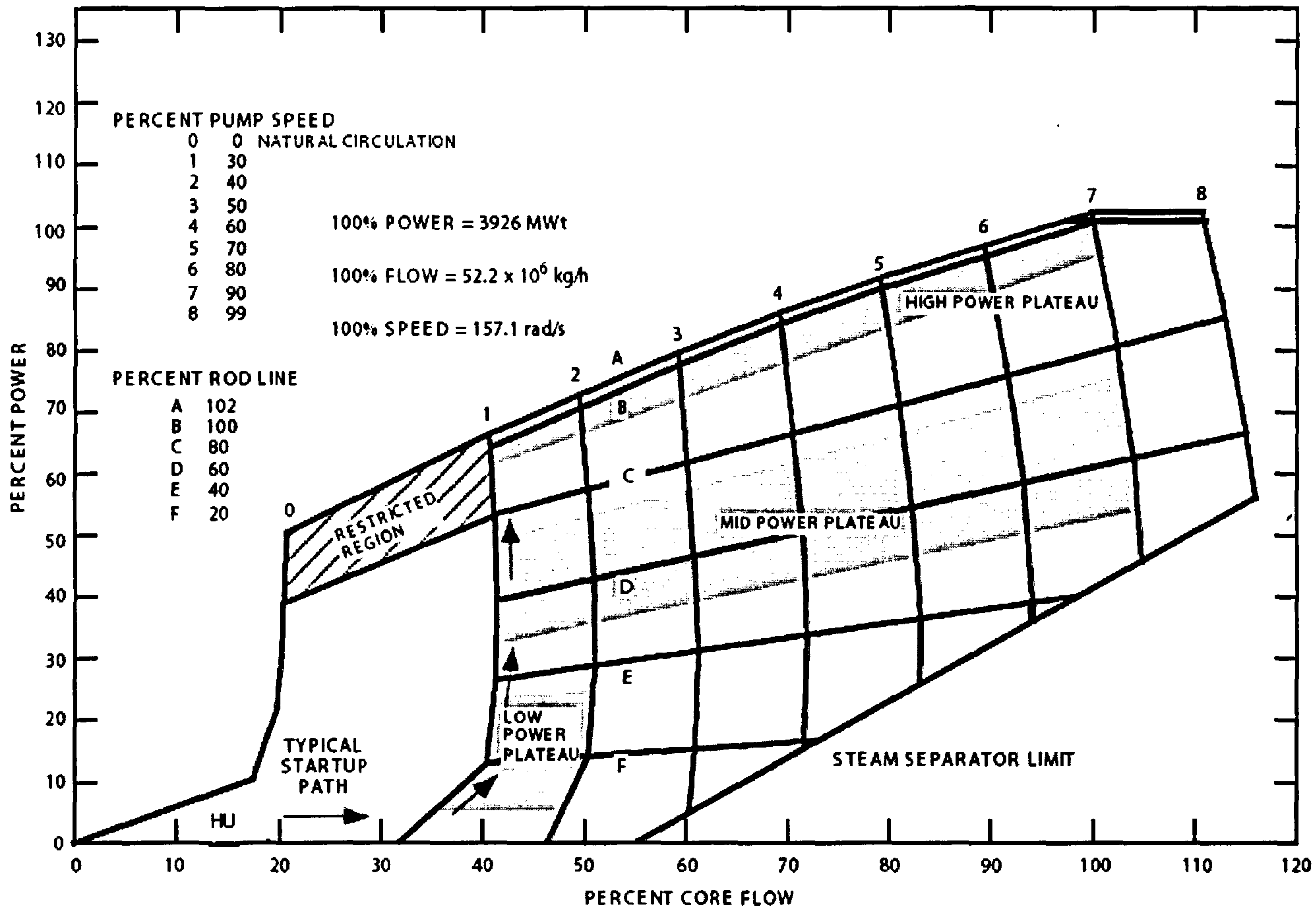


Figure 1 Power/Flow Operating Map

Table 1 Test Plateau Definitions

Test Plateau	Power Flow Map Region and Note
Open Vessel (**OV**)	With the RPV head removed, from initiation of fuel loading to cold conditions with a fully loaded core.
Heat-Up (**HU**)	During nuclear heat-up, from ambient conditions and 0 kPaG to rated temperature and pressure within the RPV, with reactor power typically less than 5% of rated.
Lower Power (**LP**)	Between 5% and 25% of rated thermal power, with the RIPs operating within 10% of minimum speed.
Mid Power (**MP**)	Between approximately 50% and 75% power rod lines, with the RIPs operating between minimum and rated speeds, with the lower power corner within the capacity of the bypass valves.
High Power (**HP**)	Along and just below (+0, -5%) the 100% power rod line, from minimum RIP speed to rated core flow.

Table 2 Detailed Description of Tests and Test Conditions

TEST NUMBER 1 – CHEMICAL & RADIOCHEMICAL MEASUREMENTS

Action	Test Conditions
1. Reactor water chemistry and radiochemistry.	A. Prior to fuel loading. B. During heatup. C. Steady state power operation at major power levels in LP, MP, and HP.
2. Process radiation monitor calibration.	A. Prior to fuel loading. B. During heatup. C. Steady state power operation at major power levels in LP, MP, and HP.
3. Condensate polisher measurements.	A. During heatup.
4. Gaseous and liquid effluents activity monitor.	A. During heatup. B. Steady-state power operation at major power levels in LP, MP, and HP.
5. CUW filter/demineralizer performance test (i.e., No Cleanup Test.)* (*Total steam quality measurement in Test. No. 21 will be performed in conjunction with this test at rated conditions.)	A. At high power and high flow corner of MP and HP defined in the Power-Flow operating map.

TEST NUMBER 2 – RADIATION MEASUREMENTS

Action	Test Conditions
1. Background radiation level surveys.	A. Prior to fuel loading and repeated prior to reaching initial criticality.
2. Complete standard radiation survey.	A. During heatup, approximately 50% power at MP and near 100% power at HP.
3. Limited standard radiation survey.	A. At approximately 15% power at LP, 25 - 35% power at MP, and 75% power at HP.

TEST NUMBER 3 – FUEL LOADING

Action	Test Conditions
1. Core loading.	A. Control rods, neutron sources and SRNM detectors installed, inspected and tested.
2. Subcriticality checks.	A. During fuel loading. B. Control rod, neutron sources, and SRNM detectors are installed and tested.
3. Control rod functional test. * (* Items 1, 2, 3 and 5 of Test 5 can be performed in conjunction with this test.)	A. During fuel loading. B. Control rods, neutron sources, and SRNM detectors are installed and tested.
4. Partial core shutdown margin demonstration.	A. During fuel loading. B. Control rod, neutron sources, and SRNM detectors are installed and tested.
5. Full core verification.	A. Fully loaded core.

TEST NUMBER 4 – FULL CORE SHUTDOWN MARGIN DEMONSTRATION

Action	Test Conditions
1. Full core shutdown margin demonstration at the beginning-of-cycle.	A. Xenon-free fully loaded core and all control rods in their full-in configuration.

TEST NUMBER 5 – CONTROL ROD DRIVE SYSTEM PERFORMANCE

Action	Test Conditions
1. Drive position checks at the completion of rod movement in step driving mode.	A. During fuel loading.
2. Drive position checks at the completion of selected rod movement in notch driving mode.	A. During reactor heatup to near rated reactor temperature and pressure.
3. Drive position checks at the completion of rod movement and the rated driving speed during continuous drive movement.	A. During approach to initial criticality and reactor heatup to rated reactor temperature and pressure.
4. Gang rod motion verification in response to RCIS (and APR during automatic rod movement) commands.	A. During reactor heatup to near rated reactor temperature and pressure.
5. Coupling checks by verifying actuation of the separation switches and confirmation of synchro position indication at the overtravel-out position.	A. During fuel loading.
6. Individual CRD friction tests.	A. During fuel loading , or B. After the completion of fuel loading.
7. Paired and unpaired CRD scram tests.	A. Prior to fuel loading; and B. After heatup to near rated reactor temperature.
8. Four slowest CRDs * scram tests. (* Refers to four CRDs selected for continuous monitoring based on slow scram times as determined in Action 6.A. The four selected CRDs must be compatible with requirements of both the withdrawal sequence and the rod movement limitation systems.)	A. At 4.14 MPaG* and 5.51 MPaG* pressure during reactor heatup. * ± 0.34 MPaG
9. Full core scram performance tests.	A. In conjunction with the planned full core scram tests (e.g., Test No. 29 at LP and Test No. 32 at HP).

TEST NUMBER 6 – NEUTRON MONITORING SYSTEM PERFORMANCE TESTS

6A – SRNM Performance and Control Rod Sequence

Action	Test Conditions
1. SRNM functional testing	A. Prior to and during fuel loading. B. Prior to initial criticality.
2. Adjust SRNM gains to ensure the continuity of the SRNM signals between counting and	A. During heatup from ambient to rated reactor temperature.

Action	Test Conditions
MSV flux ranges.	
3. Adjust SRNM gains, if necessary, for proper SRNM-APRM overlap.	A. After first APRM calibration based on a heat balance.
4. Verify SRNM response to rod withdrawal and confirm the adequacy of control rod withdrawal sequence.	A. Operational neutron sources installed. B. SRNM minimum count rate and minimum signal-to-noise ratio criteria satisfied. C. During approach to initial criticality after fuel loading, heatup from ambient to rated reactor temperature, and power ascension to LP.

6B – LPRM Calibration

Action	Test Conditions
1. Verify LPRM flux responses to the control rod movement or the ATIP predicted power shape.	A. During heatup when LPRMs on-scale.
2. Take data and calibrate LPRM system.	A. Approximately 20% power at LP, and repeated at high power /flow ends of power-flow map within MP and HP. B. All systems in NORM mode.

6C – APRM Calibration

Action	Test Conditions
1. Calibrate APRM system based on the CTP value determined by the average of the bypass valve power and the SRNM power calculations.	A. During heatup from ambient to rated reactor temperature.
2. Calibrate APRM system based on the CTP value determined by the normal heat balance calculation.	A. Approximately 25% power at LP, 50% & 75% power at MP, 100% power at HP and repeated as necessary.

