TABLE OF CONTENTS

TABLE OF CONTENTS

LIST OF TABLES

LIST OF TABLES

LIST OF TABLES

LIST OF TABLES

LIST OF TABLES

LIST OF TABLES

LIST OF TABLES

LIST OF TABLES

Section **Title**

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LIST OF FIGURES

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14.0 INITIAL TEST PROGRAM

14.1 SPECIFIC INFORMATION TO BE INCLUDED IN PRELIMINARY SAFETY ANALYSIS REPORT

This section is not required for the Final Safety Analysis Report; see Section 14.2 for applicable information.

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14.2 TEST PROGRAM

14.2.1 Summary of Test Program and Objectives

The purpose of the initial test program for the Watts Bar Nuclear Plant (WBNP) is to assure that the installed plant structures, systems, and components will be subjected to tests as required to verify that the plant has been properly designed and constructed and is ready to operate in a manner that will not endanger the health and safety of the public, and to provide assurance of total plant reliability for operation. The test program will also ensure, to the extent practical, that the procedures for operating the plant safely have been evaluated and demonstrated and that the operating organization is knowledgeable about the plant and procedures.

The necessary procedures to control, implement, and document the test program are established by startup and plant administrative procedures and summarized in the following sections. Tennessee Valley Authority (TVA), as the applicant, has responsibility for overall direction and management of the test program.

The initial test program is divided into two phases, the preoperational test phase and the power ascension test phase. Preoperational phase testing will be performed prior to fuel load and power ascension testing will be performed during and following fuel loading activities. During each of these two phases, tests will be performed to verify design requirements of safety-related and selected non-safety-related components, systems, and structures. A graded approach, based on criteria provided in Regulatory Guide 1.68, Revision 2, will be used for selection of plant structures, systems, components, and design features to be included in the initial test program. During the preoperational testing phase, three types of tests will be performed: (1) component tests, (2) preoperational tests, and (3) acceptance tests.

Component (preliminary) tests will be performed on safety-related and non-safety-related components and consist of items such as instrument calibration, flushing, cleaning, and functional tests of individual components to demonstrate conformance with design requirements. Component testing will prepare individual components for system level testing. Verification that appropriate component testing is complete is prerequisite to performance of preoperational (system level) tests as described in Section 14.2.12.1.

Preoperational test instructions will be performed on safety-related and selected non-safety-related structures, systems, components, and design features as required to demonstrate their capability to perform in accordance with design requirements. Preoperational tests are intended to demonstrate the proper operation of system design features through integrated operation of components under normal and transient conditions, where practical, including system and component interactions. Such tests also demonstrate the capability of certain component design features for which system operation is required to establish the necessary test conditions. Summaries of preoperational test instructions are provided in Table 14.2-1.

Acceptance test instructions will be performed on components and systems which do not perform a safety-related function and are not required for safe shutdown and cooldown of the reactor under normal or upset conditions as described in the applicable sections of the FSAR. Acceptance tests are not summarized in Table 14.2-1.

Components and systems which are not tested in accordance with a preoperational or acceptance test instruction will be tested and/or placed into service in accordance with procedures which are appropriate for the installation.

During the power ascension testing phase, power ascension tests, surveillance instructions, and other permanent plant tests and technical instructions will be performed to demonstrate satisfactory operation of systems. Summaries of power ascension testing to be performed during the initial test program are provided in Table 14.2-2.

Regulatory guidance will be used for development of initial test program requirements as discussed in Section 14.2.7.

Tests summarized in Table 14.2-1 will be completed and test results approved prior to commencing fuel load. Tests, or portions thereof, which can not be completed prior to fuel load will be evaluated to assure incomplete tests will not adversely affect fuel loading operations or cause features that have not been tested to be relied upon for safe plant operation. The Joint Test Group (JTG) and the Plant Operations Review Committee (PORC) or the Test Review Group (TRG) will review the technical justification for delaying test completion until after fuel load. If approved by the Plant Manager, the technical justification and schedule, including power level for completion of delayed testing, will be provided to the NRC staff prior to fuel load.

Power ascension tests will be performed beginning with activities leading to fuel loading and ending with full power operation. The intent of these tests is to assure that fuel loading is effected in a safe manner; that tests deferred from the preoperational test phase are completed satisfactorily; that the plant is safely brought to rated capacity; that plant performance is satisfactory in terms of established design criteria; and to demonstrate, where practical, that the plant is capable of withstanding anticipated transients and postulated accidents. Testing activities related to fuel load and initial criticality are further described in Section 14.2.10.

Tests will be conducted in accordance with approved test procedures. Review, approval, and revision of test procedures and the evaluation and disposition of test results will be accomplished by methods specified in the appropriate administrative procedures summarized in Section 14.2.3. Preoperational and power ascension tests are discussed in Section 14.2.12.

The initial test program will utilize, to the extent practical, operations personnel and operating procedures to provide familiarization with the plant installation and demonstrate the adequacy of operating procedures.

Preoperational testing activities will be coordinated through a Joint Test Group (JTG), as described in Section 14.2.2.5. Power ascension testing activities will be coordinated through the TRG, as described in Section 14.2.2.6.

14.2.2 Organization and Staffing

The Director, WBN Unit 2, has the overall responsibility for the initial test program through the preoperational test phase. He administratively reports to the Senior Vice President, Watts Bar Operations and Construction.

The component, acceptance, and preoperational testing will be performed under the direction of the Preoperational Startup Manager by the Preoperational Startup Engineering Organization.

The Site Vice President has the responsibility for the initial Unit 2 fuel load and the power ascension test phase of the initial test program. He administratively reports to the Senior Vice President, WBN Operations and Construction.

The power ascension testing will be performed under the direction of the Power Ascension Test Manager by the Power Ascension Test Group.

14.2.2.1 Preoperational Startup Engineering

The Preoperational Startup Engineering (PSE) organization consists of management personnel, Startup Test Engineers, and support personnel necessary to conduct preoperational test phase activities.

The Preoperational Startup Manager reports to the Completions and Startup Manager, and is responsible for management of the PSE Organization and for preoperational test-related activities. Other nontest functions of the Preoperational Startup Manager are described in the Startup Manual Procedures (SMPs). The Joint Test Group chairman reports to the Preoperational Startup Manager as do the Test Group Supervisors. The Preoperational Startup Manager may serve as Chairman on the JTG.

Responsibilities of the Preoperational Startup Manager, and Startup Test Engineers are provided below. Responsibilities of the Joint Test Group (JTG) are described in Section 14.2.2.5.

14.2.2.1.1 Preoperational Startup Manager

The Preoperational Startup Manager is responsible for the overall management of the PSE organization including coordination and implementation of component, acceptance, and preoperational test activities. The responsibilities of the Preoperational Startup Manager include:

(1) Development, approval, and implementation of the WBN SMPs;

(2) Selection of Preoperational Startup Representatives for the JTG;

(3) Development of plans and schedules for component, acceptance, and preoperational testing activities;

(4) Analysis of system completion schedules for compatibility with testing schedules and implementation of corrective actions to minimize conflicts;

(5) Submittal of proper and timely notifications and reports (pertaining to component and preoperational testing activities) to the Nuclear Regulatory Commission and other regulatory agencies;

(6) Assuring proper review of preoperational test instructions and results;

(7) Managing the overall development and conduct of individual component, acceptance, and preoperational test procedures;

(8) Assuring testing activities are conducted in accordance with the Startup Manual Procedures and applicable WBN administrative procedures;

(9) Coordinating test program activities and requirements with appropriate Engineering, Construction, Operations, and Maintenance and,

(10) Providing technical direction to Startup Test Engineers and others assigned to the PSE.

14.2.2.1.2 Startup Test Engineers

Startup Test Engineers are members of the Preoperational Startup organization and report through a Test Group Supervisor to the Preoperational Startup Manager. Their duties and responsibilities include:

(1) Preparation of assigned test instructions which direct and guide specific tests in accordance with a standard format;

(2) Performance of component, acceptance, and preoperational tests;

(3) Direction of support personnel during performance of tests including appropriate interface with plant operators;

(4) Ensuring the safety of personnel and plant equipment during testing;

(5) Familiarization of support personnel with specific tests;

(6) Identification of deficiencies that could adversely affect test performance;

(7) Assembly of test data and preparation of test reports for evaluation by others; and,

(8) Authority to disallow or terminate testing due to conditions which could endanger personnel or equipment.

14.2.2.1.3 Unit 2 Operations During Preoperational Testing Phase

The Operations Manager is responsible for proper operation of all equipment and ensuring that the conduct of test program does not place the plant in an unsafe condition. This manager will provide personnel from the operating organization, as required, to support the conduct of testing activites.

The Unit Supervisor/Senior Reactor Operator (US/SRO) report to the Operations Manager and are responsible for the safe operation of the plant during assigned shifts. They also are responsible for the implementation of appropriate clearance procedures and have the authority to disallow or terminate testing due to conditions which could endanger personnel or equipment.

14.2.2.1.4 Unit 2 Quality Assurance

Unit 2 Quality Assurance will conduct activities in accordance with Chapter 17 of the FSAR.

14.2.2.2 Plant Operating Organization

The Nuclear Power Group (NPG) Plant Operating organization is described in Chapter 13. The Plant Manager is responsible for management of the power ascension test program and directly reports to the WBN Vice President.

The WBN operating staff will be involved in the test program in several capacities throughout the initial test program. This involvement will include review of test procedures and results and the direct participation of operating personnel in test activities. Plant operators will assist test engineers in performing tests and will take over the routine operations of systems when authorized by Preoperational Startup Manager. The operating staff will direct fuel loading and will be responsible for operation and testing of the plant during the power ascension test phase.

14.2.2.2.1 Power Ascension Test Manager

The Power Ascension Test Manager reports to the Plant Manager and is responsible for supervision of power ascension testing personnel. The responsibilities of the Power Ascension Test Manager for the power ascension testing program include:

(1) Coordination and direction of power ascension testing and related activities;

(2) Development and implementation of plans and schedules for the power ascension test phase;

(3) Development of power ascension test procedures;

(4) Assuring proper review of test procedures and results;

(5) Assuring proper and timely notifications and reports pertaining to power ascension testing activities are submitted to the Nuclear Regulatory Commission and other regulatory agencies; and,

(6) Performance of the power ascension test sequence to ensure a safe and orderly power ascension program.

14.2.2.2.2 Power Ascension Test Engineers

Power Ascension Test Engineers report to the Power Ascension Test Manager and will be responsible for the conduct and direction of tests during the power ascension test phase. Their duties and responsibilities include:

(1) Preparation of assigned test procedures to direct and guide specific tests;

(2) Performance of power ascension tests;

(3) Direction to support personnel and others during performance of tests including appropriate interface with plant operators;

(4) Ensuring the safety of personnel and plant equipment during testing;

(5) Familiarization of support personnel with specific tests;

(6) Identification of deficiencies that could adversely affect test performance;

(7) Assembly of test data and preparation of test reports for evaluation by others; and,

(8) Authority to disallow or terminate testing due to conditions which could endanger personnel or equipment.

14.2.2.2.3 Operations

The Operations Manager is responsible for the proper operation of all equipment and for ensuring that the conduct of the test program does not place the plant in an unsafe condition. He will provide personnel from the operating staff as required to support the conduct of testing activities.

The US/SRO report to the appropriate Operations Manager (via the Operations Superintendent) and are responsible for the safe operation of the plant during assigned shifts. They also are responsible for the implementation of appropriate clearance tagging procedures and have authority to disallow or terminate testing due to conditions which could endanger personnel or equipment.

14.2.2.3 Site Quality Assurance

Site Quality Assurance will conduct activities in accordance with Chapter 17 of the FSAR.

14.2.2.4 Major Participating Organizations

14.2.2.4.1 Nuclear Engineering

WBN Nuclear Engineering (Unit 1 Design, Unit 2 Design, Reactor and Systems) will be responsible for assisting, to the extent required, in ensuring that tests sufficiently verify the adequacy of system design. This responsibility includes:

(1) Review of preoperational test procedures (as member of Joint Test Group (JTG)) and power ascension test procedures (as member of TRG) to assure test objectives and acceptance criteria comply with design, license commitments, and consistency with the plant safety analyses;

(2) Providing all design information necessary to ensure that detailed test procedures correspond to WBN systems; and,

(3) Review and approval of completed preoperational test results (as member of JTG) and power ascension test results (as member of TRG).

14.2.2.4.2 Construction

The Construction Manager is responsible for construction completion, performance of associated construction tests, and orderly completion of components and systems consistent with the test program schedules. This responsibility includes:

(1) Completion of construction and construction testing activities; and,

(2) Providing craft technical manpower support as required for performance of the initial test program.

14.2.2.4.3 Westinghouse Electric Corporation

Westinghouse, as the Nuclear Steam Supply System (NSSS) supplier, is responsible for providing technical direction to TVA-WBN during preoperational and power ascension testing. Technical direction is defined as technical guidance, advice, and counsel based on current engineering, installation, and testing practices. This responsibility includes:

(1) Assignment of qualified personnel to provide advice and assistance to WBN for testing and operation of all equipment and systems in the Westinghouse area of responsibility as required;

(2) Assignment of an operational physicist to the site organization during fuel loading and low power testing; and

(3) Providing test guidelines for preoperational tests and power ascension tests of Westinghouse furnished components and systems.

14.2.2.5 Joint Test Group

The Joint Test Group (JTG), functioning as an advisory group to the Preoperational Startup Manager, is responsible for reviewing assigned preoperational testing activities and advising the Preoperational Startup Manager on the disposition of those items reviewed. The primary function of the JTG is the review and recommendation for approval of preoperational test procedures, test instruction revisions, and test results. The normal chairman of the JTG is the Preoperational Startup Manager; the alternate chairmen of the JTG report to the Preoperational Startup Manager.

14.2.2.5.1 JTG Membership

The JTG will be composed of one representative from each of the organizations listed below. Representatives of other organizations will participate as requested by the JTG Chairman.

PSE
Operations
Unit 2 Site Quality Assurance
Unit 2 Design Engineering

Westinghouse (for all matters concerning preoperational testing performed on the NSSS and associated auxiliary systems)

JTG members and their alternates, including the Chairman, will be designated in writing with the concurrence of the Completions and Startup Manager.

14.2.2.6 Test Review Group

The Test Review Group (TRG) is comprised of certain plant supervisory and technical personnel. TRG is charged with reviewing power ascension testing activities and advising the Plant Manager on the disposition of those items reviewed. The responsibilities of TRG, with respect to power ascension testing activities, include final review and recommendation of approval of all power ascension test procedures, revisions, and test results. TRG is a subcommittee of the Plant Operations Review Committee (PORC).

14.2.2.6.1 TRG Membership

The TRG will be composed of one representative from each of the organizations listed below. Representatives of other organizations will participate as requested by the TRG Chairman.

Operations
Site Engineering
Power Ascension Test Group

Westinghouse (for matters concerning testing performed on Westinghouse supplied systems and components)

Site Quality Assurance will conduct activites in accordance with Chapter 17 of the FSAR.

TRG members and their alternates, including the Chairman, will be designated in writing with the concurrence of the PORC chairman.

14.2.2.7 Personnel Qualifications

The minimum qualifications of individuals that direct or supervise the conduct of preoperational tests at the time of performance of their duties, shall be: (1) Bachelor's degree in engineering or the physical sciences and one year of experience in power plant testing or operation of similar nuclear power plant components/systems, or (2) a high school diploma or equivalent and six years of experience in power plant testing or operation of similar nuclear power plant components/systems. In case (2), credit for up to two years for related technical training may be substituted for experience on a one-for-one basis.

The minimum qualifications of individuals that direct or supervise the conduct of power ascension tests at the time of performance of their duties shall be: (1) Bachelor's degree in engineering or the physical sciences and two years of experience in power plant testing or operation. Included in the two years shall be a minimum of one year of nuclear power plant testing, operating or training on a nuclear facility, or (2) a high school diploma or equivalent and five years of experience in power plant testing of which two years will be nuclear power plant experience. In case (2), credit for up to two years for related technical training may be substituted for experience on a one-for-one basis.

The minimum qualifications of individuals responsible for review and approval of preoperational test procedures and results shall be; (1) Bachelor's degree in engineering or the physical sciences and four years of applicable power plant experience; or (2) high school diploma or equivalent and eight years of applicable power plant experience. In case (2), credit for up to two years for related technical training may be substituted for experience on a one-for-one basis. A minimum of two years shall be applicable nuclear power plant experience.

The minimum qualifications of individuals responsible for final review and recommendation of approval of power ascension test procedures and results shall be the same as required for regular members of the Plant Operations Review Committee as described in Section 13.4.1.

14.2.3 Test Procedures and Instructions

14.2.3.1 General

Tests will be performed in accordance with approved instructions/procedures. The following sections describe the general methods employed to control procedure development and review, and describes the responsibilities of the various organizational units participating in this process.

The detailed controls and methods will be prescribed in the startup manual procedures for component, acceptance, and preoperational testing and plant administrative technical instructions for power ascension testing, as applicable.

14.2.3.2 Development of Procedures

Technical information required for the preparation of test procedures will be provided by the appropriate engineering organizations. Sources for this information are system descriptions, technical specifications, design drawings, and other technical documents which define the functional requirements and performance objectives for the various systems and components. Additional technical data may also be obtained from the various component vendors and other contractors as required.

The applicable functional requirements provided by the system designers will be incorporated into the acceptance criteria for each test procedure. This information will also be used by the test engineer in developing the detailed test methods which ensure that the capability of systems and components to function properly within design specifications is adequately demonstrated.

14.2.3.3 Review and Approval of Test Procedures and Instruction

An independent technical review of component and acceptance test procedures by qualified personnel will be performed prior to approval by the Preoperational Startup Manager.

Preoperational test instructions will be reviewed by personnel who are assigned to JTG member organizations and qualified for review of preoperational test instructions as described by Section 14.2.2.7. The Preoperational Startup Manager may request additional reviews to be performed as he deems appropriate. Preoperational test instructions will be approved by the Preoperational Startup Manager after recommendation for approval by the JTG.

A technical review of power ascension test procedures by qualified personnel will be performed prior to approval. Power ascension test procedures will be forwarded to the appropriate members of the Test Review Group for review and comment. Final approval of power ascension test procedures is the responsibility of the Plant Manager.

14.2.3.4 Format of Test Instructions/Procedures

Test instructions/procedures will be prepared based on formats specified by administrative procedures. These standard formats will help ensure that each procedure contains the information and instructions required to satisfactorily perform and document the test. The procedures format and content will reflect the guidance provided in Regulatory Guide 1.68. The standard format will include, as a minimum, the following:

(1) Test Objectives:

A detailed statement of the test objectives and method of system or plant operation to be demonstrated.

(2) References:

References to technical specifications, supporting procedures, vendor manuals, or other technical documents will be included as required.

(3) Precautions and Limitations:

Precautions and limitations relating to personnel safety, equipment integrity, unit to unit interactions, and overall plant safety will be specified.

(4) Prerequisites:

Prerequisites and initial conditions, including special environmental conditions, are specified and the necessary component tests and construction activities have been verified to be satisfactorily completed.

(5) Acceptance Criteria:

The performance objectives and functional requirements for system operations will be specified. The criteria used to judge the success or failure of the test shall be qualitative or quantitative.

(6) Performance:

The performance section will contain detailed step-by-step instructions for operating the system in the test configuration, performing actual test manipulations, and for use of off-normal procedures such as jumper cables or mechanical bypasses. Provisions will be made for recording all pertinent test data regarding system conditions and performance. The detailed test instructions will utilize normal and emergency plant operating procedures to the extent practical.

(7) Post-Performance Activities:

Post-Performance Activities include notification of field completion testing to the responsible Operations personnel, restoration of the plant/system to normal status including removal of jumpers, special test instrumentation and restoration of instrument settings. Also, post test accuracy test checks on plant instrumentation and M&TE used to gather quantitative Acceptance Criteria data is performed.

(8) Records

Provisions are made to determine if the completed test package is a quality related or non-quality record.

14.2.3.5 Test Instruction/Procedure Revisions/Changes

Revisions to preoperational and power ascension test procedures will be reviewed and approved in accordance with Section 14.2.3.3 prior to use.

Changes to preoperational test instructions required during test performance will be classified as intent or non-intent changes. Intent changes are changes to test methods, objectives, or acceptance criteria that affect Initial Test Program commitments as described in the FSAR. Non-intent changes are changes of any other type.

Changes to power ascension tests during test performances will be in accordance with administrative procedures.

Preoperational and power ascension test procedure changes required during test performance will be approved prior to implementation of the change, as follows:

(1) Preoperational Test Instructions

(a) Changes to test instructions that change the intent of the test will be reviewed and approved in the same manner as the original test procedure, as described in Section 14.2.3.3.

(b) Changes to test instructions that do not change the intent of the test will be approved by the assigned test engineer and another individual qualified, as described in Section 14.2.2.7, for review and approval of the test procedure.

(c) All instruction changes will be included with the instruction procedure and be subject to review with the test results as described in Section 14.2.5.

(2) Power Ascension Test Procedures

(a) Changes to power ascension tests that are minor/editorial and do not change the intent of the test will have approval of the Shift Manager and a second member of the plant management staff. This change process shall be described in administrative procedures.

(b) Changes to power ascension tests that are not minor/editorial and change the intent of the test will be reviewed and approved in the same manner as the original test.

14.2.4 Conduct of Test Program

14.2.4.1 Administrative Procedures

Conduct of the test program will be controlled by administrative procedures. These procedures will provide detailed instructions to assure adequate control of activities such as the following:

(1) Preparation, review, and approval of test procedures/instructions

(2) Turnover of systems

(3) Format and content of test procedures/instructions

(4) Test deficiency processing

14.2.4.2 Component Testing

Upon completion of construction phase activities, the Preoperational Startup Engineering organization will conduct appropriate component tests and acceptance or preoperational tests. During the system completion phase, systems and components will be reviewed for completeness, installation damage, and conformance with appropriate installation and/or design documents. Outstanding construction and test deficiencies will be identified and controlled. Component test instructions will be issued to assure that applicable prerequisites are met before testing is initiated and provide guidance on proper test performance and documentation of results. Appropriate component tests will be performed prior to the performance of preoperational or acceptance tests.

14.2.4.3 Preoperational and Acceptance Testing

Technical direction and administration, including test procedure preparation, test execution, and data recording of preoperational testing is the responsibility of the Preoperational Startup Manager. The test engineers direct support personnel in the performance of tests and provide appropriate interface with plant operators. The Shift Managers will be responsible for ensuring that the conduct of testing does not place the plant in an unsafe condition at any time. Additionally, the Shift Managers and test engineers have the authority to terminate or disallow testing at any time.

14.2.4.4 Power Ascension Testing

Technical direction and administration, including test procedure preparation, test execution, and data recording of power ascension testing is the responsibility of the Power Ascension Test Manager. The plant staff retains the responsibility for performing actual equipment operations and maintenance.

The Power Ascension Test Manager is responsible for the administration and implementation of all power ascension testing activities.

The test engineers will directly support personnel in the performance of tests and will provide appropriate interface with plant operators. The Shift Managers will be responsible for insuring that the conduct of testing does not place the plant in an unsafe condition at any time. Additionally, the Shift Managers or other licensed shift operators and test engineers have the authority to terminate or disallow testing at any time.

14.2.4.5 Test Prerequisites

Each test instruction/procedure will contain a set of prerequisites or initial conditions as prescribed by administrative procedures. The test engineer will ensure that all specified prerequisites are met prior to performing the test. The format for test instructions procedures is described in Section 14.2.3.4.

14.2.4.6 Phase Evaluation

Following each major phase of the initial test program, test results and/or test status will be reviewed to ensure that all required tests have been performed and the test results are satisfactory. This review will ensure that all required systems have been tested satisfactorily and that test results have been evaluated before proceeding to the next stage of testing. This review is described in Section 14.2.5.

14.2.4.7 Design Modifications

Modifications to the design of equipment during the test program may be initiated in order to correct deficiencies discovered as a result of testing. Any such modification will be referred to the appropriate engineering organization for approval.

Modifications made to components or systems after completion of preoperational or initial power ascension testing will be reviewed for retesting requirements on affected portions of the system.

