Chapter 14

VCS UFSAR Table of Contents

VCS UFSAR Formatting Legend

Color	Description
Black	Original Westinghouse AP1000 DCD Tier 2 & Tier 2*, Revision 19 content
Purple	Departures from AP1000 DCD Tier 2 & Tier 2*, Revision 19 content
Blue	Standard FSAR content
Green	Site-specific FSAR content
Red	Linked cross-references (chapters, appendices, sections, subsections, tables, figures, and references)

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Chapter 14 Initial Test Program

14.1 Specific Information to be Included in Preliminary/Final Safety Analysis Reports

Not applicable to the AP1000.

14.2 Specific Information to be Included in Standard Safety Analysis Reports

14.2.1 Summary of Initial Test Program and Objectives RN-15-099

The purpose of this section is to describe the initial test program that is performed for the AP1000 plant.

The overall objective of the initial test program is to demonstrate that the plant has been constructed as designed, that the systems perform consistent with the plant design, and that activities culminating in operation at full licensed power including initial fuel load, initial criticality, and power ascension are performed in a controlled and safe manner. RN-15-099

Preoperational and/or startup testing is performed on those systems that are:

a) Relied upon for safe shutdown and cooldown of the reactor under normal plant conditions and for maintaining the reactor in a safe condition for an extended shutdown period;

b) Relied upon for safe shutdown and cooldown of the reactor under transient and postulated accident conditions and for maintaining the reactor in a safe condition for an extended shutdown period following such conditions;

c) Relied upon for establishing conformance with safety limits or limiting conditions for operation that will be included in the facility technical specifications;

d) Classified as engineered safety features actuation systems (ESFAS) or are relied upon to support or ensure operation of engineered safety features actuation systems within design limits;

e) Assumed to function or for which credit is taken in the accident analysis of the AP1000 as described in this document. RN-13-095

f) Used to process, store, control, or limit the release of radioactive materials.

g) Other systems identified in Regulatory Guide 1.68, Revision 2, Appendix A that are in the AP1000 and are not captured by criteria a) through f).

The inspections, tests, analyses and acceptance criteria of 10 CFR 52.47 (b)(1) relating to the AP1000 design are found in the AP1000 Certified Design Material (see Section 14.3). RN-13-095

The Initial Test Program (ITP) is implemented in three phases, categorized as component, preoperational, and startup testing. The individual programs for these three phases, which make up the overall ITP, are discussed in Sections 14.2.1.1, 14.2.1.2, and 14.2.1.3. For clarity, it is noted that construction and installation tests are performed by the construction organization to verify that the construction, installation, and assembly processes have been properly performed. Construction and installation tests are routinely performed prior to component testing and are not considered part of the ITP. RN-14-110

- Component tests are performed following turnover from construction to prepare systems for preoperational testing. Component testing includes, as appropriate, preliminary operation of components and systems and various electrical and mechanical tests including cleaning and flushing, electrical checks, operability checks, and instrumentation calibration. RN-14-110

- Preoperational tests are performed after associated component tests, but prior to initial fuel loading to demonstrate the capability of plant systems to meet performance requirements. RN-14-110

- Startup tests begin with the initial fuel loading and are performed to demonstrate the capability of individual systems, as well as the integrated plant, to meet performance requirements.

Section 14.2 provides the requirements to be included in the ITP Administrative Manual (Procedures), as discussed in subsection 14.4.3. The information referenced in this section meets the Initial Test Program (ITP) criteria of NUREG-0800 and is formatted to follow Regulatory Guide 1.206, Part I, Section C.I.14.2. RN-14-110

The ITP is applied to structures, systems, and components that perform the functions described in the Regulatory Guide 1.68 evaluation in Section 1.9. The ITP is also applied to other structures, systems and components. The ITP Administrative Manual will reference the AP1000 structures, systems and components to which the ITP is applied. RN-14-110

14.2.1.1 Component Test Program Objectives

The objective of component testing is to prepare systems for preoperational testing. Component tests are conducted following turnover from construction and include the following, as appropriate: RN-14-110

- Cleaning and flushing RN-14-110
- Checks of electrical wiring
- Mechanical and electrical component initial setup and functional testing, including valve testing
- Initial energization and operation of equipment
- Calibration of instrumentation
- Component and digital control system interface testing

Completion of component testing demonstrates that the tested components are ready for system preoperational testing. RN-14-110

Development of component tests is based on the engineering information for the equipment and systems installed.

14.2.1.2 Preoperational Test Program Objectives

Following component testing, preoperational tests are performed to demonstrate that equipment and systems perform in accordance with design criteria so that initial fuel loading, initial criticality, and subsequent power operation can be safely undertaken. Preoperational tests at elevated pressure and temperature are referred to as hot functional tests. RN-14-110

The general objectives of the preoperational test program are the following:

- Demonstrate that essential plant components and systems, including alarms and indications, meet appropriate criteria based on the design
- Provide documentation of the performance and condition of equipment and systems
- Provide baseline test and operating data on equipment and systems for future use and reference

- Operate equipment for a sufficient period to demonstrate performance

- Demonstrate that plant systems operate on an integrated basis

Abstracts for the preoperational tests for portions of systems/components that perform safety-related functions; perform defense-in-depth functions; contain, transport, or isolate radioactive material; and for applicable systems that are specified in Regulatory Guide 1.68, Appendix A, Revision 2 are provided in subsection 14.2.9. RN-15-099

Plant operating, emergency, and surveillance procedures are incorporated into the initial test program procedures to the extent practical. These procedures are verified through use, to the extent practicable, during the preoperational test program and revised if necessary, prior to fuel loading. RN-13-095

Plant equipment used in the performance of preoperational tests is operated, to the extent practical, in accordance with appropriate operating procedures, thereby giving the plant operating staff an opportunity to gain experience in using these procedures and demonstrating their adequacy prior to plant initial criticality. RN-13-095

14.2.1.3 Startup Test Program Objectives

The startup test program begins with initial fuel loading after the preoperational testing has been successfully completed.

Startup tests can be grouped into four broad categories:

- Tests related to initial fuel loading
- Tests performed after initial fuel loading but prior to initial criticality
- Tests related to initial criticality and those performed at low power (less than 5 percent)
- Tests performed at power levels greater than 5 percent

During performance of the startup test program, the plant operating staff has the opportunity to obtain practical experience in the use of normal and abnormal operating procedures while the plant progresses through heatup, criticality, and power operations.

The general objectives of the startup test program are:

- Install the nuclear fuel in the reactor vessel in a controlled and safe manner.

- Verify that the reactor core and components, equipment, and systems required for control and shutdown have been assembled according to design and meet specified performance requirements.

- Achieve initial criticality and operation at power in a controlled and safe manner.

- Verify that the operating characteristics of the reactor core and associated control and protection equipment are consistent with design requirements and accident analysis assumptions.

- Obtain the required data and calibrate equipment used to control and protect the plant.

- Verify that the plant is operating within the limits imposed by the Technical Specifications.

Abstracts of the startup tests are provided in this subsection 14.2.10. RN-15-099

14.2.1.4 Testing of First of a Kind Design Features

First of a kind (FOAK) testing may occur in any of the phases, depending on the nature of the testing and required sequencing of the tests. When testing FOAK design features, applicable operating experience from previous test performance on other AP1000 plants is reviewed, where available, and the ITP modified as needed based on those lessons learned.

14.2.1.5 Credit for Previously Performed Testing of First of a Kind Design Features

In some cases, FOAK testing is required only for the first of a new design or for the first few plants of a standard design. In such cases, credit may be taken for the previously performed tests. A discussion is included in the test reports of the results of those tests that are credited. RN-13-095

14.2.2 Organization, Staffing, and Responsibilities

The Owner has overall responsibility for the ITP. The Owner has delegated the responsibilities to manage component and preoperational testing to the ITP Organization. The Owner has the overall responsibility for conducting startup testing. RN-15-099

The ITP Organization is described in subsection 14.2.2.1. The organization for operating and maintaining the AP1000 plant is described in Section 13.1. RN-14-110

Table 13.4-201 provides milestones for initial test program implementation.

14.2.2.1 ITP Organization

The ITP Organization is responsible for component testing, preoperational testing, and startup testing. The Owner's General Manager Operational Readiness has overall responsibility for the activities performed by the ITP Organization. The Owner has delegated authority for technical direction of ITP Organization's activities to the Manage in Charge of ITP Organization. The ITP Organization structure (organizational chart) is included in the Startup Administrative Manual. RN-13-095 RN-15-099

14.2.2.1.1 Manager In Charge of ITP Organization RN-13-095

The manager in charge of ITP Organization reports to the Owner’s General Manager Operational Readiness. The manager in charge of the ITP Organization is responsible for: RN-12-042

- Staffing the ITP Organization.
- Developing, and reviewing, the administrative and technical procedures associated with component, preoperational, and startup testing. RN-13-095
- Managing the ITP Organization.
- Implementing the component, preoperational, and startup testing schedule.
- Managing contracts associated with component, preoperational and startup testing.

14.2.2.1.2 Functional Manager In Charge of ITP Organization Support

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The functional manager in charge of ITP Organization Support reports directly to the manager in charge of ITP Organization. The functional manager in charge of ITP Organization Support plans and schedules procedure development to support component, preoperational, and startup testing. The functional manager in charge of ITP Organization Support is responsible for ITP related administrative processes and test execution support, which includes ensuring the Joint Test working group (JTWG) reviews and approves ITP administrative procedures.

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14.2.2.1.3 Functional Manager in Charge of Component Testing

The functional manager in charge of component testing reports to the manager in charge of ITP Organization and is responsible for:

RN-13-095

- Preparing and maintaining the component test schedule.
- Accepting construction turnover to the ITP Organization.
- Managing initial energization.
- Managing component checkout, calibration and digital controls interface verification for plant components.
- Coordinating vendor participation in component testing activities, as required.
- Supervising and directing component test personnel.
- Supporting administrative controls to address system and equipment configuration control.
- Ensuring the performance of final cleaning and cleanliness verification of piping and ductwork.
- Issuing periodic progress reports that identify overall progress and potential challenges.
- Verifying the adequacy of test results.
- Managing system and equipment turnover to the preoperational test group.

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14.2.2.1.4 Functional Manager in Charge of Preoperational Testing

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The functional manager in charge of preoperational testing reports to the manager in charge of ITP Organization and is responsible for:

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- Participating in the JTWG and ensuring that the JTWG reviews and approves preoperational test procedures. The JTWG structure and responsibilities are defined in subsection 14.2.2.3.
- Preparing and maintaining the preoperational test schedule.
- Accepting turnover from component testing to the preoperational testing organization.
- Preparing preoperational test procedures.
- Coordinating vendor participation in preoperational testing activities, as required.
- Supervising and directing preoperational personnel.

- Involving operations personnel in testing activities. Utilizing operations personnel, to the extent practical, as test witnesses or test performers to provide the operations personnel with experience and knowledge. RN-13-095

- Supporting administrative controls to address system and equipment configuration control.

- Issuing periodic progress reports that identify overall progress and potential challenges.

- Verifying the adequacy of preoperational test results.

- Supporting jurisdictional turnover to the Licensee. RN-15-099

14.2.2.1.5 Functional Manager In Charge of Startup Testing RN-13-095

The functional manager in charge of startup testing reports directly to the manager in charge of ITP Organization. The functional manager in charge of startup testing supports the Licensee during startup testing. The functional manager in charge of startup testing is responsible for: RN-15-099

- Participating in the JTWG and ensuring that the JTWG reviews and approves startup test procedures. The JTWG structure and responsibilities are defined in subsection 14.2.2.3.

- Preparing a detailed startup testing schedule.

- Coordinating vendor participation in startup testing activities.

- Supervising and directing the startup personnel.

- Supporting operations personnel in testing activities.

- Implementing administrative controls to address system and equipment configuration control.

- Maintaining the startup schedule.

- Issuing periodic progress reports that identify overall progress and potential challenges.

14.2.2.1.6 Test Engineers RN-13-095

The test engineers report directly to the functional manager in charge of conducting the individual testing phases. The test engineers are responsible for:

- Complying with administrative controls.

- Developing and performing test procedures. RN-13-095

- Identifying any special or temporary equipment or services needed to support testing.

- Coordinating testing with involved groups.

- Reviewing and evaluating test results.

14.2.2.2 ITP Organization Personnel Qualifications and Training RN-13-095

Procedures are prepared to identify the appropriate training, qualification, and certification for test personnel. Records are kept to document training, qualification and certification, as appropriate. Test personnel are qualified as described below.

- Acceptable qualifications of non-supervisory test engineers follow the guidance provided in Regulatory Guide 1.28 as discussed in Appendix 1AA, i.e., ASME NQA-1-1994, Appendix 2A-1, Nonmandatory Guidance on the Qualification of Inspection and Test Personnel.

- Acceptable qualifications of supervisory test engineers follow the guidance provided in Regulatory Guide 1.28, i.e., ANSI/ANS-3.1-1993, Selection Qualification, and Training of Personnel for Nuclear Power Plants.

The training program/procedures shall include:

- The education, training, experience, and qualification requirements for ITP Organization supervisory and test personnel responsible for managing, developing, or conducting each test phase, or development of testing procedures. Refer to Section 13.2 for training and qualification information for the organization responsible for operating and maintaining the plant. RN-13-095 RN-15-099

The ITP Administrative Manual (Procedure) shall include: RN-14-110

- The implementation of measures to verify that personnel formulating and conducting test activities are not the same personnel who designed or are responsible for satisfactory performance of the system(s) or design features(s) being tested. This provision does not preclude members of the design organization from participating in test activities. This description also includes considerations of staffing effects that could result from overlapping initial test programs at multi-unit sites.

14.2.2.3 Joint Test Working Group

The Joint Test Working Group (JTWG) consists of an organizational group of authorized representative personnel from the Plant's operations and support group functions, Westinghouse Electric Company (WEC), Responsible Design Organization and other test support groups as identified below. RN-13-095

The Licensee has the overall responsibility for conduct of the ITP. The manager in charge of ITP Organization may be assigned overall responsibility and authority for technical direction of the component testing, preoperational testing, and startup testing and may act as the JTWG Chairman. RN-13-095

The JTWG Chairman reports to the Owner's General Manager Operational Readiness or qualified designee for matters of component, preoperational, or startup testing authority and acceptance. RN-13-095

The JTWG provides the following administrative oversight activities associated with the implementation of the ITP. RN-13-095

- Review and approve ITP administrative manual. RN-13-095

- Review and approve preoperational and startup test procedures. RN-15-099

- Review and approve component test procedures, as specified by the JTWG chairman.

- Oversee the implementation of component testing, preoperational testing, and startup testing, including planning, scheduling and performance. RN-13-095 RN-15-099

- Review and approve preoperational and startup test results.

The JTWG is composed of qualified representatives provided from the following organizations: RN-13-095

- Licensee's Operations Group

- Licensee's Maintenance Group

- Site Preoperational Test Group

- Site Startup Test Group

- Licensee's Engineering Group

- Responsible Design Organization RN-13-095

The following are additional generic details of the key responsibilities, authorities and interfaces of the organizations delineated above: RN-15-099

- Licensee's Operations Group RN-13-095

  The Operations Group has the overall responsibility for Plant Operations, including administrative control and tag-outs subsequent to system turnover. Their primary interfaces are with the Licensee Engineering and Technical Support organizations as well as the responsible design organization, Component, Preoperational, and Startup Testing Teams. RN-13-095 RN-15-099

- Licensee's Maintenance Group RN-13-095

  The Maintenance Group has the overall responsibility for the Maintenance of Plant systems and components subsequent to System Turnover. They are key participants and maintainers of system maintenance control and tag-outs. Their primary interfaces are with the Licensee Operations Group and Technical Support organizations, as well as the responsible design organization, Component, Preoperational, and Startup Testing Teams. RN-13-095

- Licensee's Engineering Group RN-15-099

  This group has the primary responsibility for site engineering and design oversight of the plant components and systems, as well as interfacing with the vendor engineering organization. This organization primarily interfaces with the Operations Group as well as the responsible design organization, Component, Preoperational, and Startup Testing Teams.

- Site Preoperational Test Group

  This group has the primary responsibility for the development, maintenance and performance of the site preoperational procedures at the site. The primary interfaces for this group are the Licensee Operations Group and Technical Support organizations, as well as the responsible design organization, Component and Startup Testing Teams and the Construction Services Group. Additional specific information regarding this organization's responsibilities and interfaces is described in subsection 14.2.2.5 below. Once preoperational testing is complete, this group turns systems over to the Site Licensee. RN-13-095 RN-15-099 RN-14-110

- Site Startup Test Group

  This group has the primary responsibility for the development and maintenance of the site startup procedures. The primary interfaces for this group are the Licensee's Operations Group and Technical Support organizations, as well as the responsible design organization, Preoperational Testing Team and the Construction Services Group. Additional specific information regarding this organization's responsibilities and interfaces is described in subsection 14.2.2.6 below. RN-13-095 RN-15-099 RN-14-110

- Responsible Design Organization RN-13-095

  The responsible design organization will vary and may be CB&I, WEC, and/or Owner depending on the particular area or SSC in question. This organization has the primary responsibility for the approval of the system test specification document indicating that the test and acceptance criteria are in accordance with the current approved design. Resolution of any questions from the review and use of engineering documents to support test procedures is the responsible design organizations's responsibility through interface between the site and the vendor's home offices. The responsible design organization will resolve any deficiencies that occur during testing of structures, systems, and components that cannot be resolved within the ITP Organization. The responsible design organization will interface with the Component, Preoperational, and Startup Testing Teams. As necessary, the responsible design organization may also interface with the Licensee Operations, Maintenance and Engineering groups. RN-15-099

14.2.2.4 Site Component Test Group

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The Site Component Test Group performs the following functions and scope of work, as necessary to support the Initial Test Program: RN-13-095

- Plan scope and schedule component testing. RN-15-099
- Review and evaluate component test results. RN-13-095
- Accept construction turnover to the ITP organization.
- Support component testing to preoperational testing turnover.
- Support/perform initial energization and operation of plant equipment as well as functional performance tests at the component or sub-system level.
- Perform component checkout, calibration, and digital controls interface verification for plant components.
- Perform the final cleaning and cleanliness verification of piping and ductwork.

14.2.2.5 Site Preoperational Test Group

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The Site Preoperational Test Group performs the following functions and scope of work, as necessary to support the Initial Test Program: RN-13-095

- Coordinate tagging and maintenance prior to turnover to the Licensee to support system acceptance testing.
- Accept systems for turnover from the Site Component Test Group. RN-13-095

- Plan, scope and schedule plant systems for test to support the plant Preoperational Test Program.

- Manage and oversee the testing of plant systems to support completion of hot functional testing. RN-15-099

- Resolve open items and exceptions identified during implementation of the Preoperational Test Program.

- Turn over preoperational test results to the Licensee, following JTWG review and approval. RN-15-099

- Coordinate other support tasks required during Preoperational Testing activities with responsible groups (e.g., Licensee's Organization). RN-13-095

14.2.2.6 Site Startup Test Group RN-14-110

The Site Startup Test Group performs the following functions and scope of work, as necessary to support the Initial Test Program: RN-13-095 RN-15-099

- Assist tagging and maintenance as required to support startup.

- Assist in planning, scoping, and scheduling plant systems, structures and components for testing, to support startup testing.

- Support initial fuel loading and precritical testing, initial criticality, low power, and power ascension testing. RN-15-099

- Resolve open items and exceptions identified during implementation of the Startup Test Program.

- Coordinate other support tasks required during Startup Testing activities with responsible groups (e.g., Licensee's Organization). RN-13-095

14.2.3 Test Specifications and Test Procedures

Preoperational and startup tests are performed using test procedures, developed primarily from test specifications. RN-114-110

For the preoperational and startup tests, test specifications are written to specify the following:

- Objectives for performing the test
- Test prerequisites
- Initial test conditions
- Data requirements
- Acceptance criteria RN-15-099

For each test, the test procedure specifies the following, as applicable:

- Objectives for performing the test
- Prerequisites that must be completed before the test can be performed
- Initial conditions under which the test is started
- Special precautions required for the safety of personnel or equipment
- Instructions delineating how the test is to be performed
- Identification of the required data to be obtained and the methods for documentation

- Data reduction analysis methods as appropriate
- Test acceptance criteria RN-15-099

Test specifications and procedures are developed and reviewed by personnel with appropriate technical backgrounds and experience. This includes the participation of responsible design organizations in the establishment of test performance requirements and acceptance criteria. Specifically, the responsible design organizations will provide scoping documents (i.e., preoperational and startup test specifications) containing testing objectives and acceptance criteria applicable to its scope of design responsibility. RN-13-095 RN-15-099

Available information on operating or testing experiences of operating reactors is factored into the test specifications and test procedures as appropriate.

Copies of the test specifications and test procedures for the startup tests are available to NRC inspection personnel not less than 60 days prior to the scheduled fuel loading date. RN-14-110

Copies of the test specifications and test procedures are available to NRC inspection personnel approximately 60 days prior to the scheduled performance of the following preoperational tests:

- Tests of systems/components that perform safety-related functions
- Tests of systems/components that are nonsafety-related but perform defense in-depth functions.

Test specifications and test procedures for preoperational tests described in Subsections 14.2.9.3 and 14.2.9.4 of the plant systems/components which perform no safety-related or defense-in-depth functions are available to NRC inspection personnel prior to the scheduled performance of these tests.

Preoperational and startup tests are performed with the quality assurance requirements as specified in Section 17.5.

The ITP Administrative Manual shall include the following controls: RN-14-110

- For component tests, controls to provide test procedures that include appropriate prerequisites, precautions, method to direct and control test performance, and acceptance criteria by which the test is evaluated.
- For preoperational and startup tests, controls to provide test procedures that include appropriate prerequisites, objectives, special precautions, initial test conditions, methods to direct and control test performance, and acceptance criteria by which the test is evaluated. RN-13-095 RN-14-110
- For preoperational and startup tests, controls for the format of individual test procedures to provide consistency with the guidance contained in Regulatory Guide 1.68, as applicable.
- For preoperational and startup tests, controls to provide for participation of the responsible design organizations in establishing test objectives, test acceptance criteria, and related performance requirements during the development of detailed test procedures. Each test procedure should include acceptance criteria that account for the uncertainties used in transient and accident analyses. The responsible design organization will solicit support from design agents (i.e major contractors, subcontractors, and vendors) as needed. RN-15-099

- Controls to provide for personnel with appropriate technical backgrounds and experience to develop and review test procedures. Persons filling designated management positions should perform final procedure review and approval.