6D – OPRM Surveillance

Action	Test Conditions
1. Record steady state neutron noise and OPRM operational data.	A. Below 60% core flow and above 70% rod line within MP. B. Near 80% core flow and near rated rod line within HP. C. Near rated core flow and near rated rod line within HP.

6E – ATIP Performance and Core Power Distribution

Action	Test Conditions
1. Perform ATIP system alignment.	A. Cold alignment at ambient temperature. B. Hot alignment after heatup to near rated temperature.
2. Verify ATIP system overall uncertainty.	A. Octant symmetric rod pattern. B. Approximately 50-75% power at MP and near 100% power at HP.

TEST NUMBER 7 – PLANT COMPUTER SYSTEM OPERATION

Action	Test Conditions
1. Computer/ATIP interface (Static System Test Case).	A. Reactor operation is not required.
2. Simulated dynamic input tests.	A. Reactor operation is not required.
3. ATIP alignment. * (* Action 1B of Test 6E may be performed in conjunction with this test.)	A. After heatup to near rated reactor temperature.
4. Dynamic System Test Case.	A. To be completed at major power levels within LP, MP and HP.
5. Obtain data.	A. Reactor power greater than 80% of rated. B. Core flow at 100% of rated. C. Sequence A control rod pattern. D. full core ATIP data processing functional check and LPRM calibration must be completed immediately prior to data taking.

TEST NUMBER 8 – CORE PERFORMANCE

Action

1. Evaluate core thermal power and core flow, and compute the thermal and hydraulic parameters associated with core behavior. Use plant computer system or manual calculations.

Test Conditions

A. During steady-state operation at major power levels of approximately 25% power at LP, 50% and 75% power at MP and near 100% power at HP; and

B. Additional points as necessary to assure compliance with plant Technical Specifications.

TEST NUMBER 9 – NUCLEAR BOILER PROCESS MONITORING TESTS

9A – Selected Process Temperatures

Actions

1. Monitor reactor coolant and bottom head drain temperatures.

Test Conditions

A. During rod withdrawal to criticality.

B. During and after heatup to near rated temperature and pressure.

C. Approximately 25% power at LP.

D. At 100% flow in MP and HP

E. After RIP trips and before RIP re-starts in Test No. 28.

F. After loss of T/G and offsite power in Test No. 30.

9B – Water Level Reference Leg Temperatures

Actions	Test Conditions
1. Monitor drywell and outside drywell temperatures.	A. Hot standby with steady drywell temperature.
	B. Approximately 25% power at LP, 50% power at MP and rated power and flow at HP
2. Record reactor vessel water level indications.	A. Hot standby with steady drywell temperature.
	B. Approximately 25% power at LP, 50% power at MP and rated power and flow at HP

9C – Core Flow Calibration

Actions	Test Conditions
1. Collect PDdP and CPdP characteristic data.	A. During the power ascension within MP via flow ramp along 75% Rod Line with 10 RIPs at minimum, 40%, 50%, 60%, 70%, and 80% speeds.
	B. During the power ascension within HP via flow ramp along 100% Rod Line with 10 RIPs at minimum, 40%, 50%, 60%, 70%, 90% and maximum speeds.
	C. With one RIP idled in Test No. 28A and three RIPs idled in Test No.28B.
2. Calibrate CPdP core flow by PDdP core flow.	A. Approximately 70 to 75% power and ≥ 95% of rated core flow at MP
	B. At rated power and rated flow in HP.
3. CPdP low cut setpoint and adequacy confirmation test	A. At low power and low flow conditions during power ascension to HP but after completing Action 2B.

TEST NUMBER 10 – SYSTEM EXPANSION

Action	Test Conditions
1. Visual inspection and hanger readings at accessible locations of the following system piping including Main Steam, SRV discharge, FW, RCIC, RHR, CUW, RPV head vent, RMC*, major piping inside drywell, the small branch piping attached to the above mentioned systems and Turbine Building Piping. * RMC piping test is not required if previously performed during preoperational testing	A. Baseline data at ambient temperature. B. At 1.03 MPaG (or 4.14 MPaG) reactor pressure during heatup, visual inspection only. C. At approximately 7.07 MPaG reactor pressure during heatup. D. During cold shutdown conditions after 3 to 5 complete heatup and cool down cycles.
2. Record displacement sensor readings on Main Steam and RCIC steam piping, selected SRV discharge line, FW lines, and RHR line B with CUW in service.	A. Baseline data at ambient temperature. B. Between ambient and approximately 7.07 MPaG reactor pressure at incremental pressure steps for a total of 3 complete heatup and cool down cycles C. At 20-25%, 50%, 75% and 100% of rated thermal power D. Repeat Item B for a total of 3 complete heatup and cooldown cycles
3. System walkdown and record displacement sensor readings in RCIC steam supply piping inside drywell. * (* This test may be performed in conjunction with Test No.22, i.e., RCIC performance test.)	A. Before the RCIC system is being placed in service. B. Steady-state operation with RCIC at 100% of rated flow. C. After the RCIC system is being removed from service and placed in standby.
4. System walkdown and record displacement sensor readings in CUW pump suction and discharge piping. * (* This test may be performed in conjunction with Test No.21, i.e., CUW performance test.)	A. At 20-30 °C steps up to rated system temperature while CUW is being placed in service. B. Steady-state operation with CUW at 100% of rated flow. C. Repeat Item 4.A while the tested CUW pump is being removed from service and placed in standby.
5. System walkdown, and record displacement sensor readings on the RHR pump suction and	A. At 20-30 °C steps up to rated system temperature while RHR loop is being placed in service.

discharge piping while RHR is placed in the shutdown cooling mode of operation in conjunction with RHR performance test (Test No. 19)	B. Steady-state operation while each RHR loop is operating at 100% of rated flow in shutdown cooling mode. A. While each RHR loop is being removed from service in shutdown cooling mode and placed in standby.
6. Record displacement sensor readings on the FW discharge piping inside and outside of containment to monitor the conditions and effects of a potential FW temperature stratification.	A. During reactor heatup B. At hot standby C. Following planned or unplanned reactor scrams D. During RCIC injection (Test No. 22). E. During reactor shutdown with RHR loop A being operated in shutdown cooling mode.

TEST NUMBER 11 – SYSTEM VIBRATION

Action	Test Conditions
1. Record Main Steam line, FW, RCIC steam lines, Turbine Building piping and selected SRV discharge line vibrations.	A. Approximately 25% NBR steam flow at LP, 50% and 75% NBR steam flow at MP and 100% NBR steam flow at HP. B. In conjunction with Test No.31, i.e., turbine trip and load rejection tests, at MP and HP. C. In conjunction with Test No.32, i.e., reactor full isolation test, at HP. D. In conjunction with Test No.25, i.e., SRV performance test, at LP. E. With RCIC turbine steam supply line at 100% of rated flow during Test No.22, i.e., RCIC System performance test, during heatup and LP.
2. Monitor and record vibration displacement sensor readings in RCIC steam supply and exhaust and pump suction and discharge piping. * (* This test may be performed in conjunction with Test No.22, i.e., RCIC System performance test.)	A. While RCIC is being placed in service. B. With RCIC turbine steam supply flow at 100% of rated. C. While RCIC is being removed from service and placed in standby.
3. Monitor and record vibration displacement sensor readings in CUW pump suction and discharge piping. * (* This test may be performed	A. While CUW pump is being placed in service B. With CUW pump at 100% of rated flow C. While tested CUW pump is being removed from service and placed in standby

in conjunction with Test No.21, i.e., CUW performance test.)

4. Monitor and record vibration displacement using remote sensors inside containment and hand held instrument and visual examination outside containment * (* This test may be performed in conjunction with Test No.19, i.e., RHR performance test.)	A. While RHR pump is being placed in shutdown cooling mode of operation. B. With tested RHR pump at 100% flow in shutdown cooling mode of operation. C. While tested RHR pump is being removed from shutdown cooling mode of operation and placed in standby.
5. Record vibration on FW discharge piping inside containment.	A. While FW pump is being placed in service. B. With tested FW pump at 100% flow. C. In conjunction with Test No.27, i.e., FW pump trip test, at HP. D. Hot standby following planned scram tests (e.g., Test No. 29, 30, 32) or unplanned scram events.
6. Record vibration on RMC piping.	A. In conjunction with RIP trip and recovery tests (e.g., Tests No. 28A and 28B).

TEST NUMBER 12 – RECIRCULATION FLOW CONTROL SYSTEM TESTS

12A – Individual RIP Speed Control

Actions	Test Conditions
1. 5 and 10% of rated frequency step input changes to adjust control system, if necessary, and demonstrate satisfactory response.	A. Approximately 25% power at LP; between 30 and 75% power at MP and between 60 and 100% power at HP with core flow initially at 100% rated. B. All RIPs in Individual Speed Control mode, and C. All other systems in NORM mode.

12B – Recirculation Flow Control

Actions	Test Conditions
1. 5 % of rated frequency demand step inputs to adjust control system and demonstrate satisfactory response.	A. Approximately 25% power at LP; between 30 and 75% power at MP and between 60 and 100% power at HP with core flow initially at 100% rated B. All RIPs in Gang Speed Control mode, and C. All other systems in NORM mode.

Actions	Test Conditions
2. 5% of rated core flow demand step inputs to adjust control system and demonstrate satisfactory response.* (* Perform negative step first if at or near rated core power initially.)	A. Approximately 25% power at LP; between 30 and 75% power at MP and between 60 and 100% power at HP with core flow initially at 100% rated. B. RFC in Core Flow Control mode. C. All other systems in NORM mode.
3. 5% and 10% of rated load demand and load ramp inputs to adjust control system, and demonstrate satisfactory response during load changing, maneuvering, daily load following, and ALF.	A. Between 65% and 75% power (40%-111% flow) at MP with core flow initially at 100% of rated. B. Repeat Action 3.A between 65-100% power (40-100% flow) on rated rod line at HP. C. RFC in ALF mode. D. All other systems in NORM mode.
4. Perform RIP ASD controller switchover test.	A. Initially at near rated flow and rated power conditions at HP . B. All 10 RIPs operating in Gang Speed Control mode, and all other systems in NORM mode. C. Repeat Item 4.A above with RFC in Core Flow Control mode.