14.2.5 Review, Evaluation, and Approval of Test Results

Following completion of a particular test, the responsible test engineer will assemble a test data package for evaluation. Acceptance test data packages will be independently reviewed as assigned by the Preoperational Startup Manager. Preoperational test data packages will be reviewed by appropriate members of the JTG. Power ascension test data packages will be reviewed by appropriate members of the Test Review Group.

Each test data package will be reviewed to ensure that the test has been performed in accordance with the approved procedure and that all required data and checks have been properly recorded and that system performance meets the approved acceptance criteria.

Deficiencies identified in the review process will be documented and resolved to the satisfaction of the appropriate review group. If the evaluation indicates that deficiencies in the test method are responsible for unsatisfactory test results, the test procedure will be appropriately revised or changed before retesting is initiated. Whenever an evaluation of test results indicates deficiencies in system performance, the problem will be referred to the appropriate engineering organization for evaluation.

The responsibility for final approval of acceptance and preoperational test results rests with the Preoperational Startup Manager. During power ascension, the responsibility for authorization to proceed to the next major test phase rests with the Plant Manager. The responsibility for final approval of power ascension test results rests with the Plant Manager. These major test phases are fuel load and precritical testing, initial criticality and low power (<5%) physics testing, 5% to 30%, 50%, 75% and 100% plateaus.

Following testing in each major test phase of the test program, test results and/or test status will be reviewed to ensure required tests have been performed and acceptance criteria satisfied; test deficiencies have been properly dispositioned and appropriate retesting completed; and the test results have been reviewed by appropriate designated personnel prior to proceeding to the next major test phase. This review will ensure that all required systems are operating properly and that testing for the next major test phase will be conducted in a safe and efficient manner.

14.2.6 Test Records

Test documentation such as test instructions/procedures, test results, test deficiencies and test changes relating to the initial test program will be processed, controlled, and retained as a plant historical record in accordance with the requirements of the WBN Quality Assurance program and project implementing procedures. A summary of power ascension testing will be provided in a Startup Test Report. The Startup Report shall address each of the power ascension tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

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

14.2.7 Conformance of Test Programs with Regulatory Guides

The initial test program will be developed and conducted in accordance with the following applicable NRC Regulatory Guides (RG). In certain cases, exceptions or alternate approaches to regulatory guidance are planned. Justification for these exceptions or alternate approaches are provided with the applicable regulatory guide.

(1) RG 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing," Revision 2

Discussion:WBN reactor internals are not a prototype design. Exception is taken to certain requirements of RG 1.20. Refer to FSAR Sections 3.9.2.3, 3.9.2.4, 3.9.2.5, and 3.9.2.6 for justification of exceptions and alternate plans.

(2) RG 1.41, "Preoperational Testing of Redundant On-site Electric Power Systems to Verify Proper Load Group Assignments," Revision 0

Discussion:The initial test program will comply fully with the requirements of RG 1.41 as discussed in FSAR Section 8.1.5.3.

(3) RG 1.52,"Design, Testing, and Maintenance Criteria for Atmospheric Cleanup System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants," Revision 2

Discussion:See Tables 6.5-1, 6.5-2, 6.5-3, and 6.5-4 for discussion of compliance with RG 1.52 and associated ANSI standards.

(4) RG 1.68,"Initial Test Programs for Water-Cooled Nuclear Power Plants"

Discussion:The initial test program complies with the requirements of RG 1.68, Revision 2 (8/78), with the following exceptions and/or alternate approaches:

(A) Regulatory Position C.1

(1) WBN takes specific exception to some provisions of RG 1.68. These include:

(a) Appendix A, subparagraph 1.c

Acceptance criteria for the response time of the various logic channels will be consistent with Technical Specifications requirements. The Reactor Trip System Response Time is defined in the Technical Specifications as the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The accident analysis accounts for conservative values for delay times, setpoint drift, etc. Therefore, it is not necessary to account for the response time of the associated hardware between the measured variable and the input to the sensor in the test or acceptance criteria.

(b) Appendix A, subparagraph 1.g.2

Unit 2 only emergency loads will not be tested with minimum and maximum design voltage available. Unit 2 only emergency loads will be tested to demonstrate satisfactory starting and operating characteristics with the power supply voltage within the design operating range.

Transformer taps were adjusted and tested in the Unit 1 Startup Program to obtain optimum voltage levels from no-load to full load conditions. No further testing is needed by Unit 2.

The validation of the engineering voltage calculations was performed during the Unit 1 Startup Program. The data recorded included the operating parameters of the offsite grid, Class 1E 6.9 kV, 480 volt, and 120 volt vital power busses under no-load, steady state load, and transient conditions. Data was obtained for the class 1E train having the lowest analyzed voltage, and the result was satisfactory. No further testing is needed by Unit 2.

(c,1) Appendix A, subparagraph 1.h.3

Testing of non-safety related portions of the ice condenser system will be accomplished by component tests in accordance with Reference 11. Air return fans are tested under Containment Ventilation preoperational tests.

(c,2) Appendix A, subparagraph 1.j.10

Seismic instrumentation was tested and is operational to support Unit 1.

(d) Appendix A, subparagraph 1.j.12, and 5q

The gross failed fuel detection system (GFFDS) is abandoned in place. Periodic sampling is used to detect failed fuel as described in Section 9.3.2.2

10 CFR 50, Appendix A and RG 1.97 do not require a GFFDS and, from a practical standpoint, there is no justification for the monitor. This particular design detects delayed neutrons and any type of failure that would be detected by this monitor would be putting significant fuel particles into the coolant.

To damage fuel to this extent would require either a severe transient, foreign material damage or an unusual design problem, such as baffle jet impingement. The plant is designed to operate such that clad damage will not occur. Technical Specifications limits on pressure, temperature, flow and power ensure that cladding is not damaged during normal operations, including anticipated transients. Reactor coolant is sampled frequently. Although Technical Specifications only require gross specific activity and dose equivalent I-131 analyses every 7 days and 14 days during power operations, coolant will actually be sampled for other parameters at least every 72 hours. The radiation protection surveys required to handle these samples would detect any significant activity changes. If power changes 15% or more in any one hour, a special sample is required between 2 and 6 hours after the power change.

Due to the ability to monitor the conditions that damage fuel cladding, it is easily determined if sampling should be performed for evidence of damage. Monitors for subcooling margin, incore thermocouples, and reactor vessel level are examples of monitors that would indicate clad damaging conditions.

Foreign material damage (monitored by noise monitor) and baffle jet impingement have resulted in non-catastrophic clad failures at other plants which were detected and monitored by normal chemistry sampling.

(e) Appendix A, subparagraphs 1.k.2, and 1.k.3

The subject equipment is calibrated and functionally tested as part of the WBN plant instrument calibration program. The calibration and functional testing is performed and documented in accordance with approved plant calibration procedures. Therefore, additional testing in the form of a preoperational test is not warranted.

(f) Appendix A, subparagraph 1.k.4

Refer to 14.2.7.3 for discussion of compliance with requirements for filter and in-place leak tests.

(f,1) Appendix A, subparagraph 1.l.3

Testing of the non-safety related solid waste processing system was accomplished during Unit 1 initial testing.

(g) Appendix A, subparagraph 1.m.1

Only the Unit 2 refueling water purification subsystem will be tested during the preoperational phase. Because of its non-safety related nature, the testing will be accomplished by a combination of system and/or component level testing in accordance with Reference 11.

(h) Appendix A, subparagraph 1.m.4

Static load testing at 125% of rated load for equipment and components used to handle irradiated or non-irradiated fuel originates from an ANSI B30.2-1976 (now ASME B30.2-1990) rated load test requirement. This test is purposed to verify the structural integrity of the handling equipment and is utilized to rate its capacity. Handling equipment is rated at 80% of the test load, resulting in a rating of 100% capacity when test loads are 125%. This ANSI testing is required to be performed prior to initial use, and following extensive repairs or modifications. Both hooks of the polar crane have been successfully load tested at 125% rated capacity in prior tests. Additionally, this equipment has had no extensive maintenance or modifications which would affect structural integrity. Therefore, repeated load testing of this equipment is not warranted. Common equipment previously used for handling Unit 1 fuel was load tested before initial use and will not be addressed.

Operational testing of cranes not associated with spent fuel movement was accomplished in accordance with Reference 14.

(h,1) Appendix A, subparagraph 1.n.13

Testing of the non-safety related communications system will be accomplished in accordance with reference 11.

(h,2) Appendix A, subparagraph 1.n.14

Testing of the intake pump station ventilation system was accomplished during Unit 1 initial testing.

(i) Appendix A, subparagraph 1.n.18

Because of its non-safety related, and simple functions, trace heating systems will receive component level tests, not preoperational or acceptance testing.

(j) Appendix A, subparagraphs 2.b.

Cold, no-flow, cold, full-flow and hot, no-flow rod drops do not provide any additional useful data. The drop times for these flow conditions are less conservative than for hot, full-flow conditions. WBN does not intend to perform cold, no-flow, cold, full- flow and hot, no-flow rod drops.

(k) Appendix A, subparagraphs 2.f and 5.m

WBN does not intend to perform a differential pressure measurement across the core nor a core flow measurement since these are prototype tests. WBN is not a prototype model plant, but, instead, is a well documented production model with several similar predecessor reactor units (Indian Point Unit 2, Trojan, and Sequoyah Unit 1) of its type. Proper differential pressures across the core and core flow is shown indirectly through performance of other tests that verify operating temperature and RCS flow.

(l) Appendix A, subparagraph 4.a

WBN will not measure a boron reactivity coefficient using the Advanced Digital Reactivity Computer (ADRC) during low power physics testing. Overall core reactivity will be measured during low power physics testing.

(m) Appendix A, subparagraphs 4.c and 5.e

No new design information is to be gained from the RCCA Pseudo Ejection Test. WBN is not a prototype plant for 4 loop, 12 foot core with 17 x 17 fuel design. Performance and measurement data already exists for this design. Therefore, WBN will not perform this test.

(n) Appendix A, subparagraphs 4.g and 5.z

The proper response of process and effluent radiation monitors is demonstrated under the plant calibration program and during preoperational testing for the Process and Effluent Radiation Monitoring System (Sheet 31, Table 14.2-1). It is not expected that enough leakage in fuel cladding, steam generator tubes or heat exchangers will exist to provide a meaningful comparison between radiation monitor responses and laboratory analysis of samples.

Therefore, WBN will not perform special power ascension testing of process and effluent radiation monitors.

(o) Appendix A, subparagraph 4.t

Natural circulation tests of the Reactor Coolant System will not be performed. Such tests have been successfully completed at Diablo Canyon Unit 1, McGuire Unit 1, Salem 2, Sequoyah 1 and other Westinghouse plants similar to WBN. It is unnecessary for WBN to compare flow (without pumps) and temperature data to data at these plants since no design differences exist which would significantly effect natural circulation capabilities. Typical natural circulation characteristics for 4 loop Westinghouse plants are given in WCAP-8460, "Natural Circulation Test Report for Zion Station Unit 1." However, in order to verify natural circulation cooldown and boron mixing capability per requirements of Branch Technical Position RSB 5-1, WBN has referenced test results (Refer to WCAP-12334) from Diablo Canyon Unit 1, and the WBN design is comparable to Diablo Canyon Unit 1.

(p) Appendix A, subparagraph 5.a

Power reactivity coefficients will not be determined at 25%, 50%, 75%, and 100% power levels. The power coefficient is not directly measured but is inferred. The required measurement for the test is time consuming compared with the value of the data obtained. The measurement has been previously deleted at other plants based on inferred measurements at similar plants. WBN is similar to Sequoyah Units 1 & 2 and McGuire Units 1 & 2. The Doppler power coefficient was inferred and compared with design values in three of these four units. The comparison between the inferred measurements and design values was within acceptance criteria in all cases thus demonstrating the ability to analytically predict this design parameter. However, overall core reactivity will be measured during low power physics testing and at approximately 100% power which demonstrates the adequacy of core design reactivity coefficients.

(q) Appendix A, subparagraph 5.d

WBN plant design does not include part-length control rods. The ability to control core xenon transients is a design feature of the Westinghouse Nuclear Steam Supply System and has been demonstrated in numerous operating pressurized water reactors. In addition, compliance with Technical Specifications for Axial Flux Difference helps ensure proper power and flux distributions. On these bases, WBN does not intend to perform an Initial Power Ascension Test to comply with subparagraph 5.d.

(r) Appendix A, subparagraph 5.f

WBN does not intend to perform a pseudo dropped rod test. There is no appreciable new data that would be obtained from performing this test. Previous plants have proved the design bases for typical cores.

(s) Appendix A, subparagraph 5.i

The Power Distribution Monitoring System is not designed to verify the position of all RCCA's that might be misaligned from their bank.

(t) Appendix A, subparagraph 5.u

Operability and response times of the main steam isolation valves will be verified in hot standby (mode 3) rather than at the recommended 25% power level. Testing at hot, zero power will result in more conservative results and will eliminate the unnecessary pressure and steam flow transients which would otherwise be induced.

(u) Appendix A, subparagraph 5.i.i

The performance of this test provides no new information needed to verify the plant performance during design transients. A trip of the reactor coolant pumps results in a reactor trip with flow coastdown, as verified in the Reactor Coolant System Flow Coastdown Test, providing sufficient heat removal to ensure DNBR does not decrease below WBN's limiting design value. Performing this test expends one of the analyzed transients and results in unnecessary cost and down time for the utility.

(v) Appendix A, subparagraph 5.k.k

The most influential contributor for this transient is the value of moderator temperature coefficient of reactivity, which has a relatively low value at beginning of core life. Since this parameter is determined in other startup tests, thus validating the safety analysis, the performance of this test provides no new information needed to verify the plant design. The transient does introduce the potential for thermal stress damage to the steam generator feedwater inlet nozzles and it expends one of the analyzed thermal cycles. Therefore, we do not intend to perform a test to comply with this subparagraph.

(w) Appendix A, subparagraph 5.m.m

The performance of this test provides no new information needed to verify the plant performance during design transients. Closure of all main steam isolation valves from 100% power will result in a turbine trip and a reactor trip. Turbine trip and reactor trip from 100% power will be performed during initial power ascension testing.

Closure of the MSIV's may cause operation of the pressurizer and steam generator power operated relief valves and/or safety valves which may then require repair and unnecessary down time for the utility. This test would expend one of the analyzed pressure transients for the reactor coolant system and steam generators and therefore will not be performed.

(2) Some provisions of RG 1.68 (power ascension phase) do not apply to the design of WBN. These include Appendix A, subparagraphs 4.m, 5.c, 5.d, 5.h, 5.j, 5.q, 5.w, and 5.i.i. In addition, certain portions of some provisions of RG 1.68 (power ascension phase) do not apply to the design of WBN. These include:

(a) Appendix A, subparagraph 2

Shutdown margin verifications should be performed at appropriate loading intervals (BWR) including full core shutdown margin tests. It should be established that the required shutdown margin exists, without achieving criticality.

(b) Appendix A, subparagraph 2.a

Shutdown margin verification for partially (BWR) loaded core.

(c) Appendix A, subparagraph 2.b

Friction tests of control rods after the core is fully loaded for BWRs. For facilities using more than one type of control element or control rod drive design, scram times should be compared with identical designs (e.g., two control rods attached to a single drive mechanism.)

(d) Appendix A, subparagraph 4.h

The design of WBN does not provide for automatic RCS chemical control and analysis. Chemistry specifications are established and monitored as prerequisites for testing as necessary.

(e) Appendix A, subparagraph 5

Testing with single loop reactor coolant system operation.

(f) Appendix A, subparagraph 5.a

Power vs. flow characteristics (BWR) are in accordance with design values.

(g) Appendix A, subparagraph 5.b

Maximum average planar linear heat generation rate (MAPLHGR) and minimum critical power ratio (MCPR) are in accordance with design values.

(h) Appendix A, subparagraph 5.g

Demonstrate that control rod sequencers and control rod worth minimizers operate in accordance with design.

(i) Appendix A, subparagraph 5.k

At a power level in the 25%-50% range for BWRs with steam-driven pumps and for BWRs with electric-driven pumps, if not previously conducted.

(j) Appendix A, subparagraph 5.l

Demonstrate design capability of residual heat removal (RHR) system in steam condensing mode and reactor core isolation cooling (RCIC) system.

(k) Appendix A, subparagraph 5.m

Demonstrate that reverse flows through idle loops or jet pumps are in agreement with design.

(l) Appendix A, subparagraph 5.s

Calibrate and verify performance of integrated control system and reactor coolant flow control system.

(m) Appendix A, subparagraph 5.y

Calibrate, as required, and verify the proper operation of important instrumentation systems including core flow.

(n) Appendix A, subparagraph 5.a.a

The design of Watts Bar does not provide for automatic RCS chemical control and analysis. Chemistry specifications are established and monitored as prerequisites for testing as necessary.

(o) Appendix A, subparagraph 5.e.e

Demonstrate that primary containment inerting system operates in accordance with design.

NOTE: As discussed in References [8] and [9], certain provisions of RG 1.68 (preoperational phase) do not apply to the design of WBN. NRC's review of these items is provided in Reference [10].

(3) Some provisions of RG 1.68 are not specifically tested or reviewed in power ascension. These provisions are satisfied by plant surveillance and other routine programs. These include Appendix A, subparagraphs 2.c, 2.d, 2.g, 4.1, 5.o, 5.u, 5.f.f, and 5.g.g. In addition, certain portions of some provisions of RG 1.68 are satisfied by plant surveillance and other routine programs and are not specifically tested or reviewed in power ascension. These include:

(a) Appendix A, subparagraph 2

Ensure that all applicable technical specification requirements and other prerequisites have been satisfied.

(b) Appendix A, subparagraph 2.e

Measurements of the water quality of the reactor coolant system.

(c) Appendix A, subparagraph 3

All systems required for startup or protection of the plant, including the reactor protection system and emergency shutdown system, should be operable and in a state of readiness.

(d) Appendix A, subparagraph 4.h

Chemical and radiochemistry tests and measurements to maintain water quality within limits in the reactor coolant and secondary coolant systems.

(e) Appendix A, subparagraph 5.a.a

Chemical and radiochemical sample to establish that reactor coolant system and secondary coolant system limits are not exceeded (25%, 50%, 75%, 100%).

(4) Some provisions of RG 1.68 are satisfied in the preoperational phase. These include Appendix A, subparagraphs 4.j, 4.k, 4.p, 4.r, 4.s, 4.u, 5.p, 5.t, 5.v, 5.x, 5.c.c, 5.g.g, and 5.o.o. In addition, certain portions of some provisions of RG 1.68 are satisfied in the preoperational phase. These include:

(a) Appendix A, subparagraph 2.f

Reactor coolant system flow test to establish that vibration levels are acceptable; Piping reactions to transient conditions and flows are as predicted for all allowable combinations of pump operation.

(b) Appendix A, subparagraph 4.q

Demonstration of the operability of residual or decay heat removal systems, including atmospheric steam dump valves (PWR).

(c) Appendix A, subparagraph 5.g

Demonstrate rod withdrawal block functions operate in accordance with design.

(d) Appendix A, subparagraph 5.k

Demonstrate that ECCS high-pressure coolant injection systems can start under simulated accident conditions and inject into the reactor coolant system as designed. (For PWRs, the testing should be in accordance with Regulatory Guide 1.79.)

(e) Appendix A, subparagraph 5.l

Demonstrate design capability of all systems and components provided to remove residual or decay heat from the reactor coolant system, including atmospheric steam dump valves and auxiliary feedwater system. Testing of the auxiliary feedwater system should include provisions that will provide reasonable assurance that excessive flow instabilities (e.g., water hammer) will not occur during subsequent normal system startup and operation.

(f) Appendix A, subparagraph 5.m

Sufficient measurements and evaluations should be conducted with the plant at steady-state conditions to establish that vibration levels of reactor coolant system components are in agreement with design values.

(g) Appendix A, subparagraph 5.r

Verify by review and evaluation of printouts and/or computer displays that the control room or process computer is receiving correct inputs from process variables and validate that performance calculations performed by the computer are correct.

(h) Appendix A, subparagraph 5.s

Calibrate, as necessary, and verify the performance of major or principle plant control systems, including boron addition systems (PWR), main, auxiliary and emergency feedwater control systems; hotwell level control systems; and reactor coolant makeup and letdown control systems.

(i) Appendix A, subparagraph 5.y

Calibrate, as required, and verify the proper operation of important instrumentation systems including reactor coolant level.

(j) Appendix A, subparagraph 5.d.d

Demonstrate the potential capability for placing the reactor in a cold shutdown condition.

(k) Appendix A, subparagraph 5.e.e

Demonstrate that primary containment purge system operates in accordance with design.

(5) Some provisions of RG 1.68 are satisfied at other plant conditions than specified. These include:

(a) Appendix A, subparagraph 4.e

Determination made up to 30% power rather than from 0 to 5% power.

(b) Appendix A, subparagraph 4.i

Demonstration done in mode 5 and mode 3 rather than from 0 to 5% power.

(c) Appendix A, subparagraph 4.o

Testing performed in mode 3 rather than from 0 to 5% power.

(d) Appendix A, subparagraph 5.s

Calibration and performance verification for the pressurizer control system is conducted in mode 3 rather than at power, and, for the automatic reactor control system, is conducted at 50% only.

(e) Appendix A, subparagraph 5.u

Operability and response times are verified in mode 3 rather than at 25% power.

(f) Appendix A, subparagraph 5.h.h

Demonstration done at 50% and 100% power only.

(1) RG 1.68.2, "Initial Startup Test Program to Demonstrate Remote

Shutdown Capability for Water Cooled Nuclear Power Plants"

Discussion:The testing activities conducted as a part of the Initial Test Program will comply with the applicable requirements of RG 1.68.2, Revision 1 (7/78).

(6) RG 1.79,"Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors"

Discussion:Preoperational testing of the Emergency Core Cooling Systems will comply fully with RG 1.79, Revision 1, (9/75) with exception to Regulatory Position C.1.b(2) Recirculation Test - Cold Conditions. A satisfactory in-plant test of the containment sump to demonstrate vortex control and acceptable pressure drops across screening and suction lines and valves is not practical. Justification for this exception is provided in FSAR Section 6.3.4.1.

(7) RG 1.68.3,"Preoperational Testing of Instrument and Control Air Systems"

Discussion:Preoperational testing of the Instrument and Control Air Systems will comply with RG 1.68.3, April 1982, with the following exceptions:

(a) Regulatory Position C.8

Auxiliary Control Air System loads will be tested on an individual basis to verify their response to a sudden loss of system pressure. NRC concurrence with this exception is reflected in correspondence from R. C. Lewis to H. G. Parris dated February 28, 1984.

(b) Regulatory Position C.8

Control air system loads (safety related only) will be tested to verify their response to a loss of system pressure. The sudden loss of air pressure will be performed on an individual load basis. Non-safety related air operated loads will be tested on a component basis to verify proper response to a loss of air pressure.

(c) Regulatory Position C.11

Functional testing to demonstrate operability of compressed air system loads under increased pressure conditions will not be performed. The safety evaluation of the system indicates that the system is adequately designed to prevent system overpressure as described in Section 9.3.1.3. The maximum pressure rating of the most limiting component of the system piping, valves, and equipment (up to the end user pressure regulator) was determined to be at least 20 psi higher than the system safety valve setpoint plus 10% accumulation. This ensures that there is adequate protection against overpressurization.

(8) RG 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," Revision 1, January 1977.

Discussion:The potential for chlorine to pose a hazard to main control room (MCR) operators due to onsite storage spills or transportation incidents in the vicinity of the site is analyzed in FSAR Section 6.4.4.2. The analysis concluded that no hazard to control room habitability is posed by chemicals stored on site, offsite within a 5-mile radius, or transported by the site by barge, rail, or road within a 5-mile radius. Therefore, the requirements of RG 1.95, Revision 1, (January 1977) "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release" do not apply.

(9) RG 1.9, "Selection, Design, Qualification and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants".

Discussion:See RG 1.9 discussion in Section 8.1.5.3.

(10) RG 1.118, "Periodic Testing of the Electric Power and Protection Systems"

Discussion:Testing of the Reactor Protection System will be in compliance with the requirements of RG 1.118, Revision 2 (6/78).

(11) RG 1.139, "Guidance for Residual Heat Removal," for comment, May 1978

Discussion:This regulatory guide will not be used for development or conduct of the WBN initial test program.