- Controls to make the approved preoperational and startup test procedures for satisfying FSAR testing commitments available to NRC inspectors approximately 60 days prior to their intended use and prior to fuel load, respectively. RN-14-110 RN-15-099

14.2.3.1 Conduct of Initial Test Program

Administrative procedures and requirements that govern the activities of the conduct of the initial test program include the following:

- Format and content of test procedures

- Process for both initial issue and subsequent revisions of test procedures

- Review process for test results

- Process for resolution of failures to meet performance criteria and of other operational problems or design deficiencies

- Various phases of the initial test program and the requirements for progressing from one phase to the next, as well as requirements for moving beyond selected hold points or milestones within a given phase

- Controls to monitor the as-tested status of each system and modifications including retest requirements deemed necessary for systems undergoing or already having completed testing

- Qualifications and responsibilities of test personnel within the ITP Organization RN-13-095 RN-15-099

The ITP Administrative Manual (procedures) supplements normal plant administrative procedures by addressing those administrative issues that are unique to the ITP. RN-14-110

The ITP Administrative Manual (procedure) governs the initial testing and is issued no later than 60 days prior to the beginning of the component test phase. Testing during all phases of the test program is conducted using approved test procedures.

14.2.3.1.1 Preoperational and Startup Test Procedure Verification RN-15-099

Since procedures may be approved for implementation weeks or months in advance of the scheduled test date, a review of the approved test procedure is required before commencement of testing. The test engineer is responsible for verifying:

- Drawing and document revision numbers listed in the reference section of the test procedure agree with the latest revisions.

- The procedure text reflects any design and licensing (i.e., FSAR and Technical Specifications) changes made since the procedure was originally approved for implementation.

- Any new (since preparation of the procedure) Operating Experience lessons learned are incorporated into individual test procedures.

Procedures require signoff verification for prerequisites and instruction steps. This signoff includes identification of the person doing the signoff and the date of completion. RN-13-095

Test engineers maintain logs of test status to facilitate turnover and aid in maintaining configuration control. These logs become part of the test documentation. RN-13-095

There is a documented turnover process to make known the test status and equipment configuration when personnel transfer responsibilities, such as during a shift change.

Test briefings are conducted for each test in accordance with administrative procedures. When a shift change occurs before test completion, another briefing occurs before resumption or continuation of the test.

Data collected is marked or identified with test, date, and person collecting data. This data becomes part of the test documentation.

Test issues (collectively including deficiencies, discrepancies, exceptions, non-conformances, delays, etc.) will be documented and addressed in accordance with the requirements stipulated in the ITP Administrative Manual. Those determined by the ITP Administrative Manual criteria and the test results that do not meet test acceptance criteria are entered into the applicable corrective action program. RN-13-095 RN-14-110

The plant manager approves proceeding from one test phase to the next during the ITP. RN-13-095

Administrative procedures detail the test documentation review and approval process.

14.2.3.1.2 Work Control

The group having jurisdictional control is responsible for initiating work requests when assistance is required from the Construction or Licensee's organization. Work requests are issued in accordance with site specific procedures governing the work management process. RN-13-095 RN-15-099

After turnover to the Licensee, activities requiring Construction organization work efforts are performed under plant equipment clearance procedures. Equipment clearance requests are governed by a site-specific procedure for protection of personnel and equipment. RN-13-095 RN-15-099

The group having jurisdictional control is responsible for supervising minor repairs and modifications, changing equipment settings, and disconnecting and reconnecting electrical terminations as stipulated in a test or administrative procedure. Test Engineers may perform independent verification of changes made in accordance with approved procedures. RN-13-095 RN-15-099

14.2.3.1.3 System Turnover

During the construction phase, systems, subsystems, and equipment are completed and turned over in an orderly and well-coordinated manner. Guidelines are established to define the boundary and interface between related system/subsystem and are used to generate boundary scope documents; for example, marked-up piping and instrument diagrams (P&IDs) and electrical schematic diagrams are provided for scheduling and subsequent development of component and system turnover packages. The system turnover process includes requirements for the following:

- Documenting inspections performed by the construction organization (e.g., highlighted drawings showing areas inspected).

- Documenting results of construction testing.

- Determining the construction-related inspections and tests that need to be completed before component testing begins. RN-13-095
- Evaluating any open items for acceptability before commencing component, preoperational, or startup testing. RN-13-095
- Developing and implementing plans for correcting adverse conditions and open items, and means for tracking such conditions and items.
- Verifying completeness of construction and documentation of incomplete items.

14.2.3.1.4 Conduct of Modifications and Configuration Changes During the Initial Test Program RN-15-099

Temporary configuration changes may be required to conduct certain tests. These configuration changes are documented in the test procedures or in accordance with the ITP Administrative Manual. The test procedures contain restoration steps and retesting necessary to confirm satisfactory restoration to the required configuration. Modifications may be performed by the Construction organization or the plant staff processes prior to NRC issuance of the 10 CFR 52.103(g) finding. If the modification invalidates a previously completed ITAAC, then that ITAAC is re-performed. Each modification is reviewed to determine the scope of post-modification testing that is to be performed. Testing is conducted and documented to maintain the validity of preoperational testing and ITAAC. Modifications and configuration changes made following NRC issuance of the 10 CFR 52.103(g) finding are in accordance with plant processes and meet license conditions. Modifications that require changes to ITAAC require NRC approval of the ITAAC change. RN-13-095 RN-14-110 RN-15-099 RN-15-099

14.2.3.1.5 Conduct of Maintenance During the Initial Test Program

Corrective or preventive maintenance activities are reviewed to determine the scope of postmaintenance testing to be performed. Prior to NRC issuance of the 10 CFR 52.103(g) finding, post-maintenance testing is conducted and documented to maintain validity of associated preoperational testing and ITAAC remain valid. Maintenance performed following NRC issuance of the 10 CFR 52.103(g) finding is in accordance with plant staff processes and meets license conditions.

14.2.3.2 Review of Test Results

Final review of the individual tests is discussed in Section 14.4.

14.2.3.2.1 Review and Approval Responsibilities

Upon completion of a test, the test engineer is responsible for: RN-13-095

- Reviewing the test data.
- Evaluating the test results.
- Verifying that the acceptance criteria are met.
- Verifying that the test results that do not meet acceptance criteria are entered into the applicable corrective action program. RN-15-099
- Verifying that the results of retesting do not invalidate ITAAC acceptance criteria.

Preoperational and startup test results are reviewed and approved by the JTWG. Test deficiencies which do not meet acceptance criteria are identified to the affected and responsible design organizations, as appropriate, and entered into the applicable corrective action program. Implementation of corrective actions and retests are performed as required. RN-13-095 RN-15-099

Prior to initial fuel load, the results of the preoperational test phase are comprehensively reviewed by the ITP Organization and the JTWG to verify the results indicate that the required plant structures, systems, and components are capable of supporting the initial fuel load and subsequent startup testing. The plant manager approves fuel loading. RN-13-095

Completed startup tests are reviewed and evaluated by the ITP Organization and the JTWG. The test results at each power ascension testing power plateau are reviewed and evaluated by the ITP Organization and the JTWG and approved by the plant manager before proceeding to the next plateau. Startup test reports are prepared in accordance with the guidance in position C.9 of Regulatory Guide 1.68, “Initial Test Programs for Water-Cooled Nuclear Power Plants.” RN-13-095

14.2.3.2.2 Technical Evaluation

Each completed test is reviewed by technically qualified personnel to confirm satisfactory demonstration of plant, system or component performance and compliance with design and license criteria. RN-15-099

14.2.3.3 Test Records

Retention periods for test records are based on considerations of their usefulness in documenting initial plant performance characteristics, and are retained in accordance with Regulatory Guide 1.28.

14.2.3.3.1 Startup Test Reports

Startup test reports are generated describing and summarizing the completion of tests performed during the ITP. A startup report is submitted at the earliest of:

1) 9 months following initial criticality,

2) 90 days after completion of the ITP, or

3) 90 days after start of commercial operations. If one report does not cover all three events, then supplemental reports are submitted every three months until all three events are completed. These reports:

- Address each ITP test described in the FSAR.
- Provide a general description of measured values of operating conditions or characteristics obtained from the ITP as compared to design or specification values.
- Describe any corrective actions that were required to achieve satisfactory operation.
- Include any other information required by license conditions.

14.2.4 Compliance of Initial Test Program with Regulatory Guides

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Subsection 1.9.1 and Table 1.9-1 discuss compliance with the applicable NRC regulatory guides.

14.2.5 Utilization of Reactor Operating and Testing Experience in the Development of Initial Test Program RN-15-099

The design, testing, startup, and operating experience from previous pressurized water reactor plants is utilized in the development of the preoperational and startup test program for the AP1000 plant. RN-15-099
Other sources of experience reported and described in documents such as NRC reports, including Inspection and Enforcement bulletins and Institute of Nuclear Power Operations (INPO) reports, including Significant Operating Event Reports (SOER), are also utilized in the AP1000 preoperational RN-15-099
and startup test program.

Special tests to further establish a unique phenomenological performance parameter of the AP1000 design features beyond testing performed for Design Certification of the AP600 and that will not change from plant to plant, are performed for the first plant only. Because of the standardization of the AP1000 design, these special tests (designated as first plant only tests) are not required on follow plants. These first plant only tests are identified in the individual test descriptions. (See Subsections 14.2.9 and 14.2.10.) The following is a listing of the first plant only tests, and the corresponding section in which they appear

First Plant Only Test	Section
IRWST Heatup Test	14.2.9.1.3 Item (h)
Pressurizer Surge Line Stratification Evaluation	14.2.9.1.7 Item (d)
Reactor Vessel Internals Vibration Testing	14.2.9.1.9 – Prototype Test
[*Natural Circulation Tests*]*	14.2.10.3.6, [*14.2.10.3.7*]*
Rod Cluster Control Assembly Out of Bank Measurements	14.2.10.4.6
Load Follow Demonstration	14.2.10.4.22

Other special tests which further establish a unique phenomenological performance parameter of the AP1000 design features beyond testing performed for Design Certification for the AP600 and that will not change from plant to plant, are performed for the first three plants. Because of the standardization of the AP1000 design, once these special tests have affirmed consistent passive system function they are not required on follow plants. These tests required on the first three plants are identified in the individual test descriptions (See subsection 14.2.9). The following is a listing of the tests required on the first three plants, and the corresponding section in which they appear.

[*First Three Plant Tests*	*Section*
Core Makeup Tank Heated Recirculation Tests	*14.2.9.1.3 Items (k) and (w)*
ADS Blowdown Test	*14.2.9.1.3 Item (s)*

For subsequent plants, the COL holder shall either perform the subject test, or justification shall be provided that the results of the first-plant-only tests or first-three-plant tests are applicable to the subsequent plant.]*

The justifications for the first-plant-only tests and the first-three-plant tests are provided below:

IRWST Heatup Test (14.2.9.1.3 item (h))

During preoperational testing of the passive core cooling system, a natural circulation test of the passive residual heat removal (PRHR) heat exchanger is conducted (item f). For the first plant only, thermocouples are placed in the IRWST to observe the thermal profile developed during the heatup of the IRWST water during PRHR heat exchanger operation. This test will be useful in confirming the results of the AP600 Design Certification Program PRHR tests with regards to IRWST mixing, and is useful in quantifying the conservatism in the Chapter 15 transient analyses.

Due to the standardization of the AP1000, the heatup and thermal stratification characteristics of the IRWST will not vary from plant to plant. The PRHR heat exchanger design, and the size and configuration of the IRWST are standardized, such that the heatup characteristics will not significantly change from plant to plant.

Therefore, since the phenomenon to be tested (i.e., heatup and mixing characteristics of the IRWST) will not vary significantly from plant to plant due to standardization, a first plant only test of the IRWST heatup characteristics is justified.

Core Makeup Tank Heated Recirculation Tests (14.2.9.1.3 Items (k) and (w))

During preoperational testing of the passive core cooling system, a test is performed for each plant to verify the CMT inlet piping resistances. In addition, cold draining tests of the CMTs are conducted that verify the discharge piping resistance and proper drain rate of the CMTs for each plant. For the first three plants, two additional CMT tests are conducted during hot functional testing of the RCS. These tests are a natural circulation heatup of the CMTs followed by a test to verify the ability of the CMTs to transition from a recirculation mode to a draindown mode while at elevated temperature and pressure.

Operation of the CMTs in their natural circulation mode is conducted on the first three plants only for the following reasons:

- Natural circulation of the CMTs will not vary from plant to plant, provided that the other verifications discussed above are performed as specified.

- Natural circulation testing of the CMTs was extensively tested as part of the Design Certification Tests.

- Performance of this test results in significant thermal transients on Class 1 components including the CMTs and the direct vessel injection nozzles.

ADS Blowdown Test (14.2.9.1.3 Item (s))

During preoperational testing of the passive core cooling system, the resistance of the automatic depressurization system Stage 1, 2, 3 flow path(s) is verified. For the first three plants only, an automatic depressurization blowdown test is performed to verify proper operation of the ADS valves, and demonstrate the proper operation of the ADS spargers to limit the hydrodynamic loads in containment to less than design limits. This test is performed on only the first three plants for the following reasons:

- The operation of the ADS, and the resultant hydrodynamic loads will not vary significantly from plant to plant.

- Full scale automatic depressurization testing was performed in the AP600 Design Certification Program. Testing was conducted to conservatively bound ADS flow rates and resultant hydrodynamic loads that will be experienced by the plant during ADS operation.

- Performance of this test results in significant thermal transients on Class 1 components including the primary components. It also results in hydrodynamic loads in containment including the IRWST.

*NRC Staff approval is required prior to implementing a change in this information.

Pressurizer Surge Line Stratification Evaluation (14.2.9.1.7 Item (d))

As part of the AP1000 conformance to NRC Bulletin 88-11, a monitoring program will be implemented by the COL Applicant for the first AP1000 to record temperature distributions and thermal displacements of the surge line piping during hot functional testing and during the first fuel cycle, as discussed in subsection 3.9.3.

Reactor Vessel Internals Vibration Testing (14.2.9.1.9)

The preoperational vibration test program for the reactor internals of the AP1000 conducted on the first AP1000 is consistent with the guidelines of Regulatory Guide 1.20 for a comprehensive vibration assessment program. This program is discussed in subsection 3.9.2.

Natural Circulation Tests (14.2.10.3.6, 14.2.10.3.7)

Natural circulation tests using the steam generators and the passive residual heat removal heat exchanger are performed at low core power during the startup test phase of the initial test program for the first AP1000. This testing of the heat removal systems meets the intent of the requirement to perform natural circulation testing and the results of this testing is factored into the operator training as discussed in subsection 1.9.4, Item I.G.1. This test is only required to be performed once because its purpose is to obtain data to benchmark the operator training simulator.

Rod Cluster Control Assembly Out of Bank Measurements (14.2.10.4.6)

Rod cluster control assembly out of bank measurements are performed during power ascension tests. The test is performed at the 30-percent to 50-percent power level so the plant does not exceed peaking factor limits. The test is required to be performed only for the first plant because its purpose is to validate calculation tools and instrument responses.

Load Follow Demonstration (14.2.10.4.22)

A load follow demonstration test is not required by Regulatory Guide 1.68. However, the AP1000 performs load follow with grey rods, as opposed to current Westinghouse PWRs which manipulate RCS boron concentration to perform load follow operations. Therefore, Westinghouse has included a load follow test for the first AP1000, to demonstrate the ability of the AP1000 plant to load follow.

Utilization of Operating Experience

Administrative procedures provide methodologies for evaluating and initiating action for operating experience information (OE). This subsection describes the general use of operating experience by WEC in the development of the test program.

14.2.5.1 Use of OE During Test Procedure Preparation

Administrative procedures require review of recent internal and external operating experience when preparing test procedures.

14.2.5.2 Sources and Types of Information Reviewed for ITP Development

Multiple sources of operating experience were reviewed to develop this description of the ITP administration program. These included INPO Reports, INPO Guidelines, INPO Significant Event Reports, INPO Significant Operating Experience Reports and NRC Regulatory Guide 1.68.

14.2.5.3 Conclusions from Review

The following conclusions are a result of the OE review conducted to develop this ITP administration program description:

- The test procedures should provide guidance as to the expected plant response and instructions concerning what conditions warrant aborting the test. Errors and problems with the procedures should be anticipated. A means for prompt but controlled approval of changes to test procedures is needed. Critical test procedures should provide specific criteria for test termination and specific steps to conduct termination are conducted in a safe and orderly manner. Providing procedural guidance for aborting the test could prevent delays in plant restoration. Conservative guidance for actions to be taken should be included in the procedures. RN-13-095

- Plant simulators may prove useful in preparing for special tests and verifying procedures.

- Appropriate component/system operability should be verified prior to critical tests.

- The need to perform physics tests that can produce severe power tilts should be evaluated, particularly if tests at other similar reactors have provided sufficient data to verify the adequacy of the nuclear physics analysis.

- Compensatory measures should be implemented in accordance with guidance for infrequently performed tests or evolutions, where appropriate.

14.2.5.4 Summary of Test Program Features Influenced by the Review

The conclusions from the preceding section were incorporated in Section 14.2.

14.2.5.5 Use of OE during Conduct of ITP

Administrative procedures require discussion of operating experience when performing pre-job briefs immediately prior to the conduct of a test.

14.2.6 Use of Plant Operating and Emergency Procedures

As appropriate and to the extent practicable, plant normal, abnormal, and emergency operating procedures are used when performing preoperational tests. RN-15-099

The use of these procedures is intended:

- To demonstrate the adequacy of the specific procedure or to identify changes that may be required

- To increase the level of knowledge of plant personnel on the systems being tested

A test procedure using a normal, abnormal, or emergency operating procedure references the procedure directly or extracts a series of steps from the procedure in the way that accomplishes the operator training goals while safely and efficiently performing the specified testing.

These procedures are used extensively in the Human-Machine Interface Testing, which is integrated as a part of the Control Room Design finalization. Additionally, the AP1000 plant operating and emergency procedures are developed to support the following design finalization activities:

- Human Factors Engineering

- Operational Task Analysis

- Training Simulator Development

- Verification and Validation of the Procedures and the Training Material

The AP1000 emergency, abnormal and some normal operating procedures, along with some Alarm Response Procedures and surveillance procedures, are exercised and verified in the processes delineated above and in the Control Room design finalization process. RN-13-095

14.2.6.1 Operator Training and Participation during Certain Initial Tests (TMI Action Plan Item I.G.1, NUREG-0737)

The objective of operator participation is to increase the capability of shift crews to operate facilities in a safe and competent manner by assuring that training for plant changes and offnormal events is conducted.

Operators are trained on the specifics of the ITP schedule, administrative requirements and tests. Specific Just In Time training is conducted for selected startup tests.

The ITP may result in the discovery of an acceptable plant or system response that differs from the expected response. Test results are reviewed to identify these differences and the training for operators is changed to reflect them. Training is conducted as soon as is practicable in accordance with training procedures.

14.2.7 Initial Fuel Loading and Initial Criticality

Initial fuel loading and subsequent initial criticality and power ascension to full licensed power are performed during the startup test program. Prior to the initiation of these operations, the systems and conditions necessary to bring the plant into compliance with the Technical Specifications must be operable and satisfied. These operations are performed in a controlled and safe manner by using test procedures that specify:

- Required prerequisite testing
- Operational status of required systems
- Step-by-step instructions
- Precautions which must be observed
- Actions to be taken in the event of unanticipated or abnormal response

14.2.7.1 Initial Fuel Loading

The minimum conditions for initial core loading include:

- The composition, duties, and emergency procedure responsibilities of the fuel handling crew are established.

- Radiation monitors, nuclear instrumentation, manual initiation controls, and other devices to actuate alarms and ventilation controls are tested and verified to be operable.

- The status of systems required for fuel loading is established and verified.

- The status of protection systems, interlocks, alarms, and radiation protection equipment is established and verified for fuel loading.

- Inspections of fuel and control rods have been made.

- Containment integrity has been established to the extent required by the Technical Specifications.

- The reactor vessel status has been established for fuel loading. Components are verified to be in place or out of the vessel as required for fuel loading.

- Required fuel handling tools are available, operational, and calibrated to include indexing of the manipulator crane with a dummy fuel element. The fuel handling tools have been successfully tested.

- Reactor coolant water quality requirements are established and the reactor coolant water quality is verified.

- The reactor vessel is filled with water to a level approximately equal to the center of the vessel outlet nozzles. The reactor coolant water is circulating at a rate which provides uniform mixing.

- The boron concentration in the reactor coolant is verified to be equal to or greater than required by the plant Technical Specifications for refueling and is being maintained under a surveillance program.

- Sources of unborated water to the reactor coolant system have been isolated and are under a surveillance program.

- At least two neutron detectors are calibrated, operable, and located in such a way that changes in core reactivity can be detected and recorded. One detector is connected to an audible count rate indicator and a containment alarm.

- A response check of nuclear instruments to a neutron source is required within 8 hours prior to loading (or resumption of loading if delayed for 8 hours or more).

Fuel assemblies together with inserted components (control rods, burnable poison assemblies, primary and secondary neutron sources) are placed in the reactor vessel, according to an established and approved sequence.

During and following the insertion of each fuel assembly, until the last fuel assembly has been loaded, the response of the neutron detectors is observed and compared with previous fuel loading data or calculations to verify that the observed changes in core reactivity are as expected. Specific instructions are provided if unexpected changes in reactivity are observed.

Because of the unique conditions that exist during initial fuel loading, temporary neutron detectors may be used in the reactor vessel to provide additional reactivity monitoring. Credit for the use of temporary detectors may be taken in meeting Technical Specifications requirements on the number of operable source range channels.

14.2.7.2 Initial Criticality

Following initial fuel loading, the reactor upper internals and the pressure vessel head are installed. Additional mechanical and electrical tests are performed in preparation for critical and power operations. The following conditions exist prior to initial criticality:

- The reactor coolant system is filled and vented.

- Tests are completed on the control rod drive system that demonstrate that the control rods have been latched, that the control and position indication systems are functioning properly, and that the rod drop time under hot full flow conditions is less than the Technical Specifications limit.

- Tests are completed that demonstrate that plant control and protection systems are operable and that the reactor trip breakers respond as designed to appropriate trip signals.

- The reactor coolant system is at hot no-load temperature and pressure. The reactor coolant boron concentration is such that the shutdown margin requirements of the Technical Specifications are satisfied for the safe shutdown condition.

Initial criticality is achieved in an orderly, controlled fashion by the combination of shutdown and control bank withdrawal and reactor coolant system boron concentration reduction.

During the approach to initial criticality, the response of the source range nuclear instruments is used as an indication of the rate of reactivity addition and the proximity to a critical condition so that criticality is achieved in a controlled, predictable fashion.

Rates for rod withdrawal and boron reduction are specified in such a way that the startup rate is less than one decade per minute.

Following criticality and prior to operation at power levels greater than 5 percent of rated power, physics tests are performed to verify that the operating characteristics of the reactor core are consistent with design predictions. During these tests, values are obtained for the reactivity worth of control and shutdown rod banks, isothermal temperature coefficient, and critical boron concentration for selected rod bank configurations.