TEST NUMBER 13 – FEEDWATER CONTROL SYSTEM

Actions	Test Conditions
1. Small and large step input changes to adjust control system, if necessary.	A. Approximately 5% power during heatup; between 5 and 25% power at LP; between 30 and 75% power at MP; and between 60% and 100% power at HP.
2. Small and large manual flow steps on each feedwater pump.	A. Between 30 and 75% power at MP and between 60 and 100% power at HP
	B. When two feedpumps are operating, with one feedpump in AUTO mode and the other feedpump to be tested in MAN mode.
	C. All RIPs Gang Speed Control mode or RFC in Core Flow Control mode.
3. Small and large level setpoint changes to demonstrate satisfactory responses.	A. Between 5 and 25% power at LP; between 30 and 75% power at MP and between 60 and 100% power at HP.
	B. FWC in 1-element and 3-element modes.
	C. All RIPs in Gang Speed Control mode or RFC in Core Flow Control mode, and
	D. All other systems in NORM mode.
4. Transfer of RFPs, LFCV and CUW blowdown valve in and out of service.	A. CUW blowdown valve off-line (> 2.86 MPaG, < 4.9 MpaG).
	B. RFP on-line with LFCV (< 4.9 MpaG).
	C. Lead RFP on-line LFCV off-line (< 20% power).
	D. Lag RFP on-line (> 20%, < 45% power).
	E. Lag RFP off-line (> 20%, < 45% power).
	F. LFCV on-line (< 20% power).
	G. CUW blowdown valve off-line (> 2.86 MPaG, < 4.9 MpaG).

TEST NUMBER 14 – PRESSURE CONTROL SYSTEM

Actions	Test Conditions
1. Pressure setpoint changes to adjust control system, if necessary.	A. Between 5 and 25% power at LP; between 30 and 75% power at MP and between 60 and 100% power at HP.
2. Pressure setpoint changes to demonstrate satisfactory response, TBVs in control.	A. Between 5 and 15% power before T/G synchronization at LP; between 30 and 75% power at MP and between 60% and 100% power at HP. B. All RIPs in Gang Speed Control mode, RFC in Core Flow Control mode and ALF mode.
3. Pressure setpoint changes to demonstrate satisfactory response, TCVs in control.	A. Between 30 and 75% power at MP and between 60 and 100% power at HP. B. All RIPs in Gang Speed Control mode, RFC in Core Flow Control mode and ALF* mode. (* Perform only downward pressure transient in ALF mode.)
4. Pressure setpoint changes to demonstrate satisfactory response, TBVs incipient, i.e., both TCVs and TBVs in control.	A. At approximately 70% power at MP. B. All RIPs in Gang Speed Control mode only.
5. Demonstrate fault tolerant capability by simulating failure consecutively of each regulator channel .	A. Between 5 and 15% power before T/G synchronization in LP, and B. Repeat Item 5 at 100% power in HP.

TEST NUMBER 15 – PLANT AUTOMATION AND CONTROL

Actions	Test Conditions
1. Plant startup operation using PGCS and Automatic Power Regulator (APR) function.	A. During heatup. B. APR in Criticality mode and Heatup/Standby mode. C. All other systems in NORM mode.
2. Power range operation using PGCS and APR function. * (* Portion of Test No. 12B to be performed in conjunction with this test)	A. Between 5 and 25% power at LP, between 30 and 75% power at MP, and between 60 and 100% power at HP. B. APR in Turbine Roll/Synch mode, and Power mode, and all RIPs in Gang Speed Control mode, RFC in Core Flow Control mode or ALF mode. C. All other systems in NORM mode.
3. Plant shutdown operation using PGCS and APR function.	A. In conjunction with plant shutdown and/or cooldown. B. APR in Shutdown mode, and Cooldown mode, and all RIPs in Individual Speed Control or Gang Speed Control C. All other systems in NORM mode.

TEST NUMBER 16 – REACTOR RECIRCULATION SYSTEM PERFORMANCE

Actions	Test Conditions
1. Record steady state operating data. (Portion of Test No. 9C can be done in conjunction with this test)	A. During the power ascension within MP via flow ramp along 75% Rod Line with RIPs at minimum, 40%, 50%, 60%, 70%, and 80% speeds. B. During the power ascension within HP via flow ramp along 100% Rod Line with RIPs at minimum, 40%, 50%, 60%, 70%, 90% and maximum speeds. C. All systems in NORM mode.
2. Record recirculation system responses to planned plant transients.	A. In conjunction with Test No. 28A and 28B, i.e., RIP trip tests at MP and HP. B. In conjunction with Test No.30, i.e., loss of T/G and offsite power at LP. C. In conjunction with Test No.31, i.e., turbine trip and load rejection tests at MP and HP. D. In conjunction with Test No.32, i.e., reactor full isolation, at HP. E. In conjunction with Test No.27, i.e., FW pump trip test at HP.

TEST NUMBER 17 – FEEDWATER SYSTEM PERFORMANCE

Actions	Test Conditions
1. Record steady state operating data.	A. Between 5 and 25% power at LP, between 30 and 75% power at MP, and between 60 and 100% power at HP. B. All systems in NORM mode.
2. Maximum runout flow determination: a. Record master controller output, FW pump suction, discharge, and reactor pressure, RFP speed, and actual locations of the high and low speed stops.	A. Four or more equally spaced FW flow points within MP, or any high power between 60 and100% of rated achieved within HP prior to commercial operation. B. Maximum number of Condensate System and FW pumps normally operated at 100% power shall be running, and C. All systems in NORM mode.
b. Determine sensitivity of FW flow to reactor pressure over a 206.8 kPa (30 psi) range and 34.5 kPa (5 psi) increments.	A. Approximately 80 to 90% power at HP. B. Maximum number of Condensate System and FW pumps normally operated at 100% power shall be running, and C. All systems in NORM mode.

TEST NUMBER 18 – MAIN STEAM SYSTEM PERFORMANCE TESTS

18A – Main Steam System Performance Data

Actions

1. Record steady state operating data.

Test Conditions

A. Between 40 and 75% power along 75% rod line within MP.

B. Between 60 and 100% power along 100% rod line within HP.

C. All systems in NORM mode.

18B – Main Steamline Flow Calibration

Actions

1. Record steady-state operating data while increasing and while decreasing power .

Test Conditions

A. Between 40 and 75% power along 75% rod line within MP.

B. Between 60 and 100% power along 100% rod line within HP.

C. All systems in NORM mode.

TEST NUMBER 19 – RHR SYSTEM PERFORMANCE

Actions	Test Conditions
1. Take heat exchanger capacity data. Demonstration of RHR system operation in shutdown cooling mode.	A. After trip or cool down from high power operation in HP in order to provide sufficient decay heat. B. RHR in shutdown cooling mode.
2. Take heat exchanger capacity data . Demonstration of RHR system operation in suppression pool cooling mode	A. In conjunction with Test No. 25, i.e., SRV performance, at LP, or B. After testing which adds heat to the suppression pool, for example, RCIC system performance (Test No.22), during HU or LP. C. RHR in suppression pool cooling mode. D. All other systems in NORM mode.

TEST NUMBER 20 – GUARANTEE PLANT PERFORMANCE WARRANTY RUN

Actions	Test Conditions
1. Demonstrate reactor power, steam pressure, steam quality and steam flow rate under steady-state conditions.	A. At conditions within HP prescribed in the Nuclear Steam System Warranty. B. Operate continuously for 100 hours. C. All systems in NORM mode.

TEST NUMBER 21 – REACTOR WATER CLEANUP SYSTEM PERFORMANCE

Actions	Test Conditions
1. Take steady-state system operating data.	A. Reactor at rated temperature and pressure during heatup. B. CUW in Normal Operation, Startup Operation, and Hot Standby Operation modes.
2. Take heat balance data for heat exchanger performance confirmation test.	A. Reactor at near rated temperature and pressure during heatup. B. CUW in rated operating conditions in Normal mode.
3. Take water samples at inlet and outlet of the filter demineralizer for filter demineralizer performance confirmation test.	A. Reactor at near rated temperature and pressure during heatup. B. CUW in rated operating conditions in Normal mode C. In conjunction with water quality confirmation test , i.e., portion of Test No.1.
4. Take temperature data at various locations on the RPV head during RPV head spray confirmation test.	A. Shutdown following major transients in LP and HP, as needed. B. CUW operating in RPV Head Spray mode.

TEST NUMBER 22 – RCIC SYSTEM PERFORMANCE

Actions	Test Conditions
1. RCIC SP Injection, Manual start.	A. At 1.03 MPaG reactor pressure during heatup[3].
	B. At near rated reactor pressure during heatup.
	C. RCIC discharge approximately 0.69 MPaG above reactor pressure.
	D. RCIC in Full Flow Test Mode.
	E. RFC and all other controllers in NORM mode for all RCIC testing.
2. RCIC SP Injection and turbine speed adjustments.	A. Immediately after Item 1.D above.
	B. RCIC in manual and automatic control modes.
3. RCIC SP Injection; extended operation demonstration.	A. In conjunction with Item 2 above.
	B. Operate RCIC continuously for 2 hours.
4. RCIC SP Injection, hot quick start followed by RCIC turbine speed adjustments.	A. At near rated reactor pressure during heatup.
	B. RCIC discharge approximately 0.69 MPaG above reactor pressure.

[3] (Optional demonstration prior to RCIC turbine speed adjustments per Item 4 below.)

TEST NUMBER 22 – RCIC SYSTEM PERFORMANCE (Continued)

5. Reactor vessel injection, manual start for RCIC turbine speed adjustments.	A. At near rated reactor pressure during heatup.
6. Reactor vessel injection, hot quick start.	A. At near rated reactor pressure during heatup (or LP as needed subsequent to reactor heatup).
7. Reactor vessel injection, hot or cold[4] quick start.	A. During heatup (HU) at 1.03 MPaG reactor pressure.
8. Confirmatory reactor vessel injection, cold* quick start.	A. At near rated reactor pressure during heatup (or LP as needed subsequent to reactor heatup).
9. Second consecutive confirmatory reactor vessel injection, cold* quick start.	A. Same as Item 8 above.
10. RCIC SP injection for surveillance test base data, cold* quick start.	A. At near rated reactor pressure during heatup (or LP as needed subsequent to reactor heatup); B. RCIC discharge approximately 0.69 MPaG above reactor pressure C. RCIC in full flow test mode D. At 1.03 MPaG reactor pressure during heatup. E. RCIC discharge approximately 0.69 MPaG above reactor pressure.