A final report of a comparison between WBN and Sequoyah was submitted to NRC by Reference [3]. This study addressed requirements of Regulatory Guide 1.139 and Branch Technical Position RSB 5-1. This report indicated that preoperational test results to demonstrate natural circulation cooldown capability at Diablo Canyon were applicable to the WBN design and satisfy NRC requirements for natural circulation tests. Reference [5] confirmed acceptability of this approach, pending review of Diablo Canyon test results.

More recently, Reference [4] provided an assessment of the applicability of the Diablo Canyon natural circulation test to WBN. This assessment concluded that natural circulation tests are not required at WBN, and was accepted by the NRC in Reference [6]. Further review on this issue is documented in Sections 5.4 and 14 of Reference [7].

Exception has been taken to RG 1.68, Revision 2, requirements for performing natural circulation tests of the reactor coolant system, as described in this section and in Section 14.2.7.4.A(1)(n).

(12) RG 1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," Revision 1 (10/79)

Discussion:Testing of normal (non-safety related) ventilation exhaust filtration systems will be performed in full compliance with guidance provided by this Regulatory Guide.

14.2.8 Utilization of Reactor Operating and Testing Experience in Development of Test Program

The WBN initial test program will utilize information gained from operating and testing experience in other similar nuclear plants. This information will be used to provide guidance in developing test procedures and schedules and to alert personnel to potential problem areas.

TVA has an established Nuclear Experience Review (NER) Program. The NER Program identifies and evaluates experience gained from TVA nuclear plants, Institute of Nuclear Power Operations (INPO), Nuclear Regulatory Commission (NRC), from other nuclear utilities, architect-engineers and contractors, equipment suppliers, and others within the nuclear industry. The program ensures that significant operational experience and events are reviewed and integrated into applicable programs/ procedures to enhance nuclear safety and reliability. Items that could impact testing will be reviewed by the appropriate testing organization to determine appropriate actions such as procedure revisions, review of completed test, retest, etc.

Additionally, testing and operating experience gained from Unit 1 will be incorporated into the Unit 2 test program.

14.2.9 Trial Use of Plant Operating and Emergency Procedures

To the extent necessary, the plant operating, emergency, surveillance and periodic test instructions/procedures will be use-tested during the initial test program and will also be used in the development of preoperational and initial power ascension procedures to the extent practical. The trial use of plant instructions serves to familiarize operating personnel with systems and plant operation during the testing phase and also serves to assure the adequacy of the procedures under actual or simulated operating conditions before plant operation begins.

Prior to fuel load, draft operating instructions may be utilized for equipment operation. The operating instructions will be revised as necessary to reflect experience gained during the actual testing.

During the power ascension test phase, power escalation tests normally performed once each cycle during startup will be performed to demonstrate their adequacy.

14.2.10 Initial Fuel Loading, Postloading Tests, Initial Criticality, Low Power Tests and Power Ascension

Fuel load will begin after review and approval of the results of tests summarized in Table 14.2-1. At the completion of fuel load, the reactor upper internals and pressure vessel head will be installed and additional mechanical and electrical tests will be performed to prepare the plant for nuclear operation. All precritical tests will be performed prior to initial criticality and the results evaluated. Then nuclear operation of the reactor will begin. This phase of testing includes initial criticality, low power testing and power level escalation and testing. The purpose of these tests is to verify the operational characteristics of the unit and core, to acquire data for the proper calibration of instruments, and to ensure that operation is within license requirements.

14.2.10.1 Fuel Loading

Core loading instructions will specify the condition of certain fluid systems to prevent inadvertent dilution of the reactor coolant; specify the movement of fuel to preclude the possibility of mechanical damage; prescribe the conditions under which loading can proceed; designate responsibility and authority and provide for continuous and complete fuel and core component accountability.

Prior to starting actual fuel load operations, the reactor containment structure will be completed and the containment building penetrations will be in the condition required by Technical Specifications. Fuel-handling tools and equipment will have been checked out and dry runs conducted in the use and operation of equipment. Inspections of fuel assemblies, associated core components, and the reactor vessel will be satisfactorily completed. The reactor vessel and associated components will be in a state of readiness to receive fuel. The overall process of initial core loading will be directed by a licensed Senior Reactor Operator and in accordance with applicable Technical Specification requirements. Standard procedures for the control of personnel will be established prior to fuel load.

The as-loaded core configuration is specified as part of the core design studies conducted well in advance of unit startup and as such is not subject to change at fuel load. In the event mechanical damage is sustained during core loading operations to a fuel assembly of a type for which no spare is available on site, an alternate core loading scheme whose characteristics closely approximate those of the initially prescribed pattern will be determined by Westinghouse and TVA.

The fuel assemblies with inserted core components will be moved through the filled transfer canal and inserted into the reactor vessel containing reactor grade water with adequate dissolved boric acid to maintain a calculated core effective multiplication factor of 0.95 or lower. Core moderator chemistry conditions (particularly boron concentration) will be prescribed and will be verified periodically by chemical analysis of moderator samples taken prior to and during core loading operations.

Core loading instrumentation consists of the installed, calibrated source range nuclear channels. The installed source range channels will be operable per Technical Specifications for monitoring by operations personnel in the control room. At least one of the two channels in the control room will be equipped to provide an audible count rate. Both channels in the control room will have the capability of displaying the neutron flux level on MCR instruments.

Following the installation of the initial nucleus of 8 fuel assemblies, in front of each source range detector, two channels will be monitoring subcritical multiplication. At least 2 artificial neutron sources located in fuel assemblies will be introduced into the vessel at specified points to ensure detector response. A response check of nuclear instruments shall be performed within 8 hours prior to loading of the core, or with the first source bearing fuel assembly. If a delay in fuel loading greater than 8 hours occurs in the nuclear instrumentation will be evaluated to determine acceptability of the detectors for continuation of fuel loading.

Fuel assemblies, together with inserted core components, will be placed in the reactor vessel one at a time according to an approved sequence. Plant instructions include detailed tabular check-sheets which prescribe and verify the movements of each fuel assembly and its core component from its initial position in the storage racks to its final position in the core. Checks are made of the fuel assembly ID number to guard against possible inadvertent substitutions of fuel assemblies. A display will be provided for indicating the status of the core throughout the core loading operation.

An initial nucleus of eight fuel assemblies (with inserted core components), one of which contains a neutron source, is the minimum source-fuel nucleus which permits subsequent meaningful inverse count rate monitoring. Calculations indicate that this initial nucleus is markedly subcritical ($K_{eff} < 0.95$) under the required conditions of loading. Each subsequent fuel assembly addition is accompanied by detailed neutron count rate monitoring to determine that the just loaded fuel assembly does not excessively increase the count rate. The results of each fuel assembly loading will be evaluated before the next prescribed fuel assembly is loaded in the reactor vessel.

Criteria for safe loading require that loading operations stop immediately if:

(1) An unanticipated increase in the neutron count rate by a factor of 2 occurs on all responding nuclear channels during any single loading step after the initial nucleus of 8 fuel assemblies is loaded (excluding anticipated change due to detector and/or source movement).

(2) The neutron count rate on any individual nuclear channel increases by a factor of 5 during any single loading step after the initial nucleus of 8 fuel assemblies is loaded (excluding anticipated change due to detector and/or source movement or detector-to-core coupling).

(3) A decrease in boron concentration greater than 20 ppm is determined from two successive samples of reactor coolant until the decrease is explained.

Alarms in the containment and main control room are coupled to either of the two source range channels with a setpoint equal to or less than five times the current count rate. This alarm automatically alerts the loading operation personnel of high count rate and requires an immediate stop of all operations until the situation is evaluated. Normally, the alarm used for this purpose is the containment evacuation alarm. In the event that the evacuation alarm is actuated, core loading personnel will be evacuated. The situation will be evaluated before core loading is continued. After it has been determined that no hazards to personnel exist, personnel will be permitted to reenter the containment.

14.2.10.2 Postloading Tests

Upon completion of core loading, the reactor upper internals and the pressure vessel head will be installed. A test is conducted after filling and venting are completed to check the integrity of the vessel head installation.

Mechanical and electrical tests will be performed on the control rod drive mechanisms. These tests will include a complete operational checkout of the mechanisms and calibration of the individual rod position indication.

Tests will be performed on the reactor trip circuits to test manual trip operation. The actual control rod assembly drop times will be measured for each control rod assembly.

At all times that the control rod drive mechanisms are being tested, the boron concentration in the moderator will be maintained such that the shutdown margin requirements specified in the Technical Specifications are met. During individual RCCA or RCC bank motion, source range instrumentation is monitored for unexpected changes in core reactivity.

A functional electrical and mechanical check will be made of the Power Distribution Monitoring System. After evaluation of precritical tests, nuclear operation of the reactor will begin.

14.2.10.3 Initial Criticality

Initial criticality will be achieved by a combination of shutdown and control bank withdrawal and reactor coolant system boron concentration reduction.

Initially, all shutdown and control banks will be withdrawn incrementally in the normal withdrawal sequence, until all shutdown and control banks are at the full ARO configuration. The boron concentration in the reactor coolant system will then be reduced by the addition of primary water.

Inverse count rate ratio monitoring, using data from the source range instrumentation, will be used as an indication of the proximity and rate of approach to criticality. Inverse count rate ratio data will be evalutated during control bank withdrawal and subsequent reactor coolant system boron concentration reduction.

Criticality will be achieved during boron dilution. The rate of primary water addition, and hence, the rate of approach to criticality will be reduced as the reactor approaches criticality to ensure that effective control is maintained. Throughout this period, samples of the primary coolant are obtained and analyzed for boron concentration.

Written procedures specify the plant conditions, precautions and specific instructions for the approach to criticality.

14.2.10.4 Low Power Tests

Following initial criticality, a program of reactor physics measurements will be undertaken to verify that the basic static and kinetic characteristics of the core are as expected and that the values of the kinetic coefficients in the accident analysis are conservative.

Procedures will specify the sequence of tests and measurements to be conducted and the conditions under which each is to be performed in order to ensure both safety of operation and the validity and consistency of the results obtained. If test results deviate significantly from design predictions, if unacceptable behavior is revealed, or if unexplained anomalies develop, the unit will be brought to a safe stable condition and the situation reviewed to determine the course of subsequent unit operation.

These measurements will be made at low power and primarily at or near normal operating temperature and pressure. Measurements will be made in order to verify the calculated values of rod bank reactivity worths, isothermal temperature coefficient and the critical ARO, hot zero power boron concentration.

14.2.10.5 Power Ascension

After the operating characteristics of the reactor have been verified by low power testing, a program of power level escalation will bring the unit to its full rated power level in successive stages. Power range nuclear instrumentation high flux trips will be set less than or equal to 20% of RTP above each power plateau before escalating to the plateau. The minimum test requirements for each successive plateau of power escalation will be specified in the power ascension test procedures.

Measurements will be made to determine the relative power distribution in the core as functions of power level and rod bank position.

Secondary system heat balance measurements will ensure that the indications of power level are consistent and provide bases for calibration of the power range nuclear channels. The ability of the reactor coolant system to respond effectively to signals from primary and secondary instrumentation under a variety of conditions encountered in normal operations will be verified.

At prescribed power levels the dynamic response characteristics of the primary and secondary systems will be evaluated. System response characteristics will be measured for step load changes, rapid load reduction, and plant trips.

Adequacy of radiation shielding will be verified by gamma and neutron radiation surveys at preselected locations throughout the plant at various power levels. Normal periodic sampling will be performed to verify the chemical and radio-chemical analysis of the reactor coolant.

14.2.11 Test Program Schedule

Test sequence schedules for preoperational and power ascension testing are shown in Figures 14.2-1 and 14.2-2, respectively. These schedules show certain milestones at which time tests (or portions of tests) should be completed and the time frame in which the tests are expected to be performed.

These schedules represent a plan for completion of the initial test program which, from time-to-time, may be altered due to unforeseen circumstances. Detailed schedules for testing will be prepared, reviewed and revised on a continuing basis as plant construction progresses. Plant structures, systems, and components which are relied upon to prevent or mitigate consequences of postulated accidents will be fully tested to the extent practical prior to exceeding the 5% power level. This testing includes an RCS flow measurement using RCS elbow tap differential pressure measurements and engineering calculations prior to initial criticality. However, those testing activities which cannot reasonably be performed below the 5% power level will be fully tested to the extent practical prior to exceeding the 30% power level with the exception of the RCS Calorimetric Flow Measurement test which is performed at the 75% power level consistent with Westinghouse recommendations.

Preoperational testing of the various systems and components, which will continue up to fuel loading, is planned to commence approximately 12 months prior to fuel load. Fuel load, initial criticality, and power ascension tests are scheduled to be accomplished over a period of approximately 5 months.

Certain systems will have part of the preoperational testing performed after fuel load due to system configuration (e.g., control rod drive mechanisms require fuel in the vessel to be fully tested). Such systems will be sufficiently tested prior to fuel load to provide reasonable assurance of successful testing after fuel load.

The development of test procedures will be an ongoing process, which will consist of preparation, review, and revision. Approved preoperational test procedures for satisfying FSAR testing commitments will be made available to NRC regional personnel a minimum of 60 days prior to their intended use. Power ascension test procedures will be made available 60 days prior to fuel load.

14.2.12 Individual Test Descriptions

14.2.12.1 Preoperational Tests

The test summaries for each of the preoperational tests to be performed, along with an index to these summaries, are provided in Table 14.2-1. These summaries describe the various tests which are specified as preoperational tests in Regulatory Guide 1.68. The scope and titles of these summaries may not in all cases correspond directly to the actual test procedures which will be used during the initial test program. Certain test procedures may include more than one test as described in these summaries, and in some cases tests described in one summary may be covered under more than one procedure. The overall scope and content of the tests described in these summaries will be addressed in final procedures.

The preoperational test summaries in Table 14.2-1 include general test methods such as the verification of instrumentation, controls, indications, interlocks, and protective devices. The functionality of such devices will be demonstrated during component testing. The completion of appropriate component tests will be verified prior to performance of preoperational tests as described by general prerequisite number two (2) below. It is not intended to reverify the functionality of all individual devices (e.g., instruments, relays, contacts, etc.) which have been previously verified by component tests. However, the important to safety design features (component response such as open, close, start, stop, etc.) which result from the operation of such devices will be demonstrated by system level preoperational tests.

There will be certain prerequisites which will apply in general to all preoperational tests. These general prerequisites are listed here rather than included in each summary.

General Prerequisites:

(1) Construction activities have been completed (or exceptions justified) on the system and on the necessary portions of supporting systems and any deficiencies have been properly dispositioned.

(2) Functional operability of individual components or subsystems has been demonstrated for items such as valves, pumps, motors, instrumentation, and controls.

(3) Test equipment necessary for test performance is available and calibrated.

(4) Electrical power and air supplies are available as required for test performance.

14.2.12.2 Power Ascension Tests

The Power Ascension phase of the initial test program consists of preoperational type tests deferred to the power ascension phase, power ascension tests performed prior to fuel loading, fuel load, precritical tests, initial criticality, low power physics tests (0-5%), and power ascension tests above 5% power. Test summaries providing descriptions of major power ascension testing are provided in Table 14.2-2, along with an index. The scope and titles of these summaries may not, in all cases, correspond directly to the actual test procedures which will be used during the power ascension test program. Certain test procedures may include more than one test, as described in these summaries, and in some cases, tests described in one summary may be covered under more than one procedure. The scope and content of the tests described in these summaries will be addressed in final procedures.

REFERENCES

(1) Letter from TVA to NRC dated April 16, 1993, "Preoperational Test Program Commitment Clarification."

(2) Letter from TVA to NRC dated July 2, 1993, "Testing of the Reactor Coolant System."

(3) Letter from TVA to NRC dated August 17, 1981, concerning WBN in comparison to Sequoyah Nuclear Plant.

(4) Letter from TVA to NRC dated July 11, 1991, "Natural Circulation Testing - Comparison to Diablo Canyon - Safety Evaluation Report Confirmatory Issue 15."

(5) NUREG 0847, dated June 1982, "Safety Evaluation Report related to operation of WBN, Units 1 and 2."

(6) Letter from NRC to TVA dated July 14, 1992, "Completion of Review on Natural Circulation Cooldown."

(7) NUREG 0847, Supplement 10, dated October 1992, "Safety Evaluation Report related to operation of WBN, Units 1 and 2."

(8) Letter from TVA to NRC dated January 13, 1993, "Initial Test Program - Response to NRC Request for Additional Information (RAI) on FSAR Chapter 14, Amendment 69."

(9) Letter from TVA to NRC dated April 2, 1993, "Initial Test Program - FSAR Chapter 14."

(10) NUREG 0847, Supplement 12, dated October 1993, "Safety Evaluation Report related to operation of WBN, Units 1 and 2."

(11) Letter from TVA to NRC dated August 19, 1994, "Final Safety Analysis Report (FSAR) Chapter 14 - Initial Test Program."

(12) NUREG 0847, Supplement 14, dated December 1994, "Safety Evaluation Report related to operation of WBN, Units 1 and 2."

(13) Letter from TVA to NRC dated January 5, 1995, "Reply to NRC staff position on Watts Bar FSAR Chapter 14."

(14) Letter from TVA to NRC dated July 13, 1995, "Fuel Handling and Vessel Servicing Equipment Testing."

Table 14.2-1 (Sheet 1 of 89) PREOPERATIONAL TESTS SUMMARIES INDEX

Table 14.2-1 (Sheet 2 of 89) PREOPERATIONAL TESTS SUMMARIES INDEX

Table 14.2-1 (Sheet 3 of 89) PREOPERATIONAL TESTS SUMMARIES INDEX

Table 14.2-1 (Sheet 4 of 89) ESSENTIAL RAW COOLING WATER SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the capability of each train of the Essential Raw Cooling Water System to supply required cooling water flow to assigned loads in all modes of operation.

PREREQUISITES

1. Verify system is configured as needed for unit/plant operation.
2. Intake channel from the river is open to supply water.
3. AC and DC electrical power supplies are available.
4. Control air supply is available.

TEST METHOD

1. Verify manual and automatic operation of valves.
2. Verify proper operation of pressure controllers.
3. Check for proper functioning of instrumentation, alarms, and interlocks.
4. Confirm that the isolation valves will properly respond to an isolation signal.
5. Verify proper system flow balancing.
6. Verify valve time response meets design requirements.
7. Verify proper assignment of train related equipment and proper operation independent of other trains.

ACCEPTANCE CRITERIA

1. Essential raw cooling water flow through system loads, where measurable with installed instrumentation, meets the design requirements described in FSAR Section 9.2.1 and appropriate design documents.
2. Manual and automatic controls, interlocks, alarms, and instrumentation function in accordance with design documents and as described in FSAR Section 9.2.1.

Table 14.2-1 (Sheet 5 of 89) ESSENTIAL RAW COOLING WATER SYSTEM TEST SUMMARY

ACCEPTANCE CRITERIA (Cont'd)

. 3. Train related equipment is assigned to the proper train and operates properly independent of the other trains.

Table 14.2-1 (Sheet 6 of 89) PRIMARY MAKEUP WATER SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the capability of the Primary Makeup Water System to furnish demineralized water for use as reactor coolant.

PREREQUISITES

1. The applicable primary makeup water storage tank has been filled with demineralized water to a level adequate to perform the test.
2. Adequate preparations for receipt/disposal of water at distribution points.

TEST METHOD

1. Verify proper operation of the primary makeup water pumps and confirm their capability to supply reactor makeup water to all distribution points.
2. Demonstrate proper functioning of instrumentation, interlocks, and alarms.

ACCEPTANCE CRITERIA

1. The hydraulic performance of the primary makeup water pumps meets or exceeds performance characteristics described by applicable design documents.
2. Primary makeup water can be supplied to the locations shown in design drawings.
3. Instrumentation, controls, annunciators and interlocks function properly in response to simulated or normal input signals in accordance with design documents.

Table 14.2-1 (Sheet 7 of 89) COMPONENT COOLING WATER SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the capability of each train of the Component Cooling Water System to supply required cooling water flows to its associated safety-related loads for the subsystem associated with unit being tested. To demonstrate the capability of the system to respond and be manually aligned to the safety-related configuration.

PREREQUISITES

1. Essential Raw Water System is operational.
2. An acceptable water supply is available to component cooling water surge tanks.
3. Control air is available.
4. AC and DC electrical power supplies are available.
5. The system is filled and vented to support performance of the test.
6. System loads are available to receive flow.

TEST METHOD

1. Demonstrate manual modes of operation, including most restrictive pump/loop combinations.

2 Verify automatic starting of the 2B-B CCS pump upon loss of CCS discharge pressure.

3. Demonstrate automatic isolation of the non-safety-related loop from the rest of the subsystem, and isolation between the two safety-related loops.
4. Verify instrumentation, controls, alarms and interlocks operate per design.
5. Verify proper system flow balancing of the safety and non-safety-related loops.
6. Verify proper assignment of train related equipment and proper operation independent of other trains.

Table 14.2-1 (Sheet 8 of 89) COMPONENT COOLING WATER SYSTEM TEST SUMMARY

ACCEPTANCE CRITERIA

1. Cooling water flow distribution to the non-safety related and safety related loads is in accordance with design as described in FSAR Section 9.2.2 and design documents.

2. The hydraulic performance of the thermal barrier booster pumps meets or exceeds design requirements described in FSAR Section 9.2.2 and applicable design documents.

3. Manual and automatic controls, including system isolation features, interlocks, alarms, and instrumentation, function as described in FSAR Section 9.2.2 and in accordance with design documents.

4. Isolation valve closure times meet or exceed design and Technical Specification requirements.

5. Train related equipment is assigned to the proper train and operates independent of the other train.

Table 14.2-1 (Sheet 9 of 89) PROCESS SAMPLING SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the capability of the Process Sampling System to provide liquid and gas samples through the correct flow path from sample points in non-secondary systems and to determine sample line holdup times.

PREREQUISITES

1. Plant conditions are established as necessary to facilitate drawing of liquid and gas samples from the required sampling locations.
2. Necessary tanks and sampling devices are available for receiving sample effluents and relief valve discharge.
3. Cooling water is available to sample coolers as required.
4. AC and DC electrical power supplies are available.
5. Required instruments and analyzers are calibrated.
6. Sample hood ventilation systems are operational.

TEST METHOD

1. Demonstrate proper system operation with regard to flow paths, flow capacity, and mechanical operability using grab samples and sample analyzers.
2. Verify the operation of the sample analyzers, sample coolers, sample selection valves, and pressure reducing and regulating equipment.
3. Verify operation of instrumentation, detectors, interlocks, and alarms.
4. Demonstrate, by flow rate vs. line length, that the sample line holdup time is within allowable limits.
5. Establish required purge times.
6. Demonstrate isolation valves operate properly on receipt of isolation signal.

Table 14.2-1 (Sheet 10 of 89) PROCESS SAMPLING SYSTEM TEST SUMMARY

ACCEPTANCE CRITERIA

1. Instrumentation and controls, including automatic isolation valves, interlocks, and alarms, operate properly in response to simulated or normal operating inputs as described in FSAR Section 9.3.2 and applicable design documents.

2. Sample line delay times are in accordance with FSAR Section 9.3.2.3.

3. Samples from non-secondary systems (including waste disposal, primary makeup water, steam generator blowdown, etc.) can be satisfactorily collected from designated sample points described in FSAR Section 9.3.2.2.

Table 14.2-1 (Sheet 11 of 89) Deleted by Amendment 99

Table 14.2-1 (Sheet 12 of 89) LIQUID WASTE DRAINS, COLLECTION AND TRANSFER SYSTEM TEST SUMMARY

OBJECTIVE

Demonstrate the capability of the floor and equipment drains to direct drainage from areas housing safety related equipment, radioactive and potentially radioactive liquids, chemicals and oils to designated collection points for transfer to storage tanks or processing systems.

PREREQUISITES

1. Floor and equipment drain lines and sumps have been cleaned of construction debris and are capable of receiving and transferring liquids.

2. Drain collector tanks and associated transfer pumps are operable.

3. AC and DC electrical power supplies are available.

TEST METHOD

1. Deliver water to each floor and equipment drain and verify capability of the drains to remove the water.

2. Verify setpoints for sump levels and pump actuation.

3. Verify flood detection instrumentation and alarm actuation.

ACCEPTANCE CRITERIA

1. All floor and equipment drains are clear of obstruction and direct waste liquids to the proper location.

2. RHR and Containment Spray Pump compartment drains transfer liquid waste at design flowrates as described in FSAR Section 9.3.3.