Other tests at low power include verification of the response of the nuclear instrumentation system and radiation surveys.

14.2.7.3 Power Ascension

After the operating characteristics of the reactor have been verified by low-power testing, a power ascension program brings the unit to its full rated power level in successive stages. At each successive stage, hold points are provided to evaluate and approve test results prior to proceeding to the next stage. The minimum test requirements for each successive stage of power ascension are specified in the applicable startup test procedures.

During the power ascension program, tests are performed at various power levels as follows:

- Statepoint data, including secondary system heat balance measurements, are obtained at various power levels up to full licensed power. This information is used to project plant performance during power escalation, provide calibration data for the various plant control and protection systems, and provide the bases for plant trip setpoints.

- At prescribed power levels, the dynamic response characteristics of the primary and secondary systems are evaluated. System response characteristics are measured for design step load changes, rapid load reductions, and plant trips.

- Adequacy of the radiation shielding is verified by gamma and neutron radiation surveys. Periodic sampling is performed to verify the chemical and radiochemical analysis of the reactor coolant.

- Using the incore instrumentation as appropriate, the power distribution of the reactor core is measured to verify consistency with design predictions and Technical Specifications limits on peaking factors.

14.2.8 Initial Test Program Schedule RN-15-099

The schedule for the initial fuel load and for each major phase of the initial test program includes the timetable for generation, review, and approval of procedures as well as the actual testing and analysis of results.

Preoperational testing is performed as system and equipment availability allows. The interdependence of systems is also considered.

Sequencing of the startup tests depends on specified power and flow conditions and intersystem prerequisites. The startup test schedule establishes that, prior to core load, the test requirements are met for those plant structures, systems, and components that are relied upon to prevent, limit, or mitigate the consequences of postulated accidents. Testing is sequenced so that the safety of the plant is not dependent on untested systems, components, or features.

A site-specific initial test program schedule will be provided to the NRC after issuance of the COL. This schedule will address each major phase of the test program (including tests that are required to be completed before fuel load), as well as the organizational impact of any overlap of first unit initial testing with initial testing of the second unit.

The sequential schedule for individual startup tests should establish that testing is completed in accordance with plant technical specification requirements for structures, systems and components (SSC) operability before changing plant modes. Additionally, the schedule establishes that the safety of the plant is not dependent on the performance of untested SSCs. Guidance provided in Regulatory Guide 1.68 is used for development of the schedule.

The ITP Administrative Manual shall include the following controls: RN-14-110

- Test Procedure Development Schedule:
  - Controls to establish a schedule for the development of detailed testing procedures. These procedures, to the extent practical, are trial-tested and corrected during the initial test program prior to fuel loading in order to establish their adequacy. RN-13-095
  - Controls to confirm that approved test procedures are in a form suitable for review by NRC inspectors at least 60 days prior to their intended use, or at least 60 days prior to fuel loading for fuel loading and startup test procedures.
  - Controls to provide timely notification to the NRC of changes in approved preoperational and startup test procedures previously available for NRC review. RN-14-110
- Initial Test Program Schedule:
  - Controls to establish a schedule to conduct the major phases of the initial test program, relative to the expected fuel loading date. This is covered in License Conditions in Part 10 of the COL Application.
  - Controls to allow at least 9 months for conducting preoperational testing.
  - Controls to allow at least 3 months for conducting startup testing, including fuel loading, low-power tests, and power-ascension tests.

- Controls to overlap test program schedules (for multi-unit sites) such that they do not result in significant divisions of responsibilities or dilutions of the staff provided to implement the test program.

- Controls to sequence the schedule for individual startup tests, insofar as is practicable, such that testing is completed prior to exceeding 25 percent power for the plant SSCs that are relied upon to prevent, limit, or mitigate the consequences of postulated accidents. The schedule should establish that, insofar as is practicable, testing is accomplished as early in the test program as is feasible and that the safety of the plant is not dependent on the performance of untested SSCs.

The milestone schedule for developing plant operating procedures is presented in Table 13.4-201. The operating and emergency procedures are available prior to start of licensed operator training and, therefore, are available for use during the ITP. Required or desired procedure changes may be identified during their use. Administrative procedures describe the process for revising plant operating procedures.

14.2.9 Preoperational Test Descriptions

During preoperational testing, it may be necessary to return system control to Construction organization to repair or modify the system or to correct new problems. Administrative procedures include direction for:

- Means of releasing control of systems and or components to construction.
- Methods used for documenting actual work performed and determining impact on testing.
- Identification of required testing to restore the system to operability/functionality/availability status, and to identify tests to be re-performed based on the impact of the work performed. RN-13-095
- Verifying retests stay in compliance with ITAAC.

Test abstracts are provided for the preoperational testing of systems/components that perform safety-related functions; that are nonsafety-related but perform functions designated to provide defense in-depth; systems/components that may contain radioactive material; and other applicable nonsafety-related systems in accordance with Regulatory Guide 1.68, Revision 2, Appendix A. A limited number of these testing abstracts establish performance parameters of AP1000 design features that will not change from plant to plant. Because the AP1000 design is standardized, these tests need only be performed on the first AP1000 plant. These testing abstracts are clearly identified.

14.2.9.1 Preoperational Tests of Systems with Safety-Related Functions

14.2.9.1.1 Reactor Coolant System Testing

Purpose

The purpose of the reactor coolant system testing is to verify that the as-installed reactor coolant system properly performs the following safety-related functions:

- Provide reactor coolant system pressure boundary integrity as described in Section 5.2
- Provide core cooling and boration in conjunction with the passive core cooling system as described in Sections 5.1 and 6.3

- Measure process parameters required for safety-related actuations and safe shutdown as described in Sections 7.2, 7.3 and 7.4

- Measure selected process parameters required for post-accident monitoring as described in Section 7.5

- Vent the reactor vessel head as discussed in subsection 5.4.12

Testing is also performed to verify that the system properly performs the following defense-in-depth functions described in Section 5.2:

- Provide forced circulation cooling of the reactor core in conjunction with heat removal by the steam generator(s) as described in Section 5.1

- Provide core cooling by natural circulation of coolant in conjunction with heat removal by the steam generator(s) as described in Section 5.1

- In conjunction with the steam generator(s) and normal residual heat removal system, provide the capability to remove core decay heat and cool the reactor coolant to permit the reactor to be refueled and started up in a controlled manner

- Provide pressurizer pressure control during normal operation

- Provide pressurizer level control in conjunction with the chemical and volume control system

- Provide pressurizer spray

Prerequisites

The component testing of the reactor coolant system has been successfully completed. The pre-operational testing of the component cooling water system, service water system, chemical and volume control system, main ac power electrical power system, and required interfacing systems is completed to the extent sufficient to support the specified testing. The reactor coolant system is filled, vented, and pressurized above the minimum required pressure for reactor coolant pump operation, and component cooling water flow to the reactor coolant pumps is initiated prior to starting the pumps. RN-14-110

In preparation for the hydrostatic test of the reactor coolant system, the reactor vessel lower and upper internals and the closure head are installed. The closure head studs are properly tensioned for the hydrostatic test pressure. The pressurizer safety valves and instrumentation within the test boundary are either removed, recalibrated or verified to be able to withstand the hydrostatic test pressure. Welds within the test boundaries are verified as ready for hydrostatic testing. A hydrostatic test pump is available for the pressure boundary integrity testing.

General Test Method and Acceptance Criteria

Reactor coolant system performance is observed and recorded during a series of individual component and system tests. The following testing demonstrates that the reactor coolant system can perform the functions described above and in appropriate design specifications:

a) The integrity and leaktightness of the reactor coolant system and the high-pressure portions of associated systems is verified by performing a cold hydrostatic pressure test in conformance with Section III of the American Society of Mechanical Engineers (ASME) Code. The reactor coolant system is pressurized in stages by operation of the temporary hydrostatic test pump, while monitoring system welds, piping, and components for leaks at each stage. The hydrostatic test verifies that there are no leaks at welds or piping within the test boundaries during the final inspection. Any identified pressure boundary leaks (i.e. piping walls, vessel walls, welds, valve bodies, etc.) are repaired and the hydrostatic test repeated.

Leakage through valve seats, valve packing, flanges, and threaded or mechanical fittings is acceptable during the hydrostatic test as long as the hydrostatic test pump can maintain the proper test pressure. Leakage through these items may, as necessary and practical, be isolated, repaired, and retested at a later date.

b) Proper operation of the safety-related reactor coolant system and reactor coolant pressure boundary valves is verified by the performance of baseline in-service tests as described in subsection 3.9.6.

c) The operability of the pressurizer safety valves is demonstrated by a bench test at temperature and pressure with steam as the pressurizing fluid or with a suitable in-situ test. This testing verifies that each pressurizer safety valve actuates at the required set pressure, with appropriate tolerance as specified in the Technical Specifications. The safety valve rated capacity, as recorded on the valve vendor code plates, is verified to be greater than or equal to that described in Section 5.4.

d) During hot functional testing, reactor coolant system leakage is verified to be within the limits specified in the Technical Specifications. Proper calibration and operation of instrumentation controls, actuations, and interlocks related to reactor coolant system leak detection are verified. The pressurizer water level is set to the no-load level, the chemical and volume control system makeup pumps and letdown line do not operate, and no primary system samples are taken. During this test, the identified and unidentified reactor coolant system leakage rates are determined by monitoring the reactor coolant system water inventory, reactor coolant drain tank level, containment sump level, and other leak detection instrumentation as described in subsection 5.2.5 over a specified period of time.

e) The leakage across individual valves between high pressure and low pressure systems, as specified in the Technical Specifications, is verified to be less than design requirements.

f) The as-installed safety valve discharge chamber rupture disks are inspected to verify the manufacturer’s stamped set pressure is within the limits specified in the appropriate design specifications.

g) Proper calibration and operation of safety-related instrumentation, controls, actuation signals and interlocks are verified. This testing includes the following:

- Hot leg and cold leg resistance temperature detectors
- Flow instrumentation at selected locations in the reactor coolant loop
- Reactor coolant system wide range pressure transmitters
- Hot leg level instruments
- Pressurizer pressure and level instruments
- Reactor coolant pump bearing water temperature detectors
- Reactor coolant pump speed sensor instruments
- Reactor vessel head vent valve controls

This testing includes demonstration of proper actuation of safety-related functions from the main control room.

h) Automatic trip of the reactor coolant pumps following appropriate safety-related actuation signals is demonstrated.

i) Proper operation of the reactor vessel head vent valves is verified with the reactor coolant system pressurized.

The following testing demonstrates that the system properly performs the defense-in-depth functions described above and in appropriate design specifications:

j) The pressurizer spray valves are verified to operate properly over the range of reactor coolant system operating temperatures and with the reactor coolant pumps operating.

k) Proper calibration and operation of defense-in-depth related instrumentation, controls, actuation signals and interlocks are verified. This testing includes actuation of the pressurizer spray valves on receipt of appropriate signals, as well as actuation from the main control room.

l) Reactor coolant pump and motor performance and operating characteristics are initially verified with the reactor coolant system at cold conditions. This testing includes verification of the proper flow through the reactor coolant system when all four reactor coolant pumps are operated in various combinations and speeds as specified in the appropriate design specifications and operating procedures. In addition, the proper operation of the pump motor instrumentation, alarms, and interlocks is verified including:

- Motor current
- Motor power
- Pump vibration
- Motor Stator temperature
- Proper transfer from variable speed startup operation

m) The reactor coolant system is heated from cold conditions to hot standby conditions by operating the reactor coolant pumps and the pressurizer heaters. The reactor coolant system is operated at full flow conditions for at least 240 hours prior to core loading. The reactor coolant temperature is maintained at or above 515°F for at least one-half of this operating time. In addition to facilitating the reactor coolant system tests that are required to be performed hot and pressurized, these hot functional testing conditions allow the plant operators to control the plant using the plant operating procedures for the reactor coolant system, secondary side systems, and auxiliary systems.

Other preoperational tests that require these hot and/or dynamic conditions are conducted during this hot functional testing period.

n) During hot functional testing, the reactor coolant pump and motor operating characteristics are measured and recorded at various temperature plateaus during reactor coolant system heatup to verify proper operation over their operating temperature range. This testing includes verification of the proper pump flow; proper motor current, power, and stator temperature; and pump vibration level.

o) The pressurizer spray continuous flow rate is established, and the proper spray line temperature is verified for each pressurizer spray line.

p) The proper operation of the pressurizer heaters, pressurizer spray, and pressure control functions and alarms is verified during the heatup, operation at hot functional test conditions, and cooldown of the reactor coolant system.

q) The proper operation of the pressurizer level control functions and alarms is verified during the heatup, operation at hot functional test conditions, and cooldown of the reactor coolant system.

r) The pressure drops across the major components of the reactor coolant system are measured and recorded using temporary instrumentation during flow testing, and verified to be in accordance with appropriate design specifications.

Tests associated with the automatic depressurization functions of reactor coolant system components are described in subsection 14.2.9.1.3.

14.2.9.1.2 Steam Generator System Testing

Purpose

The purpose of the steam generator system testing is to verify that the as-installed components properly perform the following safety-related functions as described in Sections 5.4, 10.3 and 10.4:

- Provide steam generator isolation, including isolation of the main steam lines, feedwater lines, and blowdown lines
- Remove heat from the reactor coolant system and provide secondary side overpressure protection
- Measure process parameters required for safety-related actuations as described in Sections 7.2, 7.3, and 7.4
- Measure process parameters required for post-accident monitoring as described in Section 7.5

This testing also verifies that the as-installed components properly perform the following defense-in-depth functions as described in Section 10.4:

- Provide heat removal from the reactor coolant system
- Provide overpressure protection for the steam generators to minimize required actuations of the spring-loaded safety valves
- Measure process parameters and provide actuation signals for the diverse actuation system

Prerequisites

The component tests of the as-installed system have been completed. The reactor coolant system as well as other systems used in power generation are functional since portions of the steam generator system testing is performed during the plant hot functional tests. Prerequisite testing of required interfacing systems are completed to the extent sufficient to support the specified testing and the appropriate system configuration. Component testing of the special monitoring system has been completed to the extent necessary to support preoperational testing. Required electrical power supplies are energized and operational. RN-14-110 RN-14-110

General Test Method and Acceptance Criteria

The performance of the steam generator system is observed and recorded during a series of individual component and integrated system testing that characterizes its modes of operation. The following testing demonstrates that the steam generator system operates as specified in Sections 10.3 and 10.4, and appropriate design specifications:

a) Proper operation of the steam generator system safety-related valves is verified by the performance of baseline in-service tests as described in subsection 3.9.6. In addition, the ability of these valves to perform their safety related functions is verified during hot functional testing with the steam generators at normal operating pressure and temperature. The following valves are tested:

- Steam line condensate drain control and isolation valves
- Main steam line isolation valves
- Main and startup feedwater isolation valves
- Steam generator blowdown isolation valves
- Steam generator power-operated relief valves
- Main steam isolation valve bypass isolation valves
- Main and startup feedwater control valves

This testing includes verification of the capability of the steam generator power operated relief valves to provide the required heat removal rate from steam generators/reactor coolant system.

b) Proper operation of safety-related and defense-in-depth instrumentation, controls, actuation signals, and interlocks is verified. This testing includes actuation of equipment from the main control room.

c) The proper operation of the steam generator safety valves is demonstrated in a bench test at temperature and pressure with steam as the pressurizing fluid or with suitable in-situ testing. The safety valve rated capacity recorded on the valve vendor code plates is verified to be greater than or equal to the required relief capacity.

Heat transfer performance of the steam generator system is verified by startup testing of the reactor coolant system described in other sections.

14.2.9.1.3 Passive Core Cooling System Testing

Purpose

The purpose of the passive core cooling system testing is to verify that the as-installed components and their associated piping and valves properly perform the following safety functions, described in Section 6.3:

- Emergency core decay heat removal
- Reactor coolant system emergency makeup and boration
- Safety injection
- Containment pH control

Prerequisites

The component testing of the passive core cooling system, or of a specific portion of the system to be tested, is successfully completed. The preoperational testing of the reactor coolant system, normal residual heat removal system, chemical and volume control system, the refueling cavity, the Class 1E dc and uninterruptible power supply, the ac electrical power and distribution systems, and other interfacing systems required for operation of the above systems is completed as needed to support the specified testing and system configurations. A source of water, of a quality acceptable for filling the passive core cooling system components and the reactor coolant system, is available. RN-14-110

General Test Method and Acceptance Criteria

The performance of the passive core cooling system is observed and recorded during a series of individual component testing and testing with the reactor coolant system. The following testing demonstrates that the passive core cooling system operates as described in Section 6.3 and appropriate design specifications.

a) Proper operation of safety-related valves is verified by the performance of baseline in-service tests as described in subsection 3.9.6. Also, the proper operation of non-safety-related valves is verified including manual valve locking devices. This testing does not include actuation of the squib valves, which is discussed in Item t, below.

b) Proper calibration and operation of safety-related instrumentation, controls, actuation signals, and safety related interlocks as specified in Section 7.6, is verified. This testing includes the following:

   - Passive residual heat removal heat exchanger flow
   - Core makeup tank level
   - In-containment refueling water storage tank level
   - Containment floodup level
   - Core makeup tank inlet/outlet valve controls
   - Passive residual heat removal heat exchanger inlet/outlet valve controls
   - In-containment refueling water storage tank outlet valve controls
   - Containment recirculation valve controls
   - Automatic depressurization valve controls
   - In-containment refueling water storage tank gutter isolation valve controls

   This testing includes demonstration of proper actuation of safety-related functions from the main control room.

c) Proper calibration and operation of instrumentation, controls, and interlocks required to demonstrate readiness of a safety-related component is verified. This testing includes the following:

   - Accumulator pressure and level and alarms
   - Passive residual heat removal heat exchanger temperatures
   - Passive residual heat removal heat exchanger high point vent level
   - Core makeup tank inlet line temperatures
   - Core makeup tank inlet line high point levels
   - Direct vessel injection line temperatures
   - In-containment refueling water storage tank level and temperatures

d) Proper calibration and operation of temporary instrumentation and data recording devices used in this testing is verified. This testing includes the following:

   - CMT level
   - CMT flow and balance line temperatures
   - PRHR supply line temperatures
   - Accumulator wide range level
   - In-containment refueling water storage tank and sump-recirculation flow
   - ADS piping differential pressure

The passive core cooling system emergency core decay heat removal function is verified by the following testing of the passive residual heat removal heat exchanger.

e) During hot functional testing of the reactor coolant system, the heat exchanger supply and return line piping water temperatures are recorded to verify that natural circulation flow initiates.

f) The heat transfer capability of the passive residual heat removal heat exchanger is verified by measuring natural circulation flow rate and the heat exchanger inlet and outlet temperatures while the reactor coolant system is cooled to ≤ 420°F. This testing is performed during hot functional testing with the reactor coolant system initial temperature ≥ 540°F and the reactor coolant pumps not running. The acceptance criteria for the PRHR HX heat transfer under natural circulation conditions are that the heat transfer rate is ≥ 1.78 E+08 Btu/hr based on a 520°F hot leg temperature and ≥ 1.11 E+08 Btu/hr based on 420°F hot leg temperature with 80°F IRWST temperature and the design number of tubes plugged. These plant conditions are selected to be close to the expected test conditions and are different than those listed in Table 6.3-2. The PRHR HX heat transfer rate has been adjusted to account for these different conditions. The heat transfer rate measured in the test should be adjusted to account for differences in the hot leg and IRWST temperatures and number of tubes plugged.

g) The proper operation of the passive residual heat removal heat exchanger and its heat transfer capability with forced flow is verified by initiating and operating the heat exchanger with all four reactor coolant pumps running. This testing is performed during hot functional testing with the reactor coolant system at an elevated initial temperature ≥ 350°F. The heat exchanger heat transfer is determined by measuring the heat exchanger flow rate and its inlet and outlet temperatures while the reactor coolant system is cooled to ≤ 250°F. The acceptance criteria for the PRHR HX heat transfer under forced circulation conditions are listed in Table 3.9-17. The heat transfer rate measured in the test should be adjusted to account for differences in the hot leg and IRWST temperatures and number of tubes plugged.

h) The heatup characteristics of the in-containment refueling water storage tank water are verified by measuring the vertical water temperature gradient that occurs in the in-containment refueling water storage tank water at the passive residual heat removal heat exchanger tube bundle and at several distances from the tube bundle, during testing in Item e), above. **Note that this verification is required only for the first plant.** The acceptance criterion for the IRWST heatup characteristics is that they support meeting the RCS safe shutdown temperature criteria (refer to subsection 19E.4.10.2).