[4]Minimum of 72 hours without RCIC operation

TEST NUMBER 23 – TURBINE VALVE PERFORMANCE

Actions	Test Conditions
1. Individually close turbine stop valves.	A. At a power level between 45 and 55% of rated within MP. B. Along the 100% rod line, perform data gathering tests at 65% power, and again between 75 and 90% power, then perform final demonstration test at the desired maximum power level chosen for all subsequent surveillance tests. C. All other systems in NORM mode.
2. Individually close turbine control valves.	A. At a power level between 45 and 55% of rated within MP. B. Along the 100% rod line, perform data gathering tests at 65% power, and again between 75 and 90% power, then perform final demonstration test at the desired maximum power level chosen for all subsequent surveillance tests. C. All other systems in NORM mode.

TEST NUMBER 23 – TURBINE VALVE PERFORMANCE (Continued)

Actions	Test Conditions
3. Individually open turbine bypass valves.	A. At a power level between 45 and 55% of rated within MP. B. Along the 100% rod line, perform data gathering tests at 65% power, and again between 75 and 90% power, then perform final demonstration test at the desired maximum power level chosen for all subsequent surveillance tests. C. All other systems in NORM mode.
4. Measure the total capacity of turbine bypass valves.	A. In conjunction with Item 3 above and further verified in Test No.31 at HP.

TEST NUMBER 24 – MSIV PERFORMANCE

Actions	Test Conditions
1. Individually close each MSIV in the fast mode.	A. At rated temperature and pressure during initial heatup and within LP. B. All systems in NORM mode.
2. Close fastest MSIV, fast mode.	A. Along the maximum rod line, perform data gathering tests at 65% power, and again between 70 and 85% power. Perform third and final demonstration test at the desired maximum power level chosen for all subsequent Technical Specification surveillance tests.
3. Slow closure of individual MSIVs to 90% open.	A. To be done prior to Items 1 and 2 above, and repeated at maximum power determined in Action 2 above. B. All systems in NORM mode.

TEST NUMBER 25 – SRV PERFORMANCE

Actions	Test Conditions
1. Manually open each SRV for functional and flow demonstration tests, plant response, and valve reseating checks. Test No.11 to be done in conjunction with this test.	A. At reactor pressure ≥ 6.55 MPaG during reactor heatup, or after T/G synchronization within LP. B. In conjunction with reactor full isolation test (Test No. 32). C. RFC in Core Flow Control mode. D. All other systems in NORM mode.

TEST NUMBER 26 – LOSS OF FEEDWATER HEATING

Actions

1. Initiate single failure event that causes largest decrease in feedwater temperature.

Test Conditions

A. Between 80 and 90% power and core flow initially at near 100% rated within HP.

B. RFC in Core Flow Control mode.

C. All other systems in NORM mode.

TEST NUMBER 27 – FEEDWATER PUMP TRIP

Actions

1. Trip one of the normally operating RFP to demonstrate scram avoidance with the successful auto-start of Standby RFP and all operating RIP speed runback *

   (* RFP speeds must have been limited in accordance with Test No. 17.)

Test Conditions

A. Within HP Plateau.

B. All systems in NORM mode.

TEST NUMBER 28 – RIP TRIP TESTS

28A – One RIP Trip Test

Actions

1. Trip one RIP. *

   (* Vibration and strain measurements in Test No. 11 can be performed in conjunction with this test.)

Test Conditions

A. In MP at 95 % or more of rated core flow.

B. Within HP Plateau.

C. All RIPs in Gang Speed Control mode.

D. All other systems in NORM mode.

Actions	Test Conditions
2. Restart the tripped RIP. * (* Vibration and strain measurements in Test No. 11 can be performed in conjunction with this test)	A. Immediately following Item 1 within MP. Remaining 9 RIPs in the Gang Speed Control mode at minimum speed; all other systems in NORM mode. B. Immediately following Item 1 within HP. Remaining 9 RIPs with maximum speed determined in prior restart testing in MP; all RIPs in Gang Speed Control mode, and all other systems in NORM mode.

28B – Three RIPs Trip Test

Actions	Test Conditions
1. Trip three RIPs. * (* Vibration and strain measurements in Test 11 can be done in conjunction with this test).	A. In MP at 95% or more of rated core flow. B. Repeat Item 1 prior to turbine trip and load rejection test (Test No. 31) at HP. C. All systems in NORM mode. D. Water level may be lowered to avoid possible turbine trip.

TEST NUMBER 29 – SHUTDOWN FROM OUTSIDE THE MAIN CONTROL ROOM

Actions

1 To shutdown and cooldown an operating reactor from outside the main control room test.*

(* Test No. 5 is to be performed in conjunction with this test).

Test Conditions

A. Shutdown from a steady state power operation at approximately 10 to 25% power at LP. Cooldown portion may be demonstrated at a later test condition when sufficient decay heat is available.

B. Reactor initially critical.

C. T/G on line.

D. RFC in Core Flow Control mode, and

E. All other systems in NORM mode.

TEST NUMBER 30 – LOSS OF TURBINE /GENERATOR AND OFFSITE POWER

Actions

1. After transferring auxiliary loads to the Unit Auxiliary Transformer and starting main and feedwater turbine's DC oil pumps; disconnect both normal and alternate preferred offsite power sources and use trip relays to trip main generator.

Test Conditions

A. Approximately 10% to 20% of rated power at LP.

B. T/G on line.

C. All RIPs in Individual Speed Control mode.

D. All other systems in NORM mode.

TEST NUMBER 31 – TURBINE TRIP AND LOAD REJECTION

Actions	Test Conditions
1. Manually trip main turbine by depressing T/B trip pushbutton. * (* Vibration and strain measurements in Test No. 11 are to be done in conjunction with this test).	A. At < 40% power within MP. B. At high end of MP with 95% or more of rated core flow. C. At 100% power within HP. D. All systems in NORM mode.
2. Initiate load rejection by opening main switchyard breakers. * (* Calibration and strain measurements in Test No. 11 are to be done in conjunction with this test).	A. At high end of MP with 95% or more of rated core flow. B. At 100% power within HP. C. All systems in NORM mode.

TEST NUMBER 32 – REACTOR FULL ISOLATION

Actions	Test Conditions
1. Close all MSIVs. * (* Test No. 5, and the vibration and strain measurements in Test No. 11 are to be done in conjunction with this test).	A. At 100 % power and prior to the 100% Turbine /Generator trip test (Test No. 31) within HP. B. All systems in NORM mode.

TEST NUMBER 33 – LOOSE PARTS MONITORING SYSTEM BASELINE DATA

Actions	Test Conditions
1. Collect initial baseline data.	A. At 5% power during heatup. B. Approximately 20% power in LP. C. At four or more equally spaced flow points between minimum and maximum core flow along 95% to 100% rod line within HP. D. LPMS in manual mode. E. All other systems in NORM mode .
2. Establish setpoints for channel alarm functions and automatic activation mode.	A. At 100% power within HP. B. All systems in NORM mode.

TEST NUMBER 34 – STEAM SEPARATOR/DRYER PERFORMANCE TEST

Actions	Test Conditions
1. Insert rods gradually and determine the moisture carryover in the exit steam at each power reduction step, until either the steam separator limit line as shown on the Power/Flow map is reached or excessive moisture carryover is determined, whichever occurs first. * (* Actual runback to minimum speeds of all operating RIPs may not be required during this test).	A. Power level just above the steam separator limit line within MP, with normal reactor water level B. 10 RIPs at maximum permissible speeds with RFC in Core Flow Control mode, and C. All other systems in NORM mode.

TEST NUMBER 35 – CONCRETE PENETRATION TEMPERATURE SURVEY

Actions	Test Conditions
1. Monitor concrete wall temperature surrounding the selected high energy piping penetrations.	A. At 5% power during heatup. B. Approximately 20% power at LP, 50% power at MP, and 100% power at HP. C. All systems in NORM mode.

TEST NUMBER 36 – REACTOR BUILDING COOLING/SERVICE WATER SYSTEMS PERFORMANCE

Actions	Test Conditions
1. Take RCW/RSW heat exchanger capacity data during steady-state power operation.	A. At 5% power during heatup. B. Approximately 20% power at LP, 50% power at MP, and 100% power at HP. C. All systems in NORM mode
2. Take RCW/RSW heat exchanger capacity and system operational data when RHR heat exchanger(s) are in operation.	A. In conjunction with RHR System Performance tests (Test No. 19) during RHR suppression pool cooling mode of operation, and B. In conjunction with shutdown following planned or unplanned transients. C. All systems in NORM mode.

TEST NUMBER 37 – HVAC SYSTEM PERFORMANCE

Actions	Test Conditions
1. Record pressure, humidity and temperature readings in the rooms and areas served by the RBHV system.	A. At 5% power during heatup. B. Approximately 50% power at MP, 100% power at HP and repeat in spaces where equipment is operated during a major plant transient test at HP. C. All systems in NORM mode.
2. Record pressure, humidity and temperature readings in the rooms and areas served by the CBHV system.	A. At 5% power during heatup. B. Approximately 50% power at MP, 100% power at HP and repeat in spaces where equipment is operated during a major plant transient test at HP. C. All systems in NORM mode.
3. Record pressure, humidity and temperature readings in the rooms and areas served by the TBHV system.	A. At 5% power during heatup. B. Approximately 50% power at MP, 100% power at HP and repeat in spaces where equipment is operated during a major plant transient test at HP. C. All systems in NORM mode.

Turbine Building Cooling/Service Water Systems Performance

Actions	Test Conditions
1. Take TCW/TSW heat exchanger capacity data during steady-state power operation.	A. At 5% power during heatup. B. Approximately 20% power at LP, 50% power at MP, and 100% power at HP. C. All systems in NORM mode.