3. Automatic controls, interlocks and alarms operate in accordance with design drawings.

4. Sump and/or drain pumps operate in accordance with design drawings to control sump or tank level as described in FSAR Section 9.3.3.

5. Flood detection instrumentation and alarm operates as described in FSAR Section 6.3.2.11.3.

Table 14.2-1 (Sheet 13 of 89) FIRE PROTECTION SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the operability of the Fire Protection System, including its fire detection and fire suppression functions, in accordance with design requirements.

PREREQUISITES

1. The Fire Protection System pumps, piping, controls and associated valves and dampers are operational.

2. Fire protection and detection instrumentation is calibrated.

TEST METHOD

1. Verify the proper functioning of the fire detection devices to activate the automatic fire protection system, alert the appropriate control location, initiate fire alarms, and to activate automatic closure of fire dampers, as required.

2. Demonstrate the automatic start feature of the Fire Protection System pumps.

3. Demonstrate the capability of the Fire Protection System and pumps to supply water to required areas of the Unit 2 annulus, reactor building, Unit 1 and Unit 2 Auxiliary Feedwater systems in Flood Mode and the penetration room on 713' elevation at the purge air area.

4. Verify system vibrations are within design limits.

5. Demonstrate proper operation of system instrumentation, alarms, controls and interlocks.

Table 14.2-1 (SHEET 14 OF 89) FIRE PROTECTION SYSTEM TEST SUMMARY

ACCEPTANCE CRITERIA

1. The automatic fire detection system provides indication, annunciation and suppression system actuation outputs in accordance with design documents as described in FSAR Section 9.5.1.

2. The Fire Protection System automatic and manual controls, interlocks, alarms, and instrumentation operate properly and in accordance with design drawings.

Table 14.2-1 (Sheet 15 of 89) Deleted by Amendment 88

Table 14.2-1 (Sheet 16 of 89) Deleted by Amendment 88

Table 14.2-1 (Sheet 17 of 89) RESIDUAL HEAT REMOVAL SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the operability of the Residual Heat Removal (RHR) System and its capability to provide recirculation flows required to remove heat from the Reactor Coolant System (RCS).

PREREQUISITES

1. The RCS is filled with water for portions of the test, as required.
2. The RWST is filled with water as required to perform tests.
3. Hot functional testing is in progress or the RCS is hot as required to demonstrate RHR cooldown capability.

TEST METHOD

1. Verify the logic, controls, and interlocks associated with the valves in the RHR system.
2. Demonstrate acceptable pump performance while operating on the miniflow bypass line with the discharge isolation valves closed.
3. Demonstrate recirculation capability within the isolated RHR.
4. Demonstrate RHR operation during heatup with letdown from the RHR system through the chemical and volume control system.
5. Demonstrate RHR operation during plant cooldown following hot functional testing.
6. Demonstrate RHR mid-loop operation.
7. Verify proper operation of alarms, controls, and interlocks.

   NOTE: The safety injection functions of RHR system will be demonstrated during testing of Safety Injection System.

ACCEPTANCE CRITERIA

1. The hydraulic performance of the RHR pumps meets or exceeds design requirements as described in FSAR Section 5.5.7.
2. Automatic and manual controls, interlocks, and alarms operate in accordance with design drawings.
3. The RHR system provides low pressure letdown and heat removal for RCS cooldown in accordance with design requirements as described in FSAR Section 5.5.7.

Table 14.2-1 (Sheet 18 of 89) CHEMICAL AND VOLUME CONTROL SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the operability of the Chemical and Volume Control System (CVCS), including the capability to maintain charging and letdown flows, to maintain seal-water injection flow to the reactor coolant pumps, to maintain water chemistry conditions, to provide reactor makeup control, and to adjust reactor coolant boron concentration.

PREREQUISITES

1. Applicable portions of the Reactor Coolant System are capable of operationally interfacing with the Chemical and Volume Control System as required.

2. A cooling water supply is available for the heat exchangers as necessary for test performance.

3. Systems required to supply cover gas to the Volume Control Tank (VCT) are operational, and adequate supplies of gas are available.

TEST METHOD

1. Verify proper functioning of charging and letdown system components, including the hydraulic performance of the charging pumps, and operability of heat exchangers, letdown orifices, and control valves.

2. Demonstrate the capability to maintain seal water flow to the reactor coolant pumps.

3. Verify proper flows and pressure drops for seal injection and reactor coolant filters.

4. Verify proper operation of the volume control tank level and pressure control, including testing of the automatic makeup, dilution, alternate dilute, borate, and manual modes of reactor makeup control and cover gas system.

5. Check proper operation of instrumentation, interlocks, and alarms.

6. Verify proper operation of valves in the boric acid subsystem.

7. Demonstrate the chemical control function of the CVCS by verifying the capability of the system to introduce chemicals into the charging flow for pH and oxygen control, and that the system is capable of maintaining a gas pressure in the volume control tank as required during the applicable modes of operation.

Table 14.2-1 (Sheet 19 of 89) CHEMICAL AND VOLUME CONTROL SYSTEM TEST SUMMARY

TEST METHOD (Cont'd)

8. Verify proper flows to the in-service mixed bed and the cation demineralizer, and determine pressure drops and effectiveness of demineralizers and filters.

9. Verify that boric acid can be transferred to other systems as required.

ACCEPTANCE CRITERIA

1. The hydraulic performance of the charging pumps meets or exceeds design requirements as described in FSAR Section 9.3.4.

2. Charging and letdown normal and alternate flowpaths, including heat exchangers, letdown orifices, and control valves, operate in accordance with design requirements as described in FSAR Section 9.3.4.

3. Automatic and manual controls, including chemical and automatic reactor makeup water control, interlocks, and alarms, operate in accordance with design drawings.

4 Boric acid can be transferred in accordance with design requirements as described in FSAR Section 9.3.4.

Table 14.2-1 (Sheet 20 of 89) FLOOD MODE BORATION SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate that the Auxiliary Charging equipment is capable of supplying makeup water flow to the Reactor Coolant System as required during flood mode operations.

PREREQUISITES

1. All necessary support systems are operational as required.
2. All equipment required for flood mode preparation is available.
3. Auxiliary Charging filters are installed and demineralizer loaded.

TEST METHOD

1. Verify proper operation of instrumentation
2. Demonstrate the equipment's ability to provide required flow to the Reactor Coolant System from the various makeup water sources.
3. Install all equipment required for flood mode operation.
4. Transfer makeup water from the preferred sources to the auxiliary makeup tank.
5. Connect fire hose between the High Pressure Fire Protection System and the auxiliary makeup tank. (Fire water is not to be introduced into the tank).

ACCEPTANCE CRITERIA

1. The hydraulic performance of the auxiliary charging pumps meets or exceeds design requirements as described in FSAR Section 9.3.6.
2. Equipment required for flood mode operation can be properly installed.
3. All automatic controls, interlocks and alarms function in accordance with design drawings.

Table 14.2-1 (Sheet 21 of 89) Deleted by Amendment 97

Table 14.2-1 (Sheet 22 of 89) SAFETY INJECTION SYSTEM TEST SUMMARY

OBJECTIVE

Demonstrate the ability of the centrifugal charging pumps, the safety injection pumps and Residual Heat Removal (RHR) pumps to deliver required flows to the RCS during injection phase and recirculation modes of operation.

Verify the discharge characteristics and proper system actuation for each of the SI accumulators.

Adjust flow to all SI branch lines for even flow distribution and total flow rate to prevent charging and safety injection pumps from exceeding runout conditions.

PREREQUISITES

1. The RCS is drained down and the reactor head and internals are removed as required for portions of system tests.
2. The RCS is at nominal operating pressure and temperature for the check valve operability portion of system tests.
3. A supply of nitrogen or compressed air is available for pressurizing the accumulators.
4. Demineralized water or borated water is available to fill the accumulators.
5. The Refueling Water Storage Tank (RWST) is filled with demineralized water or borated water at refueling concentration.

TEST METHOD

Injection Mode

1. Verify that all valves and components required for SIS operation are sequenced properly and are actuated within minimum required times.
2. With the RHR, centrifugal charging and safety injection pumps aligned to take suction from the RWST, sufficient data will be taken to ensure satisfactory operation in the miniflow mode.
3. With the reactor vessel head and internals removed, and the pumps taking suction from the RWST, perform full flow tests for each pump. Measure flow and discharge pressure and adjust system as necessary to ensure pumps do not exceed their maximum runout conditions. Measure flows in branch lines and adjust as necessary to ensure that flow distributions for all injection lines are within limits.
4. Align RHR and safety injection pumps to deliver to RCS hot legs. Measure and adjust flows as applicable.

Table 14.2-1 (Sheet 23 of 89) SAFETY INJECTION SYSTEM TEST SUMMARY

TEST METHOD (Cont'd)

5. Verify proper operation of controls, interlocks and alarms.
6. Verify the primary safety injection to reactor coolant loop check valves will open with the RCS at nominal operating pressure and temperature.

Accumulators

1. Each accumulator is partially pressurized and its discharge valve is then opened allowing discharge to the RCS. Level and pressure measurements are used to calculate line resistance.
2. Verify that the check valves in the accumulator discharge lines will open with the RCS in a hot condition.
3. Verify proper operation of controls, interlocks and alarms.
4. Verify operability of nitrogen fill, venting, and accumulator makeup system.

Recirculation Mode

1. Verify auto operation of RHR pump containment sump suction valves swapover.
2. Align safety injection and centrifugal charging pumps to take suction from the discharge of the RHR pumps and deliver to the RCS cold legs with RHR taking suction from the RWST or the Reactor Coolant System hot leg. Measure flows and adjust as necessary to ensure that RHR pumps deliver adequate flow to safety injection and centrifugal charging pump suctions under runout conditions.
3. Verify proper operation of all controls and interlocks, RWST immersion heaters, and alarms.
4. Verify proper operation of containment sump wide and narrow range water level instrumentation used in conjunction with post accident monitoring.

ACCEPTANCE CRITERIA

1. The hydraulic performance of the safety injection pumps meets or exceeds design requirements as described in FSAR Section 6.3.
2. The flow rates and flow distribution for all branch lines are in accordance with design requirements.

Table 14.2-1 (Sheet 24 of 89) SAFETY INJECTION SYSTEM TEST SUMMARY

ACCEPTANCE CRITERIA (Cont'd)

3. Accumulator discharge performance is in accordance with design requirements as described in FSAR Section 6.3.

4 Automatic and manual controls, interlocks and alarms function properly in response to normal or simulated input signals in accordance with design drawings. RWST temperature maintenance functions as described in accordance with design drawings.

5. Primary SIS to RCS check valves will open with the RCS at nominal operating conditions.

Table 14.2-1 (Sheet 25 of 89) CONTAINMENT SPRAY SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the hydraulic performance of the containment spray pumps, proper function of the spray nozzles and headers and the proper operation and actuation of the system components.

PREREQUISITES

1. A water supply is available for the containment spray pumps.
2. An air supply is available for testing the Containment Spray and RHR spray nozzles.
3. The Containment Spray pumps discharge lines and the RHR spray lines are isolated to prevent spraying water into the Containment Building.

TEST METHOD

1. Through test connections, pass air under pressure to the containment spray and RHR spray nozzles and ring headers to ensure that they are free of obstructions.
2. Operate the Containment Spray pumps through the Refueling Water Storage Tank (RWST) recirculation path to verify pump hydraulic performance.
3. Verify proper sequencing of valves and pumps in response to simulated safeguards actuation signals.

ACCEPTANCE CRITERIA

1. The hydraulic performance of the containment spray pumps meets or exceeds design requirements described in FSAR Section 6.2.2.
2. Containment and RHR spray headers and nozzles are not obstructed.
3. Manual and automatic controls, interlocks, alarms, and instrumentation function as described in FSAR Section 6.2.2 and applicable design documents.

Table 14.2-1 (Sheet 26 of 89) INTEGRATED ENGINEERED SAFETY FEATURES ACTUATION SYSTEM TEST SUMMARY

OBJECTIVE

Demonstrate proper automatic actuation, alignment and operation of all ESF components controlled by the Engineered Safety Features Actuation System (ESFAS) with and without offsite power.

PREREQUISITES

1. The Reactor Coolant System (RCS) is cold and drained down and the reactor vessel head and internals are removed.
2. The Refueling Water Storage Tank has an adequate supply of water.
3. The Containment and RHR spray lines to containment are isolated to prevent spraying into the Containment Building. The containment sump has been isolated to prevent back filling.
4. Actuation circuitry has been tested and is capable of actuating all equipment upon manual ESFAS actuation.
5. All ESF systems and equipment required to actuate on a ESFAS signal are operational and are aligned for normal operation.

TEST METHOD

1. Conduct the ESF test under normal power conditions, one load group at a time, to demonstrate ESF system independence and redundance.
2. Verify the proper actuation, alignment and operation of all ESF equipment in response to the ESFAS signal.
3. Verify the ESF component position following reset of the ESFAS signal.
4. Verify all train-related ESF equipment operates correctly, including emergency diesel start and load sequence, pump start times and valve actuation upon initiation and reset of an ESFAS signal coincident with a simulated station blackout.

ACCEPTANCE CRITERIA

1. The ESF components operate and properly align in response to a ESFAS signal in accordance with design as described in FSAR Section 7.3.1.
2. The diesel generators start and sequence loads when offsite power is not available as described in FSAR Section 8.3.1.
3. Components actuated by a ESFAS signal remain in the actuated condition after reset of the initiating signal.

Table 14.2-1 (Sheet 27 of 89) INTEGRATED ENGINEERED SAFETY FEATURES ACTUATION SYSTEM TEST SUMMARY

ACCEPTANCE CRITERIA (Cont'd)

4. During single load group actuation tests, all components function in accordance with the requirements of FSAR Section 7.3.1.

Table 14.2-1 (Sheet 28 of 89) LIQUID WASTE PROCESSING SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the capability of the Liquid Waste Processing System to process liquid waste and transfer these wastes to their respective disposal points.

PREREQUISITES

1. All necessary supporting equipment is operational.

TEST METHOD

1. Demonstrate proper operation of all components in the system.
2. Verify the proper operation of the reactor coolant drain tank and associated equipment.
3. To demonstrate that the system pumps are capable of providing design flow rates.
4. To verify the operability of the system instrumentation, controls, and interlocks, including effluent isolation features and applicable alarms and setpoints associated with the liquid waste processing.

ACCEPTANCE CRITERIA

1. The system will process waste liquids from all assigned collection points through system filters to designated discharge and storage locations in accordance with design requirements as described in FSAR Section 11.2.
2. Automatic and manual controls, interlocks, including process effluent isolation, and alarms operate in accordance with design drawings and vendor documents.

Table 14.2-1 (Sheet 29 of 89) Deleted by Amendment 88

Table 14.2-1 (Sheet 30 of 89) GASEOUS WASTE PROCESSING SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the capability of the gaseous waste portion of the Waste Disposal System (WDS) to collect and store gases from various plant sources and discharge them to the environment in a controlled manner.

PREREQUISITES

1. The gaseous waste components, piping, all valves and instrumentation are available as required.

TEST METHOD

1. Demonstrate proper operation of the system and its components.
2. Check for proper functioning of controls, interlocks and alarms.

ACCEPTANCE CRITERIA

1. Automatic and manual controls, interlocks, and alarms operate in accordance with design drawings.

Table 14.2-1 (Sheet 31 of 89) PROCESS AND EFFLUENT RADIATION MONITORING SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the capability of the Process and Effluent Radiation Monitoring System to continuously monitor radiation levels of plant liquid and gaseous processes and effluent, record radiation levels, and initiate radiation alarm and isolation signals.

PREREQUISITES

1. A remotely operable check source is available for each radiation detector as required.
2. Electrical power is available at each radiation detector, control panel and associated equipment.
3. System leak tightness has been verified as applicable.
4. Calibration of detector instrumentation is complete, including alarm setpoints, using a radioactive calibration source.
5. Calibration at associated isokinetic sampling systems has been completed.

TEST METHOD

1. Verify operability of sample coolers.
2. Verify proper operation of associated isokinetic sampling systems.
3. Verify operation and response of the process and effluent radiation monitors to the appropriate check sources, including local and control room indicating and recording devices, as applicable.
4. Verify the applicable alert, high radiation and circuit malfunction annunciators function properly.
5. Verify proper operation of interlock functions and devices.
6. Verify pump or fan operation, flow indication and control, and proper valve operation.

Table 14.2-1 (Sheet 32 of 89) PROCESS AND EFFLUENT RADIATION MONITORING SYSTEM TEST SUMMARY

ACCEPTANCE CRITERIA

1. Liquid and gaseous monitors provide high radiation and instrument malfunction alarms and indication in accordance with design requirements as described in FSAR Section 11.4.
2. Automatic interlocks operate in accordance with design drawings.
3. Sample flowrates are in accordance with design requirements and vendor documents.
4. Sample and recirculation pumps, indicators, and control valves operate in accordance with design drawings and vendor documents.

Table 14.2-1 (Sheet 33 of 89) AREA RADIATION MONITORING SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the capability of area radiation and airborne radioactivity monitors to provide continuous surveillance of radiation levels throughout accessible areas of the plant site, alert personnel of excessive dose rate levels, and provide direct reading indication and/or recording of dose rates at local and/or central (Main Control Room) monitoring locations.

PREREQUISITES

1. Calibration of each monitor including detectors, indicators, recorders, alarm setpoints and indicating lights is complete.
2. A check source is available for each radiation detector as required.
3. Electrical power is available to each radiation detector, control panel and associated equipment.

TEST METHOD

1. Verify operation and response of the area and airborne radiation monitors with the appropriate check source, including local and control room indicating and recording functions.
2. Verify the applicable alert, high radiation level and malfunctions annunciators function properly.
3. Verify the operation of pumps, flow control instrumentation, and control valves for all airborne radioactivity monitors.

ACCEPTANCE CRITERIA

1. Area radiation and fixed airborne radioactivity monitors provide radiation indication and alarms in accordance with design requirements as described in FSAR Section 12.3.4.
2. Sample pumps, indicators and control valves operate in accordance with design drawings and vendor documents.

Table 14.2-1 (Sheet 34 of 89) Deleted by Amendment 97

Table 14.2-1 (Sheet 35 of 89) Deleted by Amendment 97

Table 14.2-1 (Sheet 36 of 89) AUXILIARY BUILDING VENTILATION SYSTEM

OBJECTIVE

To demonstrate the proper operation of the Auxiliary Building HVAC systems including north and south valve vault exhaust and turbine driven auxiliary feedwater pump room exhaust, air handling units, fans, cooling coils, and air flow.

PREREQUISITES

1. AC and DC electrical power supplies are available.
2. Cooling water is available to cooling coils.

TEST METHOD

1. Demonstrate proper operation of the turbine driven auxiliary feedwater pump room exhaust fans, and air handling units.
2. Demonstrate proper operation of the cooling units in the building equipment rooms.
3. Ensure proper operation of instrumentation, interlocks, and alarms.

Table 14.2-1 (Sheet 37 of 89) AUXILIARY BUILDING VENTILATION SYSTEM

ACCEPTANCE CRITERIA

1. Safety Features Equipment Room Coolers maintain air and water flows as described in FSAR Section 9.4.5.3.

2. Manual and automatic controls, interlocks, auto start and isolation features, and alarms function in accordance with design documents.

Table 14.2-1 (Sheet 38 of 89) CONTAINMENT VENTILATION SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the capability of the Containment Ventilation System to provide air recirculation, proper cooling of upper and lower containment spaces, CRDM cooling, reactor vessel cavity and nozzle support cooling, and proper air purging and filtration.

PREREQUISITES

1. Cooling water is available for the air handling units.
2. Control air is available for valve and damper operations.
3. AC and DC electrical power supplies are available.
4. Access hatches, plugs, gates and shields are in their proper positions.

TEST METHOD

1. Verify the Containment Purge Supply and Exhaust fans are capable of delivering design air flow capacities and filtration portions of system operate in accordance with design requirements.
2. Verify the Upper Compartment Cooling units maintain upper containment spaces within design temperatures.
3. Verify the Lower Compartment Cooling, in conjunction with CRDM cooling units, maintain lower containment spaces within design temperatures.
4. Verify return air fans can deliver design air flow including that drawn from the hydrogen collection header.
5. Verify instrument room cooling system and associated chilled water equipment will operate per design documents.
6. Operate all fans, dampers, and flow control equipment to demonstrate proper manual control and automatic response to start, stop and isolation input signals.

Table 14.2-1 (Sheet 39 of 89) CONTAINMENT VENTILATION SYSTEM TEST SUMMARY

ACCEPTANCE CRITERIA

1. CRDM Cooling units, in conjunction with the Lower Compartment Cooling units, will maintain design temperatures in the lower containment areas as described in FSAR Section 9.4.7.

2. Containment Purge Supply and Exhaust System maintains containment atmospheric conditions to control release of contamination through HEPA and carbon adsorber beds described in Section 6.5 and provides for isolation of airflow paths to primary containment as described in FSAR Section 9.4.6.

3. Upper Containment Cooling units, will maintain design temperatures in the upper containment areas as described in FSAR Section 9.4.7.

4. Lower Containment Cooling units, in conjunction with CRDM Cooling units, will maintain design temperatures in the lower containment areas as described in FSAR Section 9.4.7.

5. Air return fans will recirculate air, including that from the hydrogen collection ductwork system, and discharge into the return air fan suction.

6. Incore Instrument room cooling units and associated chilled water systems will maintain the Incore Instrument room environment at design temperatures and provide proper containment isolation as described in FSAR Section 9.4.7.

7. Manual and automatic controls, including isolation features, interlocks, alarms, and instrumentation, function as described in FSAR Section 9.4.6 or 9.4.7, as applicable, and in accordance with design documents.

Table 14.2-1 (Sheet 40 of 89) COMBUSTIBLE GAS CONTROL SYSTEMS TEST SUMMARY

OBJECTIVE

To demonstrate the proper operation of the Hydrogen Mitigation System (HMS) igniters.

To demonstrate the ability of the hydrogen sampling system to detect the presence of hydrogen in the primary containment atmosphere and give indication in the control room.

PREREQUISITES

1. Each igniter has been energized for the minimum required duration and allowed to cool prior to test conduct.

2. Containment Ventilation Systems are operational.

TEST METHOD

1. Verify sample system controls, interlocks, instrumentation, and alarms operation.

2. Demonstrate the hydrogen analyzer is capable of drawing and analyzing a sample and returning it to containment. Verify alarms and instrumentation function properly.

3. Energize each HMS igniter circuit and verify the voltage, current, and igniter surface temperature.

ACCEPTANCE CRITERIA

1. The hydrogen analyzer system is capable of sampling and analyzing the containment atmosphere in accordance with design requirements as described in FSAR Section 6.2.5.

2. All HMS igniters are operational at a minimum required surface temperature without exceeding a maximum power requirement calculated from circuit current and voltage readings, in accordance with the applicable design documents.

3. Controls, interlocks, instrumentation, and alarms operate in accordance with FSAR Section 6.2.5 and design drawings.

Table 14.2-1 (Sheet 41 of 89) SECONDARY CONTAINMENT VENTILATION SYSTEM TEST SUMMARY

OBJECTIVE

The Secondary Containment System is comprised of the Shield Building of each reactor unit and that portion of the Auxiliary Building which serves as the Auxiliary Building Secondary Containment Enclosure (ABSCE). The Emergency Gas Treatment System (EGTS) is provided for ventilation control and cleanup of the atmosphere inside the annulus between the Shield Building and the Primary Containment Building of each reactor unit.

To demonstrate the ventilation systems will maintain shield building conditions that prevent potentially radioactive gases leaking from primary containment or auxiliary building secondary containment enclosure from directly reaching the outside environment.

PREREQUISITES

1. AC and DC electrical power supplies are available.
2. Compressed air is available.
3. Ventilation systems are sufficiently complete to support testing.
4. Building pressure boundary is complete as required to support testing.

TEST METHOD

1. Verify capability of the Emergency Gas Treatment System Annulus Vacuum Control Subsystem to maintain the correct Shield Building pressure at design-flow.
2. Verify capability of the Emergency Gas Treatment System Air Cleanup Units to maintain the correct annulus pressure at design flow.