The passive core cooling system emergency makeup and boration function is verified by the following testing of the core makeup tanks.

i) The resistance of the core makeup tank cold leg balance lines is determined by filling the core makeup tanks with flow from the cold legs. This testing is performed by filling the cold, depressurized reactor coolant system using a constant, measured discharge flow from the normal residual heat removal pumps. The reactor coolant system is maintained at a constant level above the top of the cold leg balance line(s). The normal residual heat removal system flow rate and the differential pressure across the cold leg balance lines are used to determine the resistance of the balance lines. The acceptance criterion for the resistance of these lines is $\leq 7.21 \times 10^{-6}$ ft/gpm^2.

j) During hot functional testing of the reactor coolant system, the core makeup tank cold leg balance line piping water temperature at various locations is recorded to verify that the water in this line is sufficiently heated to initiate recirculation flow through the CMTs.

k) [*Proper operation of the core makeup tanks to perform their reactor water makeup and boration function is verified by initiating recirculation flow through the tanks during hot functional testing with the reactor coolant system at≥ 530°F. This testing is initiated by simulating a safety signal which opens the tank discharge isolation valves, and stops reactor coolant pumps after the appropriate time delay. The proper tank recirculation flow after the* *pumps have coasted down is verified. Based on the cold leg temperature, CMT discharge temperature, and temporary CMT flow instrumentation, the net mass injection rate into the reactor is verified.* ***Note that this verification is required only for the first three plants.***]*

The passive core cooling system safety injection function is verified by the following testing of the core makeup tanks, accumulators, in-containment refueling water storage tank, containment sump, automatic depressurization, and their associated piping and valves.

l) Proper flow resistance of each of the core makeup tank injection lines is verified by gravity draining each tank filled with cold water through the direct vessel injection flow path, while measuring the CMT level (driving head) and discharge flow rate. Air enters the top of the draining tank from the reactor coolant system cold leg via the cold leg balance line. If necessary, the flow limiting orifice in the core makeup tank discharge line is to be resized, and the core makeup tank retested to obtain the required line resistance. The acceptance criteria for the resistance of these lines are $\leq 2.25 \times 10^{-5}$ ft/gpm^2 and $\geq 1.81 \times 10^{-5}$ ft/gpm^2 with all valves open.

m) The proper flow resistance of each of the accumulator injection lines is verified by performing a blowdown from a partially pressurized accumulator through the direct vessel injection flow path, while measuring the change in accumulator level and pressure. If necessary, the flow orifice in the accumulator discharge line is to be resized and the accumulator retested to obtain the required discharge line resistance. The acceptance criteria for the resistance of these lines are $\leq 1.83 \times 10^{-5}$ ft/gpm^2 and $\geq 1.47 \times 10^{-5}$ ft/gpm^2.

n) The proper flow resistance of each of the in-containment refueling water storage tank injection lines is verified by gravity draining water from the tank through the direct vessel injection flow path, while measuring the water level (driving head) and discharge flow rate using temporary instrumentation. A test fixture with prototypical resistance may be used to simulate the squib valves in the flow paths tested. The acceptance criteria for the resistance of these lines are $\leq 9.20 \times 10^{-6}$ ft/gpm^2 and $\geq 5.53 \times 10^{-6}$ ft/gpm^2 for line A and $\leq 1.03 \times 10^{-5}$ ft/gpm^2 and $\geq 6.21 \times 10^{-6}$ ft/gpm^2 for line B with all valves open.

o) The flow resistance of each of the flow paths from the in-containment refueling water storage tank to each containment sump, and from each containment sump to the reactor is verified by a series of tests. These tests gravity drain water from the in-containment refueling water storage tank to the containment sump, and from the sump through the direct vessel injection flow path, while measuring the storage tank water level (driving head) and injection flow rate using temporary instrumentation. This testing is performed using temporary piping to prevent flooding of the containment. A test fixture with prototypical resistance may be used to simulate the squib valves in the flow paths tested. The acceptance criteria for the resistance of the lines between each containment sump and the reactor are $\leq 1.11 \times 10^{-5}$ ft/gpm^2 for line A and $\leq 1.03 \times 10^{-5}$ ft/gpm^2 for line B with all valves open. The acceptance criterion for the resistance of the lines between the IRWST and each containment sump is $\leq 4.07 \times 10^{-6}$ ft/gpm^2.

p) The resistance of each automatic depressurization stage 1, 2, and 3 flow path and flow path combination is verified by pumping cold water from the in-containment refueling water storage tank into the cold, depressurized, water-filled reactor coolant system; and back to the in-containment refueling water storage tank using the normal residual heat removal pump(s). The resistances are determined by measuring the residual heat removal pump flow rate and the pressure drop across the flow paths tested using temporary instrumentation. The acceptance criteria for the resistance of these lines is $\leq 2.91 \times 10^{-6}$ ft/gpm^2 for each ADS stage 1, 2, 3 group with all valves open.

q) The resistance of each automatic depressurization stage 4 flow path and their flow path combinations is verified by pumping cold water from the in-containment refueling water storage tank into the cold, depressurized, water-filled reactor coolant system using the normal residual heat removal pump(s). The resistances are determined by measuring the residual heat removal pump flow rate and the pressure drop across the flow paths tested using temporary instrumentation. A test fixture with prototypical resistance may be used to simulate the squib valves in the flow paths tested. The acceptance criteria for the resistance of these lines are $\leq 1.70 \times 10^{-7}$ ft/gpm^2 for ADS stage 4 on loop 1 and $\leq 1.57 \times 10^{-7}$ ft/gpm^2 for ADS stage 4 on loop 2 with all valves open.

r) The proper operation of the vacuum breakers in the automatic depressurization discharge lines is verified.

s) [*During hot functional testing of the reactor coolant system, proper operation of automatic depressurization is verified by blowing down the reactor coolant system. This testing verifies proper operation of the stage 1, 2, and 3 components including the ability of the spargers to limit loads imposed on the in-containment refueling water storage tank by the blowdown. Proper operation of the stage 1, 2 and 3 valves is demonstrated during blowdown conditions.* ***Note that this verification is required only for the first three plants****.*]*

t) The proper operation of at least one of each squib valve size and type including a containment recirculation, in-containment refueling water storage tank injection, and a stage 4 automatic depressurization squib valve is demonstrated. Squib valve functionality is validated via squib valve lot acceptance testing. Flow through the actuated squib valves is validated via the squib valve qualification program. This test does not have to be performed in the plant. RN-15-078

u) The proper operation of the containment sump instrumentation is demonstrated by simulating the containment flood-up water levels.

v) The proper operation of the CMT level instrumentation is demonstrated during the draindown testing of the CMTs, specified in Item l) above.

w) [*In conjunction with the verification of the core makeup tanks to perform their reactor water makeup function and boration function described in item k) above, the proper operation of the core makeup tanks to transition from their recirculation mode of operation to their draindown mode of operation after heatup will be verified. This testing will also verify the proper operation of the core makeup tank level instrumentation to operate during draining of the heated tank fluid. The in-containment refueling water storage tank initial level is reduced to at least 3 feet below the spillway level as a prerequisite condition for this testing in order to provide sufficient ullage to accept the mass discharged from the reactor coolant system via the automatic depressurization stage 1.*

The recirculation operation in Item k) above, should be continued until the core makeup tank fluid has been heated to≥ 350°F. The core makeup tank isolation valves are then closed, the reactor coolant pumps are started, and the reactor coolant system is reheated up to hot functional testing conditions. This testing is initiated by shutting off the reactor coolant pumps, opening the core makeup tank isolation valves, and by opening one of the automatic depressurization stage 1 flow paths to the in-containment refueling water storage tank. This will initiate a large loss of mass from the reactor coolant system, depressurization of the reactor coolant system to the bulk fluid saturation pressure, and additional recirculation through the core makeup tank. Core makeup tank draindown initiates in response to the continued depressurization and mass loss from the reactor coolant system. The automatic depressurization stage 1 flow path is closed after the core

*NRC Staff approval is required prior to implementing a change in this information.

makeup tank level has decreased below the level at which stage 4 actuation occurs. ***Note that this verification is required only for the first three plants.***]*

14.2.9.1.4 Passive Containment Cooling System Testing

Purpose

The purpose of the passive containment cooling system testing is to verify that the as-installed components perform properly to accomplish their safety-related functions to transfer heat from inside the containment to the environment, as described in subsection 6.2.2. The passive containment cooling water storage tank also provides a safety-related source of makeup water for the spent fuel pool, and provides a seismically qualified source of water for the fire protection system. Testing of these functions are discussed in Subsections 14.2.9.2.7 Spent Fuel Pool Cooling System Testing, and 14.2.9.2.8 Fire Protection System Testing.

Prerequisites

The component testing of the passive containment cooling system is successfully completed. The preoperational testing of the Class 1E dc electrical power and uninterruptible power supply systems, the non-Class 1E electrical power supply system, the compressed and instrument air system, and other interfacing systems required for operation of the above systems is available as needed to support the specified testing and system configurations. Additionally, a sufficient quantity of acceptable quality water for filling the passive containment cooling water storage tank and draining onto the containment is available, and a means of filling the tank is available. RN-14-110

General Test Acceptance Criteria and Methods

Passive containment cooling system performance is observed and recorded during a series of individual component testing that characterizes passive containment cooling system operation. The following testing demonstrates that the passive containment cooling system operates as described in Section 6.2 and appropriate design specifications:

a) Proper operation of safety-related valves is verified by the performance of baseline in-service tests as described in subsection 3.9.6.

b) Proper calibration and operation of safety-related, defense-in-depth, and system readiness instrumentation, controls, actuation signals and interlocks as discussed in Sections 7.3 and 7.5 are verified. This testing includes the following:

- Normal range containment pressure
- High range containment pressure
- Passive containment cooling water flow rate
- Passive containment cooling water storage tank level
- Passive containment cooling water isolation valve instrumentation and controls
- Diverse actuation system passive containment cooling initiation
- Passive containment cooling water storage tank water temperature
- Air inlet and shield plate freeze protection heater controls

This testing includes demonstration of proper actuation of these functions from the main control room.

*NRC Staff approval is required prior to implementing a change in this information.

c) Flow testing is performed to demonstrate proper system flow rates by draining the passive containment cooling system water storage tank. This testing demonstrates the proper resistance of the four passive containment cooling water storage tank delivery flow paths. This testing also demonstrates that water is supplied at the specified flow rates and times for 72 hours consistent with the design basis analyses presented in subsection 6.2.1.

d) The proper operation of the passive containment cooling water distribution bucket and weirs is verified and proper wetting of the containment is observed and recorded during draindown testing in Item c, above. Water delivery and coverage is verified at the initial minimum water level and as each of the first two standpipes is uncovered. Water coverage is measured at the spring line and the base of the upper annulus as described in subsection 6.2.2.4.2.

e) The proper operation of the drains in the upper containment/shield building annulus to drain the containment cooling water from the annulus floor is verified.

f) The resistance of the passive containment cooling air flow path is verified by measuring the wind induced driving head developed from the air inlet plenum region of the shield building to the air exhaust at several locations along the flow path and at several circumferential locations, and measurement of the induced air flow velocity. Temporary instrumentation is used for this testing.

g) Sample coupons from the containment shell with and without an appropriate coating of paint are laboratory tested to determine their conductivity.

h) The proper operation of each of the PCS water storage tank recirculation/makeup pumps to makeup sufficient water to the PCS water storage tank from the PCS ancillary water storage tank is verified.

14.2.9.1.5 Chemical and Volume Control System Isolation Testing

Purpose

The purpose of the chemical and volume control system isolation testing is to verify that the as-installed components properly perform the following safety-related isolation functions, described in Section 9.3:

- Termination of inadvertent dilution of the reactor coolant boron concentration
- Isolation of unborated water sources for reactor makeup
- Reactor coolant system pressure boundary isolation
- Isolation/termination of excessive makeup to the reactor

Prerequisites

The component testing of the chemical and volume control system has been successfully completed. RN-14-110
The required preoperational testing of appropriate support and interfacing systems is completed. Data collection is available as needed to support the specified testing and system configurations.

General Test Acceptance Criteria and Methods

Performance of the chemical and volume control system isolation functions is observed and recorded during a series of individual component and integrated system testing that characterizes the system isolation modes of operation. The following testing demonstrates that the chemical and volume control system properly performs the safety-related isolations as specified in Section 9.3 and appropriate design specifications:

*NRC Staff approval is required prior to implementing a change in this information.

a) Proper operation of the safety-related valves is verified by the performance of baseline in-service tests as described in subsection 3.9.6, including:

- Purification loop isolation valves
- Letdown isolation valves
- Demineralized water isolation valves
- Makeup isolation valves
- Auxiliary spray isolation valve

b) Proper calibration and operation of safety-related instrumentation, controls, actuation signals and interlocks is verified. This testing includes the following:

- Purification isolation valve controls
- Letdown isolation valve controls
- Demineralized water isolation controls
- Makeup isolation valve controls

This testing includes demonstration of proper actuation of safety-related functions from the main control room.

14.2.9.1.6 Main Control Room Emergency Habitability System Testing

Purpose

The purpose of the main control room emergency habitability system testing is to verify that the as-installed components properly perform the safety-related functions described in Section 6.4, including the following:

- Provide sufficient breathable quality air to the main control room
- Maintain the main control room at positive pressure
- Provide passive cooling of designated equipment

In addition, the following safety-related functions performed by the nuclear island nonradioactive ventilation system described in subsection 9.4.1 are tested:

- Provide isolation of the main control room from the surrounding areas and outside environment during a design basis accident if the nuclear island nonradioactive ventilation system becomes inoperable.

- Monitor the radioactivity in the main control room normal air supply and provide signals to isolate the incoming air and actuate the main control room emergency habitability system.

In addition, the following safety-related functions performed by the potable water system, described in subsection 9.2.5; the sanitary drainage system, described in subsection 9.2.6; and the waste water system, described in subsection 9.2.9, are tested:

- Provide isolation of the main control room from the surrounding areas and outside environment during a design basis accident.

Prerequisites

The component testing of the main control room habitability system has been successfully completed. The required preoperational testing of the compressed and instrument air system, Class 1E electrical power and uninterruptible power supply systems, normal control room ventilation system, and other interfacing systems required for operation of the above systems is available as RN-14-110 needed to support the specified testing and system configurations. The main control room air supply tanks are filled with air acceptable for breathing. The main control room construction is complete and its leak-tight barriers are in place.

General Test Acceptance Criteria and Methods

Performance of the main control room habitability system is observed and recorded during a series of individual component and integrated system testing. The following testing demonstrates that the habitability system operates as specified in Section 6.4 and as specified in the appropriate design specifications:

a) Proper operation of safety-related valves is verified by the performance of baseline in-service tests as described in subsection 3.9.6.

b) Proper calibration and operation of safety-related and system readiness instrumentation, controls, actuation signals and interlocks is verified. This testing includes the following:

   - Air storage tank pressure
   - Refill line connection pressure
   - Main control room differential pressure
   - Air supply line flow rate
   - Controls for the main control room pressure relief valves
   - Controls for the air supply isolation valves
   - Controls for the main control room air inlet isolation valves
   - Air intake radiation
   - Passive filtration line flow rate
   - Filter performance
   - Sanitary drainage system vent isolation valves

c) The proper flow rate of emergency air to the main control room is verified, demonstrating proper sizing of each air flow limiting orifice, proper operation of each air supply pressure regulator, and the ability to maintain proper control room air quality. The MCR passive filtration system flow rate and filter performance will also be verified at this time to ensure a filtration flow rate of at least 600 cfm. This testing demonstrates the control room pollutant concentrations during the first 6 hours of operation. To determine the control room air quality at 72 hours, the CO_2 concentrations can be predicted based on calculations. The other pollutants described in Table 1 and Appendix C, Table 1 of ASHRAE Standard 62-1989 can be predicted by extrapolating their concentrations for the entire 72-hour period.

d) The ability of the emergency air supply to maintain the main control room at the proper positive pressure is demonstrated, verifying proper operation of the main control room pressure relief dampers.

e) The ability of the emergency air supply to limit air inleakage to the main control room is verified by inleakage testing as specified in subsection 6.4.5.4.

f) The ability to maintain the main control room environment within specified limits for 72 hours (Reference subsection 6.4.3.2) is verified with a test simulating a loss of the nuclear island nonradioactive ventilation system. This testing demonstrates the control room heatup from 0 to 6 hours with the actual heat loads from the battery powered equipment and personnel specified for this time period. This testing period includes the high 0 to 3 hour heat load and subsequent control room temperature changes versus time that occur when the equipment heat load is decreased when the 2 hour batteries are expended, for the 3 to 6 hour testing time period. The control room temperature versus time versus heat load data are used to verify the analysis basis used to assure that the control room conditions remain within specified limits for the 72 hour time period. Periodic grab samples will be taken of the control room air environment to support analyses to confirm that specified limits would not be exceeded for 72 hours.

g) The ability to maintain temperatures in the protection and safety monitoring system cabinet and emergency switchgear rooms within specified limits for 72 hours (Reference subsection 6.4.3.2) is verified with a test simulating a loss of the nuclear island nonradioactive ventilation system. This testing demonstrates the room heatup from 0 to 6 hours with the actual heat loads from battery powered equipment. The room temperature versus time versus heat load data are used to verify the analysis basis used to assure that the room temperature will not exceed the specified limit for the 72 hour time period.

14.2.9.1.7 Expansion, Vibration and Dynamic Effects Testing

Purpose

The purpose of the expansion, vibration and dynamic effects testing is to verify that the safety-related, high energy piping and components are properly installed and supported such that expected movement due to thermal expansion during normal heatup and cooldown, and as a result of transients; thermal stratification and thermal cycling; as well as vibrations caused by steady-state or dynamic effects do not result in excessive stress or fatigue to safety-related plant systems and equipment, as described in Section 3.9.

Prerequisites

The component testing and preoperational testing of the reactor coolant system at cold conditions has been successfully completed. Required portions of the chemical and volume control system, passive core cooling system, normal residual heat removal system, main feedwater system, startup feedwater system, steam generator system, and steam generator blowdown system are operational. Piping and components within the reactor coolant system and steam generator system pressure boundaries and their associated supports and restraints have been inspected and determined to be installed as designed. Permanently installed support devices have been verified to be in their expected cold, static positions and temporary restraining devices such as hanger locking pins have been removed. The instrumentation required for this testing is installed. RN-14-110

General Test Method and Acceptance Criteria

During hot functional testing, verifications that ASME Code Class 1, 2, and 3 high-energy piping system components, piping, support, and restraint deflections are unobstructed and within design basis functional requirements. The systems to be monitored during preoperational vibration and dynamic effects tests include:

- ASME Code, Class 1, 2, and 3 piping
- High-energy piping systems inside seismic Category I structures
- High-energy portions of systems whose failure could reduce the functioning of seismic Category I plant features to an unacceptable safety level
- Seismic Category I portions of moderate-energy piping systems located outside the containment

The high-temperature portions of the following systems are considered for inclusion in this test:

- Reactor coolant system

- Chemical and volume control system

- Passive core cooling system

- Steam generator system (including the safety-related portions of main steam system, main and startup feedwater systems, and steam generator blowdown system)

- Normal residual heat removal system

a) Thermal expansion testing during the preoperational testing phase consists of displacement measurements on the above systems during heatup and cooldown of the reactor coolant system and associated systems (including heatup and cooldown of the passive core cooling system). The testing is performed in accordance with subsection 3.9.2.1.2 and consists of a combination of visual inspections and local and remote displacement measurements. This testing includes the inspection and measurement of deflection data associated with support thermal movements to verify support swing clearance at specified heatup and cooldown intervals; that there is no evidence of blocking of the thermal expansion of any piping or components, other than by installed supports, restraints, and hangers; that spring hanger movements remain within the hot and cold setpoints; that moveable supports do not become fully retracted or extended; and that piping and components return to their approximate baseline cold positions. RN-15-022

b) Vibration testing is performed on safety-related and high-energy system piping and components during both cold and hot conditions to demonstrate that steady-state vibrations are within acceptable limits. See subsection 3.9.2.1.1 for the acceptable standard for alternating stress intensity due to steady-state vibration. This testing includes visual observation and local and remote monitoring in critical steady-state operating modes. Results are acceptable when visual observations show no signs of excessive vibration and when measured vibration amplitudes are within acceptable levels.

c) Testing for significant vibrations caused by dynamic effects is conducted during hot functional testing and may be performed as part of other specified preoperational tests. This testing is conducted to verify that stress analyses of safety-related and high-energy system piping under transient conditions are acceptable. See subsection 3.9.2.1.1 for the acceptable standard for alternating stress intensity due to dynamic effects vibration. These tests are performed to verify that the dynamic effects caused by transients such as pump starts and stops, valve stroking, and significant process flow changes are within expected values. These tests include anticipated normal operating evolutions with system differential temperatures, such as startup, which could induce dynamic effects. Suitable instrumentation is used to monitor for the occurrence of water hammer noise and vibration. Visual inspections are performed to confirm the integrity of system piping and supports.

   Deflection measurements during various plant transients are recorded and compared to acceptance limits and it is confirmed that no effects due to water hammer are detected.

d) As described in subsection 3.9.3, temperature sensors are installed on the pressurizer surge line and pressurizer spray line for monitoring thermal stratification and thermal cycling during power operation. Testing is performed to verify proper operation of these sensors. **Note that this verification is required only for the first plant.**

The main control room habitability system is classified as a high energy system based on the pressure criteria not temperature. Tests that measure thermal movements are not required. Vibration testing of the high pressure portion of the main control room habitability system is performed during testing of the air delivery rate provided to the control room. See subsection 14.2.9.1.6 for information on the testing of the main control room habitability system.

14.2.9.1.8 Control Rod Drive System

Purpose

The purpose of the control rod drive system testing is to verify the proper operation of the control rod drive mechanisms, motor-generator sets and system components as described in subsection 3.9.4 and Section 4.6, and in appropriate design specifications.

Prerequisites

The component tests of the control rod system have been completed. Required interfacing systems, as needed, are completed to the extent sufficient to support the specified testing and the appropriate system configuration. Required electrical power supplies are energized and operational. RN-14-110

For the control rod drive mechanism cooling test, the plant is at or near normal operating temperature and pressure, and post-core hot functional testing is in progress. The integrated head and control rod drive mechanism cooling system are in their normal operational alignment.

For the control rod drive mechanism motor-generator sets tests, a three-phase load bank is available for motor generator set testing under loaded conditions.

General Test Methods and Acceptance Criteria

Performance is observed and recorded during a series of individual component and integrated system tests. The following tests verify that the control rod drive system operates properly:

a) Tests are conducted to verify the current command sequence, timing, and rod speed signal voltages by initiating control rod drive mechanism withdrawal and insertion. Proper operation of the bank overlap unit to control rod bank sequence and movement is verified.

b) Tests are conducted to verify the adequacy of the integrated head and control rod drive mechanism cooling system for maintaining control rod drive mechanism temperature. This test is conducted by measuring control rod drive mechanism coil resistances and calculating the coil temperatures.

c) Tests are conducted to verify control rod drive mechanism motor-generator set and system component control circuits, including interlock and alarm functions.

d) Tests are conducted to verify generator phasing for parallel generator operation. Operation of the control rod drive mechanism motor generator sets and control system during starting, running, and parallel operations is verified.

14.2.9.1.9 Reactor Vessel Internals Vibration Testing

Purpose

The AP1000 reactor internals testing is part of a comprehensive vibration assessment program performed in accordance with Regulatory Guide 1.20 as discussed in subsection 3.9.2.4. This testing obtains data to verify the structural integrity of the AP1000 reactor internals with regard to flow-induced vibrations, as part of an internals vibration assessment program. This program also includes visual examination of the reactor internals after testing is completed, and analysis of the test data. Testing is performed for the first plant only.

AP1000 plants subsequent to the first plant are visually inspected before and after the hot functional test to confirm that the internals are functioning correctly. The major features of the reactor internals outlined in subsection 3.9.2.4 are visually inspected for signs of abnormal wear and structural changes.

Prerequisites

The component testing of the reactor coolant system has been completed. The testing and calibration of the required test instrumentation has been completed. The test instrumentation has been installed on the internals as specified in Table 3.9-4 and the internals pre-test visual inspection has been completed. The internals, test instrumentation, and instrumentation lead wires are installed in the reactor vessel. The reactor vessel head is installed in preparation for the cold hydrostatic test of the reactor coolant system and instrument leads have been properly sealed. The proper operation and calibration of the test instrumentation and recording equipment is verified during the hydrostatic testing of the reactor coolant system. RN-14-110

General Test Method and Acceptance Criteria

Reactor vessel internals testing is performed for the first plant only by measuring and recording strains or accelerations of components in order to determine actual displacements that occur with the reactor coolant pumps operating. This testing is performed at several reactor coolant system temperatures during the system hot functional test. The analysis of data obtained from this testing, combined with a pre-test and post-test visual inspection of the internals, are intended to confirm that the stresses and wear on the AP1000 internals, due to flow induced vibration during plant operation, are acceptably low. The criteria for evaluating testing results are established in the AP1000 reactor internals flow-induced vibration assessment program (see Section 7 of WCAP-15949), and appropriate design specifications.