Liquid Radwaste System Performance

Actions	Test Conditions
1. Collect data on system flow rates, tank levels and process temperatures during normal operation in support of plant startup.	A. At 5% power during heatup. B. Approximately 20% power at LP, 50% power at MP, and 100% power at HP. C. During outage periods.

Gaseous Radwaste System Performance

Actions	Test Conditions
1. Collect data on system process parameters during system startup and at normal operations.	A. When system is placed in service and at 5% power during heatup. B. Approximately 20% power at LP, 50% power at MP, and 100% power at HP.

Steam and Power Conversion System Performance

Actions	Test Conditions
1. Collect data on system parameters during system startup and at normal operations.	A. When components are placed in service and at 5% power during heatup. B. Approximately 20% power at LP, 50% power at MP, and 100% power at HP.

Table 3 Control System Operating Modes

Master Controller	Core Flow Controller	Individual / Gang Selection Station	Description of the Operational Mode
Manual	Manual	Individual	**Individual RIP Speed Control Mode** The individual RIP speed command is from the control signal at the Individual/Gang Selection Station.
Manual	Manual	Gang	**Gang Speed Control Mode** All RIPs (in the Gang mode) speed command is from the manual control signal at the [core flow controller]
Manual	Automatic	Gang	**Core Flow Control Mode** The core flow control command is from the manual control signal at the master controller
Automatic	[Automatic]	Gang	**Automatic Load Following Mode** The load demand command is from the APR load demand error input

<u>Notes:</u>

1. The pressure control system has only a single operating mode, and therefore, is generally not mentioned as a unique condition under the Detailed Description of Test and Test Plateaus, Table 2.
2. The Feedwater Control System shall be in the 3-element mode during all tests, except as specifically noted in a portion of the feedwater controller testing.
3. Unless otherwise specified, the Recirculation Flow Control System must be in the Core Flow Control mode (or Master Manual mode) for all startup tests.
4. In all cases, the abbreviation "NORM" indicates that the reactor systems are in the mode that is appropriate to the power and flow conditions of the test.
5. Since operating modes of the Recirculation Flow Control System can vary from test to test, a description of each mode is given above.

Table 4 Startup Test Signal List

Signal Name or Source Required (R) or Useful (U)	Number of Signals	Sample Plan (sample/sec)	Test Numbers for Signal to be Recorded
APRM Neutron Flux (R)	4	25	6C, 12; 13; 14; 15; 22; 23; 24; 25; 26; 27; 28; 29; 30; 31; 32; 34
LPRM Flux (R)	208	10	6B; 6D; 23; 24; 26; 29; 30; 31; 32
Simulated Fuel Surface Heat Flux (R)	4	10	12; 13; 14; 22; 23; 24; 25; 26; 27; 28; 29; 30; 31; 32
SRNM Neutron Flux (R)	10	25	12; 13; 14; 15; 22; 25; 29; 30
SRNM Period (R)	10	25	12; 13; 14; 15; 22; 25; 29; 30
Control Rod Positions (R)	205	10	5; 29; 32
Reactor Scram (R)	4	20	24; 26; 27; 28; 29; 30; 31; 32
Total Isolation (R)	4	20	24; 29; 30; 31; 32
ADS Initiation (U)	4	20	29; 30; 31; 32
MSIV Inboard and Outboard Status or Positions (R)	8	20	15; 24; 29; 30; 32;
SRV Status (R)	18	20	15; 25; 29; 30; 31;32
All-Rods-In (U) (At Scram Test Panel)	1	100	15; 29; 32
Main Steam Line Flow (R)	4	10	18; 23; 24; 29; 30; 31; 32
Main Steam Line Flow Nozzle DP (R)	4	10	18; 23; 24
Total Reactor Steam Flow (R)	1	10	12; 13; 18; 23; 24; 27; 28; 29; 30; 31; 32
Wide Range Reactor Dome Pressure (R)	3	40	14; 15; 22; 29; 30; 31; 32

Table 4 Startup Test Signal List (Continued)

Signal Name or Source Required (R) or Useful (U)	Number of Signals	Sample Plan (sample/sec)	Test Numbers for Signal to be Recorded
Narrow Range Reactor Pressure (R)	3	40	14; 15; 22; 23; 24; 25; 27; 28; 29; 30; 31;.32
Wide Range Reactor Water Level (R)	1	10	13; 14; 15; 27; 28; 29; 30; 31; 32
Narrow Range Reactor Water Level (R)	3	10	13; 14; 15; 27; 28; 29; 30; 31; 32
Upset Range Reactor Water Level (U)	1	10	29; 30; 31; 32
Moderator Temperature (R)	1	5	9; 15; 16; 26; 28A; 28B
RCIC Turbine Supply Steam Inlet Pressure (U)	1	5	22
RCIC Turbine Exhaust Pressure (U)	1	5	22; 30; 32
RCIC Pump Discharge Pressure (U)	1	5	22; 30; 32
RCIC Pump Flow (R)	1	20	22; 29; 30; 32
RCIC Initiation (R)	1	20	22; 30; 32
RCIC Turbine Control Valve Positions (U)	1	20	22; 30; 32
RCIC Turbine Steam Admission Valve Positions	1	20	22; 30; 32
RCIC Steam Line DP (R)	4	20	22; 30; 32
RCIC Turbine Speed (U)	1	20	22; 30; 32
RHR System Flow (U)	3	20	15; 19
RHR Pump Discharge Pressure (U)	3	20	15; 19

Table 4 Startup Test Signal List (Continued)

Signal Name or Source Required (R) or Useful (U)	Number of Signals	Sample Plan (sample/sec)	Test Numbers for Signal to be Recorded
B/C HPCF Initiation (U)	2	10	29; 30; 32
B/C HPCF Flow (U)	2	10	29; 30; 32
B/C HPCF Pump Discharge Pressure (U)	2	10	29; 30; 32
B/C HPCF Injection Valve Position (U)	2	10	29; 30; 32
Main Condenser Vacuum (U)	3	10	15; 23; 29; 30; 31; 32
Generator Breaker Position (R)	1	10	15; 29; 30; .31; 32
Main Switchyard Breaker Position (R)	2	10	15; 29; 30; 31; 32
Main Turbine Trip (R)	1	100	15; 23; 29;.30; 31; 32
Generator Trip (U)	1	100	15; 29; 30; 31; 32
OPC Signal	1	100	15; 29; 30; 31; 32
Individual Turbine Governor Valve Positions (R)	4	100	12; 14; 15; 23; 29; 30; 31; 32
Total Turbine Governor Valve Positions (R)	1	100	12; 14; 15; 23; 29; 30; 31; 32
Individual Turbine Stop Valve Positions (Analog) (R)	4	100	12; 14; 15; 23; 29; 30; 31; 32
Individual Turbine Bypass Valve Positions (R)	10	100	12; 14; 15; 23; 29; 30; 31; 32
Total Turbine Bypass Valve Position (U)	1	100	12; 14; 15; 23; 29; 30; 31; 32
ICV Valve Positions (U)	6	100	15; 29; 30; 31; 32
Main Turbine Speed (U)	1	100	14; 15; 29; 30; 31; 32
Turbine Steam Flow (U)	1	25	14; 15; 25; 30

Table 4 Startup Test Signal List (Continued)

Signal Name or Source Required (R) or Useful (U)	Number of Signals	Sample Plan (sample/sec)	Test Numbers for Signal to be Recorded
Turbine Speed Setpoint (U)	3	25	14; 15; 30; 31;
Turbine Inlet Pressure (U)	1	25	14; 15; 29; 30; 31; 32
Main Generator Gross MWe (U)	1	25	12; 14; 15; 25; 29; 30; 31; 32
Main Generator Voltage (U)	1	25	12; 14; 15; 25; 29; 30; 31; 32
Main Generator Current (U)	1	25	12; 14; 15; 29; 30; 31; 32
Main Generator Frequency (U)	1	25	12; 14; 15; 29; 30; 31; 32
Total Turbine Steam Flow Demand (R)	2	50	14; 15; 31; 32
Turbine Bypass Valve Servo Current (U)	3	25	14; 15; 30; 31; 32
Turbine Bypass Valve Position Error (U)	3	25	14; 15; 30; 31; 32
Total Steam Bypass Demand (U)	3	10	14; 15; 30; 31; 32
Limited Speed Regulator Output	3	10	14; 15; 31; 32
Turbine First Stage Pressure (U)	3	25	14; 15; 30; 31
Pressure Setpoint (R)	3	50	12; 14; 15; 31
Pressure Setpoint Adjuster Output (U)	3	50	12; 14; 15; 31
Pressure Setpoint Adjuster Feed Forward Output (U)	3	50	12; 14; 15; 31
Load Limit (U)	1	5	12; 14; 15; .31

Table 4 Startup Test Signal List (Continued)

Signal Name or Source Required (R) or Useful (U)	Number of Signals	Sample Plan (sample/sec)	Test Numbers for Signal to be Recorded
Load Demand (U)	1	5	12; 14; 15; 23; 31
Load Demand Error (U)	1	5	12; 14; 15; 31
Load Set (R)	1	5	12; 14; 15; 23; 24; 31
Pressure Regulator Output (R)	3	25	12; 14; 15; 31
Core Plate Differential Pressure (R)	4	5	9; 12; 16; 28; 31
Pump Deck Differential Pressure (R)	4	5	9; 12; 16; 28; 31
Validated Core Flow (U)]	1	10	9; 12; 16; 28; 24; 30; 31; 32
CPdP Core Flow (R)	4	40	9; 12; 15; 16; 23; 24; 28; 29; 30; 31; 32;34
RIP MG Set Voltage (U)	2	20	12; 15; 16; 27; 28; 30; 31; 32; 34
RIP MG Set Current (U)	2	20	12; 15; 16; 27; 28; 29; 30; 31; 32; 34
RIP MG Set Speed (U)	2	5	12; 15; 16; 27; 28; 29; 30; 31; .32; 34
RIP Frequency Demand (U)	10	10	12; 14; 15; 16; 27; 28; 29; 30; 31; 32; 34
RIP Speed Demand (U)	1	10	12; 14; 15; 27; 28; 29; 30; 31; 32; 34
RIP Gang Speed Controller Demand (U)	1	10	12; 14; 15; 27; 28; 29; 30; 31; 32; 34
Core flow Demand (U)	1	10	12; 14; 15; 28; 29; 31; 32; 34
APR Thermal Power Controller Output (U)	1	10	15; 26; 28; 31; 32; 34