3 Verify that instrumentation, valve, damper and train separation interlocks operate per design.

Table 14.2-1 (Sheet 42 of 89) SECONDARY CONTAINMENT VENTILATION SYSTEM TEST SUMMARY

TEST METHOD (Cont'd)

4. Verify all setpoints, alarms, and indication.

.5 Verify that EGTS modulation dampers operate properly to maintain building pressure differentials within design limits.

ACCEPTANCE CRITERIA

1. Emergency Gas Treatment System (EGTS) maintains annulus atmospheric conditions to control the release of contamination through HEPA filters and carbon adsorbers. Maintains air flow and pressure differentials within the annulus area as described in FSAR Section 6.5.

2. Manual and automatic controls, including isolation features, valve and damper closure times, interlocks, alarms, and instrumentation, function in accordance with design documents.

Table 14.2-1 (Sheet 43 of 89) Deleted by Amendment 97

Table 14.2-1 (Sheet 44 of 89) DIESEL GENERATORS TEST SUMMARY

OBJECTIVE

To demonstrate the operability and reliability of the Unit 2 diesel generator units and associated auxiliaries, including proper starting and load sequencing.

PREREQUISITES

1. DC and AC power is available for controls, alarms, protective relays, air starting solenoid valves and generator field flashing.
2. Compressed air is available for air starting of the diesel generator unit.
3. Fuel and Lubrication Oil Systems are available to support testing.
4. Cooling water is available and operational.
5. Diesel generator building ventilation is operational.

TEST METHOD

1. Verify the capability of the diesel generator unit to accept the sequenced equipment by utilizing actual plant loads appropriate for existing plant conditions. The loads will be added at the proper sequence and time duration, while maintaining an acceptable voltage and frequency.
2. Demonstrate ability to synchronize each diesel generator unit with the offsite power source while the unit is loaded, transfer the load from the unit to the offsite source, isolate the diesel generator unit, and place it in standby status.
3. Demonstrate the WBN Unit 2 diesel generators automatic trip are bypassed on automatic or emergency start signal except for engine over speed and generator differential current.

Table 14.2-1 (Sheet 45 of 89) DIESEL GENERATORS TEST SUMMARY

TEST METHOD (Cont'd)

4. Perform load rejection tests, including loss of the largest single load and complete loss of load, and verify proper voltage is maintained and overspeed limits are not exceeded.

Table 14.2-1 (Sheet 46 of 89) DIESEL GENERATORS TEST SUMMARY

ACCEPTANCE CRITERIA

1. Each diesel generator unit will accept assigned shutdown loads in the required sequence and maintain speed, voltage and frequency within design limits as described in FSAR Section 8.3.1.
2. Each diesel generator will maintain voltage and speed upon rejection of largest single load and full load as described in FSAR Section 8.3.1.

Table 14.2-1 (Sheet 47 of 89) AC POWER DISTRIBUTION SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the capability of the two offsite power sources (normal and alternate) to independently provide power to the 6.9 kV and 480 Volt AC Distribution System. The load carrying capability of system components (transformers, switchgear, breakers, etc.) will be demonstrated by proper normal operation of loads. Other design features, including redundance, independence, interlocks, load shedding, and manual and automatic transfers, will be demonstrated.

PREREQUISITES

1. The common and unit station service transformers are energized or available for energization as required.
2. The shutdown boards are energized or available for energization as required.
3. Metering and relaying circuits have been calibrated and are operational.
4. DC control power is available as required.
5. Diesel generator sets and supporting auxiliaries are operational as required.
6. Transformer tap settings are in accordance with design requirements.

TEST METHOD

1. Confirm that all Unit 2 Train A and Train B ESF controlled equipment (valves, pumps, fans, etc.) operate correctly upon initiation of ESFAS signal and after reset with a loss of trained offsite power.
2. Verify operation of alarms, tripping devices, and interlocks are in accordance with design drawings.
3. Demonstrate proper operation of remote and local control and indicating devices for shutdown boards and safety related motor control centers.

Table 14.2-1 (Sheet 48 of 89) AC POWER DISTRIBUTION SYSTEM TEST SUMMARY

TEST METHOD (Cont'd)

4. Verify proper load shedding and/or undervoltage protection of all 6.9 kV shutdown boards and 480 Volt shutdown boards and confirm diesel generator units receive proper start signals.

5. Verify the capability of each common station service transformer to carry the load required to supply ESF loads for its respective load group under Unit 2 loss of coolant accident conditions, in addition to power required for shutting down the non-accident unit.

Table 14.2-1 (Sheet 49 of 89) AC POWER DISTRIBUTION SYSTEM TEST SUMMARY

ACCEPTANCE CRITERIA

1. The AC Distribution System provides power at the design voltage level to safety related loads when supplied power from the normal, alternate, or onsite (standby) power supplies as described in FSAR Section 8.3.1.

   Power supply to safety related loads will automatically and manually transfer to the onsite (standby) diesel units from the normal or alternate supply or manually from the diesel generator units back to the normal or alternate supply as described by FSAR Section 8.3.1.

2. Safety related load groups will operate independent of each other.

3. Interlocks, alarms, controls, tripping, and lockout devices function in accordance with design drawings.

4. AC Distribution System components operate in accordance with vendor and design documents and will carry loads approximating normal operating conditions.

5. Unit 1 hot standby loads will be less than rating of each transformer secondary winding.

6. Unit 2 ESF loads under accident conditions will be less than their associated transformer ratings.

Table 14.2-1 (Sheet 50 of 89) Deleted by Amendment 97

Table 14.2-1 (Sheet 51 of 89) Deleted by Amendment 97

Table 14.2-1 (Sheet 52 of 89) Deleted by Amendment 97

Table 14.2-1 (Sheet 53 of 89) Deleted by Amendment 97

Table 14.2-1 (Sheet 54 of 89) Integrated Computer System (ICS) Test Summary

OBJECTIVE

The purpose of this test is to verify the ICS has been internally wired properly and that the internal CPU, I/O and analog converters function properly.

To verify the operation of the computer for conversion and computer printout of process parameters.

To verify the proper operation of the ICS software.

PREREQUISITESı

1. The latest Operating System Software and Data files are loaded and operational.

TEST METHOD

1. Verify the software functions are processed accurately by the hardware.

2. Verify that the control processing and peripheral hardware operates in a manner to satisfy all requirements. (Note: Loop calibration is performed in the component test program.)

3. Verify the ICS will appropriately process input signals.

ACCEPTANCE CRITERIA

1. The ICS has been internally wired properly and the internal CPU, peripheral devices, Input/Output (I/O) and analog converters function properly.

2. The calibration and operation of the elements of the ICS results in accurate processing and display of analog and digital input signals using the ICS.

3. Installed application programs perform as designed.

Table 14.2-1 (Sheet 55 of 89) Deleted by Amendment 110

Table 14.2-1 (Sheet 56 of 89) Deleted by Amendment 88

Table 14.2-1 (Sheet 57 of 89) REACTOR PROTECTION SYSTEM TEST SUMMARY

OBJECTIVE

The purpose of this test is to verify that the operability of the Reactor Protection System (RPS) to perform protective functions and to verify satisfactory RPS response times.

PREREQUISITE

1. Electrical power must be available to operate the reactor plant and electrical systems, instruments and reactor protection system equipment and recorders.
2. Lead/Lag coefficients will be set according to Precautions, Limitations and Setpoints requirements and the circuitry is operable. There will be no attempt to defeat or desensitize the effect of lead/lag compensation for this test.
3. Associated instrumentation systems will be complete, aligned and calibrated in accordance with the Precautions, Limitations, and Setpoints Document.

TEST METHOD

1. Demonstrate proper operation of the Reactor Protection System and Engineered Safety Features Actuation System.
2. Determine the process fluid variable to sensor input delay time analytically.
3. Determine Reactor Protection System sensor response time using simulated input signals.
4. Verify the Eagle 21 delay time is within design requirements.
5. Verify the response of the logic channels for all logic combinations.
6. Determine reactor trip and bypass breaker delay time.
7. Compile the total reactor trip and ESFAS time response.

ACCEPTANCE CRITERIA

1. The Reactor Protection System and Engineered Safety Features Actuation System function in response to logic initiation signals in accordance with the design requirements as described in Sections 7.2 and 7.3 and design drawings.
2. Response time of the RPS logic channels including RPS sensors is in accordance with design requirements and the Technical Specifications and Requirements Document.

Table 14.2-1 (Sheet 58 of 89) ROD CONTROL SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate operability of the Rod Control System as installed prior to fuel load.

PREREQUISITES

1. The control rod drive mechanisms (CRDM) are connected without drive shafts installed.
2. The control rod drive system has been aligned in accordance with vendor instructions.
3. Electrical power supplies are available for operation.

TEST METHOD

1. Verify proper operation of the control rod drive motor-generator (MG) sets.
2. Verify correct rod drive mechanism magnetic coil polarity.
3. Verify correct sequencing of the control rod drive mechanisms.
4. Verify system operation in both manual and automatic modes up to maximum operating speeds using simulated input signals as applicable.
5. Verify operation of rod position step counters and bank overlap.
6. Verify proper operation of alarms and interlocks using simulated input signals as applicable.
7. Record the response time between a reactor trip breaker opening and the loss of gripper coil voltage.

ACCEPTANCE CRITERIA

1. The control rod drive system functions in the manual and automatic modes in accordance with vendor manuals, applicable design documents, and as described in FSAR Section 7.7.1.2.
2. The control rod drive MG sets operate in accordance with vendor technical manuals and applicable design documents.
3. The rod drive mechanisms properly sequence in accordance with vendor technical manuals.
4. The magnetic coil polarity for each CRDM is in accordance with vendor technical manuals.

Table 14.2-1 (Sheet 59 of 89) REACTOR PRESSURE BOUNDARY LEAKAGE DETECTION SYSTEM TEST SUMMARY

OBJECTIVE

Demonstrate that the instrumentation used to detect reactor coolant leakage is functional.

PREREQUISITES

1. Instruments are installed and available for testing.

TEST METHOD

1. Verify calibration and sensitivity of the humidity monitor and its proper annunciation and recording.
2. Verify proper calibration and annunciation of the reactor vessel flange leakoff and temperature monitors.
3. Verify proper operation and annunciation of an abnormal rate of rise in the Reactor Building floor and equipment drain sump.
4. Verify proper operation of containment temperature detectors and instrumentation.
5. Provide actual or simulated input signals to verify operation of alarms, controls, interlocks, and instrumentation.

ACCEPTANCE CRITERIA

1. Instrumentation and alarms operate properly and in accordance with design requirements as described in FSAR Section 5.2.7.

   Note: Containment air particulate and radioactive gas monitors are addressed in the test summaries for the Area and Process Radiation Monitoring System.

Table 14.2-1 (Sheet 60 of 89) EXCORE NUCLEAR INSTRUMENTATION TEST SUMMARY

OBJECTIVE

To demonstrate the operability of the Nuclear Instrumentation System including its ability to provide required indications, alarms, control and protective functions.

PREREQUISITES

1. The Nuclear Instrumentation System is installed and the calibration and alignment has been completed according to the manufacturer's instructions.

2. System has been operational for at least 1 hour.

3. The Solid State Protection System, Auxiliary Relay Panels, and the Annunciator System are operational as required to support this test.

TEST METHOD

1. Demonstrate the capability of source, intermediate and power range circuitry to respond to a simulated test signal.

2. Verify that the source range detectors properly respond to a neutron source.

3. Demonstrate proper operation of the auctioneered circuits and flux channel deviation signals.

4. Check all channels to verify high level trip functions, alarm setpoints, and audible count rates where applicable.

ACCEPTANCE CRITERIA

1. The control and indication functions and associated setpoints of the Nuclear Instrumentation System function in accordance with design requirements as described in FSAR Section 7.2 and the Precautions, Limitations and Setpoint Document.

2. The Nuclear Instrumentation System output signals to the reactor protection system function in accordance with FSAR Section 7.2.

3. The source range detectors respond properly to a neutron source.

4. The source range detectors are not adversely affected by electrical noise.

Table 14.2-1 (Sheet 61 of 89) LOOSE PARTS MONITORING SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the capability of the Loose Parts Monitoring System to detect and record acoustic disturbances which may be indicative of loose parts in the Reactor Coolant System.

PREREQUISITES

1. Electrical power supplies are available for operation.

2. Plant conditions are established to allow operation of the reactor coolant pumps.

TEST METHOD

1. Operate the system to verify proper operation of installed sensors, alarms, recorders, and audio monitors.

2. Verify channel calibration using a controlled mechanical input to verify proper function of each recorded channel.

ACCEPTANCE CRITERIA

1. The system operates in accordance with vendor instructions, applicable design documents, and as described in FSAR Section 7.6.7.

2. During operation, the signals are clearly discernable/detectable in the presence of background noise.

Table 14.2-1 (Sheet 62 of 89) Deleted by Amendment 88

Table 14.2-1 (Sheet 63 of 89) MAIN STEAM SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the operability and performance of Main Steam System components during hot functional testing.

PREREQUISITES

1. Steam dump control system is operational.
2. Main condenser and circulating water system are in operation.
3. Plant conditions during hot functional testing are established as necessary for test conduct.
4. All support systems required to perform the test are in service.

TEST METHOD

1. Demonstrate proper operation of the turbine bypass system (condenser steam dump valves) to actual or simulated signals.
2. Verify Main Steam components, controls, interlocks and alarms function properly in response to simulated or actual signals.

ACCEPTANCE CRITERIA

1. Main Steam components, controls, interlocks and alarms function in accordance with the requirements of FSAR Section 10.3.4 and associated design drawings.
2. The Turbine Bypass system (condenser steam dump valves) operates in accordance with the requirements of FSAR Section 10.4.4 and associated design drawings.

Table 14.2-1 (Sheet 64 of 89) STEAM GENERATOR SAFETY & ATMOSPHERIC RELIEF VALVES TEST SUMMARY

OBJECTIVE

To demonstrate the operability of the main steam atmospheric relief valves and to verify the setpoints of main steam safety valves associated with each steam generator.

PREREQUISITES

1. Plant conditions are established as necessary for test performance.
2. A pressure-assist device is available for use in lifting the safety valves and equipment is available to measure lift pressures.
3. Test equipment is available to measure the atmospheric relief valve stroke time.

TEST METHOD

1. Verify proper actuation and operation of the atmospheric relief valves at normal system operating pressure.
2. Lift each safety valve with a pressure-assist device and measure the lifting pressure to ensure setpoints are as required.
3. Determine the full stroke time of the atmospheric relief valves at normal system operating pressure.

ACCEPTANCE CRITERIA

1. Operation of the steam generator atmospheric relief valves and setpoints of the safety valves is in accordance with design requirements as described in the FSAR section 10.3.2.

2 The relief and safety valves reseat properly and do not chatter.

Table 14.2-1 (Sheet 65 of 89) MAIN STEAM AND FEEDWATER ISOLATION VALVES TEST SUMMARY

OBJECTIVE

To demonstrate the operability of the Main Steam and Feedwater Isolation and Bypass isolation valves including their ability to close automatically as required.

PREREQUISITES

1. Plant conditions are established as required for test performance.
2. Electrical power and control air is available.
3. The solid state protection systems is operable.
4. Test equipment is available to measure the valve closure time.

TEST METHOD

1. Demonstrate remote and control room operation of the isolation valves.
2. Verify the capability of the isolation valves to close upon receipt of an isolation signal.
3. Measure the closure time of isolation and bypass valves.
4. Verify that the isolation valves will close upon receipt of a simulated signal from either power train.

ACCEPTANCE CRITERIA

1. The main steam and feedwater isolation valves and bypass valves close upon receipt of an isolation signal in accordance with FSAR Sections 6.2.4, 10.3 and 10.4.7, as applicable.
2. Isolation valve closure time is in accordance with the Technical Specification requirements.
3. Controls, interlocks, alarms and operation of the main steam, feedwater isolation valves and bypass valves is in accordance with FSAR Sections 10.3 and 10.4.7, and the associated design drawings.

Table 14.2-1 (Sheet 66 of 89) STEAM GENERATOR BLOWDOWN SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the capability and performance of the Steam Generator Blowdown (SGBD) system and components to adequately control steam generator water chemistry.

PREREQUISITES

1. AC and DC electrical power is available.
2. Control air is available for component operation.
3. The condenser and cooling towers are available.
4. Plant conditions are established as necessary for test performance.

TEST METHOD

1 Demonstrate the SGBD isolation valves close upon required isolation signals.

2. Demonstrate SGBD flowpaths in normal and flood mode conditions.

3. Verify the SGBD system can maintain proper steam generator water chemistry during hot functional testing. Verify design blowdown rates can be achieved.

4. Verify SGBD controls divert blowdown from the cooling tower upon a high radiation signal.

5. Verify controls, interlocks and alarms function properly in response to actual or simulated signals.

ACCEPTANCE CRITERIA

1. SGBD components, control, including isolation features, interlocks and alarms function in accordance with FSAR Section 10.4.8 and associated design documents.
2. The SGBD system can achieve design blowdown rates and maintain steam generator chemistry as required by FSAR Section 10.4.8.

Table 14.2-1 (Sheet 67 of 89) FEEDWATER SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the capability of the Main Feedwater System to provide a regulated supply of water to the steam generators in response to steam generator level control demand.

PREREQUISITES

1. System sufficiently complete to support testing.
2. AC and DC electrical power supplies are available.
3. Control air is available for component operation.
4. The Condensate System is available and an adequate supply of clean water is available in the condenser hotwell and condensate storage tank.
5. The system is aligned for normal operation.
6. The steam generator feedwater pumps and associated lube oil and seal water auxiliaries are operational.
7. The automatic steam generator level controls for feedwater regulating and bypass valves have been calibrated and are operational.

TEST METHOD

1. Verify proper operation of the feedwater system controls, interlocks and alarms by normal operation or simulation of required input signals.
2. Operate the main (turbine driven) and standby (motor driven) feedwater pumps and verify performance on recirculation path during Integrated Hot Functional Tests.

ACCEPTANCE CRITERIA

1. The minimum flow capability of the main and standby feedwater pumps meets or exceeds design requirements as described in FSAR Section 10.4.7.
2. Automatic and manual controls, interlocks, and alarms operate properly in accordance with design drawings.

Table 14.2-1 (Sheet 68 of 89) CONDENSATE AND CONDENSER VACUUM SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the Condensate and Condenser Vacuum Systems are capable of providing an adequate supply of high quality deaerated water to the Feedwater System and maintain condenser water levels within operational limits.

PREREQUISITES

1. System sufficiently complete to support testing.
2. AC and DC electrical power supplies are available.
3. Control air is available for component operation.
4. A adequate supply of clean water is available in the condenser hotwell and condensate storage tank.
5. The vacuum pumps are operable and support systems including seal and cooling water are available.
6. The Condensate and Condenser Vacuum system is aligned for normal operation.

TEST METHOD

1. Verify proper operation and minimum flow capabilities, as applicable, of the condenser hotwell pumps, condensate booster pumps, and condensate demineralizer pumps.
2. Demonstrate proper differential pressure control and flow through the condensate polishing demineralizers and verify water quality of condensate polisher effluent.
3. Demonstrate level controls will maintain proper hotwell level.
4. Demonstrate the capability of the condenser vacuum system to draw and maintain a vacuum on the condenser and verify proper operation of the vacuum pumps.
5. Verify proper operation of the condensate and condenser vacuum system controls, interlocks and alarms by normal operation or simulation of required input signals.

Table 14.2-1 (Sheet 69 of 89) CONDENSATE AND CONDENSER VACUUM SYSTEM TEST SUMMARY

ACCEPTANCE CRITERIA

1. The minimum flow capability of the hotwell pumps and condensate booster pumps meets or exceeds design requirements as described in FSAR Section 10.4.7. Flow capabilities for the demineralizer condensate pumps are in accordance with vendor requirements.

2 Each condensate polishing demineralizer produces condensate quality water at 3300 to 3500 gpm and less than 60 psi differential pressure.

3. Each condensate vacuum pump can draw and maintain a vacuum of 1 inch Hg absolute at greater than or equal to 15 scfm.

4. Automatic and manual controls, interlocks, and alarms operate properly in accordance with design drawings.

Table 14.2-1 (Sheet 70 of 89) CONDENSER CIRCULATING WATER SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the capability of the Condenser Circulating Water (CCW) system to provide adequate cooling to the main turbine condensers for removing and dissipating waste heat from the power generation cycle.

PREREQUISITES

1. All instrumentation and electrical equipment associated with the CCW system has been tested and calibrated.
2. The tube side of the main turbine condenser is available.
3. The cooling tower is available with adequate water in the basin.
4. The CCW system is filled and vented.

TEST METHOD

1. Operate the CCW system with various pump configurations and verify pump hydraulic performance and proper flow to the condensers.
2. Verify all CCW components controls, interlocks and alarms function properly to actual or simulated signals.

ACCEPTANCE CRITERIA

Condenser Circulating Water system and components operate at design conditions in accordance with FSAR Section 10.4.5 and associated design drawings.

Table 14.2-1 (Sheet 71 of 89) AUXILIARY FEEDWATER SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the capability and reliability of the Auxiliary Feedwater System to supply feedwater to the steam generators and to maintain steam generator water inventory as required.

PREREQUISITES

1. Verify system is configured as necessary for unit/plant operation.
2. Steam supply is available for the turbine-driven Auxiliary Feedwater pump.
3. An acceptable water supply is available for the Condensate Storage Tank.
4. Interfacing systems needed to support testing of the Auxiliary Feedwater System are available.

TEST METHOD

1. Demonstrate proper manual and automatic operation of suction and discharge valves on the motor and turbine driven pumps and the steam supply to the turbine-driven pump.
2. Verify proper functioning of local and remote means of component control.
3. Verify that the hydraulic performance of each pump meets design requirements.
4. Demonstrate that the motor driven and turbine-driven pumps are capable of delivering rated flow to the steam generators within the acceptable time after an initiating signal.
5. Check for proper operation of the steam driven AF Pump instrumentation, control interlocks and alarms including governor valve and turbine speed control.
6. Verify the capability of the flow limiter to prevent pump runout.
7. Verify the operability of the Essential Raw Cooling Water System interface with the Auxiliary Feedwater Suction Header (essential raw cooling water will not be introduced into the Auxiliary Feedwater System).
8. Feedwater bypass line check valves operate properly.
9. Start the steam-driven auxiliary feedwater pump five consecutive times from a cold condition.

Table 14.2-1 (Sheet 72 of 89) AUXILIARY FEEDWATER SYSTEM TEST SUMMARY

ACCEPTANCE CRITERIA

1. The hydraulic performance of the auxiliary feedwater pumps meets or exceeds design requirements as described in FSAR Section 10.4.9.

2. Automatic and manual (local and remote) controls, interlocks, and alarms operate in accordance with design drawings.

3. Pump flow does not exceed run-out conditions.

4. The auxiliary feedwater pumps deliver rated flow within the time required by design as described in FSAR Section 10.4.9.

5. The steam driven auxiliary feedwater pump successfully starts and achieves rated speed within the time required five consecutive times from a cold condition.

Table 14.2-1 (Sheet 73 of 89) THERMAL EXPANSION TEST SUMMARY

OBJECTIVE

To demonstrate that Reactor Coolant System components and specified safety-related systems designated as ASME Class 1, 2, and 3, with operating temperatures greater than 200EF, experience thermal expansion consistent with design. Additionally, to verify that specified system support components do not interfere with the pipe thermal growth.

PREREQUISITES

1. The specific piping and support components selected for thermal expansion have been reviewed to ensure that all hangers which allow or restrict thermal growth have been installed and adjusted.
2. Preliminary shimming has been completed and shim crevice and support surfaces have been inspected to ensure debris has been removed.
3. The required heat load for thermal expansion testing is available.
4. Pipe whip restraints have been properly set to allow specified piping to move without obstruction or binding.
5. Stroke testing of all snubbers on specific systems is completed.

TEST METHOD

1. While monitoring at predetermined points in discrete temperature step increments, verify that the selected system can expand without obstruction or interference during system heatup from ambient to no-load operating conditions.
2. Inspect snubbers and spring cans at specified temperature intervals to ensure their thermal movements are within the criteria per applicable design drawings.
3. Verify that the selected system piping and components are capable of returning to their cold position within specified tolerance limits.