For the first plant only, the internals are instrumented to obtain data during the following reactor coolant system operating conditions:

a) Background noise in the instrumentation and recording equipment is recorded with no reactor coolant pumps running

b) Data is recorded during the initial startup of the reactor coolant pumps and with all four pumps operating and with the reactor coolant at cold temperature

c) Data is recorded at several increasing coolant temperatures with the pumps operating

d) Data is recorded at the hot functional testing temperature with all four pumps operating

e) Data is recorded at the hot functional testing temperature with the appropriate combinations of reactor coolant pumps operating, including pump start and stop transients

For all plants subsequent to the first plant, visual inspections are performed before and after the hot functional test. When no indications of harmful vibrations or signs of abnormal wear are detected and no structural damage or changes are apparent, the core support structures are considered to be structurally adequate and sound for operation. If such indications are detected, further evaluation is required.

14.2.9.1.10 Containment Isolation and Leak Rate Testing

Purpose

The purpose of the containment isolation and leak rate testing is to demonstrate that the as-installed containment isolation valves, piping and electrical containment penetrations, and hatches, and the containment vessel properly perform the following safety functions as described in Section 6.2:

- Automatic isolation of the piping penetrating containment required to assure containment integrity
- The containment vessel, penetration, and isolation valve leakage is less than the design basis leakage at or near the containment design pressure consistent with 10 CFR 50, Appendix J pressure test requirements.

Prerequisites

The component testing of the containment, containment hatches/airlocks and containment penetrations including the containment pressure test as specified in subsection 3.8.2.7 has been completed. The component testing of the piping and isolation valves or electrical wiring through the penetrations, has been completed. The instrumentation to be used in performing the Type A, B, and C testing is calibrated and available, including their associated data processing equipment. The required preoperational testing of the protection and safety monitoring system, plant control system, the Class 1E electrical power uninterruptible power supply, and other interfacing systems required for operation of the containment isolation devices and data collection is available. RN-14-110

General Test Acceptance Criteria and Methods

Containment isolation functions, leak rate, and structural integrity performance are observed and recorded during a series of individual component and integrated system testing. The following testing demonstrates that the containment functions as described in Section 6.2 and the appropriate design specifications are achieved. The testing is in accordance with the Containment System Leakage Testing Program and is discussed in subsection 6.2.6, which meets the requirements of ANSI/ANS-56.8-1994, as appropriate.

a) Proper operation of safety-related containment isolation valves, listed in Table 6.2.3-1, is verified by the performance of baseline in-service tests as specified in subsection 3.9.6.

b) Proper calibration and operation of safety-related containment isolation instrumentation, controls, actuation signals and interlocks is verified. This testing includes actuation of the containment isolation valves from the main control room, and upon receipt of a containment isolation signal.

c) The appropriate Type C leakage testing is performed for each piping path penetrating the containment boundary, verifying the leakage for each containment isolation valve (listed in Table 6.2.3-1) or set of isolation valves. This testing for individual isolation valves may be performed in conjunction with the associated system test.

d) The appropriate Type B leakage testing is performed for each containment penetration whose design incorporates seals, gaskets, sealants, or bellows. This testing includes door or hatch operating mechanisms and seals.

e) A baseline in-service test/inspection of the accessible interior and exterior surfaces of the containment structure and components is performed as specified in subsection 3.8.2.

f) A Type A integrated leak rate test is performed to verify that the actual containment leak rate does not exceed the design basis leak rate specified in the Technical Specifications.

14.2.9.1.11 Containment Hydrogen Control System Testing

Purpose

The purpose of the containment hydrogen control system testing is to verify that the system properly performs the following safety-related and non-safety defense-in-depth functions described in Section 6.2:

- Prevent the concentration of hydrogen in containment from reaching the flammability limit.
- Prevent the concentration of hydrogen in containment from reaching the detonation limit.
- Monitor the containment hydrogen concentration as required by Regulatory Guide 1.97.

Prerequisites

The component testing of the containment hydrogen control system is completed. The Class 1E dc electrical power and uninterruptible power supply systems, the non-Class 1E electrical supply system, and other interfacing systems required for operation of the above systems and calibrated data collection instrumentation are available as needed to support the specified testing. RN-14-110

General Test Acceptance Criteria and Methods

Performance of the containment hydrogen control system is observed and recorded during a series of individual component testing. The following testing verifies that the system operates as described in subsection 6.2.4 and as specified in the appropriate design specifications:

a) Proper operation of both the Class 1E safety-related and non Class 1E containment hydrogen concentration instrumentation and alarms is verified.

b) The ability of the passive autocatalytic recombiners to properly respond to a known inlet hydrogen/air mixture is verified by removing and testing one plate or cartridge from each manufacturing lot of catalyst material, contained in each recombiner unit. This verification is performed in accordance with the guidance provided in subsection 6.2.4.5.1 using a manufacturer's standard test device and test procedure. Plate performance is verified to be consistent with the response obtained in manufacturer's tests.

c) Manual actuation and operation of the hydrogen igniters confirm that the igniters are supplied by two power groups from two subsystems of the non-Class 1E dc and UPS system. Operability of the igniters is confirmed by verification that the igniter surface temperature exceeds the temperature specified in subsection 6.2.4.

14.2.9.1.12 Protection and Safety Monitoring System Testing

Purpose

The purpose of the protection and safety monitoring system preoperational testing is to verify that the as-installed components properly perform the following safety-related functions, described in Section 7.1:

- Receive and analyze sensor inputs required for reactor trip and automatically initiate reactor trip signals when plant conditions reach the appropriate setpoints

- Provide actuation signals to the engineered safety features to limit the consequences of design basis accidents

- Provide instrumentation and display systems to monitor the safety-related functions of the plant during and following the occurrence of design basis accidents in accordance with Regulatory Guide 1.97

Preoperational testing is also performed to verify proper operation of the following defense-in-depth functions, described in Section 7.1:

- Provide data from the safety-related sensors to the plant control system
- Provide information to the data display and processing system
- Provide data to the monitor bus for use by other systems within the plant

Prerequisites

Related system interfaces are available or simulated as necessary to support the specified test configurations. Component testing and instrument calibrations have been completed. Programming has been completed and the initial software diagnostics tests have been completed. Required electrical power supplies and control circuits are energized and operational. Plant systems or components which are to be operated during testing are specifically identified in the preoperational test procedures, are properly aligned, and have proper support systems operating prior to actuation of the particular system or component. Equipment or components which can not be actuated without damage or upsetting the plant are isolated using the test switches provided by the Protection and Safety Monitoring System to block device actuation. Continuity of wiring up to the actuation equipment is verified. RN-14-110

General Test Methods and Acceptance Criteria

Performance of the protection and safety monitoring system is observed and recorded during a series of individual component and integrated tests designed to verify operation of the system components. The following testing verifies that the system operates as described in Section 7.1 and appropriate design specifications:

a) Processing of the analog and digital signals is verified by injecting reference signals and verifying the outputs at various locations in the system.

b) Capability to process sensor data and main control room manual inputs resulting in the initiation of appropriate reactor trip signals is demonstrated by simulating inputs for each of the trip functions. Response times are verified by demonstrating that the applicable trip, actuate, permissive or interlock signal reaches the actuated equipment within the maximum allowable period following a defined step change in the applicable simulated input, above or below the trip, actuate, permissive or interlock setpoint. Operation of the protection cabinet trip/normal/bypass switches and indicators for each of the reactor trip functions is demonstrated by verifying appropriate outputs. Verification that the reactor trip bypass logic satisfies the single failure criteria is demonstrated by operating the bypass switches while simulating channel failures. Proper operation of the reactor trip reset function which is a nonsafety function of the Plant Control System (PLS) will be verified.

c) Operation of the reactor trip breakers, including breaker interlock, alarm, and tripping functions and verification that reactor trip response times are less than the specified maximum allowable response times is performed by initiating a manual reactor trip from the main control room. The capability of the undervoltage coil and the shunt trip coil functions to independently trip the reactor trip breakers is verified during this test using the test capabilities provided by the reactor trip switchgear interface.

d) The capability to trip the reactor from the remote shutdown workstation is demonstrated by verifying actuation of the reactor trip breaker undervoltage and shunt trip attachments upon initiation of a reactor trip at the remote shutdown workstation location.

e) The capability of the protection and safety monitoring system to process sensor data and manual inputs, resulting in appropriate engineered safety features actuation at design setpoints, is demonstrated by verifying that injection of simulated inputs for each of the engineered safety features actuation functions results in the proper output as indicated by contact operation, component actuation, or electrical test. Response times associated with the engineered safety features actuation functions are evaluated during these tests to provide verification that the applicable trip, actuate, permissive or interlock signal reaches the actuated equipment within the maximum allowable period following a defined step change in the applicable simulated input above or below the trip, actuate, permissive or interlock setpoint. Operation of the manual actuation/bypass switches and indicators for each of the engineered safety features functions is verified by demonstrating appropriate system outputs. Verification that the engineered safety features bypass logic satisfies the single failure criteria is demonstrated by operating the bypass switches while simulating channel failures. Correct input processing and calculational accuracy of the redundant actuation equipment and operator interface features is verified for each defined engineered safety features actuation function using simulated inputs. Proper operation of the engineered safety features reset functions will be verified.

f) Correct processing of inputs by redundant equipment and operation of the processing, permissive, interlock, display and operator interface features is verified by demonstrating that simulated command inputs result in correct output or actuation functions as indicated by contact operation, component actuation, or electrical test.

g) Accurate processing of component-level manual actuation commands from the main control room to the protection logic cabinets is verified by simulating main control room commands. Processing of component status information is demonstrated by simulating protection logic cabinet outputs to the main control room.

h) Processing of component-level actuation commands from the remote shutdown workstation to the protection logic cabinets is verified by simulating remote shutdown workstation commands. Processing of component status information is verified by simulating protection logic cabinet outputs to the remote shutdown workstation.

i) Operation of the automatic testing features provided in the protection and safety monitoring system is verified by observing the automatic test functions while simulating component failures and utilizing man-machine interface capabilities to evaluate system performance.

j) The capability of the protection and safety monitoring system to provide the plant operator with correct equipment status, component position indication, component control modes and abnormal operating conditions is verified by evaluating system response to simulated inputs representing feedback from actuation devices and position indicators. Communication of information via the plant monitor bus/data display and processing system, such as channel input quality, neutron flux detector high voltage, partial trip/actuation, permissive, interlock, block, reset, bypass, automatic test, reactor trip switchgear and system level actuation status, from the protection and safety monitoring system to external systems is verified by evaluating system response to injected reference signals and operating applicable block and bypass controls.

k) Operation of the qualified data processing equipment is verified by monitoring outputs and qualified display indications generated in response to simulated inputs representing data from the integrated protection cabinets and sensor inputs to the qualified data processing I/O cabinets.

l) Operation of the isolated data links and data highways used for communication between the engineered safety features actuation cabinets, main control room multiplexer cabinets, remote workstation multiplexer cabinets and protection logic cabinets is verified.

m) Preoperational testing of plant sensors used to provide data related to plant equipment monitored by the protection and safety monitoring system is performed in conjunction with testing of the respective systems in which these sensors are located.

n) The capability of the protection and safety monitoring system to provide data from the safety-related sensors to the plant control system is verified by injecting reference signals into the integrated protection cabinets and monitoring the plant control system signal selector outputs.

14.2.9.1.13 Incore Instrumentation System Testing

Purpose

The purpose of the incore instrumentation system preoperational testing is to verify that the as-installed components properly perform the following safety-related functions, described in Section 7.1:

- Provide reactor coolant system pressure boundary integrity for the incore instrumentation thimble assemblies which penetrate the upper head of the reactor vessel

- Provide the protection and safety monitoring system with the core exit temperature signals required for post-accident monitoring

Testing is also performed to verify the following nonsafety-related defense-in-depth functions, described in subsection 4.4.6:

- Provide core exit temperature signals to the diverse actuation system dedicated display in the main control room

Prerequisites

Related system interfaces are available or simulated as necessary to support the specified test configurations. Component testing and instrument calibrations have been completed. Required electrical power supplies are energized and operational.

General Test Methods and Acceptance Criteria

Performance of the incore instrumentation system is observed and recorded during a series of individual component and integrated tests designed to confirm operation of the system components outside the reactor vessel. The following testing verifies that the system operates as described in Section 7.1 and the appropriate design specifications:

a) Reactor coolant system pressure boundary integrity at the incore instrumentation reactor vessel head penetrations is verified during hydrostatic testing of the reactor coolant system.

b) Processing of the incore thermocouple signals is verified by thermocouple signals at the incore instrumentation thimble assembly connectors and verifying the thermocouple signal paths.

14.2.9.1.14 Class 1E DC Power and Uninterruptible Power Supply Testing

Purpose

The purpose of the Class 1E dc power and uninterruptible power supply testing is to verify that the as-installed components properly perform the following safety-related functions described in Section 8.3:

- Provide the electrical power required for the operation of the plant safety-related equipment, equipment controls, and instrumentation
- Provide the required safety-related electrical power for at least 72 hours following a design basis event, independent of both offsite and onsite ac electrical power supplies
- Provide separation and independence of Class 1E power divisions from other Class 1E divisions and non-Class 1E systems

Testing is also performed to verify proper operation of the following defense-in-depth functions described in subsection 8.3.2:

- The capability to recharge the batteries from the onsite or offsite ac electrical sources is verified so that safety-related functions can be supported for an indefinite time

Prerequisites

The component testing of the Class 1E dc power and interruptible power supply components has been completed. The necessary permanently installed and test instrumentation is calibrated and operational. The 480V ac electrical power system is in operation and supplying power to the battery chargers and regulating transformers. A test load is available for the performance of battery capacity tests. RN-14-110

General Test Methods and Acceptance Criteria

Performance of the Class 1E dc power and interruptible power supply is observed and recorded during a series of individual component and integrated system tests that characterize the operation of the system. The following testing verifies that this system operates as described in Section 8.3 and appropriate design specifications:

a) The capability of each of the seven Class 1E batteries to provide the required momentary and continuous load is verified by a battery service test performed in accordance with IEEE Standard 450. Following this discharge testing, the voltage of each cell is verified to be greater than or equal to the specified minimum cell voltage.

b) The capacity of each of the seven Class 1E batteries is verified to meet or exceed the required ampere-hour rating by a battery performance test performed in accordance with IEEE Standard 450. Following this discharge testing, the voltage of each cell is verified to be greater than or equal to the specified minimum cell voltage.

c) The capability of each of the seven battery chargers to charge its associated battery at the required rate is verified. This testing includes verification that the individual voltage of each cell is within the specified limits for a charged battery.

d) The capability of each of the six inverters to provide the required output current, frequency, and voltage is verified.

e) The capability of each of the four regulating transformers to provide the proper ac current to the Class 1E ac distribution panels is verified.

f) The capability of each of the static transfer switches to automatically transfer the electrical loads supplied by each inverter to its associated regulating transformer is verified.

g) The separation and independence of each redundant division of the Class 1E dc power and interruptible power supply is verified by successively powering only one division at a time and verifying power to the proper loads and the absence of voltage at the bus and loads not under test.

h) The proper calibration and operation of instrumentation and alarms, electrical ground detection, and permissive and prohibitive interlocks is verified.

14.2.9.1.15 Fuel Handling and Reactor Component Servicing Equipment Test

Purpose

To verify proper operation of the fuel-handling and reactor component servicing equipment as described in Section 9.1. This includes the refueling machine, fuel handling machine, fuel transfer system, and refueling tools used to lift, transport, or otherwise manipulate fuel, control rods and other incore instruments.

Prerequisites

The component tests have been completed. Prerequisites of the required interfacing systems are completed to the extent sufficient to support the specified testing. Required electrical power supplies are energized and operational. Compressed air, as required for tool operation, is available. The reactor vessel head has been removed, the reactor vessel and refueling cavity are drained, the refueling cavity gate is open, and the area in which the refueling machine moves is free of structures or components that could interfere with fuel handling operations. RN-14-110

The spent fuel pool and fuel transfer canal are drained, and the area in which the fuel handling machine moves is free of any structures or components that interfere with design fuel handling operations.

The fuel transfer system is operable and capable of transporting a dummy fuel assembly from the spent fuel pool to containment. A dummy fuel assembly, resembling an actual fuel assembly in weight, envelope, and mating hardware, is available for use. The fuel transfer system and new fuel elevator are operable as required to permit testing of fuel handling machine functions.

General Test Methods and Acceptance Criteria

The following tests are performed to verify the refueling machine operation:

a) The refueling machine is operated to simulate actual refueling operations, using a dummy fuel assembly. This testing includes manual and automatic modes of operation, displays, interlocks, and limits. These tests verify:

- The ability to move a fuel assembly from the fuel transfer system to the reactor vessel and back

- The consistency of measured trolley, bridge, and hoist speeds with each mode of operation
- The operability of interlocks limiting motion, speed, and weight, including interlocks with other plant equipment
- The operability of displays indicating position, mode, alarm status, and load
- The adequacy of indexing (by placing the dummy fuel assembly in selected core locations)

b) A known weight or a calibrated spring scale is used to calibrate and set the load limits for the refueling machine load cells. A static load test or the manufacturer's test results are used to verify the ability of the refueling machine hoists to support 125 percent of their rated loads.

The following tests are performed to verify the operation of the fuel handling machine:

c) The fuel handling machine is operated to simulate actual refueling operations, using a dummy fuel assembly. These tests verify:

- The ability to transfer fuel assemblies between the new fuel elevator, fuel transfer system, fuel storage racks, and other areas of the pool where fuel is serviced or stored
- The consistency of measured trolley, bridge, and hoist speeds with each mode of operation
- The operability of interlocks limiting motion, speed, and weight, including interlocks with other plant equipment
- The operability of displays indicating position, mode, alarm status, and load

d) The fuel handling machine is operated to verify its capability to transfer fuel between the new fuel elevator, fuel transfer system, fuel storage racks, and other areas of the pool where fuel is serviced or stored.

e) A known weight or a calibrated spring scale is used to calibrate and set the load limits for the fuel handling machine load cells. A static load test or the manufacturer's test results are used to verify the ability of the fuel handling machine hoists to support 125 percent of their rated loads. RN-14-087

The following tests are performed to verify the proper operation of the fuel transfer system and refueling tools:

f) Using appropriate plant operating procedures, the operability of the new fuel elevator is verified. Testing is performed to demonstrate the proper operation of controls, displays, and limit switches, including operation of the interlock that prevents raising the elevator when it contains a fuel assembly.

g) Using appropriate plant operating procedures, the fuel transfer system is operated to simulate actual refueling operations, using a dummy fuel assembly. During these operations, the following items are verified:

- The ability to move fuel assemblies between the fuel building and containment, including proper operation of upenders in both locations

- The operability and setpoints of limit switches and of interlocks between stations and with other plant equipment
- The operability of displays indicating mode of operation and status

h) Tests are performed to verify that the refueling tools operate properly. Included are tools for handling new fuel assemblies, fuel assembly inserts, irradiation specimens, control rod drive shafts, as well as tools for such operations as control rod drive shaft latching and reactor vessel stud tensioning. As applicable, power is applied to each tool to verify proper operation of controls, limit switches, actuators, and indicators. Stud tensioning equipment is checked when assembling the reactor for hot functional testing. The new fuel handling tool is tested with the dummy fuel assembly during the test of the new fuel elevator.

14.2.9.1.16 Long-Term Safety-Related System Support Testing

Purpose

The purpose of this testing is verify the capability to perform the following functions for maintaining the extended operation of the safety-related systems and components as described in Section 1.9:

- Supply makeup water to the passive containment cooling system.
- Supply makeup water to the spent fuel pool.
- Provide electrical power for post-accident instrumentation, control room lighting and ventilation, division B and C I&C room ventilation, passive containment cooling system pumps, ancillary generator room lights, ancillary generator tank heaters.
- Provide ventilation cooling to the main control room.
- Provide ventilation cooling to the Class 1E cabinets for post-accident instrumentation.

Prerequisites

The component tests of the safety-related systems and/or components designed for long-term actions have been successfully completed. The preoperational testing of these systems and/or components, including instrument calibrations, has been completed as required for the specified testing, system configurations, and operations. Equipment required for data collection is available and operable. Water used in this testing should be of a quality suitable for filling the specified components. Equipment used to provide the required long-term actions is available. RN-14-110

General Test Method and Acceptance Criteria

The ability to perform the required long-term actions is observed and recorded during a series of individual component and integrated system testing. The following testing verifies that the long-term actions can be performed as discussed in Section 1.9 and as specified in appropriate design specifications:

a) The ability to provide makeup water to the passive containment cooling water storage tank as described in subsection 6.2.2 is verified.

b) The ability to provide electrical power to the post-accident monitoring instrumentation, control room lighting and ventilation, division B and C I&C room ventilation, passive containment cooling system pumps, ancillary generator room lights, ancillary generator tank heaters, using the ancillary diesel generators as described in Section 8.3 is verified.

c) The ability to provide main control room ventilation cooling using ancillary fans as described in subsection 9.4.1 is verified.

d) The ability to provide ventilation cooling to post-accident monitoring instrumentation equipment rooms using ancillary fans as described in subsection 9.4.1 is verified.

e) The ability to provide makeup water to the spent fuel pool via the safety-related makeup connection from the passive containment cooling system water storage tank, as described in subsection 9.1.3, is verified.

14.2.9.2 Preoperational Testing of Defense-in-Depth Systems

14.2.9.2.1 Main Steam System Testing

Purpose

The purpose of the main steam system testing is to verify that the as-installed system properly performs the following defense-in-depth function, as described in Section 10.3 and appropriate design specifications:

- Provide backup isolation of the steam lines to prevent blowdown of steam from the steam generators following an event where steam line isolation is required

Prerequisites

The component tests of the as-installed main steam system have been completed. Prerequisites of the required interfacing systems are completed to the extent sufficient to support the specified testing and the appropriate system configuration. RN-14-110

General Test Method and Acceptance Criteria

Main steam system performance is observed and recorded during a series of individual component and integrated system testing. The following testing demonstrates that the system operates as described in Section 10.4 and appropriate design specifications:

Proper operation of the following system valves is verified.

- Turbine steam stop valves
- Turbine bypass valves
- Auxiliary steam system supply header isolation valve
- Main steam moisture separator reheater 2nd stage steam isolation valve
- Extraction steam isolation and non-return valves

This testing includes actuation of these valves from the main control room. The ability of these valves to isolate steam flow is verified during hot functional testing.