Table 4 Startup Test Signal List (Continued)

Signal Name or Source Required (R) or Useful (U)	Number of Signals	Sample Plan (sample/sec)	Test Numbers for Signal to be Recorded
APR Generator Power Controller Output (U)	1	10	15; 26; 28; 31; 32; 34
APR to RCIS Insert (U)	1	10	15; 26; 28; 31; 32; 34
APR to RCIS Withdraw (U)	1	10	15; 26; 28; 31; 32; 34
RIP ASD Operational Status (R)	10	20	12; 15 .26; 27; 28; 29; 30; 31; 32; 34
RIP ASD Frequency Demand Ringback (R)	10	10	12; 15; 16; 7; 28; 29; 31; 32; 34
SCADA Power Demand (U)	1	10	15; 26; 28; 31; 32; 34
Load Demand Error to APR (U)	1	10	15; 26; 28; 31; 32; 34
RIP ASD Current (R)	10	20	9; 12; 15; 16; 27; 28; 29; 31; 32; 34
RIP ASD Voltage (R)	10	20	9; 12; 15; 16; 27; 28; 29; 31; 32; 34
RIP Speed (R)	10	20	9; 12; 15; 16; 27; 28; 29; 30; 31; 32; 34
RIP Circuit Breaker Position (R)	10	20	15; 28; 29; 30; 31; 32
RPT Initiation (R)	4	10	29; 31; 32
SCRRI Initiation (R)	2	2	15; 26; 28; 31
Fast Load Winddown Initiation (R)	1	20	31
Individual FW Pump Flow (R)	4	10	13; 15; 27; 28; 29; 30; 31; 32
Total FW Flow (R)	1	10	13; 15; 18; 27; 28; 29; 30; 31; 32
FW Line Flow to Vessel (U)	2	10	18; 20
FW Line Flow to Vessel Venturi DP (U)	2	10	18; 20

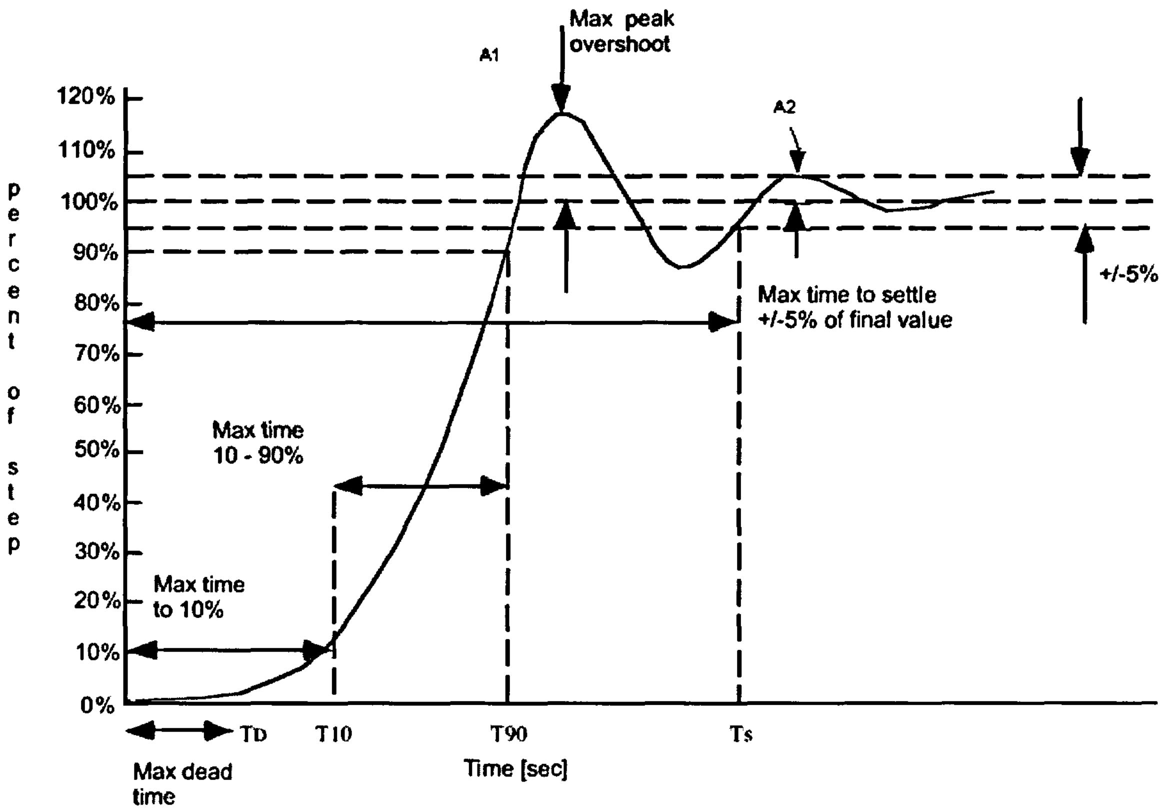
Table 4 Startup Test Signal List (Continued)

Signal Name or Source Required (R) or Useful (U)	Number of Signals	Sample Plan (sample/sec)	Test Numbers for Signal to be Recorded
FW Line Temperature (R)	2	5	26
MS/FW Flow Mismatch (U)	1	25	13; 27; 28; 29; 30; 31; 32
Individual RFP Turbine Speed (U)	3	25	13; 15; 27;28; 29; 30; 31; 32
RFP Start/Trip Status (R)	3	50	13; 15; 27; 28; 29; 30; 31; 32
RFP Trip Signal (R)	3	50	13; 15; 27; 28; 29; 30; 31; 32
CUW Dump FCV Position (U)	1	10	13; 15
Combined LFCV Position (U)	1	10	13; 15; 27; 28; 29; 30; 31 32
RFP Suction Pressure (U)	4	10	27; 28; 29; 32
RFP Discharge Pressure (U)	4	10	17; 27; 28; 29; 32
Level Setpoint (R)	1	10	13; 15; 27; 28; 29; 30; 31; 32
Master Level Controller Output (U)	1	10	13; 15; 27; 28; 29; 30; 31; 32
FW Loop Flow Controller Output (U)	4	10	13; 15; 27; 28; 29; 30; 31; 32
FW Master Flow Controller Output (U)	1	10	13; 15; 27; 28; 29; 30; 31; 32
RFP Position Demand (U)	1	10	13; 15; 27; 28; 29; 30; 31; 32

Table 4 Startup Test Signal List (Continued)

Signal Name or Source Required (R) or Useful (U)	Number of Signals	Sample Plan (sample/sec)	Test Numbers for Signal to be Recorded
RFP Speed Demand (R)	3	10	13; 15; 27; 28; 29; 30; 31; 32
RFP Function Generator Output (U)	3	10	13; 15; 27; 28; 29; 30; 31; 32
CUW Dump Flow Controller Output (R)	1	10	13; 15
Bus Breaker Positions for Auxiliary Power (U)	[3]	20	30
Manual Event Marker (R)	1	[100]	as required
CTG Start Relay (R)	1	20	30
CTG Generator Voltage (U)	1	20	30
CTG Generator Current (U)	1	20	30
CTG Frequency (R)	1	20	30
CTG Power (R)	1	20	30
CTG Breaker Position (U)	1	20	30
EDG Start Relay (R)	3	20	30
EDG Generator Voltage (U)	3	20	30
EDG Generator Current (U)	3	20	30
EDG Frequency (R)	3	20	30
EDG Power (R)	3	20	30
EDG Breaker Position (U)	3	20	30
PIP Bus Power (U)	3	20	30
PIP Bus Voltage (U)	3	20	30
ESF Bus Power (U)	3	20	30
ESF Bus Voltage (U)	3	20	30

Performance Measurements



Decay Ratio

Decay Ratio = A2/A1

Appendix B, Initial Fuel Loading Master Checklist

		Signature of Responsible Person
1.	Required preoperational testing completed.	______
2.	Operating procedures required for initial loading issued.	______
3.	Initial loading test procedures issued.	______
4.	Communications between Control Room Operator and Refueling Platform established.	______
5.	Dunking chambers installed in vessel (if necessary) and connected (non-coincidence) to SRNM System; checks complete.	______
6.	Personnel restrictions established for Control Room, Refueling Floor, Drywell, and Reactor Building.	______
7.	Initial loading source installed in vessel.	______
8.	Checkout of underwater lights completed.	______
9.	Layout boards of core and fuel storage areas provided as required on refueling floor and in Control Room.	______

Appendix B

INITIAL FUEL LOADING MASTER CHECK LIST

		Signature of Responsible Person
10.	Sign provided at entrance to reactor cavity specifying clothing and working regulations.	______________
11.	Refueling floor access control and checkout station provided.	______________
12.	Water chemistry procedures in effect.	______________
13.	Radiation protection procedures in effect.	______________
14.	Refueling operators qualified for use of fuel handling equipment.	______________
15.	Required consumable materials available.	______________
16.	Preventative maintenance program in effect.	______________
17.	Surveillance program in effect.	______________
18.	Special test equipment for initial loading and low power test provided.	______________
19.	Required logbooks and data sheets provided.	______________

Appendix B

INITIAL FUEL LOADING
MASTER CHECK LIST

Signature of Responsible Person

20. Fuel accountability procedures and forms in use. ____________

21. Shift routine system in effect. ____________

22. Valves and instruments status up to date. ____________

23. Fuel handling and reactor servicing tool storage provided on Refueling Floor. ____________

24. Spare parts storage organized and catalogued. ____________

25. Checklists of systems required for fuel loading complete. ____________

26. Cold functional test document is signed off. ____________

27. Final review by Management completed. ____________

28. Compliance authority permit to load issued.

Appendix C, Surveillance Tests Required Prior To and During Initial Fuel Loading

After Reactor Building access control is established and prior to initial fuel loading, the Surveillance Program will be initiated on those systems listed below so that, at the time of initial loading, these systems will have received the applicable portions of the following general functional testing procedures that have been written to cover all requirements of the Technical Specifications.