ACCEPTANCE CRITERIA

The piping systems defined by Section 3.9 are free to expand thermally without restriction other than by design. Spring hanger movements remain within their working range and snubbers have not become fully extended or retracted. The measured thermal movement data shall be within the design analytical predicted value, or reconciled by the Engineering Department.

Table 14.2-1 (Sheet 74 of 89) FUEL HANDLING EQUIPMENT TEST SUMMARY

OBJECTIVE

To demonstrate the operability of the fuel handling equipment, including the handling tools and equipment, cranes and fuel transfer system.

To provide for final indexing of the manipulator crane and to establish reference marks for the manipulator crane bridge using a verified dimensionally correct dummy assembly.

To provide the opportunity for training fuel handlers prior to actual fuel loading.

PREREQUISITES

1. The refueling cavity, refueling canal and spent fuel pool are clean and areas adjacent to the system equipment are clear.

2. Dummy assembly and test fixtures are available as required for testing the fuel handling equipment.

3. Load testing of the reactor head and internals lifting fixtures has been completed.

TEST METHOD

1. With the use of a dummy assembly demonstrate the proper operation of all system components, including the manipulator crane, fuel transfer system, rod cluster control changing fixture, various handling tools, and indexing of the system.

2. For the equipment listed in Test Method 1, verify operation of interlocks and proper setting of limit switches.

3. For the equipment listed in Test Method 1, demonstrate proper operation of crane and hoist controls including overspeed, overloads and travel limits, and warning devices.

4. For the equipment listed in Test Method 1, demonstrate hoist, bridge and trolley travel is acceptable.

5. For the equipment listed in Test Method 1, verify the operation of the hoist braking systems.

6. Perform a Manipulator Crane static load test using a test fixture and an operational load test with a dummy assembly.

Table 14.2-1 (Sheet 75 of 89) FUEL HANDLING EQUIPMENT TEST SUMMARY

ACCEPTANCE CRITERIA

1. Fuel transfer system operation, controls and interlocks function in accordance with FSAR Section 9.1.4 and the vendor technical manual and drawings.

2. All Manipulator Crane operation, controls and interlocks function in accordance with FSAR Section 9.1.4 and the vendor technical manual and drawings.

3. The RCC change fixture's function to remove, store, and load RCC's has been successfully demonstrated in accordance with FSAR Section 9.1.4.

4. Final Manipulator Crane indexing is completed and the Manipulator Crane has proven repeatability at various core locations.

5. Manipulator Crane has been successfully load tested.

Table 14.2-1 (Sheet 76 of 89) REACTOR COOLANT SYSTEM COLD HYDROSTATIC TEST TEST SUMMARY

OBJECTIVE

To verify the integrity and leak-tightness of the Reactor Coolant System (RCS), and high pressure portion of systems connected to the RCS, by performing a hydrostatic test of the system in accordance with Section III of the ASME Boiler and Pressure Vessel Code.

PREREQUISITES

1. A positive displacement pump is available.
2. A water supply within acceptable chemistry and temperature limits is available for pressurizing the reactor coolant system.
3. The reactor coolant pumps are operational.
4. The reactor lower internals, upper internals, and vessel head are installed and the head studs are tensioned for hydrostatic test pressure.
5. Temporary temperature and pressure test instrumentation is calibrated and installed or available.

TEST METHOD

1. The reactor coolant pumps will be operated as required to properly vent the RCS and establish the required temperature prior to pressurizing the RCS to test pressure.
2. Pressurize the RCS within the maximum rate and in the prescribed increments until the desired test pressure is obtained, and stabilize at the test pressure for the required time.
3. Perform inspection of reactor coolant pumps at test pressure.
4. Reduce system pressure to design pressure.
5. Perform inspections of welds, joints, piping, and components within the test boundary at design pressure.

ACCEPTANCE CRITERIA

1. No leakage from welds within the test boundary is observed.
2. Test pressure is maintained for the time required by Section III of the ASME Boiler and Pressure Vessel Code.
3. Examination for leakage is performed at required pressure.

Table 14.2-1 (Sheet 77 of 89) INTEGRATED HOT FUNCTIONAL TESTS TEST SUMMARY

OBJECTIVE

To establish initial plant conditions required for performance of preoperational and acceptance tests of primary, secondary, and auxiliary systems which require normal operating temperature and pressure of the reactor coolant system. The HFT procedure will control plant conditions as required to properly sequence these tests during heatup, while at normal operating temperature and pressure, and during cooldown.

To perform operational checks of the reactor coolant system components including pumps, motors, valves, instruments used during plant heatup from cold shutdown conditions to normal operating temperature and pressure prior to core loading. This heatup/cooldown also affords the opportunity to demonstrate the adequacy of the plant operating procedures.

To collect any foreign material, having a cross section greater than 1/16 inch, which may be present in the reactor coolant system.

To operate the reactor coolant system at full flow conditions for a minimum of 240 cumulative hours to achieve one million vibration cycles of the reactor internals.

PREREQUISITES

1. The RCS Cold Hydrostatic Test has been completed.
2. All systems, or portions of systems, and components whose adequacy or proper operations are to be verified under hot plant conditions, are operational.
3. The Reactor vessel internals vibration baseline inspection has been completed and the internals are installed.
4. Full flow filters have been installed in the reactor vessel.
5. The reactor coolant system has been filled and vented.
6. Prior to demonstrating remote shutdown capability, establish the prerequisites required by Regulatory Position C.2 and minimum crew size required by Regulatory Position C.3 of Regulatory Guide 1.68.2, Revision 1.

Table 14.2-1 (Sheet 78 of 89) INTEGRATED HOT FUNCTIONAL TESTS TEST SUMMARY

TEST METHOD

1. Heat the RCS to normal operating temperature and pressure using heat from the reactor coolant pumps and pressurizer heaters. Verify proper operation of the reactor coolant pumps, motors, and seals.

2. Verify the thermal expansion of system components and piping.

3. Perform isothermal cross-calibration of reactor Coolant Loop Resistance Temperature Detectors and Thermocouples.

4. Verify capability of the Chemical and Volume Control System to provide charging water at rated flow against normal reactor coolant pressure, check letdown design flow rate for each applicable operating mode, and check response of the system to changes in pressurizer level.

5. Verify proper operation of steam generator instrumentation to changes in steam generator level.

6. Demonstrate proper functioning of the main steam isolation and bypass valves under normal operating pressures and temperature conditions.

7. Operate the RC pumps for a minimum of 240 hours at full flow in order to achieve greater than one million cycles on the vessel internals. Following hot functional testing, the internals are to be removed and inspected for vibration effects.

8. Perform periodic vibrations measurements on RCS components as required.

9. Verify acceptability of the excess letdown and seal water flows.

10. Demonstrate proper operation of the pressurizer pressure control system.

11. Demonstrate proper operation of the steam dump control system.

12. Demonstrate proper operation of the RV head vent system.

13. Demonstrate proper operation of the motor and steam driven auxiliary feedwater pumps.

14. Demonstrate the ability to cooldown the plant in a controlled manner from the Main Control Room and from outside the Main Control Room (Auxiliary Control Room). During cooldown from outside the control room, the RCS pressure and temperature will be reduced to allow operation of the RHR system and subsequent cooldown of at least 50 degrees using RHR in accordance with Regulatory Position C.4 of Regulatory Guide 1.68.2, Revision 1.

Table 14.2-1 (Sheet 79 of 89) INTEGRATED HOT FUNCTIONAL TESTS TEST SUMMARY

ACCEPTANCE CRITERIA

1. The Reactor Coolant System has been operated at full flow conditions for a minimum of 240 cumulative hours.
2. Tests requiring the RCS to be at normal operating pressure and temperature have been completed.
3. Automatic controls, alarms, and interlocks operate in accordance with design drawings.
4. Reactor coolant pumps operate in accordance with vendor documents.

Table 14.2-1 (Sheet 80 of 89) OPERATIONAL VIBRATION TESTS TEST SUMMARY

OBJECTIVE

To verify acceptable vibration levels exist for (1) Class 1, 2, and 3 piping, (2) other high-energy piping systems inside Seismic Category I Structures, (3) high-energy portions of systems, whose failure could reduce the functioning of any Seismic Category I Plant Feature to an unacceptable level.

PREREQUISITES

1. Systems are operational as required.
2. Instrumentation is in place for testing as required.

TEST METHOD

1. Subject the specified piping systems to various flow modes and transients such as pump trips and valve closures as required.
2. Visually inspect and/or measure the vibration level of the piping and components at the specified locations.
3. Following completion of the system transient test, visually inspect the piping and supports, including snubbers for damage, looseness of parts, etc.

ACCEPTANCE CRITERIA

The vibration level for piping and components is within acceptable limits in accordance with the requirements of FSAR Section 3.9.2.1.

Table 14.2-1 (Sheet 81 of 89) CONTAINMENT LOCAL LEAK RATE TESTS-TEST SUMMARY

OBJECTIVE

To determine the leakage rate across each pressure-containing or leakage-limiting boundary for primary containment penetrations as applicable, including containment isolation valves.

PREREQUISITES

1. Equipment is available and calibrated to provide measurement of the local leak rate at penetrations, containment isolation valves, and air locks.

TEST METHOD

1. Examine the individual containment penetrations and containment isolation valves for leakage in accordance with Option B of Appendix J to 10 CFR Part 50.

2. Measure the leakage rate across each containment penetration pressure-containing or leakage-limiting boundary, as applicable, in accordance with Option B of Appendix J to 10 CFR Part 50.

3. Determine path leakage rates for valves in secondary containment bypass leakage paths.

ACCEPTANCE CRITERIA

The Containment Local Leak Rate Tests meet the requirement of 10 CFR Part 50, Appendix J, Option B.

Table 14.2-1 (Sheet 82 of 89) CONTAINMENT INTEGRATED LEAK RATE TEST SUMMARY

OBJECTIVE

To verify the primary reactor containment overall integrated leakage rate is within acceptable limits.

PREREQUISITES

1. Fluid system conditions are established as applicable to simulate post accident conditions which extend the boundary of the Containment Building.
2. Containment pressure retaining boundary, leakage limiting boundary, and isolation valve leak tests have been satisfactorily performed.
3. All containment isolation valves have been closed by normal actuation methods.
4. The vendor containment over pressurization (structural integrity) test has been successfully completed.

TEST METHOD

1. Perform the containment integrated leak rate test per Option B of Appendix J to 10 CFR Part 50.
2. Perform the leakage rate calculation by using the mass-point methodology as described by ANSI/ANS 56.8-1994.
3. If during the performance of a Type A test, excessive leakage occurs through locally testable penetrations or isolation valves, these leakage paths may be isolated and the Type A test continued until completion. The sum of the post repaired minimum pathway local leakage rate values will be added to the UCL.
4. Verify proper operation of containment narrow and wide range pressure instrumentation and alarms used in conjunction with post accident monitoring.

ACCEPTANCE CRITERIA

1. The Containment Integrated Leak Rate Test meets the requirements of Option B of Appendix J to 10 CFR Part 50.

Note: The containment structural integrity test described in FSAR Section 3.8 may be performed concurrently with the Integrated Leak Rate Test.

2. The containment pressure instrumentation operates in accordance with design, as described in FSAR Section 7.5.

Table 14.2-1 (Sheet 83 of 89) CONTAINMENT ISOLATION SYSTEM TEST SUMMARY

OBJECTIVE

Demonstrate the capability of various plant system components to properly respond to Phase A and Phase B containment isolation signals and to containment ventilation isolation signals.

PREREQUISITES

1. All systems with applicable automatic containment isolation valves are in a status allowing operation of these valves.

TEST METHOD

1. Verify manual operation of all containment isolation valves.
2. Verify all valves designed to isolate containment in response to Phase A and Phase B containment isolation and containment ventilation isolation signals properly actuate in response to these signals.
3. Demonstrate the operation and independence of redundant trains of containment isolation components.
4. Verify air operated valves fail to the accident position upon loss of air or electrical power.
5. Measure the time required for isolation valves to close.
6. After isolation valve closure, reset the isolation signal and verify the isolation valve remains in the accident position.
7. Verify proper operation of main control room alarms and indicators.

ACCEPTANCE CRITERIA

1. All containment isolation valves designed to isolate on receipt of a Phase A or Phase B containment isolation signal or containment ventilation isolation signal properly respond to receipt of the appropriate signal as described in FSAR Section 6.2.4.
2. The closure time of containment isolation valves is equal to or less than the time requirements specified in FSAR Table 6.2.4-1.
3. Containment isolation valves remain in the accident position after reset of the isolation signal.
4. Control room alarms and indicators operate in accordance with design.

Table 14.2-1 (Sheet 84 of 89) ANTICIPATED TRANSIENT WITHOUT SCRAM MITIGATION SYSTEM ACTUATION CIRCUITRY TEST SUMMARY

OBJECTIVE

Demonstrate the capability of the Anticipated Transients Without Scram Mitigation System Actuation Circuitry (AMSAC) to respond properly to initiation signals.

PREREQUISITES

1. Interfacing systems such as the Annunciator System are available.

2. Instrument loops providing input to AMSAC.

TEST METHOD

1. Simulating transmitter input, verify AMSAC functions properly with the appropriate turbine impulse pressure (simulated power), steam generator level, and block switch position conditions.

2. Demonstrate AMSAC setpoints and time response requirements.

3 Verify proper annunciation from AMSAC System.

ACCEPTANCE CRITERIA

1. When AMSAC is armed and Steam Generator level coincidence logic is obtained, a Main Turbine trip signal and start signal for all Auxiliary Feedwater Pumps is generated. AMSAC test/block switch can block an AMSAC signal.

2. With AMSAC armed at ≥40% simulated power, Steam Generator low-low level logic setpoint is 12% of narrow range level.

3. AMSAC logic and output relay actuation response time is #1.0 seconds. AMSAC actuation time delay including sensor, logic and output relay actuation response time is #30 seconds.

4. AMSAC status lights and annunciators respond as designed.

Table 14.2-1 (Sheet 85 of 89) COMPRESSED AIR SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the capability of the Compressed Air System to provide regulated air that is clean, dry and oil-free to instrumentation and control loads during normal plant operation and to vital equipment required for safe shutdown under design basis event conditions.

PREREQUISITES

1. AC and DC electrical power supplies are available.

2. The system has been blown-down and verified to be clean in accordance with approved cleanliness standards.

3. Station and Auxiliary Air Compressors are available.

TEST METHOD

1. Perform a sudden and gradual loss of Auxiliary Control Air System pressure, with the affected loads in their normal operating position, to verify response of supplied loads.

2. With the Auxiliary Control Air System operating in a steady state condition, operate at least two of the highest demand loads supplied from the same train, simultaneously.

3. Perform a sudden and gradual loss of Control Air System pressure, with the affected loads in their normal operating position, to verify response of supplied loads.

Table 14.2-1 (Sheet 86 of 89) COMPRESSED AIR SYSTEM TEST SUMMARY

ACCEPTANCE CRITERIA

1. Automatic controls including system isolation features, interlocks, and alarms operate in accordance with design drawings.
2. Auxiliary Control Air loads respond properly to a loss of system air pressure as described by design documents.
3. Simultaneous operation of the two highest demand loads supplied from each train will not cause unacceptable system pressure transients as described in applicable design documents.
4. Control Air System loads (safety related only) respond properly to a loss of System air pressure as described by design documents.

Table 14.2-1 (Sheet 87 of 89) ICE CONDENSER SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the operability of the ice condenser and its doors, and ensure that gross bypass leakage paths between the upper and lower containment areas are not present.

PREREQUISITES

1. Ice condenser lower inlet doors have been prepared for testing.
2. All modifications to the boundary between upper and lower containment areas are complete.
3. All system components have been installed.
4. The Raw Cooling Water System and Demineralized Water System are available.
5. System relief valve setpoints have been verified.
6. Heat tracing is installed on Air Handling Unit drains and is available for use.
7. Air handling units, glycol circulation and refrigeration equipment are operable.

TEST METHOD

1. Verify all doors and associated alarms function properly.
2. By a combination of physical measurements and/or testing, determine that gross bypass leakage paths are not present.
3. Demonstrate the ice condenser can be adequately cooled to and maintained at design conditions.
4. Verify the ice condenser has been loaded with the proper quantity and quality of ice.

ACCEPTANCE CRITERIA

1. The ice condenser, its doors, and associated alarms function in accordance with FSAR Section 6.7 and the appropriate design basis documents.
2. Gross bypass leakage paths between the upper and lower containment areas as described in FSAR Section 6.2.1 are not detected.
3. The Ice Condenser has been loaded with the proper quantity and quality of borated ice as required in FSAR Section 6.7.

Table 14.2-1 (Sheet 88 of 89) PRESSURIZER SAFETY AND RELIEF VALVES TEST SUMMARY

OBJECTIVE

To demonstrate the proper operation of the pressurizer power-operated relief valves and pressurizer relief tank.

PREREQUISITES

1. Plant conditions are established as necessary for test performance.
2. The pressurizer relief tank is filled to its normal level.
3. Primary makeup water supply is available as required.
4. The pressurizer safety valves have been bench tested to verify their setpoints and installed.

TEST METHOD

1. Verify proper actuation and operation of the power-operated relief valves and isolation valves.
2. Check operation of the discharge header leak detection devices.
3. Verify proper operation of the pressurizer relief tank control systems, instrumentation, interlocks, and alarms.
4. Confirm that cooling spray to the pressurizer relief tank meets design requirements.

ACCEPTANCE CRITERIA

1. The pressurizer power-operated relief valves and isolation valves operate in accordance with design as described in FSAR Section 5.2.2.
2. The pressurizer relief tank controls and alarms operate in accordance with design as described in FSAR Section 5.5.11 and applicable design documents.
3. Controls, instrumentation, interlocks and alarms operate properly in response to simulated or normal input signals as described in FSAR Section 5.2.2 and applicable design documents.

Table 14.2-1 (Sheet 89 of 89) Deleted by Amendment 88

Table 14.2-2 (Sheet 1 of 39) POWER ASCENSION TEST SUMMARIES INDEX

Table 14.2-2 (Sheet 2 of 39) POWER ASCENSION TEST SUMMARIES INDEX

Table 14.2-2 (Sheet 3 of 39) INITIAL FUEL LOADING TEST SUMMARY

OBJECTIVE

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

PREREQUISITES

1. Testing prior to initial fuel loading is completed sufficiently to demonstrate the operability of required systems and components as defined in Technical Specifications.

2. The nuclear instrumentation system source range channels are installed and calibrated and have been response checked using the source fuel assembly as the source prior to being unlatched.

3. At least one path for boron addition to the reactor coolant system is available and a uniform boron concentration in the reactor coolant system is maintained by circulation with at least one residual heat removal pump and is sufficient to assure K_{eff} less than or equal to 0.95 during fuel loading.

TEST METHOD

Fuel assemblies containing specified core component inserts are placed into the reactor vessel in accordance with the specified loading sequence. Neutron count rate is monitored on responding source range detectors. Core reactivity is monitored through the installed source range instrumentation. A minimum of two source range channels will be responding to primary source neutrons before additional assemblies are loaded. Then following the loading of the initial nucleus of eight assemblies near each of the two installed source range channels each channel will be adequately responding to subcritical multiplication of additional assemblies loaded. If a delay in fuel loading greater than 8 hours occurs the nuclear instrumentation will be evaluated to determine acceptability of the detectors for continuation of fuel loading.

ACCEPTANCE CRITERIA

1. Neutron instrumentation is operational and indicates a positive (negative) change in count rate as the neutron level detected from a source is increased (decreased).

2. The core is assembled in accordance with the configuration specified in the Westinghouse Nuclear Parameters and Operations Package (NuPOP) Watts Bar.

Table 14.2-2 (Sheet 4 of 39) REACTOR SYSTEM SAMPLING FOR CORE LOADING TEST SUMMARY

OBJECTIVE

To verify the boron concentration in the reactor coolant system and directly connected auxiliary systems is uniform in order to prevent inadvertent dilution during core loading.

PREREQUISITES

1. The reactor coolant system boron concentration is at a concentration sufficient to assure K_{eff} less than or equal to 0.95.
2. The auxiliary systems are available and operational as needed to perform the filling and circulation required to meet uniform boron concentrations.
3. The water level in the reactor vessel (reactor cavity) has been established.

TEST METHOD

Borated water is circulated and sampled in each system until the required boron concentration is obtained. Systems include the reactor vessel and coolant loops, the charging pumps and volume control tank, the cold leg accumulators, the RWST, RHR pumps and associated lines. Boric acid concentration is sampled in the boric acid tanks as well.

ACCEPTANCE CRITERIA

1. Boron concentration of samples meet requirements of the Technical Specifications.

Table 14.2-2 (Sheet 5 of 39) THERMAL EXPANSION OF PIPING SYSTEMS TEST SUMMARY

OBJECTIVE

To demonstrate that piping systems defined in Section 3.9.2.1, with operating temperatures greater than 200°F, experience thermal expansion consistent with design and verify that specified system support components do not interfere with the pipe thermal growth.

PREREQUISITES

1. System pipe supports are installed and adjusted.
2. Preservice examinations of all snubbers on specified systems are complete.
3. Required test equipment is available.

TEST METHOD

Selected systems, monitored at predetermined points, will be verified to expand and then return without obstruction or interference. This will be done for initial ambient conditions and at the 30%, 50%, 75%, 100% power test plateaus and at final ambient conditions. Specified snubbers and spring hangers are inspected to ensure their movements remain within their working range.

ACCEPTANCE CRITERIA

1. Piping systems are free to expand thermally without restriction other than by design.
2. Spring hanger movements remain within their working range and snubbers have not become fully extended or retracted.
3. The measured thermal movement data is within the design analytical predicted value, or reconciled by Site Engineering.

Table 14.2-2 (Sheet 6 of 39) PIPING VIBRATION MONITORING TEST SUMMARY

OBJECTIVE

To verify that the vibration level of the piping systems defined in Section 3.9.2.1 is acceptable under steady state and operational transient conditions.

PREREQUISITES

1. System pipe supports are installed and adjusted.
2. Preservice examinations of all snubbers on specified systems are complete.
3. Required test equipment is available.

TEST METHOD

Selected locations at various flow modes and transients shall be observed to ensure that severe vibrations do not exist.

ACCEPTANCE CRITERIA

1. The vibration level for piping and components are within acceptable limits in accordance with Section 3.9.2.1.

Table 14.2-2 (Sheet 7 of 39) CONTROL ROD DRIVE MECHANISM TIMING TEST SUMMARY

OBJECTIVE

To verify: (a) that each rod control system slave cycler provides its associated power cabinet with the appropriate command signals to obtain proper sequence timing of current supplied to the CRDM coils, (b) the CRDM coil current amplitudes are within acceptable ranges, (c) the operability of each shutdown and control rod drive mechanism, and (d) the stepping rate for shutdown and control rods.

PREREQUISITES

1. The reactor vessel upper internals and head are installed and each rod cluster control assembly is latched to its control rod drive shaft.

2. The NIS source range channels are operable and the CRDM cooling fans are available to provide ventilation.

3. The RCS is filled and vented or is at hot standby conditions at nominal operating temperature and pressure.

4. The RCS boron concentration is equal to or greater than the refueling boron concentration.

TEST METHOD

This test will monitor the time-variant response of the lift, stationary gripper, and moveable gripper coil currents during rod withdrawal and insertion operations. The data will be used to verify proper: (a) slave cycler timing, (b) CRDM coil current amplitudes, (c) CRDM operation, and (d) MANUAL mode rod stepping rates with the reactor in a cold shutdown condition. The test will be repeated with the reactor at normal operating temperature and pressure prior to initial criticality.

ACCEPTANCE CRITERIA

1. Each slave cycler provides its associated power cabinet with the appropriate command signals to obtain proper sequence timing of current supplied to the associated CRDM coils during rod withdrawal and insertion operations.

2. The current amplitudes for the CRDM coils fall within acceptable ranges during withdrawal and insertion operations, as recommended by the NSSS vendor.

3. The rod stepping rate is approximately 48 steps/minute for control rods and 64 steps/minute for shutdown rods.

4. The operability of each shutdown and control rod drive mechanism is demonstrated by the ability to withdraw and insert rods.