14.2.9.2.2 Main and Startup Feedwater System

Purpose

The purpose of the main and startup feedwater system testing is to verify that the as-installed system properly performs the following nonsafety-related defense-in-depth function, as described in Subsections 10.4.7 and 10.4.9:

- Provide startup feedwater to the steam generators to remove heat from the reactor coolant system following the loss of normal feedwater

Prerequisites

The component testing of the main and startup feedwater system components and instruments, or specific portion to be tested has been completed. Required interfacing systems are available. RN-14-110

General Test Method and Acceptance Criteria

The main and startup feedwater system performance is observed and recorded during a series of individual component and integrated system testing. The following defense-in-depth testing demonstrates that the system operates as described in Subsections 10.4.7 and 10.4.9 and appropriate design specifications:

a) Proper operation of defense-in-depth instrumentation, controls, actuation signals and interlocks is verified. This testing includes actuation of startup feedwater pumps and remotely-operated valves from the main control room including isolation of the main feedwater system.

b) The capability of the startup feedwater pumps to operate properly when performing their defense-in-depth function and main feedwater pumps are verified with the steam generator at normal operating pressure.

c) The capability of the startup feedwater pumps to operate properly with miniflow to the condensate storage tank is verified.

d) The capability to restore normal steam generator water level from the low narrow range water level, without causing unacceptable feedwater or steam generator water hammer, is demonstrated (refer to Subsections 14.2.9.1.7 and 14.2.10.4.18).

14.2.9.2.3 Chemical and Volume Control System Testing

Purpose

The purpose of the chemical and volume control system testing is to verify that the as-installed system properly performs the following defense-in-depth functions described in subsection 9.3.6 and appropriate design specifications:

- Provide makeup water to the reactor coolant system
- Provide boration of the reactor coolant system
- Provide auxiliary pressurizer spray

Prerequisites

The component testing of the as-installed chemical and volume control system is completed. The following interfacing and support systems are available as necessary to support testing: component cooling water system; service water system; reactor coolant system; electrical power and distribution systems. Data collection is available as needed to support the specified testing and system configurations. RN-14-110

General Test Acceptance Criteria and Methods

Chemical and volume control system performance is observed and recorded during a series of individual component and integrated system testing. The following testing verifies the system properly performs the defense-in-depth functions described in subsection 9.3.6 and appropriate design specifications:

a) Operation of pumps and valves which perform defense-in-depth functions is verified, including:

- Makeup pumps
- Boric acid mixing control valve
- Makeup flow control valve

b) Calibration and operation of defense-in-depth related instrumentation, controls, actuation signals and interlocks is verified, including:

- Automatic makeup pump actuation and shutoff
- Automatic alignment of the boric acid tank
- Pressurizer auxiliary spray initiation and termination
- Letdown/purification isolation

This testing includes actuation of defense-in-depth pumps and remotely-operated valves from the main control room. Pressurizer level control testing is described in subsection 14.2.9.1.1.

c) The capability of the makeup pumps to operate when performing their normal makeup and pressurizer spray functions is verified with the reactor coolant system at normal operating pressure.

d) The capability of the makeup pumps to operate at miniflow and the operation of the miniflow heat exchanger is verified.

e) The proper purification loop flowrate through the demineralizers and filters is verified.

14.2.9.2.4 Normal Residual Heat Removal System Testing

Purpose

The purpose of the normal residual heat removal system testing is to verify that the as-installed components and associated piping, valves, and instrumentation properly perform the following defense-in-depth functions, as discussed in Section 5.4:

- Remove reactor core decay heat and cool the reactor coolant system during shutdown operations at low pressure and temperature

- Remove reactor core decay heat from the reactor coolant system during reduced reactor coolant inventory operations in Modes 5 and 6

- Following actuation of the automatic depressurization system, provide makeup to the reactor coolant system at low pressure

- Circulate and cool water from the containment after draindown of the in-containment water storage tank

- Provide low temperature overpressure protection for the reactor coolant system

- Remove reactor core decay heat and cool the spent fuel pool during refueling operations when the core is off-loaded from the reactor vessel to the spent fuel pool.

Prerequisites

The component testing of the normal residual heat removal system is completed. The required preoperational testing of the in-containment refueling water storage tank, reactor coolant system, passive core cooling system, component cooling water system, service water system, ac electrical power and distribution systems, and other interfacing systems required for operation of the above RN-14-110 systems and data collection is available as needed to support the specified testing and system configurations. The reactor coolant system and the in-containment refueling water storage tank have an adequate water inventory to support testing.

General Test Acceptance Criteria and Methods

Normal residual heat removal system performance is observed and recorded during a series of individual component and system testing, that characterizes system operation. The following testing verifies that the normal residual heat removal system performs its defense-in-depth functions as described in subsection 5.4.7.6.1 and appropriate design specifications:

a) Operation of valves to open, to close, or to control flow as required to perform the above defense-in-depth functions is verified.

b) Operation of system controls, alarms, instrumentation, and interlocks associated with performing the above defense-in-depth functions is verified. In addition, the proper operation of the normal residual heat removal system/reactor coolant system isolation valve interlocks specified in Section 7.6 is verified.

c) The normal residual heat removal system pumps testing includes verification that the pump flow rate corresponds to the expected system alignment, proper pump miniflow operation, and verification that adequate net positive suction head is available for the configurations tested. The following system configurations are tested with each pump operating individually and with two pumps operating:

   - Recirculation from and to the reactor coolant system with the reactor coolant system at mid-loop hot leg water level and atmospheric pressure
   - Makeup to the reactor from the in-containment refueling water storage tank with approximately 4 feet of water in the tank
   - Makeup to the reactor from the cask loading pit with water in the pit at a sufficient level to support pump operation
   - Recirculation from and to the spent fuel pool with the pool at normal minimum level.

d) During the verifications of normal residual heat removal system flow to the reactor coolant system, verify that the pumped flow provides sufficient back pressure to maintain a water level in the CMT.

e) The capability of the normal residual heat removal heat exchangers to provide the required heat removal rate from the reactor coolant system is verified by testing performed with flow from and to the heated reactor coolant system, with each normal residual heat removal pump/heat exchanger operating individually.

f) The capability of the normal residual heat removal heat exchangers to provide the required heat removal rate from the spent fuel pool is verified. Since the spent fuel pool is not heated during pre-operational testing, this verification can be made based on the flowrate from Item c and heat removal capability from Item e, above.

g) Operation of the normal residual heat removal system relief valve which provides low temperature overpressure protection for the reactor coolant system is verified by the performance of baseline in-service testing, as specified in subsection 3.9.6. The acceptance criteria are based on the valve performance criteria specified in subsection 5.4.9.

h) Operation of the system to facilitate draining the reactor coolant system water level to near the centerline of the hot leg for reduced inventory operations is verified. This test is performed in conjunction with the chemical and volume control system, and is used to demonstrate the performance of the reactor coolant system hot leg level instruments as discussed in subsection 14.2.9.1.1.

14.2.9.2.5 Component Cooling Water System Testing

Purpose

The purpose of the component cooling water system testing is to verify that the as-installed system properly performs the following defense-in-depth functions as described in subsection 9.2.2:

- Provide cooling water to defense-in-depth components and transfer heat to the service water system. In addition, this system provides cooling water to other nonsafety-related components for heat removal.

Prerequisites

The component testing of the component cooling water system is completed. Preoperational testing of the cooled components has been completed as necessary to support testing of the component cooling water system. Required support systems are available, including applicable portions of the service water system and electrical power and distribution systems. Data collection is available as needed to support the specified testing and system configurations. RN-14-110

General Test Acceptance Criteria and Methods

Component cooling water system performance is observed and recorded during a series of individual component and integrated system testing that characterizes the various modes of system operation. The following testing demonstrates that the system operates as described in subsection 9.2.2 and in appropriate design specifications:

a) Proper operation of the component cooling water pumps is verified.

b) Proper operation of defense-in-depth related instrumentation, controls, actuation signals and interlocks is verified, including:

   - Automatic pump actuation if an operating pump stops
   - Pump flow rate
   - Pump discharge pressure
   - Surge tank water level and control
   - Surge tank pressure and control
   - Water flow rate to defense-in-depth components

   This testing includes actuation of the system pumps and remotely-operated valves from the main control room as appropriate.

c) The capability to provide the expected cooling water flow rates to and from the required components with both pumps operating, and with either individual pump and heat exchanger operating as specified in the appropriate design specifications is verified.

d) In conjunction with Item c above, the pump(s) runout flow rate is verified to be properly limited, and adequate net positive suction head is verified to be available during its operating modes.

e) The capability of the heat exchanger(s) to transfer heat properly to the service water system is verified under simulated plant conditions during plant hot functional testing. Testing conditions assume both pumps/heat exchangers in operation and with either one of the pumps/heat exchangers operating.

14.2.9.2.6 Service Water System Testing

Purpose

The purpose of the service water system testing is to verify the capability of the as-installed system to perform the following defense-in-depth function as described in subsection 9.2.1:

- Transfer heat from the component cooling water heat exchangers to the environment

Prerequisites

The component testing of the service water system is completed. Preoperational testing of the component cooling water heat exchangers so that they can receive service water has been completed, as well as the electrical power and distribution systems, and other interfacing systems required for operation of the service water system. Data collection is available as needed to support the specified testing and system configurations. The component cooling water system and components it cools are functional and hot preoperational testing of the reactor coolant system is in progress in order to confirm the service water system heat removal and heat rejection capability. RN-14-110

General Test Acceptance Criteria and Methods

Service water system performance is observed and recorded during a series of individual component and integrated system testing. The following testing demonstrates that the service water system properly performs its defense-in-depth functions, as described in subsection 9.2.1 and appropriate design specifications:

a) Proper operation of the service water pumps, valves, strainers, cooling tower fans, and freeze protection provisions are verified.

b) Proper operation of the defense-in-depth related instrumentation, controls, actuation signals and interlocks is verified, including:

   - Automatic pump actuation if an operating pump stops
   - Pump flow rate
   - Pump discharge pressure
   - Cooling tower water level and control
   - Cooling tower basin water temperature and control
   - Water supply and return temperature
   - Cooling tower fan control

   This testing includes actuation of defense-in-depth pumps and remotely-operated valves from the main control room as appropriate.

c) The capability of the pumps to provide the expected cooling flow rates to and from the component cooling water heat exchangers is verified. Testing conditions include both pumps operating and either individual pump operating.

d) In conjunction with Item c above, the pump(s) runout flow rate is verified to be properly limited, and adequate net positive suction head is verified to be available during appropriate operating modes.

e) The heat removal and heat rejection capability of the service water system during the conditions of the plant hot functional testing is verified. Testing conditions include both pumps/cooling towers cells in operation and with either one of the pumps/cooling tower cells operating.

14.2.9.2.7 Spent Fuel Pool Cooling System Testing

Purpose

The purpose of the spent fuel pool cooling system testing is to verify that the system properly performs the following defense-in-depth function described in subsection 9.1.3:

- Remove heat from the spent fuel stored in the spent fuel pool
- Prevent back flow through refueling canal drain lines when other in-containment compartments have been flooded

Prerequisites

The component testing of the spent fuel pool cooling system has been completed. The spent fuel pool is filled with water of acceptable quality and chemistry. The ac electrical power and distribution systems and other interfacing systems required for operation of the pumps and for data collection are available as needed to support the specified testing and system configurations. RN-14-110

General Test Acceptance Criteria and Methods

Spent fuel pool cooling system performance is observed and recorded during a series of individual component and integrated system testing. The following testing demonstrates that the system properly performs its defense-in-depth function as described in subsection 9.1.3 and appropriate design specifications:

a) Proper operation of the spent fuel pool cooling pumps, valves, and strainers is verified.

b) Proper operation of the instrumentation, controls, actuation signals, and interlocks is verified, including:

- Automatic pump actuation if an operating pump stops
- Pump flow rate
- Pump discharge pressure
- Spent fuel pool water level and control
- Spent fuel pool water temperature
- Water return temperature

This testing includes operation of the system pumps from the main control room.

c) The capability of the pumps to provide the expected cooling flow rates to and from the pool is verified; with both pumps operating, with either individual pump operating, and with either heat exchanger operating.

d) In conjunction with Item c above, the pump(s) runout flow rate is verified to be properly limited, and adequate net positive suction head is verified to be available during the appropriate operating modes.

e) The proper operation of the spent fuel pool siphon breakers is verified.

f) The proper operation of the spent fuel pool post-72 hour gravity drain flowpaths from the cask washdown pit and the passive containment cooling water storage tank is verified.

g) The gates, drains, bellows, and gaskets in the refueling canal and fuel storage pool are checked for unacceptable leakage.

14.2.9.2.8 Fire Protection System Testing

Purpose

The purpose of the fire protection system testing is to verify the system properly performs the following defense-in-depth function as described in subsection 9.5.1:

- Provide equipment for manual fire fighting in areas containing safe shutdown equipment
- Provide automatic fire suppression in areas containing selected non-safety-related equipment.
- Provide a nonsafety-related containment spray to reduce offsite dose following a severe accident

Prerequisites

The component tests of the fire protection system have been completed. Required preoperational testing of the ac power and distribution systems and other interfacing systems required for operation of the fire protection system. Data collection is available as needed to support the specified testing and system configurations. RN-14-110

General Test Method and Acceptance Criteria

Fire protection system performance is observed and recorded during a series of individual component and integrated system testing to verify the system performs its defense-in-depth function. The following testing demonstrates that the system performs its defense-in-depth functions specified in subsection 9.5.1 and as specified in appropriate design specifications:

a) The capability of the seismic standpipes to supply the required fire water quantity and adequate water pressure for effective hose streams as the required flow rate is verified.

b) The operability of the fire detection equipment is verified to be able to properly detect fires and alert personnel.

c) The proper installation and operation of fire barriers, fire walls, and portions of HVAC systems used for smoke control and exhaust is verified.

d) The proper operation of the fire pumps, fire water storage tank, and fire water supply piping, valves, and instrumentation to provide the as-designed fire water supply is verified.

e) The proper installation and operation of automatic fire suppression equipment is verified.

f) The proper installation and operation of electrical isolation devices for non-safety related equipment in opposite divisional fire areas is verified.

g) Operation of the containment spray remotely operated valve and the continuity of a flow path through the containment spray piping is verified.

14.2.9.2.9 Central Chilled Water System Testing

Purpose

The purpose of the central chilled water system testing is to verify that the as-installed low capacity portion of this system properly performs the following defense-in-depth function, as described in subsection 9.2.7:

- Provide chilled water to cool air used to cool safety-related or defense-in-depth equipment rooms

The proper function of the high capacity portion of this system is also verified.

Prerequisites

The component testing of the low capacity subsystem of the central chilled water system has been completed. The required preoperational testing of the component cooling and service water systems, ac electrical power and distribution systems, and other interfacing systems required for operation of the central chilled water system has been completed. Data collection is available as needed to support the specified testing and system configurations. RN-14-110

General Test Acceptance Criteria and Methods

Central chilled water system performance is observed and recorded during a series of individual component and integrated system testing. The following testing demonstrates that the central chilled water system performs its defense-in-depth functions described in subsection 9.2.7 and appropriate design specifications:

a) Proper operation of the low capacity portion of the central chilled water system equipment is verified, including chillers, pumps, and valves.

b) Proper calibration and operation of defense-in-depth related instrumentation, controls, actuation signals and interlocks are verified, including:

   - Temperature control of the chilled water
   - Chiller and chilled water pump actuation
   - Chilled water pump flow and discharge pressure
   - Chilled water flow control to air handling units

   This testing includes actuation of the defense-in-depth pumps and remotely operated valves from the main control room.

c) The proper chilled water flow rate to each of the nuclear island nonradioactive ventilation system air handling units is established, and the capability of each pump to provide this chilled water flow rate is verified.

d) In conjunction with Item c above, the pump(s) runout flow rate is verified to be properly limited, and adequate net positive suction head is verified to be available during the appropriate operating modes.

e) The heat removal capability of the air-cooled chillers is verified when the component areas cooled by the nuclear island nonradioactive ventilation system air handling units are operating.

In addition, the operability of the high capacity portion of the central chilled water system described in subsection 9.2.7 and appropriate design specifications, is verified.

14.2.9.2.10 Nuclear Island Nonradioactive Ventilation System Testing

Purpose

The purpose of the nuclear island nonradioactive ventilation system testing is to verify that the as-installed system properly performs the following defense-in-depth functions, as described in subsection 9.4.1:

- Protect the main control room and control support area from smoke infiltration
- Provide the capability to remove smoke from the main control room, control support area, and Class 1E electrical equipment rooms
- Provide heating, ventilation, and cooling for the main control room, control support area, and Class 1E electrical equipment rooms
- Provide air filtration to limit radioactivity in the main control room and control support area
- Maintain passive heat sinks at acceptably low initial temperatures
- Maintain the main control room and control support area at positive pressure

The safety-related functions associated with this system are tested as part of the main control room emergency habitability testing described in subsection 14.2.9.1.6.

Prerequisites

The component testing of the nuclear island nonradioactive ventilation system has been completed. RN-14-110
The required preoperational testing of central chilled water system, the hot water heating system, the ac electrical power and distribution systems, and other interfacing systems required for operation of the above systems has been completed. Data collection is available as needed to support the specified testing and system configurations.

General Test Acceptance Criteria and Methods

Nuclear island nonradioactive ventilation system performance is observed and recorded during a series of individual component and integrated system testing to verify the system performs its defense-in-depth functions. The following testing demonstrates that the system performs its defense-in-depth functions as described in subsection 9.4.1 and appropriate design specifications:

a) Proper function of the fans, filters, heaters, coolers, and dampers is verified.

b) Proper operation of instrumentation, controls, actuation signals, and alarms and interlocks is verified. This testing includes the following:

   - Smoke detectors and alarms
   - Air handling unit and fan flows, controls, and alarms
   - Differential air pressures and alarms
   - Air and air filtration unit charcoal temperatures, controls, and alarms
   - Air relative humidity measurements, controls, and alarms
   - Isolation/shutoff damper controls
   - Fire/smoke damper controls

   This testing includes operation from the main control room.

c) The proper air flows from and through each air handling unit, as well as to and from the main control room, control support area, and other equipment rooms is established for each mode of operation.

d) The main control room and control support area are verified to be maintained at the proper positive pressure.

e) The main control room, control support area, class 1E equipment rooms, and passive heat sink areas are verified to be maintained at their proper temperature during hot functional testing.

f) Air inleakage into the main control room and control support area is measured using a tracer gas.

14.2.9.2.11 Radiologically Controlled Area Ventilation System

Purpose

The purpose of the radiologically controlled area ventilation system testing is to verify that the as-installed system properly performs the following defense-in-depth function, as described in subsection 9.4.3:

- In conjunction with the low capacity portion of the central chilled water system, maintain the normal residual heat removal system and chemical and volume control system pump rooms at proper temperature during pump operation

Prerequisites

The component testing of the radiologically controlled area ventilation system has been completed. RN-14-110
The required preoperational testing of the central chilled water system, the ac electrical power and distribution systems, and other interfacing systems required for operation of the radiologically controlled area ventilation system has been completed. Data collection is available as needed to support the specified testing and system configurations.

General Test Acceptance Criteria and Methods

Radiologically controlled area ventilation system performance is observed and recorded during a series of individual component and integrated system testing to verify the system performs its defense-in-depth function as described in subsection 9.4.3 and appropriate design specifications:

a) Proper function of the defense-in-depth fans, filters, heaters, and coolers is verified.

b) Proper operation of defense-in-depth instrumentation, controls, actuation signals, alarms, and interlocks is verified. This testing includes operation of the normal residual heat removal and chemical and volume control pump room cooler/fans from the main control room.

c) The proper air flow and cooling capability of the normal residual heat removal and chemical and volume control pump room cooler/fans is verified.

d) The proper actuation of the normal residual heat removal and chemical and volume control pump room cooler fans in response to pump operation or high room temperature is verified.

14.2.9.2.12 Plant Control System Testing

Purpose

The purpose of the plant control system testing is to verify that the as-installed components perform the following nonsafety-related defense-in-depth functions, described in Section 7.1:

- Provide control and coordination of the plant during startup, ascent to power, power operation and shutdown conditions by integrating the automatic and manual control of the reactor, reactor coolant and reactor support processes required for normal and off-normal conditions. This includes rod control, pressurizer pressure and level control, steam generator water level control, steam dump (turbine bypass) control and rapid power reduction.

- Provide control of other defense-in-depth systems and components.

Prerequisites

Related system interfaces are available or simulated as necessary to support the specified test configurations. Component testing and instrument calibrations have been completed. The reactor vessel integrated head package is in place, all control rod drive mechanism cables are connected and the integrated head and control rod drive mechanism cooling system is operational. Programming has been completed and the initial software diagnostics tests have been completed. Required electrical power supplies and control circuits are energized and operational. Required plant control system field wiring is electrically isolated to prevent operation of components controlled by the plant control system. Equipment or components that cannot be operated without damage or upsetting the plant are isolated, either by using test switches provided by the Plant Control System or by racking out power circuit breakers, to block device operation. Continuity of wiring up to the equipment is verified. RN-14-110

General Test Methods and Acceptance Criteria

Performance of the plant control system hardware and software is observed and recorded during a series of individual component and integrated tests designed to verify operation of defense-in-depth functions. The following testing demonstrates that the system operates as described in Section 7.1 and applicable design specifications:

a) Processing of analog and digital signals is verified by injecting reference signals and monitoring the outputs of the plant control system.

b) Interfaces with other applicable plant equipment and systems such as reactor power control, feedwater control and turbine control are verified by demonstrating that injection of simulated inputs for each of the control functions provided in the main control room results in the proper output as indicated by contact operation, component actuation, or electrical test.

c) Interfaces with applicable plant equipment and systems are verified by demonstrating that injection of simulated inputs for selected control functions provided at the remote shutdown workstation results in the proper output as indicated by contact operation, component actuation, or electrical test.

d) Proper operation of defense-in-depth processing, signal selector processing, monitoring, display and operator interface features provided by the plant control system is demonstrated by monitoring system outputs in response to simulated inputs, including simulated device or data highway failures, and utilization of provided self-test functions.

e) Proper functioning of the rod control system is verified by evaluating response to simulated demands from the plant control system and protection and safety monitoring system, including group selection and interlocking functions.

f) Proper calibration and operation of the rod position indication system is demonstrated by evaluating system response to simulated rod control logic inputs, utilizing applicable displays, annunciators and alarms.

g) Proper operation of logic and controls for the pressurizer level and pressure control functions, including interlocks and equipment protective devices, is demonstrated by injecting simulated input signals representing anticipated pressurizer level and pressure transients.