Long interval surveillance items, such as primary and secondary containment leak testing, which have been satisfied by successful preoperational testing prior to Reactor Building access restrictions, will not be repeated prior to initial loading.

Surveillance Requirements

Applicable Mode	Procedure Title/Content of Test	Frequency	SR No.
5	Control Rod Not Full-in Indication Verification	CR not full-in	3.9.4.1
2, 5	SRNM Sensor Channel Check	12 Hours	3.3.1.1.1
1, 2, 3, 4, 5	Reactor Vessel Water Level 8, 3, 2, 1.5, and 1 Sensor Channel Check	12 Hours	3.3.1.1.1
1, 2, 3, 4, 5	Drywell Pressure High Sensor Channel Check	12 Hours	3.3.1.1.1
1, 2, 5	CRD Charging Water Header Pressure Low Sensor Channel Check	12 Hours	3.3.1.1.1
1, 2, 3, 4, 5	Condensate Storage Tank Level Low Sensor Channel Check	12 Hours	3.3.1.1.1
1, 2, 3, 4, 5	Suppression Pool Water Level High Sensor Channel Check	12 Hours	3.3.1.1.1
1, 2, 3, 5	Reactor Building Area Exhaust Air Radiation High Sensor Channel Check	12 hours	3.3.1.1.1
1, 2, 3, 5	Fuel Handling Area Exhaust Air Radiation High Sensor Channel Check	12 hours	3.3.1.1.1
1, 2, 3, 4, 5	RCW/RSW Heat Exchanger Room Water Level High Sensor Channel Check	12 hours	3.3.1.1.1
1, 2, 5	Seismic Activity High Sensor Channel Check	12 hours	3.3.1.1.1

Applicable Mode	Procedure Title/Content of Test	Frequency	SR No.
1, 2, 3, 4, 5	Low Pressure Core Flooder Actuation Sensor Channel Check	12 Hours	3.3.1.4.1
1, 2, 3, 4, 5	High Pressure Core Flooder Actuation Sensor Channel Check	12 Hours	3.3.1.4.1
1, 2, 3, 4, 5	Diesel Generator Actuation Sensor Channel Check	12 Hours	3.3.1.4.1
1, 2, 3, 4, 5	RSW/RCW Actuation Sensor Channel Check	12 Hours	3.3.1.4.1
2, 3, 4, 5	Startup Range Neutron Monitor Sensor Channel Check	12 Hours	3.3.2.1.1
5	Startup Range Neutron Monitor Operability Verification	12 Hours	3.3.2.1.2
5 2	SRNM Count Rate Verification	12 Hours 24 Hours	3.3.2.1.3
4, 5	Suppression Pool Water Lever Verification for LPFL	12 Hours	3.5.2.1
4, 5	Suppression Pool Water Level and CST Water Level Verification for HPCF	12 Hours	3.5.2.2
5	Reactor Mode Switch Position Verification	12 Hours	3.9.2.1
5	Control Rod Full-in Verification	12 Hours	3.9.3.1
5	RHR SDC In-service Verification	12 Hours	3.9.7.1, 3.9.8.1
1, 2, 3, 5	CHRA EF Unit Radiation Monitors Channel Check	24 Hours	3.3.7.1.1
1, 2, 3, 5	Secondary Containment Vacuum Verification	24 Hours	3.6.4.1.1
1, 2, 3, 4, 5	RSW Pump Intake Structure Level Verification	24 Hours	3.7.1.1 3.7.2.1
1, 2, 3, 4, 5	RSW Water Temperature at RCW Heat Exchanger Inlet Verification	24 Hours	3.7.1.2 3.7.2.2
5	Reactor Cavity water Level Verification	24 Hours	3.9.6.1
1, 2, 5	RPS Manual Scram Channel Functional Test	7 Days	3.3.1.2.1
5, 2, 3, 4	SRNM Channel Functional Test	7 Days	3.3.2.1.4 3.3.2.1.5
3, 4, 5	Reactor Coolant Temperature Monitoring Channel / RHR SDC Channel Check	7 Days	3.3.8.2.1

Applicable Mode	Procedure Title/Content of Test	Frequency	SR No.
5	Fuel Pool Water Level Verification	7 Days	3.7.8.1
1, 2, 3, 4, 5	Offsite Power Breaker Alignment Verification	7 Days	3.8.1.1 3.8.11.1
1, 2, 3, 4, 5	Battery Voltage Verification	7 Days	3.8.4.1 3.8.5.1
1, 2, 3, 4, 5	Battery Pilot Cell Parameters Verification	7 Days	3.8.6.1
1, 2, 3, 4, 5	Inverter Voltage, Frequency, and Alignment Verification	7 Days	3.8.7.1 3.8.8.1
1, 2, 3, 4, 5	Distribution Systems Breaker Alignment Verification	7 Days	3.8.9.1 3.8.10.1
5	Refueling Equipment Channel Functional Test	7 Days	3.9.1.1
5	Refueling Mode Rod-out Channel Functional Test	7 Days	3.9.2.2
5	Withdrawn Control Rod Exercise	7 Days	3.9.5.1
5	Withdrawn Control Rod Accumulator Pressure Verification	7 Days	3.9.5.2
1, 2, 5	SRNM Division Functional Test	31 Days	3.3.1.1.4
1, 2, 3, 4, 5	LPFL Keep Fill Subsystem and Valve Lineup Verification	31 Days	3.5.1.1, 2 3.5.2.3, 4
1, 2, 3, 4, 5	HPCF Keep Fill Subsystem and Valve Lineup Verification	31 Days	3.5.1.1, 2 3.5.2.3, 4
1, 2, 3, 4, 5	RCIC Keep Fill Subsystem and Valve Lineup Verification	31 Days	3.5.1.1, 2 3.5.2.3, 4
1, 2, 3, 4, 5	RSW/RCW Valve Lineup Verification	31 Days	3.7.1.3 3.7.2.3
11, 2, 3, 4, 5	Turbine Control Valve Functional Test Verification	31 Days	SIL 413
1, 2, 3, 4, 5	Diesel Generator Operability Demonstration Test	31 Days 184 Days	3.8.1.2, 3 3.8.11.1
1, 2, 3, 4, 5	Diesel Generator Day Tank Fuel Oil Verification and Water Removal	31 Days	3.8.1.4, 5 3.8.11.1
1, 2, 3, 4, 5	Diesel Generator Fuel Oil Storage Tank Fuel Oil Inventory and Properties Verification	31 Days	3.8.3.1 3.8.3.3
1, 2, 3, 4, 5	Diesel Generator Lube Oil Inventory Verification	31 Days	3.8.3.2

Applicable Mode	Procedure Title/Content of Test	Frequency	SR No.
1, 2, 3, 4, 5	Diesel Fuel Oil Property Analyses	(31 Days) DFOTP	3.8.3.3
1, 2, 3, 4, 5	Diesel Generator Air Start Receiver Pressure Verification	31 Days	3.8.3.4
1, 2, 3, 4, 5	Fuel Oil Storage Tank Accumulated Water Removal	31 Days	3.8.3.5
1, 2, 3, 4, 5	Reactor Water Levels 8, 3, 2, 1.5, and 1 Division Functional Test	92 Days	3.3.1.1.5
1, 2, 5	CRD Charging Water Header Pressure Low Division Functional Test	92 Days	3.3.1.1.5
1, 2, 3, 4, 5	Suppression Pool Water Temperature High Division Functional Test	92 Days	3.3.1.1.5
1, 2, 3, 4, 5	Condensate Storage Tank Level Low Division Functional Test	92 Days	3.3.1.1.5
1, 2, 3, 4, 5	Suppression Pool Water Level High Division Functional Test	92 Days	3.3.1.1.5
1, 2, 3, 5	Reactor Building Area Exhaust Air Radiation High Division Functional Test	92 Days	3.3.1.1.5
1, 2, 3, 5	Fuel Handling Area Exhaust Air Radiation High Division Functional Test	92 Days	3.3.1.1.5
1, 2, 3, 4, 5	RSW/RCW Heat Exchanger Room Water Level High Division Functional Test	92 Days	3.3.1.1.5
1, 2, 5	Seismic Activity High Division Functional Test	92 Days	3.3.1.1.5
1, 2, 5	RPS Actuation Channel Functional Test	92 Days	3.3.1.2.2
1, 2, 3, 4, 5	Low Pressure Core Flooder Actuation Division Functional Test	92 Days	3.3.1.4.3
1, 2, 3, 4, 5	High Pressure Core Flooder Actuation Division Functional Test	92 Days	3.3.1.4.3
1, 2, 3, 4, 5	Diesel Generator Actuation Division Functional Test	92 Days	3.3.1.4.3
1, 2, 3, 4, 5	RSW/RCW Actuation Division Functional Test	92 Days	3.3.1.4.3
1, 2, 3, 4, 5	RHR Suppression Pool Cooling Subsystem Actuation Division Functional Test	92 Days	3.3.1.4.3
2, 3, 5	RHR Shutdown Cooling Subsystem Actuation Division Functional Test	92 Days	3.3.1.4.3
1, 2, 3, 4, 5	EMS Operability Verification	92 Days	3.3.3.1.1