Table 14.2-2 (Sheet 8 of 39) ROD POSITION INDICATION SYSTEM TEST SUMMARY

OBJECTIVE

To verify that: (a) the rod position indication system performs the required indication function satisfactorily for each shutdown and control rod over their entire range of travel, and (b) rod position indication system alarm functions operate properly.

PREREQUISITES

1. The NIS source range channels are operable and the CRDM cooling fans are available to provide ventilation.
2. Control rod drive mechanism timing test has been completed.
3. The RCS boron concentration is equal to or greater than the refueling boron concentration.
4. The reactor is in hot standby conditions at nominal operating temperature and pressure.

TEST METHOD

Shutdown and control banks will be individually withdrawn in increments of 20 steps until they are fully withdrawn. At each 20 step increment, control room RPI indicator, and group step counter readings will be recorded. The banks will then be inserted to rod bottom. The dropped rod function will be tested by confirming rod bottom bistable setpoint during rod insertion. In addition, the bank position digital readout will also be recorded for the control banks.

Proper operation of the following annunciator alarms will also be confirmed during this test:

a) Urgent and Non-urgent failure

b) Bank D Withdrawal Limit

c) RPI System In Trouble

d) ICS computer generated rod deviation alarms.

ACCEPTANCE CRITERIA

1. The rod position indication system performs the required indication and alarm functions at hot standby conditions.
2. Each RPI agrees within 12 steps of the group demand position for the full range of rod travel.

Table 14.2-2 (Sheet 9 of 39) ROD DROP TIME MEASUREMENT AND STATIONARY GRIPPER RELEASE TIMING TEST SUMMARY

OBJECTIVE

To measure the rod drop and stationary gripper release time for each control and shutdown rod at hot standby conditions with full RCS flow.

PREREQUISITES

1. The reactor is in hot standby conditions at nominal operating temperature and pressure.
2. The NIS source range channels are operable and the CRDM cooling fans are available to provide ventilation.
3. All reactor coolant pumps are in operation.
4. The RCS boron concentration is equal to or greater than the refueling boron concentration.

TEST METHOD

The shutdown and control rods will be partially withdrawn and the reactor trip breakers will be opened to demonstrate that all CRDMs unlatch and all rods fully insert into the core. The drop time for each shutdown and control rod will be measured by monitoring the time-variant voltage response of the associated rod position indication detector secondary coil and stationary gripper coil. The rods will be dropped from the fully withdrawn position either by de-energizing the stationary gripper coils individually, or by opening the reactor trip breakers thus dropping all or a portion of the rods together. The rod drop and stationary gripper release times will be determined using a data collection system. Testing will be performed under hot full flow conditions. Rods whose drop times are greater than two standard deviations from the average drop time for all rods will be retested at least 3 times to ensure that the rod drop times are consistent. The stationary gripper release times will be measured to verify that the requirements of Chapter 4 are satisfied.

NOTE: Additional rod drops may be performed at either hot or cold RCS conditions with any RCS flow condition. These drops are not used to satisfy acceptance criteria and their times are not included in calculating averages.

ACCEPTANCE CRITERIA

1. The rod drop times for all shutdown and control rods, dropped from the fully withdrawn position, are within the limits specified in the Technical Specifications.
2. The stationary gripper release time for all rods satisfy the requirements of Chapter 4.
3. Each CRDM unlatches upon opening the Reactor Trip Breakers.

Table 14.2-2 (Sheet 10 of 39) ROD CONTROL SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate that the rod control system satisfactorily performs the required control and indication functions.

PREREQUISITES

1. CRDM timing and rod position indication system testing have been completed.
2. The reactor is in hot standby conditions at nominal operating temperature and pressure.
3. The RCS boron concentration is equal to or greater than the refueling boron concentration.

TEST METHOD

The test will be performed in hot standby and ensures that control room indicators respond properly and the control bank overlap function performs adequately. Rod bank starting and stopping positions will be compared with the control settings for verification. Each bank of shutdown rods will be operated individually in the withdraw and insert directions using the normal controls. The control banks will be operated in manual to verify the overlap function with minimum overlap settings. Sufficient travel will demonstrate drive operability, position indication, and other instrumentation.

ACCEPTANCE CRITERIA

1. The control bank overlap circuitry functions properly during sequential withdrawal and insertion of control rods in MANUAL mode.
2. The RPI indicators and group step counters function properly.
3. Rod direction indicator lights function properly to indicate the rod movement status and direction of rod movement during rod withdrawal and insertion operations.

Table 14.2-2 (Sheet 11 of 39) Deleted by Amendment 97

Table 14.2-2 (Sheet 12 of 39) INCORE INSTRUMENTATION SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate operability of the Incore Instrumentation system.

PREREQUISITES

Containment pressurization tests are complete and Incore Instrumentation System is capable of having simulated signals injected at the seal table in the order to generate a simulated power distribution measurement.

TEST METHOD

The electrical continuity of the signal paths for the Incore Instrumentation System Fixed Incore Detectors are verified and the ability to generate a power distribution measurement is simulated by injecting signals into the Incore Instrumentation System detector cables at the seal table. After connecting the cables to the Incore Instrument Thimble Assemblies (IITAs), and achieving at least 25% RTP, a power distribution measurement is produced using reactor core neutron power.

ACCEPTANCE CRITERIA

1. Electrical continuity is demonstrated between the cable connector on the seal table and Power Distribution Monitoring System (PDMS) workstation by simulating an IITA neutron signal.
2. After reaching at least 25% RTP, a core power distribution measurement is obtained by the PDMS and compared to the core design prediction of core flux.

Table 14.2-2 (Sheet 13 of 39) PRESSURIZER SPRAY CAPABILITY AND CONTINUOUS SPRAY FLOW SETTING TEST SUMMARY

OBJECTIVE

To verify the effectiveness of the pressurizer spray and to determine the throttle positions for the manual spray bypass valves.

PREREQUISITES

1. The reactor is in Mode 3 at normal operating temperature and pressure.

2. Pressurizer pressure and level instrumentation and associated control systems are calibrated.

TEST METHOD

Pressurizer spray capability or effectiveness consists of a transient initiated by full spray to reduce pressurizer pressure approximately 250 psi. Data is recorded during this pressure transient. The change in pressurizer pressure is plotted versus time, and compared to nominal NSSS performance curves.

The manual spray bypass valves are adjusted to an optimum position. The optimum position for the manual spray bypass valves is achieved when during steady-state the pressurizer backup heaters are not energized, and the spray line temperature is above the setpoint for the spray line low temperature alarm.

In addition a functional check of the spray line low temperature alarm will be performed during the adjustment of the manual spray bypass valve.

ACCEPTANCE CRITERIA

Pressurizer pressure response to the opening of both normal spray valves is within the allowable range specified by the NSSS performance curves.

Table 14.2-2 (Sheet 14 of 39) RCS FLOW MEASUREMENT TEST SUMMARY

OBJECTIVE

To determine RCS flow.

PREREQUISITES

1. The elbow tap flow instrumentation has been calibrated.
2. The reactor is stable at the test plateau.

TEST METHOD

At zero power conditions (Mode 3), RCS flow is calculated from the elbow tap differential pressure instrumentation. At the ascending power test plateaus of 50%, 75% and 100%, RCS flow is determined from a steam generator heat balance using a secondary calorimetric and RCS loop temperature data.

ACCEPTANCE CRITERIA

RCS flow will be greater than or equal to the Technical Specification limit.

NOTE:

Below 90% power, RCS flow results will be reviewed and evaluated.

Table 14.2-2 (Sheet 15 of 39) REACTOR COOLANT FLOW COASTDOWN TEST SUMMARY

OBJECTIVE

To measure the rate at which reactor coolant flow changes subsequent to a simultaneous trip of all four reactor coolant pumps, to measure the delay times associated with assumptions of the loss of flow accident and measure the decay of the RCP voltage.

PREREQUISITES

1. RCS flow measurement testing has been completed in Mode 3.
2. The reactor is at hot standby conditions with all rods inserted.
3. All RCPs are running and any pressure damping devices installed in the elbow tap differential pressure cell sensing lines for RCS flow measurement testing have been removed.

TEST METHOD

Measurements are made by tripping all reactor coolant pumps simultaneously and recording reactor coolant loop d/p, RCS low flow bistable position, reactor coolant pump breaker and reactor pump motor voltage decay data.

ACCEPTANCE CRITERIA

1. All reactor coolant pumps have been tripped within 100 msec of each other.
2. The reactor coolant flow coastdown is within the design of the coastdown flow values in Chapter 15.
3. The delay times associated with the low reactor coolant flow reactor trip are within the values assumed in Chapter 15.

Table 14.2-2 (Sheet 16 of 39) OPERATIONAL ALIGNMENT OF PROCESS TEMPERATURE INSTRUMENTATION TEST SUMMARY

OBJECTIVE

To align the RCS ΔT and T_{avg} process instrumentation during power ascension.

PREREQUISITES

1. The reactor is stable at the test plateau.

2. The final calibration constants related to the primary loop RTDs (obtained from the RTD cross calibration test) have been entered into the process protection system for each RTD.

TEST METHOD

Prior to initial criticality with the RCS at isothermal conditions, temperature data is collected and the ΔT process instrumentation alignment is verified. At the ascending power test plateaus of 30%, 50%, and 75%, temperature, pressure, and calorimetric data is collected. This data is used to determine RCS hot and cold leg enthalpies. A curve fit, using the enthalpies and associated calorimetric power, is performed to predict the enthalpies for full power. This full power information is used to predict RCS temperatures ($T_{hot,}$ $T_{cold,}$ $T_{avg,}$ and ΔT) at full power. Temperature instrumentation is aligned per these predicted values at 75% power. Then at the 90% and 100% power test plateaus, alignment checks of the ΔT and T_{avg} instrumentation are performed. If required, new values are calculated. With satisfactory alignment results at the 100% power test plateau, the reference T_{avg} parameter values may be used to rescale the ΔT trip setpoints.

ACCEPTANCE CRITERIA

1. Prior to criticality with the RCS at isothermal conditions, the process instrumentation values of ΔT indicate 0% and are within the tolerance specified by the NSSS vendor.

2. At 100% power, the process instrumentation value of ΔT for each channel agrees with reactor power as determined by a secondary calorimetric and is within the tolerance specified by the NSSS vendor.

Table 14.2-2 (Sheet 17 of 39) OPERATIONAL ALIGNMENT OF NUCLEAR INSTRUMENTATION TEST SUMMARY

OBJECTIVE

To make adjustments to the source range, the intermediate range and power range channels and determine overlap between source/intermediate range and intermediate/power range channels.

PREREQUISITES

1. The nuclear instrumentation system source range, intermediate range and power range channels are installed and operable.
2. The reactor is stable at the test plateau.
3. Initial calibration has been implemented in accordance with Technical Specifications.

TEST METHOD

Initial settings for the source range channels are determined prior to core loading. Initial trip setpoints for the intermediate and power range channels are determined prior to power ascension and then adjusted to support power ascension. The overlap between source/intermediate range and intermediate/power range channels is determined during power ascension. Reactor power determined by secondary calorimetric is used to recalibrate (as needed) the intermediate and power range channels at each major test plateau. The power range channels are verified to be linear to power.

ACCEPTANCE CRITERIA

1. All channels of source range, intermediate range and power range are operable and calibrated per Technical Specifications and Section 14.2.10.
2. Overlap between source range/intermediate range and between intermediate range/power range channels has been verified.

Table 14.2-2 (Sheet 18 of 39) RADIATION BASELINE SURVEY TEST SUMMARY

OBJECTIVE

To determine the effectiveness of the shielding by measuring radiation dose rates at preselected locations throughout the plant.

PREREQUISITES

1. Radiation work permits are issued.
2. Personnel are trained, qualified, and briefed on the survey requirements.
3. Radiation survey instruments are calibrated and source checked.

TEST METHOD

Baseline gamma and neutron dose rates are monitored at preselected locations throughout the plant at ambient conditions after fuel load. At various power levels (< 10%, 40%-60%, 90%-100%) during the power ascension testing program, gamma and neutron dose rates are monitored at preselected locations throughout the plant.

ACCEPTANCE CRITERIA

1. Radiation levels are below the upper limit for the zone designation of each area surveyed.

Table 14.2-2 (Sheet 19 of 39) REACTOR TRIP SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the proper functioning of the reactor trip system.

PREREQUISITES

1. Fuel loading is complete.
2. Electrical power is available for the reactor trip system circuitry.

TEST METHOD

With all control rods fully inserted, attempt to close both reactor trip bypass breakers and verify interlocks will momentarily permit both bypass breakers to close and then cause a reactor trip. With one reactor trip bypass breaker closed, verify that placing the opposite trip channel in test causes both reactor trip breakers and the bypass breaker to open. Confirm the ability to test a reactor trip breaker without tripping the unit when its associated bypass breaker is racked in and closed.

ACCEPTANCE CRITERIA

1. The reactor trip breakers can be opened manually (electro-mechanically) from the Main Control Room.
2. Interlocks permit momentary closure of both reactor trip bypass breakers and then cause a reactor trip.
3. The reactor trip bypass breakers maintain the rod drive mechanism energized when the associated reactor trip breaker is opened for test.
4. With one reactor trip bypass breaker closed, placing the opposite trip channel in test causes both reactor trip breakers and the bypass breaker to open.

Table 14.2-2 (Sheet 20 of 39) STARTUP ADJUSTMENTS OF REACTOR CONTROLS TEST SUMMARY

OBJECTIVE

To determine the T_{avg} program resulting in the highest possible steam pressure and thus the optimum unit efficiency without exceeding pressure limitations for the turbine or the maximum allowable T_{avg}.

PREREQUISITES

1. The reactor is at hot standby conditions prior to initial criticality.

2. Recalibration of individual RTD resistance and temperature corrections from the performance of RTD-T/C Cross Calibration has been performed, if necessary.

3. The reactor/turbine control systems have been aligned to design values for turbine pressure and design value for T_{avg}.

TEST METHOD

The test will obtain primary system temperatures, steam pressures and thermal power data at steady state conditions. Evaluation of this data will provide the basis for the necessary instrument adjustments. Temperature and pressure data is recorded in Mode 3 at nominal HZP temperature and pressure conditions, then again during 30% and 50% power test plateaus. Plots of RCS temperature versus percent power, steam generator pressure versus percent power, and turbine impulse pressure versus percent power are then prepared. Additional data is taken at the 75% power test plateau and then extrapolated to 100% power to determine if adjustments to the T_{ref} program and/or calibration of the turbine impulse pressure may be required at 100% power. This process is repeated again at the 100% power test plateau to establish the optimum T_{avg} program for the reactor control system.

ACCEPTANCE CRITERIA

1. With the Rod Control System in automatic mode, the actual full load steam generator pressure is within a prescribed acceptance band, as specified by the NSSS vendor, of the design full load steam generator pressure.

2. Full load T_{avg} shall not exceed the design T_{avg}, as specified by the NSSS vendor.

Table 14.2-2 (Sheet 21 of 39) CALIBRATION OF STEAM AND FEEDWATER FLOW INSTRUMENTATION AT POWER TEST SUMMARY

OBJECTIVE

To determine the calibration spans for each steam flow transmitter, and verify the calibration of feedwater flow and steam flow instrumentation by comparing indicated flows with calculated flows.

PREREQUISITES

1. The reactor is stable at the specified test plateau.

2. Feedwater flow and steam flow instruments have been calibrated and scaled in accordance with design parameters.

TEST METHOD

The test obtains data in Mode 3 at normal operating pressure and temperature, and at ascending power levels during the 30%, 50%, 75% and 100% power test plateaus. Secondary side parameters of pressure, temperature and flows will be collected. Feedwater flow to each steam generator is calculated using the collected data. This calculated feedwater flow is used as the basis for comparison with the permanent instrumentation. A curve fit using calculated feedwater flow and the measured differential pressures from each steam flow transmitter are used to determine the differential pressure spans for each steam flow transmitter.

ACCEPTANCE CRITERIA

At the 100% power plateau, the span for each steam flow transmitter has been normalized to feedwater flow such that there are no steam flow/feedwater flow mismatch alarms and steam generator level control can remain in automatic.

Table 14.2-2 (Sheet 22 of 39) INITIAL CRITICALITY TEST SUMMARY

OBJECTIVE

To achieve initial criticality in a cautious and controlled manner.

PREREQUISITES

1. Boron concentration in the RCS is conservative enough to preclude inadvertent criticality if the shutdown banks are at the all rods out position.

2. Source, intermediate and power range channels are calibrated and in operation. Power Range High Flux Trip setpoints are set to less than or equal to 25% power.

3. A minimum count rate of 0.5 cps and a signal to noise ratio of greater than 2 is verified on the source range channels.

4. The RCS is at hot no load pressure and temperature.

TEST METHOD

After establishing baseline count rates, the shutdown (if not previously pulled) and control banks are withdrawn in normal sequence until an All Rods Out (ARO) configuration is reached. RCS boron dilution is commenced at a rate of $\leq$ 1000 pcm per hour with the RCS boron concentration being sampled at approximately 30 minute intervals. When the Inverse Count Rate Ratio (ICRR) is approximately 0.3, dilution rate is diminished to approximately one third of the original rate. Dilution is continued until the reactor is critical or the maximum projected dilution volume has been added. Then dilution is terminated and the RCS is allowed to mix. If the reactor is slightly critical, then control bank D will be inserted to control and maintain the reactivity status of the reactor. If criticality is not achieved during mixing, RCS sampling will be used to determine the reactivity state point of the reactor and any additional dilution volume necessary to reach the projected dilution volume. Dilution will re-commence until the reactor is slightly critical or the dilution volume has been added, whichever happens first. During dilution, evaluations of ICRR versus dilution water additions are maintained.

ACCEPTANCE CRITERIA

The reactor achieves initial criticality in a safe and orderly manner.

Table 14.2-2 (Sheet 23 of 39) DETERMINATION OF CORE POWER RANGE FOR PHYSICS TESTING TEST SUMMARY

OBJECTIVE

To determine the neutron flux level at which detectable reactivity feedback effects from nuclear heating occur and establish the range of flux levels in which zero power physics testing of reactivity measurements are performed.

PREREQUISITES

1. The reactor is critical with the neutron flux level established for recording critical data.
2. The reactivity computer is calibrated, operational and available for use.
3. Temperature, neutron flux, boron concentration and reactivity are stable.

TEST METHOD

Control bank D is withdrawn to achieve a positive reactivity increase. Flux level and plant conditions are monitored and noted when nuclear heating is observed. Typical indications of nuclear heating are a decrease in reactivity and an increase in RCS temperature. Control bank D is then inserted to achieve negative reactivity. At the point of adding nuclear heat, the zero power physics testing range is then determined and set by the reactivity computer as needed based on the appropriate testing to be performed. Control bank D is then adjusted to reduce nuclear flux to within the necessary testing range.

ACCEPTANCE CRITERIA

The zero power physics testing range has been determined such that reactivity feedback from nuclear heating does not compromise the measurements.

Table 14.2-2 (Sheet 24 of 39) MODERATOR TEMPERATURE COEFFICIENT TEST SUMMARY

OBJECTIVE

To measure the isothermal temperature coefficient of reactivity and from this data, determine and confirm that the moderator temperature coefficient of reactivity is within acceptable limits.

PREREQUISITES

1. The reactor is critical with the neutron flux level within the range established for zero power testing.
2. The reactivity computer is calibrated, operational and available for use.
3. Temperature, neutron flux, boron concentration and reactivity are stable.

TEST METHOD

The reactor coolant system is heated or cooled at a constant rate. Reactivity is monitored and reactivity parameters versus temperature changes are determined. The isothermal temperature coefficient value is determined. From this data, the value of the moderator temperature coefficient is then determined.

ACCEPTANCE CRITERIA

Moderator temperature coefficient of reactivity is determined to be within limits specified in the Technical Specifications.

Table 14.2-2 (Sheet 25 of 39) ROD WORTH AND BORON MEASUREMENTS TEST SUMMARY

OBJECTIVE

To verify the design integral rod worths of the control and shutdown banks and the design All Rods Out (ARO) critical boron concentration.

PREREQUISITES

1. The reactor is critical with the neutron flux level within the range established for zero power testing.

2. The reactivity computer is calibrated, operational and available for use.

3. Temperature, neutron flux, boron concentration and reactivity are stable.

TEST METHOD

The integral worth of all control and shutdown banks will be measured with the reactivity computer, using the Dynamic Rod Worth Measurement (DRWM) technique. The ARO critical boron endpoint will be determined from data obtained during rod worth measurements.

ACCEPTANCE CRITERIA

1. The measured integral rod worths of the control and shutdown banks at hot zero power are within the limits specified by the NSSS vendor.

2. The all rods out critical boron concentration at hot zero power is within the limits specified by the NSSS vendor.

Table 14.2-2 (Sheet 26 of 39) CORE REACTIVITY BALANCE TEST SUMMARY

OBJECTIVE

To verify that core reactivity effects are in agreement with design values.

PREREQUISITES

1. The reactor is critical at hot zero power with xenon free conditions, OR
2. The reactor is critical at approximately 100% power with equilibrium xenon conditions.

TEST METHOD

The core reactivity is determined during low power PHYSICS Testing by determining the ARO boron concentration at hot zero power, xenon free critical conditions. This measurement also confirms the reactivity measurement requirements of Technical Specifications are met. At approximately 100% power, a core reactivity balance calculation is performed using the boron concentration for all rods out, equilibrium xenon, hot full power. The calculated ARO critical boron concentration will be compared to design values in the Westinghouse Nuclear Parameters and Operations Package (NuPOP) Watts Bar

ACCEPTANCE CRITERIA

1. The all rods out, hot zero power, xenon free critical boron concentration is within the limits specified by the NSSS vendor.
2. At approximately 100% power, the core reactivity balance is within the limits specified by the NSSS vendor.

Table 14.2-2 (Sheet 27 of 39) FLUX DISTRIBUTION MEASUREMENT TEST SUMMARY

OBJECTIVE

To determine the reactor core power flux distribution for comparison with distribution predictions and; thereby, provide a check for potential errors in fuel loading, fuel assembly enrichment, lumped poison elements and mispositioned or uncoupled control rods and to determine that steady-state core performance is in accordance with design.

PREREQUISITES

1. The reactor is critical with reactor power stable at the test plateau.
2. The Power Distribution Monitoring System (PDMS) is operable and the Integrated Computer System (ICS) is available to gather power distribution measurement data.

TEST METHOD

Reactor power is stabilized at greater than or equal to 25% Rated Thermal Power (RTP) and a power distribution measurement is obtained using the PDMS. The measurement is obtained with control rods controlling Axial Flux Difference (AFD) within the target band. Power distribution measurement information is then processed and calculations are performed. These measurements are repeated as a minimum at approximately 50%, 75% and 100% RTP.

ACCEPTANCE CRITERIA

1. Hot channel factors are less than or equal to design safety limits which provide a check for potential errors in fuel loading, enrichment of fuel assemblies, placement of lumped poison elements and to check for mispositioned or uncoupled control rods.
2. Hot channel factors are evaluated to ensure that limits would not be exceeded before reaching the high flux trip setpoint for the next power plateau.
3. The measured incore quadrant tilt is within the limits specified by the NSSS vendor.

Table 14.2-2 (Sheet 28 of 39) DYNAMIC AUTOMATIC STEAM DUMP CONTROL TEST SUMMARY

OBJECTIVE

To verify operation of the steam dump control system. This system has three controllers: Steam Pressure, Plant Trip and Load Rejection. Each controller is tested to demonstrate stability following a small transient.

PREREQUISITES

For the steam flow portion of the test:

1. The reactor is critical at no load temperature and pressure and able to permit increases and decreases in core power of approximately 3%.

2. The main turbine is not synchronized to the grid.

3. The reactor is less than 10% power.

TEST METHOD

With no steam dump flow, the steam dump control system control valves are functionally tested. This functional test includes modulating the valves open and closed, and tripping open all control valves using simulated signals. With steam dump flow, the Steam Pressure controller is tested by varying reactor power and observing that the controller automatically maintains steam header pressure. With steam dump flow, the Plant Trip controller is tested by simulating a reactor trip, varying reactor power, and observing controller input parameters and output. With steam dump flow, the Load Rejection controller is tested by simulating the loss of load permissive, varying reactor power and observing controller input parameters and output.