14.2.9.2.13 Data Display and Processing System Testing

Purpose

The purpose of the data display and processing system testing is to verify that the as-installed components properly perform the following nonsafety-related defense-in-depth functions, described in Section 7.1:

- Display plant parameters for normal and emergency operations
- Provide plant alarm functions for normal and emergency plant operations
- Provide operational support for plant personnel, including computerized, interactive plant procedures
- Provide analysis, logging and historical storage and retrieval of plant data
- Provide a redundant communications network for transmission of plant parameters, plant status, displays, alarms and data files

Prerequisites

Related system interfaces are available or simulated as necessary to support the specified test configurations. Component testing and instrument calibrations have been completed. Programming has been completed and the initial software diagnostics tests have been determined acceptable. Required electrical power supplies are energized and operational. Required system interfaces are connected and available or simulated as necessary to support the specified test configurations. RN-14-110

General Test Methods and Acceptance Criteria

Performance of the data display and processing system hardware and software is observed and recorded during a series of individual component and integrated tests designed to verify that the data display and processing system equipment operates as described in Section 7.1 and the applicable design specifications:

a) Initial operation of installed devices is verified by completing the diagnostics tests provided for the components and equipment.

b) Proper operation of the data display and processing system software and hardware is demonstrated by utilizing the data display and processing system to provide the processing, monitoring, display and operator interface features required during preoperational testing of associated plant instrumentation and control systems.

c) Verification that the time periods associated with accessing displays, displaying data after it has been made available on the plant monitor bus and display refresh or update rates are within the maximum allowable times is demonstrated. This verification is performed while utilizing the data display and processing system to provide the processing, monitoring, display and operator interface features required during preoperational testing of associated plant instrumentation and control systems.

14.2.9.2.14 Diverse Actuation System Testing

Purpose

The purpose of the diverse actuation system preoperational testing is to verify that the as-installed components properly perform the following nonsafety-related defense-in-depth functions, described in Section 7.7:

- Provide diverse (from the safety-related protection and safety monitoring system) automatic actuation of the following:
  - Reactor/turbine trip
  - Passive residual heat removal heat exchanger
  - Core makeup tanks/reactor coolant pump trip
  - Passive containment cooling
  - Isolation of selected containment penetrations
- Provide a diverse, alternate means for manual actuation of reactor trip and engineered safety features functions
- Provide a diverse system for monitoring selected plant parameters used to provide guidance for manual operation and confirmation of reactor trip and selected engineered safety features actuation

Prerequisites

Related system interfaces are available or simulated as necessary to support the specified test configurations. Component testing and instrument calibrations have been completed. Programming has been completed and initial system diagnostics tests have been determined acceptable. Required electrical power supplies and control circuits are energized and operational. Required field wiring is electrically isolated to prevent operation of components controlled by the diverse actuation system. Exceptions are specifically identified in the preoperational test procedures if plant systems or components are to be operated during testing and these systems or components are to be properly aligned and have proper support systems operating prior to actuation of the particular system or component. Equipment or components that cannot be actuated without damage or upsetting the plant are isolated using the test switches provided by the Diverse Actuation System to block device actuation. Continuity of wiring up to the actuation equipment is verified. RN-14-110

General Test Methods and Acceptance Criteria

Performance of the diverse actuation system is observed and recorded during a series of individual component and integrated tests designed to verify operation of the system components. The following testing demonstrates that the system operates as described in Section 7.7 and applicable design specifications:

a) Processing of the analog and digital signals is verified by injecting reference signals and verifying the outputs at various locations in the system.

b) Correct outputs or actuation functions, for the automatic actuation logic mode, are verified by demonstrating that injection of simulated inputs for each of the specified actuation functions results in the proper output as indicated by contact operation, component actuation, or electrical test.

c) Correct outputs or actuation functions, for the manual actuation logic mode, are verified by demonstrating that each manual actuation function results in the proper output as indicated by contact operation, component actuation, or electrical test.

d) Proper operation of indications and alarms for the specified inputs, including those which provide reactor trip or engineered safety features actuation status, are verified by injecting simulated input signals.

14.2.9.2.15 Main AC Power System Testing

Purpose

The purpose of the main ac power system testing is to verify that the as-installed components properly perform the following nonsafety-related function:

- Provide ac electrical power to plant nonsafety-related loads as described in subsection 8.3.1; and the following nonsafety-related function:

- Provide onsite power for post-72 hour electrical requirements.

Prerequisites

The component tests for the individual components associated with the main ac power system have been completed. The required test instrumentation is properly calibrated and operational. Additionally, the plant offsite grid connection is complete and available. RN-14-110

General Test Methods and Acceptance Criteria

The capability of the main ac power system to provide power to plant loads under various plant operating conditions is verified. The system components to be tested include the ancillary diesel generator, the medium and low voltage power system, load centers, motor control centers, and instrumentation and controls. The following tests verify that the main ac power system provides its functions as specified in subsection 8.3.1 and appropriate design specifications:

a) Verify the operability of medium-voltage supply breakers.

b) Energize the diesel-backed buses from their associated onsite standby diesel-generator supplies. Verify the bus voltages are within design limits. This test can be performed in conjunction with the testing of the standby diesel generator.

c) Energize the medium voltage buses from their associated unit auxiliary transformer. Verify the bus voltages are within design limits.

d) Energize each medium voltage bus from the reserve auxiliary transformer. Verify the bus voltages are within design limits.

e) Operate the automatic and maintenance bus transfer schemes. Verify successful transfer and return operation.

f) Verify correct operation of the manual controls, annunciation, and instrumentation for the 480 V load centers and their 6900 V feeder breakers.

g) Simulate fault conditions at the 480 V load centers and verify alarms and operation of trip devices and protective relays.

h) Energize the 480 V load centers. Verify the bus voltages are within design limits.

i) Verify the operability of motor control center supply breakers.

j) Simulate fault conditions at the motor control centers and verify alarms and operation of trip devices and protective relays.

k) Energize the motor control centers. Verify the bus voltages are within design limits.

l) Start ancillary diesel generators, energize voltage regulating transformers. Verify the input voltages to the regulating transformers are within design limits.

14.2.9.2.16 Non-Class 1E dc and Uninterruptible Power Supply System Testing

Purpose

The purpose of the non-Class 1E dc and uninterruptible power supply system testing is to verify the ability to provide continuous, reliable power for the non-Class 1E control and instrumentation defense-in-depth loads.

Prerequisites

The component tests for the individual components associated with the non-Class 1E dc and uninterruptible power supply system have been completed. Permanently installed and test instrumentation are properly calibrated and operational. The 480 V ac system is in operation to supply power to the battery chargers. Additionally, a test load is available for the performance of battery capacity tests. RN-14-110

General Test Methods and Acceptance Criteria

The non-Class 1E dc and uninterruptible power supply system consists of electrical equipment including batteries, battery chargers, inverters, static transfer switches, and associated instrumentation and alarms that is used to supply power for the non-Class 1E control and instrumentation loads. Performance is observed and recorded during a series of individual component and integrated system tests. These tests verify that the non-Class 1E dc and uninterruptible power supply system operates as specified in subsection 8.3.2 and appropriate design specifications:

a) The capability of each non-Class 1E battery serving defense-in-depth loads is verified to meet or exceed the required ampere-hour rating by a battery performance test in accordance with IEEE 450. Following this discharge, the voltage of each cell is verified to be greater than or equal to the specified minimum cell voltage. RN-12-077

b) The capability of each charger serving defense-in-depth loads to meet the rating specified by Table 8.3.2-6 is verified. This testing includes a verification that the charger output voltage is within design limits. RN-12-077

c) The capability of each inverter to meet the rating specified by Table 8.3.2-6 is verified. This testing includes a verification that the output frequency and voltage to be within the limits specified in Table 8.3.2-6.

d) The proper operation and calibration of instrumentation and alarms, electrical ground detection, and permissive and prohibitive interlocks is verified.

14.2.9.2.17 Standby Diesel Generator Testing

Purpose

The purpose of the standby diesel generator testing is to verify the capability to provide electrical power to plant nonsafety-related loads that enhance an orderly plant shutdown if off-site ac power is not available.

Prerequisites

The component tests have been completed. The necessary permanently installed instrumentation is properly calibrated and operational. Appropriate electrical power sources and diesel generator building heating and ventilation system are available for use. The plant control system is available for operation as applicable to the diesel generators. Sufficient diesel fuel is available, on site or readily accessible, to perform the tests. RN-14-110

General Test Methods and Acceptance Criteria

Performance is observed and recorded during a series of individual component and integrated tests. These tests verify that the diesel generators operate properly as specified in Sections 8.3 and 9.5 through the following testing:

a) Verify the operability of generator protection features described in subsection 8.3.1.1.2.2.

b) Simulate the loss of ac voltage and verify proper operation of undervoltage relay. Verify sequencer control logic support the description in Tables 8.3.1-1 and 8.3.1-2.

c) Verify the diesel generators fuel transfer pumps start and stop automatically in response to simulated day tank low level and high level signals.

d) Transfer fuel oil from the fuel oil storage tank to the diesel fuel oil day tanks by means of the transfer pumps. Verify flow parameters are within design limits.

e) Verify proper operation of diesel generators building heating and ventilation system fans and dampers, manual and automatic controls, alarms, and indicating instruments, as described in subsection 9.4.10.

f) Verify the air flow in the diesel generator building heating and ventilation system is acceptable.

g) Verify the diesel generator lockout features (turning gear engaged, emergency stop).

h) Verify that the diesel generator air starting system has sufficient capacity for cranking the engine for prescribed number of automatic or manual starts without recharging.

i) Start the diesel generators. Verify voltage and frequency control.

j) Verify the full load-carrying capability for a period of not less than 24 hours, of which 2 hours at are at a load equivalent to the 2-hour (Standby) rating of the diesel generators and 22 hours at a load equivalent to the continuous rating of the diesel generators. Verify the voltage and frequency requirements are maintained. Verify that the diesel generator cooling system functions within design limits.

k) Following the full-load capability test, simulate loss of ac voltage and verify proper automatic startup, sequencing, and operation of the diesel generators. Verify diesel generators bus de-energization and load shedding. Verify diesel generators attain frequency and voltage within design limits within the time described in subsection 8.3.1.1.2.3. Verify sequencer control logic meets the description in Tables 8.3.1-1 and 8.3.1-2. Verify that the diesel generators continuous rating is not exceeded. Verify voltage and frequency requirements are maintained.

l) Verify that the rate of fuel consumption and the operation of the fuel transfer pumps and associated components, while providing power to the load equivalent to those specified in Table 8.3.1-1 or 8.3.1-2, are such that the design capacity of the fuel oil storage tanks meets the subsection 9.5.4 requirement for 7-day storage inventory.

m) With each diesel generator bus supplied only by the diesel generator and supplying loads up to its continuous rating, trip a load equivalent to the largest single load in Table 8.3.1-1 or 8.3.1-2. Verify that the voltage and frequency values are maintained within design limits.

n) With each diesel generator supplying loads up to its continuous rating, trip the generator breaker that supplies power to the diesel generator bus. Verify that the diesel engine continues to run and does not trip on overspeed.

14.2.9.2.18 Radiation Monitoring System Testing

Purpose

The purpose of the radiation monitoring system testing is to verify that the as-installed radiation monitors perform their defense-in-depth function as described in Section 11.5.

Prerequisites

The component testing of the radiation monitoring system has been completed. The radiation monitors have been calibrated and the monitor check sources are installed, as appropriate. The required preoperational testing of the protection and safety monitoring system, plant control system, the electrical power and distribution systems, and other interfacing systems required for operation and data collection is available as needed to support the specified testing. RN-14-110

General Test Acceptance Criteria and Methods

Radiation monitoring system performance is observed and recorded during a series of individual component and integrated system testing to verify the system performs its defense-in-depth functions. The following testing demonstrates that the system operates as specified in Section 11.5 and as specified in appropriate design specifications:

a) The proper calibration and operation of each radiation detector assembly and associated equipment using a standard radiation source or portable calibration unit are verified.

b) Proper operation of the monitoring equipment and controls required for manually initiated operation of the monitor check sources is verified.

c) Proper operation of the local processors that process and transmit radiation monitoring data to the protection and safety monitoring system or plant control system, as appropriate, is verified.

d) Proper actuation of alarms and signals for actuation of equipment responses following receipt of a high radiation signal is verified.

The preoperational testing discussed in subsection 11.5.7 is performed following successful completion of the testing described above.

14.2.9.2.19 Plant Lighting System Testing

Purpose

The purpose of plant lighting system testing is to verify that the system can perform its defense-in-depth function of providing emergency lighting in the main control room and remote shutdown workstation area to illuminate these areas for emergency operations upon loss of normal lighting, as described in subsection 9.5.3. In addition, the operability of lighting for emergency ingress and egress is verified.

Prerequisites

The component testing of the plant lighting system is completed. The required preoperational testing of the interfacing and support systems required for testing the emergency lighting function is available as needed to support the specified testing and system configurations including the Class 1E dc and uninterruptible power supply system, and the main ac power system. RN-14-110

General Test Acceptance Criteria and Methods

Plant lighting system performance is observed during a series of individual component and integrated system testing to verify the system capability to perform its defense-in-depth functions. The following testing verifies that the system operates as described in subsection 9.5.3 and in appropriate design specifications:

a) The proper operation of the plant lighting system emergency lighting is verified when powered from the Class 1E dc and uninterruptible power supply system.

b) Self-contained emergency lighting units are verified to be operable and installed into the proper ingress and egress paths, standby diesel generator rooms, switchgear rooms (annex and turbine buildings), fire pump rooms, access route between the main control room and remote shutdown workstation, and appropriate connecting corridors and stairwells.

14.2.9.2.20 Primary Sampling System Testing

Purpose

The purpose of the primary sampling system testing is to verify that the as installed components properly perform the following nonsafety-related defense-in-depth functions described in subsection 9.3.3:

- Provide the capability to obtain samples of the reactor coolant, passive core cooling system, containment sump water, and containment atmosphere
- Provide the capability to analyze and measure samples.

Prerequisites

Component testing of the primary sampling system has been completed. Component cooling water is being provided to the sample cooler when samples are taken from the reactor coolant system when it is at elevated temperature. The systems/components to be sampled are filled and at their normal pressure and temperature. The liquid radwaste system is available to receive discharged sample fluid. Electrical power is available for operation of the system components and a source of compressed gas is available for operation of the gas sample eductor. RN-14-110

General Test Method and Acceptance Criteria

The performance of the primary sampling system is observed and recorded during a series of individual component tests and testing in conjunction with the reactor coolant system and passive core cooling system operation. The following testing demonstrates that the primary sampling system performs its defense-in-depth functions as described in subsection 9.3.3 and appropriate design specifications.

a) Proper operation of the system's remotely-operated valves and eductor supply pump is verified.

b) Proper calibration and operation of instrumentation, controls, actuation signals, and interlocks are verified.

c) Verify the capability to obtain samples from the reactor coolant system, core makeup tanks, accumulators, containment sump, and containment atmosphere.

d) Verify the ability to return the sample stream fluid to the containment sump or liquid radwaste system, as appropriate.

e) Verify the capability to route sample streams to the laboratory.

f) Verify the operability of the test laboratory equipment used to analyze or measure radiation levels and radioactivity concentrations.

14.2.9.2.21 Annex/Auxiliary Building Nonradioactive HVAC System

Purpose

The purpose of the annex/auxiliary non-radioactive HVAC system testing is to verify that the as installed system properly performs the defense-in-depth function, as described in subsection 9.4.2, to provide conditioned air to maintain the diesel bus switchgear rooms and battery charger rooms (containing DC switchgear) within their design temperature range during operation of the onsite standby power system.

Prerequisites

The component testing of the annex/auxiliary building HVAC system has been successfully completed. The required preoperational testing of the interfacing systems required for the operation of the above system is completed and these systems are available as needed to support the specified testing and system configurations. RN-14-110

General Test Acceptance Criteria and Methods

The annex/auxiliary building non-radioactive HVAC system performance is observed and recorded during a series of individual component and integrated system testing. The following testing verifies that the system functions as described in subsection 9.4.2 and appropriate design specifications:

a) Proper function of the fans, filters, and dampers is verified.

b) Proper operation of instrumentation, controls, actuation signals, and alarms and interlocks is verified. This testing includes the following:

- Air handling unit and fan flows, controls, and alarms
- Air temperatures, alarms, and controls
- Damper open, close and modulate control in response to monitored parameters

This testing includes operation from the main control room.

c) The ventilated areas are verified to be maintained at a slightly positive pressure relative to the outside air pressure and other areas of the auxiliary building.

d) The switchgear and equipment room subsystem air handling unit supply and return fans are verified to be automatically connected to the onsite standby power supplies on a loss of power to the buses powered by the standby diesels.

14.2.9.2.22 Pressurizer Surge Line Testing (First Plant Only)

Purpose

The purpose of the pressurizer surge line testing is: a) to obtain data to verify the proper operation of temperature sensors installed on the pressurizer surge line and pressurizer spray line, and b) to obtain Reactor Coolant System piping displacement measurements for baseline data, as described in Subsections 3.9.3, 14.2.5, and 14.2.9.1.7 item (d).

Prerequisites

The component tests for the individual components associated with the Reactor Coolant System have been completed. The testing and calibration of the required test instrumentation has been completed. The temporary sensors and instrumentation lead wires required for monitoring thermal stratification, cycling, and striping have been installed. The calibration of the transducers and the operability of the data acquisition equipment have been verified. Prior to testing of the piping system, a pretest walk-down shall be performed to verify that the anticipated piping movement is not obstructed by objects not designed to restrain the motion of the system (including instrumentation and branch lines). The system walk-down shall also verify that supports are set in accordance with the design. RN-14-110

General Test Methods and Acceptance Criteria

The performance of the Reactor Coolant System is observed and recorded during a series of individual tests that characterize the various modes of system operation. This testing verifies that the temperature sensors operate as described in subsection 3.9.3 and in appropriate design specifications.

a) Verify the proper operation of temperature sensors installed on the pressurizer surge line and pressurizer spray line.

b) Record sensor data at specified intervals throughout hot functional testing of the RCS system, including during the drawing and collapsing of the bubble in the pressurizer.

c) Retain the following plant parameters time history for the same data recording period:

- Hot leg temperature
- Reactor Coolant System pressure
- Reactor coolant pump status
- Pressurizer level
- Pressurizer temperature (liquid and steam)
- Pressurizer spray temperature

- Pressurizer spray and auxiliary spray flow
- Normal residual heat removal system flow rate
- Passive core cooling system – passive residual heat removal flow rate.

d) Monitor pressurizer surge line and pressurizer spray line for valve leakage.

e) Remove the transducers and associated hardware after the completion of testing.

f) Proper operation of the temperature sensors in the pressurizer surge and spray lines is verified.

14.2.9.3 Preoperational Testing of Nonsafety-Related Radioactive Systems

14.2.9.3.1 Liquid Radwaste System Testing

Purpose

The purpose of the liquid radwaste system testing is to verify that the as-installed components and associated piping, valves, and instrumentation properly perform the following safety-related function described in subsection 11.2.1.1: RN-13-030

- Prevent back flow through the drain lines from the containment sump to the chemical and volume control system compartment and the passive core cooling system compartments, in order to prevent cross flooding of these compartments

The liquid radwaste system testing is performed to verify that the as-installed components and associated piping, valves, and instrumentation properly perform the nonsafety-related functions described in subsection 11.2.1.2, including receiving and processing reactor coolant system effluents, radioactive equipment and floor drains, and other radioactive liquid wastes from the plant.

Prerequisites

The component testing of the liquid radwaste system is completed. The required preoperational testing of the interfacing and support systems required for testing has been completed. Data collection is available as needed to support the specified testing and system configurations. RN-14-110

General Test Acceptance Criteria and Methods

Liquid radwaste system performance is observed and recorded during a series of individual component and system testing that characterizes system operation. This testing verifies that the system operates as specified in Section 11.2 and appropriate design specifications.

a) The drain lines from the passive core cooling system compartments and the refueling cavity are verified to provide a flow path to the reactor compartment.

b) Proper operation of the backflow prevention check valves is verified by the performance of baseline in-service tests, as specified in subsection 3.9.6.

c) Proper operation of the system pumps and valves is verified, including:

- Effluent holdup tank pumps
- Waste holdup tank pumps
- Degasifier separator pumps
- Chemical waste tank pump
- Monitor tank pumps

- Reactor coolant drain tank pumps

d) Proper calibration and operation of the system instrumentation, controls, actuation signals, and interlocks are verified, including:

- Pump controls and alarms
- Tank level control and alarms
- Valve and pump responses to safeguards signals
- Valve and pump responses to high radiation isolation signals

e) In conjunction with the gaseous radwaste system testing in subsection 14.2.9.3.2, the proper operation of the degasifier is verified.

f) The proper operation of the liquid radwaste filters and ion exchangers is verified.

14.2.9.3.2 Gaseous Radwaste System Testing

Purpose

The purpose of the gaseous radwaste system testing is to verify that the as-installed components and associated piping, valves, and instrumentation properly perform the following nonsafety-related functions described in Section 11.3.

- Collect waste gases that contain radioactivity or hydrogen
- Provide holdup for radioactive waste gases as appropriate

Prerequisites

The component testing of the gaseous radwaste system is completed. The required preoperational testing of the interfacing and support systems required for testing is completed, and data collection is available as needed to support the specified testing and system configurations. In addition, a source of hydrogen and calibration gases is available. RN-14-110

General Test Acceptance Criteria and Methods

The performance of the gaseous radwaste system is observed and recorded during a series of individual component and system tests that characterizes the various modes of system operation. This testing verifies that the gaseous radwaste system operates as described in Section 11.3 and appropriate design specifications:

a) System and component control circuits, including response to normal control, interlock, and alarm signals are verified. The gaseous radwaste system instrumentation, controls, valves, and interlocks are verified to respond to various inputs and provide proper isolation and alarm signals. Appropriate automatic control functions are verified in response to abnormal conditions inputs.

b) Nitrogen, hydrogen, and calibration gases are routed through the system. Performance characteristics of the instrumentation and control systems are verified, and the delay bed operation is verified.

c) Moist test gas is routed through the system to verify proper moisture removal and detection.

d) The degasifer vacuum pump is verified to operate properly. Manual override of the automatic control functions of the drainpot and moisture separator drain and isolation valves is verified.

e) Sample pumps are operated and the sample flow meter indication is observed.

f) The proper operation of the degasifier moisture separator is demonstrated.

14.2.9.3.3 Solid Radwaste System Testing

Purpose

The purpose of the solid radwaste system testing is to verify that the as-installed components and associated piping, valves, and instrumentation operate properly to prepare waste generated during the normal operation of the plant for processing, packaging, and shipment as described in subsection 11.4.1.2.

Prerequisites

The component testing of the solid radwaste system is completed. The interfacing and support systems required for testing and data collection are available as needed to support the specified testing and system configurations. RN-14-110

General Test Method and Acceptance Criteria

The performance of the solid radwaste system is observed and recorded during a series of individual component and system tests that characterizes the various modes of system operation. This testing verifies that the solid radwaste system operates as described in Section 11.4 and in appropriate design specifications:

a) Tests are performed to verify that manual and automatic system controls, alarms, and instruments are functional; the system instrumentation, controls, valves, and interlocks respond properly to various inputs and provide proper isolation and alarm signals; and appropriate automatic control functions occur in response to abnormal condition inputs.

b) Tests are performed to verify proper system process rates as described in Section 11.4, and that no free liquids are present in packaged waste.

c) The capability to properly transfer and retain spent resins is verified.

d) The capability to properly handle filter cartridges in a manner that minimizes personnel radiation exposure is demonstrated.