Applicable Mode	Procedure Title/Content of Test	Frequency	SR No.
1, 2, 3, 5	CRHA MACE Radiation Monitor Channel Functional Test	92 Days	3.3.7.1.2
1, 2, 3, 5	CRHA EF Unit Flow Low Channel Functional Test	92 Days	3.3.7.1.2
1, 2, 3, 5	CRHA EF Unit Manual Switch Channel Functional Test	92 Days	3.3.7.1.2
1, 2, 3, 4, 5	Electrical Protection Assembly Channel Functional Test	92 Days	3.3.8.1.1
1, 2, 3, 4, 5	Electrical Protection Assembly Channel Calibration	92 Days	3.3.8.1.2
3, 4, 5	Reactor Coolant Temperature Monitoring - Shutdown Channel Functional Test	92 Days	3.3.8.2.2
1, 2, 3, 4, 5	Low Pressure Core Flooder Flow Rate Test	92 Days	3.5.1.4 3.5.2.5
1, 2, 3, 4, 5	High Pressure Core Flooder Flow Rate Test	92 Days	3.5.1.4 3.5.2.5
1, 2, 3, 4, 5	Fuel Oil Transfer System Operability Test (3.8.2.1)	92 Days	3.8.1.6 3.8.112.1
1, 2, 3, 4, 5	Battery Connector and Terminal Visual Inspection	92 Days	3.8.4.2 3.8.5.1
1, 2, 3, 4, 5	Battery Cell Parameters Verification	92 Days	3.8.6.2
1, 2, 3, 4, 5	Battery Electrolyte Temperature Verification	92 Days	3.8.6.3
1, 2, 3, 4, 5	Diesel Generator Cold Quick Start Test	184 Days 18 Months	3.8.1.7 3.8.11.1
1, 2, 3, 4, 5	Battery Cells, Cell Plates, and Rack Conditions Visual Inspection	12 Months	3.8.4.3 3.8.5.1
1, 2, 3, 4, 5	Battery Visual Inspection and Conditions Annual Maintenance	12 Months	3.8.4.4 3.8.5.1
1, 2, 3, 4, 5	Battery Connection Resistance Verification	12 Months	3.8.4.5 3.8.5.1
1, 2, 5	SRNM Comprehensive Functional Test Comprehensive Functional Test	18 Months	3.3.1.1.9
1, 2, 3, 4, 5	Reactor Water Level 8, 3, 2, 1.5, and 1 Comprehensive Functional Test	18 Months	3.3.1.1.9
1, 2, 3, 4, 5	Condensate Storage Tank Level Low Instrumentation Comprehensive Functional	18 Months	3.3.1.1.9
1, 2, 3, 4, 5	Suppression Pool Water Level High Instrumentation Comprehensive Functional	18 Months	3.3.1.1.9

Applicable Mode	Procedure Title/Content of Test	Frequency	SR No.
5	Reactor Building Area Exhaust Air Radiation High Comprehensive Functional Test	18 Months	3.3.1.1.9
5	Fuel Handling Area Exhaust Air Radiation High Comprehensive Functional Test	18 Months	3.3.1.1.9
1, 2, 3, 4, 5	RSW/RCW HX Room WL High Comprehensive Functional Test	18 Months	3.3.1.1.9
1, 2, 5	Seismic Activity High Comprehensive Functional Test	18 Months	3.3.1.1.9
1, 2, 3. 4, 5	Reactor Water Levels 8, 3, 2, 1.5, and 1 Instrumentation Sensor Channel Calibration	18 Months	3.3.1.1.1 0
1, 2, 3, 4, 5	Condensate Storage Tank Level Low Sensor Channel Calibration	18 Months	3.3.1.1.1 0
1, 2, 3, 4, 5	Suppression Pool Water Level Sensor Channel Calibration	18 Months	3.3.1.1.1 0
5	RX Building Area Exhaust Air Radiation High Sensor Channel Calibration	18 Months	3.3.1.1.9
5	FH Area Exhaust Air Radiation High Sensor Channel Calibration	18 Months	3.3.1.1.9
1, 2, 3,4, 5	RSW/RCW HX Room Water Level High Instrumentation Sensor Channel Calibration	18 Months	3.3.1.1.1 0
1, 2, 5	Seismic Activity High Sensor Channel Calibration	18 Months	3.3.1.1.1 0
During Fuel	Reactor Building Area Exhaust Air Radiation High Isolation Initiation Response Time Test	18 Months	3.3.1.1.1 4
During Fuel	Fuel Handling Area Exhaust Air Radiation High Isolation Initiation Response Time Test	18 Months	3.3.1.1.1 4
1, 2, 5	Reactor Mode Switch Shutdown Position Comprehensive Functional Test	18 Months	3.3.1.2.4
1, 2, 5	Manual MSIV Actuation Comprehensive Functional Test	18 Months	3.3.1.2.4
1, 2, 5	RPS Actuation Output Channel Functional Test	18 Months	3.3 1.2 5
1, 2, 5	RPS Actuation Response Time Test	18 Months	3.3 1.2 6
1, 2, 3, 4, 5	Low Pressure Core Flooder Actuation Output Channel Functional Test	18 Months	3.3.1.4.2
1, 2, 3, 4, 5	High Pressure Core Flooder Actuation Output Channel Functional Test	18 Months	3.3.1.4.2
1, 2, 3, 4, 5	Diesel Generator Actuation Output Channel Functional Test	18 Months	3.3.1.4.2

Applicable Mode	Procedure Title/Content of Test	Frequency	SR No.
1, 2, 3, 4, 5	RSW/RCW Actuation Output Channel Functional Test	18 Months	3.3.1.4.2
1, 2, 3, 4, 5	RHR SPC Actuation Output Channel Functional Test	18 Months	3.3.1.4.2
1, 2, 3, 4, 5	Low Pressure Core Flooder Actuation Comprehensive Functional Test	18 Months	3.3.1.4.4
1, 2, 3, 4, 5	High Pressure Core Flooder Actuation Comprehensive Functional Test	18 Months	3.3.1.4.4
1, 2, 3, 4, 5	Diesel Generator Actuation Comprehensive Functional Test	18 Months	3.3.1.4.4
1, 2, 3, 4, 5	RSW/RCW Actuation Comprehensive Functional Test	18 Months	3.3.1.4.4
1, 2, 3, 4, 5	RHR SPC Actuation Comprehensive Functional Test	18 Months	3.3.1.4.4
1, 2, 3, 4, 5	Low Pressure Core Flooder Actuation Response Time Test	18 Months	3.3.1.4.5
1, 2, 3, 4, 5	High Pressure Core Flooder Actuation Response Time Test	18 Months	3.3.1.4.5
1, 2, 3, 4, 5	Diesel Generator Actuation Instrumentation Response Time Test	18 Months	3.3.1.4.5
1, 2, 3, 4, 5	RSW/RCW Actuation Instrumentation Response Time Test	18 Months	3.3.1.4.5
1, 2, 3, 4, 5	Low Pressure Core Flooder Actuation Instrumentation Sensor Channel Calibration	18 Months	3.3.1.4.6
1, 2, 3, 4, 5	High Pressure Core Flooder Actuation Sensor Channel Calibration	18 Months	3.3.1.4.6
1, 2, 3, 4, 5	Diesel Generator Actuation Sensor Channel Calibration	18 Months	3.3.1.4.6
1, 2, 3, 4, 5	RSW/RCW Actuation Sensor Channel Calibration	18 Months	3.3.1.4.6
1, 2, 3, 4, 5	Low Pressure Core Flooder Manual Initiation Channel Functional Test	18 Months	3.3.1.4.7
1, 2, 3, 4, 5	High Pressure Core Flooder Manual Initiation Channel Functional Test	18 Months	3.3.1.4.7
1, 2, 3, 4, 5	Diesel Generator Manual Initiation Channel Functional Test	18 Months	3.3.1.4.7
1, 2, 3, 4, 5	RSW/RCW Manual Initiation Channel Functional Test	18 Months	3.3.1.4.7
1, 2, 3, 4, 5	RHR SPC Manual Initiation Channel Functional Test	18 Months	3.3.1.4.7

Applicable Mode	Procedure Title/Content of Test	Frequency	SR No.
2, 3, 4, 5	SRNM Channel Calibration	18 Months	3.3.2.1.6
1, 2, 3, 4, 5	EMS Comprehensive Network Performance Test	18 Months	3.3.3.1.2
4, 5	Mode Switch in Shutdown Position Channel Functional Test	18 Months	3.3.5.1.5
1, 2, 3, 5	CRHA MAC Radiation Monitor Channel Calibration	18 Months	3.3.7.1.3
1, 2, 3, 5	CRHA EF Unit Flow Low Channel Calibration	18 Months	3.3.7.1.3
1, 2, 3, 5	CRHA MACE Radiation Monitor Logic System Functional Test	18 Months	3.3.7.1.4
1, 2, 3, 5	CRHA EF Unit Flow Low Logic System Functional Test	18 Months	3.3.7.1.4
1, 2, 3, 5	CRHA EF Unit Manual Switch Logic System Functional Test	18 Months	3.3.7.1.4
1, 2, 3, 4, 5	Electric Protection Assembly System Functional Test	18 Months	3.3.8.1.3
3, 4, 5	Reactor Coolant Temperature Monitoring for RHR SDC Channel Calibration	18 Months	3.3.8.2.3
1, 2, 3, 4, 5	LPFL Automatic Initiation Test	18 Months	3.5.1.7 3.5.2.6
1, 2, 3, 4, 5	HPFC Automatic Initiation Test	18 Months	3.5.1.7 3.5.2.6
1, 2, 3, 4, 5	RSW/RCW Automatic Initiation Test	18 Months	3.7.1.4 3.7.2.4
3, 4, 5	Diesel Generator Load Rejection Test	18 Months	3.8.1.9, 10
4, 5	Loss of Offsite Power Test	18 Months	3.8.1.11 3.8.11.1
3, 4, 5	Integrated ECCS Test	18 Months	3.8.1.12 3.8.11.1
4, 5	Diesel Generator Protective Device Bypass Test	18 Months	3.8.1.13 3.8.11.1
4, 5	Diesel Generator to Offsite Power Transfer Test	18 Months	3.8.1.16 3.8.11.1
4, 5	Diesel Generator Test Mode Override Test	18 Months	3.8.1.17 3.8.11.1
4, 5	Loss of Offsite Power / Integrated ECCS Test	18 Months	3.8.1.18, 19

Applicable Mode	Procedure Title/Content of Test	Frequency	SR No.
4, 5	Battery Charger Capacity Verification	18 Months	3.8.4.6 3.8.5.1
4, 5	Battery Capacity for Emergency Loads Test Verification	18 Months	3.8.4.7 3.8.5.1
4, 5	Battery Performance Discharge Test Capacity Verification	60 Months 12 Months	3.8.4.8 3.8.5.1
1, 2, 3, 4, 5	Fuel Oil Storage Tank Cleanup	10 Years	3. 8.3.6