ACCEPTANCE CRITERIA

The steam dump control system can remain in automatic with plant parameters of steam pressure, RCS temperature, and steam dump valve demand not displaying divergent oscillations following small power transients.

Table 14.2-2 (Sheet 29 of 39) INTEGRATED COMPUTER SYSTEM TEST SUMMARY

OBJECTIVE

To verify by review of printouts and/or ICS displays that the ICS is receiving correct inputs and to validate that selected performance calculations performed by the ICS are correct.

PREREQUISITES

1. Preoperational testing of the ICS is completed.
2. The ICS is available for operation.

TEST METHOD

At approximately 0%, 30%, 50%, 75% and 100% power, selected computer inputs are verified correct by comparing printouts and/or ICS displays to main control board indications.

Selected performance calculations are performed and compared to alternate calculational methods to confirm validity. Selected application programs are performed and evaluated.

ACCEPTANCE CRITERIA

1. Selected process inputs, as indicated by main control boards, agree with ICS printouts and/or ICS displays within the accuracy of the instrumentation.
2. Selected performance calculations are confirmed by alternate calculational methods. Selected application programs perform correctly. At the 100% power test plateau, the thermal power measurement by the ICS agrees with the calculations within the uncertainty analysis.

   NOTE: Selected application programs performance is evaluated and judged acceptable by the plant staff.

Table 14.2-2 (Sheet 30 of 39) AUTOMATIC STEAM GENERATOR LEVEL CONTROL TEST SUMMARY

OBJECTIVE

To verify the proper operation and automatic response of the steam generator level control system for each steam generator during steady-state operation.

PREREQUISITES

1. The reactor is stable at the test plateau.
2. Steam generator level control system and associated subsystems have been calibrated.

TEST METHOD

This test is performed at various power levels from 5% through 100% reactor power. Major components and subsystems tested are: the feedwater bypass control valve control system, the main feedwater control valve control system, and the main feedwater pump speed control system. Steam generator parameters monitored during testing include steam pressure, steam flow, feedwater flow, steam generator level and controller output. At low power, the feedwater bypass control valves are tested by observing the response to a step change in level setpoint. Parameters are monitored during the transfer from the feedwater bypass control valves to the main feedwater control valves. At approximately 50% power, the main feedwater control valves are tested by observing the response to a step change in level setpoint. The main feedwater pump speed control is tested by observing the response to small feedwater pressure transients and evaluating main feedwater control valve position with steam flow requirements. At higher power levels (75% and 100% power test plateaus), parameters are monitored to verify proper steady state operation.

ACCEPTANCE CRITERIA

1. The feedwater bypass control valve control system and the main feedwater control valve control system automatically respond to maintain steam generator water level following a change in level setpoint.
2. The main feedwater pump speed control system:
    A. Remains in automatic with pump speed and feedwater pressure not displaying divergent oscillations following small pressure transients.
    B. Maintains adequate feedwater header pressure such that the main feedwater control valves are not fully open at 100% power.

Table 14.2-2 (Sheet 31 of 39) AUTOMATIC REACTOR CONTROL SYSTEM TEST SUMMARY

OBJECTIVE

To verify the performance of the automatic reactor control system in maintaining reactor coolant average temperature within acceptable steady state limits.

PREREQUISITES

1. The reactor is stable at approximately 50% power.

2. Pressurizer pressure and level, steam dump, SG level and FW pump speed control systems are in automatic control.

3. Reactor rod control is in manual, controlling in overlap between the insertion limits and ARO configuration.

TEST METHOD

Neutron flux, power mismatch, T_{avg}, T_{ref}, T_{error}, rod speed demand signal, steam header, turbine impulse and pressurizer pressure signals are continuously recorded. Rod control will be placed in automatic to verify T_{avg} is maintained within specified limits of T_{ref}. Rod control is switched to manual and T_{avg} will be successively increased and decreased higher and lower than the T_{ref} setpoint by control rod motion. Rod control will be switched to automatic and the transient recovery of the system will be observed. Recorded data will be analyzed to determine whether setpoint changes are required.

ACCEPTANCE CRITERIA

1. No manual intervention is required to return the plant to stable conditions for both steady-state and transient conditions.

2. The rod control system responds properly in automatic mode to position control rods and return T_{avg} to within limits, specified by the NSSS vendor, of T_{ref}.

Table 14.2-2 (Sheet 32 of 39) SHUTDOWN FROM OUTSIDE THE CONTROL ROOM TEST SUMMARY

OBJECTIVE

To demonstrate that the unit can be taken to and maintained in the hot standby condition from outside the control room. This test will be performed during the 100% power testing plateau and be initiated from approximately 30% power.

PREREQUISITES

1. The reactor is at approximately 30% power.
2. The Technical Specification minimum shift crew is ready to implement the abnormal operating instruction for abandoning the main control room.
3. The responsibilities and authority of control room observers is discussed in the test procedure.

TEST METHOD

A reactor trip will be initiated from a location outside the main control room. After confirmation of the reactor/turbine trip, operator actions will be taken to abandon the main control room in accordance with the appropriate operating instruction. Actions will be taken to achieve and maintain hot standby conditions for at least thirty minutes from locations outside the main control room. Finally, the unit is cooled down approximately 50EF to show cooldown capability and control.

ACCEPTANCE CRITERIA

1. The unit is tripped from outside the main control room and is controlled at hot standby conditions from outside the main control room for at least thirty minutes using Technical Specifications minimum shift crew.
2. The ability to cool the unit down approximately 50EF using the appropriate operating instruction(s) has been demonstrated.

Table 14.2-2 (Sheet 33 of 39) TURBINE GENERATOR TRIP WITH COINCIDENT LOSS OF OFFSITE POWER TEST SUMMARY

OBJECTIVE

To demonstrate that the unit's response to a turbine generator trip with a coincident loss of offsite power is in accordance with design. This test will be performed during the 50% power testing plateau and be initiated from approximately 30% power.

TEST METHODS

Option 1 -Test Unit 2 while Unit 1 is defueled

A dual unit, loss of offsite power will be initiated by manually tripping the U2 turbine followed immediately with opening the feeder breakers from the four common station service transformers (CSSTs). All four emergency diesel generators (EDGs) will start and connect to their respective shutdown board. The unit 2 reactor coolant system is maintained in hot standby (Mode 3) conditions for a minimum of 30 minutes using only equipment available during the loss of offsite power. Onsite power systems (emergency diesel generators and batteries) provide the necessary power to controls, indicators, and equipment for the duration of the test.

PREREQUISITES

1. Unit 1 is defueled
2. Reactor power is approximately 30% of rated thermal power.
3. The generator is synchronized to the TVA grid with a load of greater than or equal to 120 MWe.
4. Unit 2 electrical distribution is in its normal operating lineup. Unit 1 electrical distribution is as required to support the plant conditions. The unit station service transformers are carrying the RCP buses and unit buses and the four common station service transformers are carrying the common and shutdown buses. The automatic transfers to offsite alternate power supplies are defeated.
5. The emergency diesel generators are in normal standby.

Option 2- Limited Loss of Offsite Power

The main turbine will be manually tripped followed immediately with opening the breakers supplying offsite power to the 2A-A and 2B-B 6.9KV shutdown boards. All four emergency diesel generators (EDGs) will start and the 2A-A and 2B-B EDGs connect to their respective shutdown board. The unit 2 reactor coolant system is maintained in hot standby (Mode 3) conditions for a minimum of 30 minutes using equipment available during the loss of offsite power. Onsite power systems (emergency diesel generators and batteries) provide the necessary power to controls, indicators, and equipment for the duration of the test.

PREREQUISITES

1. Reactor power is approximately 30% of rated thermal power
2. The generator is synchronized to the TVA grid with a load of greater than or equal to 120 MWe
3. Unit 2 electrical distribution is in its normal operating lineup wrth the Unit 1 electrical distnbution aligned as required to support the plant conditions. The unit station service transformers are carryrng the RCP buses and unit buses and the common station servrce transformers are carryrng the common

and shutdown buses. The automatic transfers to offsite alternate power supplies are defeated.

4. The emergency diesel generators are in normal standby.

ACCEPTANCE CRITERIA

1. Safety injection is not initiated.
2. Pressurizer and steam generator safety valves do not open.
3. Monitored plant parameters can be maintained in hot standby conditions using only the equipment available during the test.
4. Diesel generators are supplying their respective shutdown boards.

Table 14.2-2 (Sheet 34 of 39) LOAD SWING TEST SUMMARY

OBJECTIVE

To demonstrate the ability of primary and secondary side systems, including automatic control systems to sustain 10% step changes in turbine generator load.

PREREQUISITES

1. The reactor is stable at the test plateau.
2. Pressurizer pressure and level, steam dump, rod control, steam generator level and feedwater pump speed control systems are in automatic.

TEST METHOD

Applicable plant parameter data will be obtained (e.g., reactor power, RCS temperature and pressure, pressurizer level, feedwater and steam flows, steam generator levels, feedwater pump speed, feedwater pressure, generator breaker position, and turbine speed) during the transient. Turbine governor valves are positioned to produce approximately a 10% step decrease in generator load. Parameters are allowed to stabilize, then turbine governor valves are positioned to produce approximately a 10% step increase in generator load. This test is performed at the 50% and 100% power test plateau.

ACCEPTANCE CRITERIA

1. Reactor and turbine do not trip.
2. Safety injection is not initiated.
3. Pressurizer and steam generator relief and safety valves do not lift.
4. Monitored plant parameters stabilize without manual intervention.

Table 14.2-2 (Sheet 35 of 39) LARGE LOAD REDUCTION TEST SUMMARY

OBJECTIVE

To demonstrate the ability of primary and secondary side systems, including automatic control systems, to sustain a 50% step decrease in turbine generator load.

PREREQUISITES

1. The reactor is stable at the 100% power test plateau.

2. Pressurizer pressure, pressurizer level, steam dump, rod control, steam generator level, and feedwater pump speed controls are in automatic.

TEST METHOD

Applicable plant parameter data will be obtained (e.g., reactor power, RCS temperature and pressure, pressurizer level, feedwater and steam flows, steam generator levels, feedwater pump speed, feedwater pressure, generator breaker position, and turbine speed) during the transient. Turbine governor valves are positioned to produce approximately a 50% step decrease in generator load. Parameters are allowed to stabilize.

ACCEPTANCE CRITERIA

1. Reactor and turbine do not trip.

2. Safety injection is not initiated.

3. Pressurizer and steam generator safety valves do not lift.

4. Monitored plant parameters stabilize without manual intervention.

Table 14.2-2 (Sheet 36 of 39) PLANT TRIP FROM 100% POWER TEST SUMMARY

OBJECTIVE

To demonstrate the ability of primary and secondary side systems to bring the unit to stable conditions following a turbine trip resulting from the opening of the generator output breaker and to determine the overall response time of the RCS narrow range hot leg RTDs.

PREREQUISITES

1. The reactor is stable at the 100% power test plateau.
2. Pressurizer pressure, pressurizer level, steam dump, steam generator level, and feedwater pump speed controls are in automatic.
3. The electrical distribution system for the unit is in normal full power alignment.

TEST METHOD

Applicable plant parameter data will be obtained (e.g., reactor power, RCS temperature and pressure, pressurizer level, feedwater and steam flows, steam generator levels, feedwater pump speed, feedwater pressure, generator breaker position, and turbine speed) during the transient. The transient is initiated by opening the generator output breaker from the control room. The turbine trips as a direct result of opening the generator output breaker. A reactor trip follows the turbine trip, and plant parameters are allowed to stabilize. Operators may take manual control of plant systems as directed by operating procedures.

ACCEPTANCE CRITERIA

1. A safety injection is not initiated.
2. Pressurizer and steam generator safety valves do not lift.
3. The overall response time for the RCS narrow range hot leg RTDs is within design requirements.
4. The reactor trips.
5. Nuclear flux rapidly decreases after the turbine trip.
6. The turbine trips without overspeed protection system initiation.

Table 14.2-2 (Sheet 37 of 39) REACTOR VESSEL LEVEL INSTRUMENTATION SYSTEM (RVLIS) TEST SUMMARY

OBJECTIVE

To collect RVLIS data during power ascension in order to determine the RVLIS scaling coefficients.

PREREQUISITES

1. The reactor core has been loaded.
2. The RCS is filled and vented.
3. Dynamic range RVLIS is operational.

TEST METHOD

At the beginning of plant heatup with RCP's running in various combinations, data is collected from the dynamic range RVLIS, wide range RTDs and wide range pressure instrumentation. Data will continue to be collected during heatup. Data collection with the various RCP combinations will be repeated at 557°F. The zero power level scaling coefficients may then be entered, if deemed necessary, and a verification that RVLIS will compensate for RCP status changes will be made. As the plant escalates to 100% power the two dynamic RVLIS trains are monitored. At the 100% power test plateau the full power scaling coefficient are determined.

ACCEPTANCE CRITERIA

After the full power scaling coefficient have been entered the two dynamic RVLIS trains are to track within the limits specified by the vendor.

Table 14.2-2 (Sheet 38 of 39) LOOSE PARTS MONITORING SYSTEM TEST SUMMARY

OBJECTIVE

Monitor the reactor coolant system for the presence of loose parts and system degradation. Verify the gain and alarm setpoints are properly adjusted for changing unit conditions while maintaining system sensitivity.

PREREQUISITES

1. Preoperational testing of the loose parts monitoring system is complete.

2. Channel calibration of the loose parts monitoring system is complete.

TEST METHOD

Testing consists of documenting gain and alarm setpoints at the initial criticality and low pressure (<5%) physics testing, 30%, 50%, 75%, and 100% power test plateaus. Baseline frequency spectrums are recorded for each channel while maintaining system sensitivity with the unit above 90% power.

ACCEPTANCE CRITERIA

1. Required changes to the gain/alarm setpoints are documented.

2. Frequency spectrum for each channel are available for conditions above 90% power. (These will be used for baselines in order to investigate apparent shifts in the frequency spectrums at later times.)

Table 14.2-2 (Sheet 39 of 39) INADEQUATE CORE COOLING MONITORING SYSTEM TEST SUMMARY

OBJECTIVE

To demonstrate the capability of the Common Q Post Accident Monitoring System (PAMS) to accurately process and display signals associated with reactor vessel level, subcooling margin, and incore thermocouples.

PREREQUISITES

1. Reactor vessel head is installed.
2. Initial Common Q PAMS checkout is complete.
3. Eagle 21 checkout procedure is complete for process protection system racks supplying inputs to Common Q PAMS.
4. The initial core load has been completed and the Incore Instrument Thimble Assemblies (IITAs) are installed.
5. Reactor Vessel Level Instrumentation System (RVLIS) fill and vent is complete.

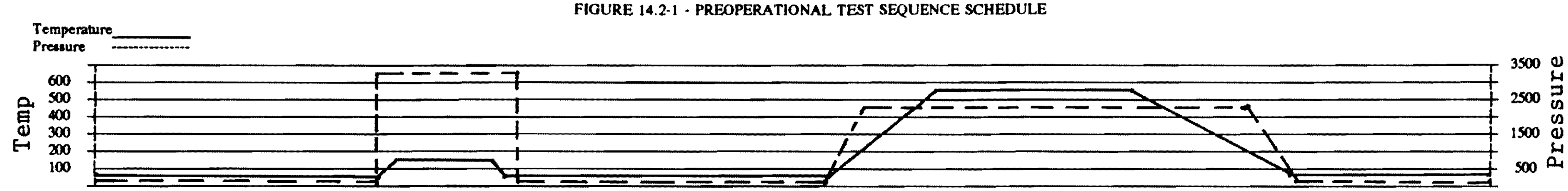
TEST METHOD

1. Monitor indicators in the main control room during varying reactor coolant system conditions.
2. Observe all pages of the screen displays to verify operability.
3. Simulate malfunction conditions to verify malfunction alarms.
4. Simulate loss of subcooling margin to verify main control room alarm is initiated.

ACCEPTANCE CRITERIA

1. Main control room indicators including all pages of the screen displays, respond to changes in reactor coolant system conditions as required by FSAR Sections 5.6, 7.5, and 7.7.
2. The main control room alarms are initiated for Common Q PAMS simulated malfunction conditions as required by approved design and vendor information.
3. The main control room alarm is initiated for simulated loss of subcooling margin as required by approved design documents.

FIGURE 14.2-1 - PREOPERATIONAL TEST SEQUENCE SCHEDULE



RCS COMPLETION AND EQUIPMENT INSTALLATION	RCS CLEANING, INITIAL RCS FILL, AND SERVICE SYSTEMS CHECKOUT	COLD FUNCTIONAL TESTING	CONTAINMENT TESTS	PREPARATION FOR HOT FUNCTIONAL TESTING	RCS HEATUP	INTEGRATED HOT FUNCTIONAL TESTING	RCS COOLDOWN	PREPARATION FOR CORE LOADING	CORE LOADING
1. Offsite Power System	1. Auxiliary Building Heating, Ventilating, and Air Conditioning	1. RCS and Associated Systems Hydrostatic Testing	1. Air Return Fans	1. Pressurizer Relief Tank	1. RCS Heatup	1. RCS at Temperature	1. RCS Cooldown	1. Fuel Transfer System Manipulator Crane	
2. Onsite AC Distribution System	2. Liquid Waste Drains, Collection and Transfer Facilities	2. Spent Fuel Pit Cooling System	2. Containment Local Leak Rate Test	2. RCS Heatup	2 Thermal expansion and Vibration	2. CVCS Functional Test	2. RHR System	2. Nuclear Instrumentation System	
3. Computer System	3. Primary Makeup Water System	3. Containment Isolation System	3. Airlock Leakage and Operational Test	3. RHR System And Safety Injection Testing	3. RHR System	3. Steam Dump Control	3. Thermal expansion and Vibration	3. Safety Injection System	
4. Environs Monitoring System	4. Station Drainage System	4. Electro-hydraulic Control System	4. Upper Conainment Ventilation System	4. Thermal expansion and Vibration	RCS Loop RTD and Theroouple Cross Calibaion	4. Initial Turbine Roll	4. Deleted	4. RCS Post Hot Functional Testing, Inspection, Cleaning and Testing	
5. Emergency Lighting System	5. Control Air System	5. Main Feedwater System	5. Lower Containment System	5. Condenser Circulating Water System	5. Containment Isolation System	5. Ice Condenser system		5. Reactor Protection Time Response	
6. Vital 120V AC Power System	6. Nitrogen Supply System	6. Sampling System	6. Containment Spray System	6. Essential Raw Cooling Water System	6 Deleted	6. Plant Sampling System Post Accident Sampling System		6. Reactor Protection Operational Test	
7. Vital 125V DC Power System	7. Compressed Air System	7. Auxiliary Feedwater	7. Post-LOCA Hydrogen Igniters	7. Onsite AC and DC Distribution System	7. Evacuation Alarm Signal	7. Steam Generator Blowdown System		7. Reactor Plant System Setpoint Verification	
8. Emergency Sound Power Telephones	8. Hydrogen Supply System	8. Annunciator Alarm System	8. Shield Building Inleakage Rate Test Emergency Gas Treatment System Functional Tests	8. Condenser Vent System	8. Auxiliary Feedwater System	8. Auxiliary Feedwater System		8. Safeguards System Operational Test Diesel Generator Testing	

Sheet 1 of 2

Figure 14.2-1 Preoperational Test Schedule (Sheet 1 of 2)

WBNP-69

FIGURE 14.2-1 (Continued)

RCS COMPLETION AND EQUIPMENT INSTALLATION	RCS CLEANING, INITIAL RCS FULL, AND SERVICE SYSTEMS CHECKOUT	COLD FUNCTIONAL TESTING	CONTAINMENT TESTS	PREPARATION FOR HOT FUNCTIONAL TESTING	RCS HEATUP	INTEGRATED HOT FUNCTIONAL TESTING	RCS COOLDOWN	PREPARATION FOR CORE LOADING	CORE LOADING
9. Status Monitoring System	9. Control Building Heating, Ventilating, and Air Conditioning	9. CVCS Boric Acid System		9. Control Rod Drive Cooling System	9. Main Steam System	9. RCS Loop RTD and Theroouple Cross Calibaion		9. Containment Vessel Pressure and Leak Test	
10. Reactor Coolant Pressure Boundary Leakage Detection System	10. Essential Raw Cooling Water System	10. Boron Recycle System		10. Diesel Generator and Supporting Auxiliaries		10. Containment Isolation System		10. Computer Data Printout Verification	
	11. Component Cooling Water System	11. Gaseous Waste Processing		11 Deleted		11. Boric Acid System		11 Deleted	
	12. Area Radiation Monitoring System	12. Liquid Waste Processing		12. Containment Isolation System		12. Main Steam System		12. ATWS Test	
	13. Process Radiation Monitoring System	13. Solid Waste Processing		13. Fire Protection System		13. Control Building Heating, Ventilating and Air Conditioning			
	14. Cranes and Heavy Equipment			14. Deleted		14. Auxiliary Building Heating, Ventilating and Air Conditioning			
	15. Evacuation Alarm Signal					15. Thermal expansio and vibration			
	16. Fuel Handling Tools and Fixtures					16. Control System for Turbine Runback			
	17. Annunciator Equipment Checkout								

Sheet 2 of 2

Figure 14.2-1 Preoperational Test Schedule (Sheet 2 of 2)

(Page 1 of 2)

Figure 14.2-2 Power Ascension Test Schedule

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                                                                                                    Commercial ..........
                                                                                                      Operation...........
                             Fuel Load.........Initial Crit...............................Power Ascension...........Approx. 5 Months ............
                                                                                                  after fuel load.................
                            ............................LowPwrTests.............................................................................

                                               0........10........20.........30.........40.........50..........60..........70........80.........90.......100..............
Initial Fuel Loading -----------------------------X
Reactor System Sampling For Core --------X
RVLIS-------------------------------------------------.X-----------------------------------------------------------------------------------------X

Control Rod Drive Mechanism Timing -----------X---------------X.
Rod Position Indication System ------------------------------------X
Rod Drop Time Measurements------------------------------------X
Rod Control System-----------------------------------------------X
IITA Detectors ---------------------------------------------------------X-----------------------------X
Fixed Incore Flux Detectors (testing
should be done after criticality and
above 20% power)
Pressurizer Spray And Heater Capab. --------------------------X
Reactor Coolant Flow Coastdown --------------------------------X
Reactor Trip System ---------------------------------X

Initial Criticality -----------------------------------------------------------X
Determin. of Core Pwr Range --------------------------------------------X
Moderator Temperature Coefficient --------------------------------------X
Rod And Boron Worth Measurements -----------------------------------X
Core Reactivity Balance ------------------------------------------------------X-----------------------------------------------------------------X

Flux Distribution Measurement --------------------------------------------------------------------------------X------------------X-------------------------X-----------------------------X
Radiation Baseline Survey -------------------------X---------------------------------X----------------------------------------------X------------------------------------------------------------X
Integrated Computer Sytem ------------------------------------------------------X-----------------------------X--------------------X-----------------------X------------------------------X
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(Continued) (Page 2 of 2)

Figure 14.2-2 Power Ascension Test Schedule

Commercial ....

Operation..........

Fuel Load......Initial Crit..........Power Ascension..........Approx. 5 Months ....

after fuel load..........

..........LowPwrTests..........

0........10........20........30........40........50........60........70........80........90.......100..........

Shutdown From Outside the Control Rm -.----------X (During the 100% plateau.)

T/G Trip With Cond. Loss Offsite Pwr ----------X (During the 50% plateau.)

Load Swing ----------X----------X

Large Load Reduction ----------X

Plant Trip From 100% Power ----------X

RCS Flow Measurment ----------X----------X----------X----------X

Automatic Steam Gen. Level Control ----------X----------X----------X----------X----------X

Dynamic Automatic Steam Dump Control ----------X

Automatic Reactor Control System ----------X

Thermal Expan. of Piping Systems ----------X----------X----------X----------X----------X

Pipe Vibration Monitoring ----------X----------X----------X----------X----------X

Oper. Align. of Nuclear Inst. ----------X----------X----------X----------X----------X----------X

Loose Parts Monitoring System ----------X----------X----------X----------X----------X

Oper. Align. of Process Temp. Inst. ----------X----------X----------X----------X----------X----------X

Startup Adjust. of Reactor Control ----------X----------X----------X----------X----------X

Calib. of Steam and Feedwater Flow ----------X----------X----------X----------X----------X