14.2.9.3.4 Radioactive Waste Drain System Testing

Purpose

The purpose of the radioactive waste drain system testing is to verify that the as-installed components and associated piping, valves, and instrumentation properly perform the following functions, described in Section 11.2 and subsection 9.3.5:

- Drain floor and equipment compartments
- Collect drainage and transfer drainage to the liquid radwaste system

Prerequisites

The component testing of the radioactive waste drain system is completed. The interfacing and support systems required for testing and data collection are available as needed to support the specified testing and system configurations, including the liquid radwaste system and compressed air supply. RN-14-110

General Test Method and Acceptance Criteria

The performance of the radioactive drain system is observed and recorded during a series of individual component and system tests that characterizes the various modes of system operation. This testing verifies that the system operates as described in Section 11.2, subsection 9.3.5, and in appropriate design specifications:

a) Proper operation of system instrumentation, controls, alarms, and interlocks is verified.

b) Proper operation of the system pumps and valves is verified.

c) Proper system and component flow paths and flowrates, including pump capacities and sump tank volumes, is verified.

d) Flow water in each drain path to verify that the drains discharge to their designated destination and that proper drain path segregation is maintained.

14.2.9.3.5 Steam Generator Blowdown System Testing

Purpose

The purpose of the steam generator blowdown system testing is to verify that the as-installed components and associated piping, valves, and instrumentation operate properly to provide an isolatable flow path for the controlled removal of water from the secondary side of the steam generators as described in Section 10.4.

Prerequisites

The component testing of the steam generator blowdown system is completed. The interfacing and support systems required for testing and data collection are available as needed to support the specified testing and system configurations. A portion of this testing is performed during the hot functional testing of the plant, when the steam generators are at or near normal operating pressure and temperature. RN-14-110

General Test Method and Acceptance Criteria

The performance of the steam generator blowdown system is observed and recorded during a series of individual component and system tests that characterize the various modes of system operation. This testing demonstrates that the system operates as described in Section 10.4 and in appropriate design specifications:

a) Proper operation of system instrumentation, controls, alarms, and interlocks is verified.

b) Proper operation of the system pump and valves is verified.

c) The proper operation of the electrodeionization units is verified.

d) The heat transfer capability of each blowdown heat exchanger is verified.

e) The automatic isolation of steam generator blowdown on low steam generator level is verified.

14.2.9.3.6 Waste Water System Testing

Purpose

The purpose of the waste water system testing is to verify that the as-installed components and associated piping, valves, and instrumentation operate properly to collect and perform appropriate processing of normally non-radioactive drains, as described in Section 11.2 and subsection 9.2.9.

Prerequisites

The component testing of the waste water system is completed. The interfacing and support systems required for testing and data collection are available as needed to support the specified testing and system configurations. RN-14-110

General Test Acceptance Criteria and Methods

Waste water system performance is observed and recorded during a series of individual component and system testing that characterizes system operation. This testing verifies that the system operates as described in Section 11.2 and subsection 9.2.9 and appropriate design specifications.

a) Proper operation of the system pumps and valves is verified.

b) Proper calibration and operation of the system instrumentation, controls, actuation signals, and interlocks is verified.

c) Proper system and component flow paths and flowrates, including pump capacities and sump tank volumes is verified.

d) Verify the ability of the waste water system radiation alarm to stop the turbine building sump pumps. RN-15-035

14.2.9.4 Preoperational Tests of Additional Nonsafety-Related Systems

14.2.9.4.1 Condensate System Testing

Purpose

The purpose of the condensate system testing is to verify that the as-installed components properly perform the system functions, described in subsection 10.4.7, of delivering the required flow of heated water from the condenser hotwell to the feedwater system.

Prerequisites

The component testing of the condensate system has been completed. The component testing of the condenser is completed and a source of water of appropriate quality is available for filling the condenser hotwell. The steam generator feedwater system is available to receive flow from the condensate system. Required electrical power supplies and control circuits are operational. RN-14-110

General Test Method and Acceptance Criteria

Condensate system performance is observed and recorded during a series of individual component and integrated system testing. The following testing verifies that the condensate system can perform its functions as described in subsection 10.4.7 and appropriate design specifications:

a) Proper operation of the condensate pumps and system valves is verified.

b) Proper calibration and operation of the system instrumentation, controls, actuation signals, and interlocks are verified.

c) Proper operation of the heater drains is verified.

d) During the plant hot functional testing, the integrated operation of the condensate system in conjunction with the feedwater system is verified with the condenser and circulating water system in operation.

14.2.9.4.2 Condenser Air Removal System Testing

Purpose

The purpose of the condenser air removal system testing is to verify that the as-installed components properly perform the system functions to establish and maintain the required vacuum in the main condenser, as described in subsection 10.4.2.

Prerequisites

The component testing of the condenser air removal system has been completed. The component testing of the condenser has been completed and a source of water of appropriate quality is available for filling the condenser hotwell. The turbine gland sealing system and exhaust blower are in operation. A source of steam such as the auxiliary boiler is available. Required support systems, electrical power supplies and control circuits are operational. RN-14-110

General Test Method and Acceptance Criteria

Condenser air removal system performance is observed and recorded during a series of individual component and integrated system testing. The following testing verifies that the condensate system can perform its functions as described in subsection 10.4.2 and appropriate design specifications:

a) Proper operation of the vacuum pumps and system valves is verified.

b) Proper calibration and operation of the system instrumentation, controls, actuation signals, and interlocks are verified.

c) The capability of the vacuum pumps to establish the required vacuum in the main condenser is verified.

14.2.9.4.3 Main Turbine System and Auxiliaries Testing

Purpose

The purpose of the main turbine system testing is to verify that the as-installed main turbine and its auxiliary components properly perform their functions, described in Sections 10.2 and 10.4. This testing includes testing of the turbine gland sealing system, lube oil system, turning gear, turbine controls and protective functions, and moisture separator reheater.

Prerequisites

The component testing of the main turbine and its auxiliaries has been completed. The component testing of the condenser is completed and a source of water of appropriate quality is available for filling the condenser hotwell. The main turbine is on turning gear and the condenser air removal system is operable. A source of steam such as the auxiliary boiler is available. Required support systems, electrical power supplies and control circuits are operational. RN-14-110

General Test Method and Acceptance Criteria

Because this testing is performed using a temporary steam source, the extent to which the turbine can be tested in preoperational testing is limited. However, the proper function of the turbine auxiliaries is verified to assure the turbine will operate properly when a greater amount of steam is provided.

Main turbine system performance is observed and recorded during a series of individual component and integrated system testing. The following testing verifies that the turbine and its auxiliaries function as described in Sections 10.2 and 10.4 and in appropriate design specifications:

a) Proper operation of the turbine lube oil pump and turning gear motor, gland seal exhaust blower, and moisture separator and gland seal valves is verified.

b) Proper operation of system valves including the turbine control and intercept valves is verified.

c) Proper calibration and operation of the system instrumentation, controls, actuation signals, and interlocks are verified.

d) Proper turbine operation during the turning gear testing is verified. The turning gear engagement and disengagement functions are verified to operate properly.

e) Proper performance of the turbine trip functions is verified.

14.2.9.4.4 Main Generator System and Auxiliaries Testing

Purpose

The purpose of the main generator system testing is to verify that the as-installed main generator and its auxiliary components properly perform their functions, described in Sections 8.2 and 10.2. This testing includes testing of the generator cooling systems, lube oil system, controls, and protective functions.

Prerequisites

The component testing of the main generator and its auxiliaries has been completed. The component testing of the condenser is completed. The turbine cooling water system is operable, and required support systems, electrical power supplies, and control circuits are operational. RN-14-110

General Test Method and Acceptance Criteria

Performance is observed and recorded during a series of individual component and integrated tests. These tests verify that the generator operated as specified in Sections 8.2 and 10.2 through the following testing:

a) Verify the operability of the generator protection features.

b) Verify proper cooling of the generator stator and rotor.

c) Verify MW, MVAR, and frequency control.

14.2.9.4.5 Turbine Building Closed Cooling Water System Testing

Purpose

The purpose of the turbine building closed cooling water system testing is to verify that the as-installed components properly perform their functions of supplying adequate cooling water to the designated turbine building components, as described in subsection 9.2.8.

Prerequisites

The component testing of the turbine building closed cooling water system has been completed. The cooled components are operational and operating to the extent possible, especially for verifying the heat exchanger capability. Required support systems, electrical power supplies and control circuits are operational. RN-14-110

General Test Method and Acceptance Criteria

Turbine building closed cooling water system performance is observed and recorded during a series of individual component and integrated system testing. The following testing verifies that the system functions as described in subsection 9.2.8 and appropriate design specifications:

a) Proper operation of the system pumps and valves is verified.

b) Proper operation of the system instrumentation, controls, actuation signals, and interlocks is verified.

14.2.9.4.6 Circulating Water System Testing

Purpose

The purpose of the circulating water system testing is to verify that the as-installed components properly perform the functions of cooling and circulating adequate cooling water to the main condenser and turbine building closed cooling water system heat exchangers as described in subsection 10.4.5.

Prerequisites

The component testing of the circulating water system has been completed. The main condenser and turbine building closed cooling water heat exchangers are operational. Required support systems, electrical power supplies and control circuits are operational. RN-14-110

General Test Method and Acceptance Criteria

Since there will be little, if any, heat rejected to the circulating water system, verification of the heat removal capability of the ultimate heat sink is performed during the startup testing of the plant when the reactor is producing power.

Circulating water system performance is observed and recorded during a series of individual component and integrated system testing. The following testing verifies that the system functions as described in subsection 10.4.5 and appropriate design specifications:

a) Proper operation of the system pumps and valves is verified.

b) Proper calibration and operation of the system instrumentation, controls, actuation signals, and interlocks are verified.

The proper operation of the system freeze protection equipment is verified, as applicable.

14.2.9.4.7 Turbine Island Chemical Feed System Testing

Purpose

The purpose of the turbine island chemical feed system testing is to verify that the as-installed components properly perform the functions of adding appropriate chemicals to the condensate, service water, and auxiliary boiler in a controlled manner, as described in subsection 10.4.11.

Prerequisites

The component testing of the chemical feed system has been completed. Required support systems, electrical power supplies and control circuits are operational. RN-14-110

General Test Method and Acceptance Criteria

Turbine island chemical feed system performance is observed and recorded during a series of individual component and integrated system testing. The following testing verifies that the system functions as described in subsection 10.4.11 and appropriate design specifications:

a) Proper operation of the system pumps and valves is verified.

b) Proper calibration and operation of the system instrumentation, controls, actuation signals, and interlocks are verified.

14.2.9.4.8 Condensate Polishing System Testing

Purpose

The purpose of the condensate polishing system testing is to verify that the as-installed components properly perform the functions of removing corrosion products, dissolved solids, and other impurities from the condensate system, as described in subsection 10.4.6.

Prerequisites

The component testing of the condensate polishing system has been completed. The ultimate heat sink water reservoir is filled with water of appropriate quality and the condensate and feedwater systems are operational. Required support systems, electrical power supplies and control circuits are operational. RN-14-110

General Test Method and Acceptance Criteria

Condensate polishing system performance is observed and recorded during a series of individual component and integrated system testing. The following testing verifies that the system functions as described in subsection 10.4.6 and appropriate design specifications:

a) Proper operation of the system valves is verified.

b) Proper calibration and operation of the system instrumentation, controls, actuation signals, and interlocks are verified.

14.2.9.4.9 Demineralized Water Transfer and Storage System Testing

Purpose

The purpose of the demineralized water transfer and storage system testing is to verify that the as-installed components properly perform the function of providing reservoirs of demineralized water and deliver deoxygenated, demineralized water to various plant users, as described in subsection 9.2.4.

Prerequisites

The component testing of the demineralized water transfer and storage system has been completed. The demineralized water treatment system is operational and the equipment which uses demineralized water is able to accept water. Required support systems, electrical power supplies and control circuits are operational. RN-14-110

General Test Method and Acceptance Criteria

Demineralized water transfer and storage system performance is observed and recorded during a series of individual component and integrated system testing. The following defense-in-depth testing verifies that the system functions as described in subsection 9.2.4 and appropriate design specifications:

a) Proper operation of the system pumps, valves, blower, and is verified.

b) Proper calibration and operation of the system instrumentation, controls, actuation signals, and interlocks are verified.

14.2.9.4.10 Compressed and Instrument Air System Testing

Purpose

The purpose of the compressed and instrument air system testing is to verify that the as-installed components properly perform the functions of providing compressed air at the required pressures to various plant users, as described in the Compressed and Instrument Air portion of Section 9.3.

Prerequisites

The component testing of the compressed and instrument air system has been completed. The component cooling water system is operational and providing cooling for the compressor units. Required support systems, electrical power supplies and control circuits are operational. RN-14-110

General Test Method and Acceptance Criteria

Compressed and instrument air system performance is observed and recorded during a series of individual component and integrated system testing. The following testing verifies that the system and its plant users, where applicable, function as described in subsection 9.3.1.4 and appropriate design specifications:

a) Proper operation of the system compressors, receivers, prefilters, air dryers, afterfilters, purifiers, and valves is verified.

b) Proper calibration and operation of the system instrumentation, controls, actuation signals, and interlocks are verified.

c) Integral testing is performed to verify that the instrument air subsystem can provide sufficient air pressure to accommodate the maximum number of air-operated valves expected to operate simultaneously.

d) Testing is performed to verify the fail-safe positioning of safety-related air-operated valves for sudden loss of instrument air or gradual loss of pressure as described in subsection 9.3.1.4.

e) Proper calibration is verified for system relief valves that protect the system from overpressure conditions.

14.2.9.4.11 Containment Recirculation Cooling System Testing

Purpose

The purpose of the containment recirculation cooling system testing is to verify that the as-installed components properly perform the functions of maintaining the proper containment air temperature during normal plant operation and during refueling and maintenance operations, as described in subsection 9.4.6.

Prerequisites

The component testing of the containment recirculation cooling system has been completed. The central chilled water system and hot water heating system are operational. Required support systems, electrical power supplies and control circuits are operational. RN-14-110

General Test Method and Acceptance Criteria

Containment recirculation cooling system performance is observed and recorded during a series of individual component and integrated system testing. The following testing verifies that the system functions as described in subsection 9.4.6 and appropriate design specifications:

a) Proper operation of the system fans and dampers is verified.

b) Proper calibration and operation of the system instrumentation, controls, actuation signals, and interlocks are verified.

14.2.9.4.12 Containment Air Filtration System Testing

Purpose

The purpose of the containment air filtration system testing is to verify that the as-installed components properly perform the functions of supplying and exhausting air to maintain the proper containment air pressure, and filter exhaust air to minimize radiation release, as described in subsection 9.4.7.

Prerequisites

The component testing of the containment air filtration system has been completed. The portions of the radiologically controlled area ventilation system connected to the air filtration system are operational. The hot water heating and chilled water systems are required for verification of the air filtration heating and cooling functions. Required support systems, electrical power supplies and control circuits are operational. RN-14-110

General Test Method and Acceptance Criteria

Containment air filtration system performance is observed and recorded during a series of individual component and integrated system testing. The following testing verifies that the system functions as described in subsection 9.4.7 and appropriate design specifications:

a) Proper operation of the system fans and dampers is verified.

b) Proper calibration and operation of the system instrumentation, controls, actuation signals, and interlocks are verified.

c) Proper operation of the containment air filtration filters is verified.

14.2.9.4.13 Plant Communications System Testing

Purpose

The purpose of the plant communications system testing is to verify that the as-installed components properly perform the functions of verifying the proper operation and adequacy of the plant communication systems used during normal and abnormal operations, as described in Section 9.5.

Prerequisites

The component testing of the communication system has been completed. Required support systems, electrical power supplies and control circuits are operational. RN-14-110

General Test Method and Acceptance Criteria

Plant communications system performance is observed and recorded during a series of individual component and integrated system testing. The inplant communications system includes the following subsystems:

- Wireless telephone system
- Telephone/page system
- Private Automatic Branch Exchange (PABX) System
- Sound Powered Phone System
- Emergency Offsite Communication System
- Security Communication System

The following testing verifies that the system functions as described in Section 9.5 and appropriate design specifications:

a) Transmitters and receivers are verified to operate without excessive interference.

b) Proper operation of controls, switches, and interfaces is verified.

c) Proper operation of the public address, including the plant emergency alarms, is verified.

d) The proper operation of equipment expected to function under abnormal conditions such as a loss of electrical power, shutdown from outside the control room, or execution of the plant emergency plan is verified. This functional testing will be performed under conditions that simulate the maximum plant noise levels being generated during the various operating conditions, including fire and accident conditions, to demonstrate system capabilities.

14.2.9.4.14 Mechanical Handling System Crane Testing

Purpose

The purpose of the mechanical handling system crane testing is to verify that the as-installed components properly perform their functions. The test ensures operation and adequacy of the containment polar crane, which is used to lift and relocate components providing access to the reactor fuel, vessel internals, and reactor components during refueling and servicing operations.

In addition, the following load handling systems described in subsection 9.1.5 are tested; the equipment hatch hoist, the maintenance hatch hoist, and the cask handling crane.

Prerequisites

The component testing of the heavy lift cranes has been completed. Required support systems, electrical power supplies and control circuits are operational. The heavy load analysis, defining the load paths, has been completed. RN-14-110

General Test Method and Acceptance Criteria

Heavy load crane performance is observed and recorded during a series of individual component and integrated system testing. The following testing verifies that the crane systems function as described in subsection 9.1.5 and in appropriate design specifications:

a) Proper operation and assembly of the various cables, grapples, and hoists including brakes, limit switches, load cells, and other equipment protective devices are verified.

b) Proper operation of control, instrumentation, interlocks, and alarms is verified.

c) Dynamic and static load testing of cranes and hoists, and associated lifting and rigging equipment are performed including a static load test at 125 percent of rated load and full operational test at 100 percent of rated load.

14.2.9.4.15 Seismic Monitoring System Testing

The seismic monitoring system testing described in this section also applies to site-specific seismic sensors.

Purpose

The purpose of the seismic monitoring system testing is to verify that the as-installed components properly perform the functions of verifying proper operation in response to a seismic event, as described in Section 3.7.

Prerequisites

The component testing of the seismic monitoring system has been completed. Required support systems, electrical power supplies and control circuits are operational. RN-14-110

General Test Method and Acceptance Criteria

Seismic monitoring system instrumentation performance is observed and recorded during a series of individual component and integrated system testing. The following testing verifies that the system functions as described in Section 3.7 and appropriate design specifications:

a) Proper calibration and response of seismic instrumentation are verified, including verification of alarm and initiation setpoints.

b) Proper operation of internal calibration and test features are verified.

c) Proper integrated system response, including actuations, alarms, and annunciations, is verified.

d) Verify the proper operation of the recording and analysis functions on a loss of AC power sourced.

14.2.9.4.16 Special Monitoring System Testing

Purpose

The purpose of the special monitoring system testing is to verify that the as-installed components properly perform the following nonsafety-related functions, described in subsection 4.4.6:

- Detect the presence of metallic debris in the reactor coolant system
- Obtain baseline data for metal impact monitoring prior to power operations

Prerequisites

Related system interfaces are available or simulated as necessary to support the specified test configurations. Component testing and instrument calibrations have been completed. Programming has been completed and initial system diagnostics tests have been determined acceptable. Required electrical power supplies are energized and operational. RN-14-110

General Test Methods and Acceptance Criteria

Performance of the special monitoring system is observed and recorded during a series of individual component and integrated tests designed to verify system operation in response to specified input conditions. The following testing demonstrates that the system operates as described in subsection 4.4.6 and the applicable design specifications:

a) Proper calibration and response of digital metal impact monitoring instrumentation are verified.

b) Proper operation of the digital metal impact monitoring system is verified by evaluating system response to simulated input signals representing the anticipated signal range.

c) Baseline response data is obtained for the metal impact monitoring system to serve as a reference for monitoring degradation of sensor response.

14.2.9.4.17 Secondary Sampling System Testing

Purpose

The purpose of the secondary sampling system testing is to verify that the as-installed components properly perform the following nonsafety-related functions, described in subsection 9.3.4:

- Provide the capability to continuously or semi-continuously monitor selected secondary water and steam process streams in order to establish and maintain proper water chemistry during plant operation RN-14-080

- Provide the capability to manually analyze additional secondary water and steam process streams

Prerequisites

Component testing of the secondary sampling system has been completed. Cooling water is being provided to the sample coolers when samples are taken from sample points with fluid temperatures exceeding 125°F. The systems/components to be sampled are filled and operating at their normal pressure and temperature. Electrical power is available for operation of the on-line chemistry analyzers. RN-14-110

General Test Method and Acceptance Criteria

The performance of the secondary sampling system is observed and recorded during a series of individual component tests and testing in conjunction with the plant in operation at normal pressure and temperature. The following testing verifies that the secondary sampling system operates as described in subsection 9.3.4 and appropriate design specifications.

a) Proper calibration and operation of on-line continuous and semi-continuous analyzers, data collection and display, controls, and actuation signals to the turbine island chemical feed system are verified. RN-14-080

b) Proper calibration and operation of the portable analyzer are verified.

c) Proper operation of the sample coolers is verified.

d) Capability to obtain grab samples from the sample points is verified.

14.2.9.4.18 Turbine Building Ventilation System

Purpose

The purpose of the turbine building ventilation system testing is to verify that the as installed system properly performs the normal air conditioning and ventilation functions, as described in subsection 9.4.9.

Prerequisites

The component testing of the turbine building ventilation system has been successfully completed. RN-14-110 The required preoperational testing of the central chilled water and hot water heating systems, and other interfacing systems required for the operation of the above systems and data collection is completed and these systems are available as needed to support the specified testing and system configurations.

General Test Acceptance Criteria and Methods

The turbine building ventilation system performance is observed and recorded during a series of individual component and integrated system testing. The following testing verifies that the system functions as described in subsection 9.4.9 and appropriate design specifications:

a) Proper function of the fans, filters, heaters, coolers, and dampers is verified.

b) Proper operation of instrumentation, controls, actuation signals, and alarms and interlocks is verified. This testing includes the following:

- Air handling unit and fan flows, controls, and alarms
- Damper open, close and modulate control

This testing includes operation from the main control room.

14.2.9.4.19 Health Physics and Hot Machine Shop HVAC System

Purpose

The purpose of the health physics and hot machine shop HVAC system testing is to verify that the as installed system properly performs the normal air conditioning and ventilation functions, as described in subsection 9.4.11.

Prerequisites

The component testing of the health physics and hot machine shop HVAC system has been successfully completed. The required preoperational testing of the central chilled water and hot water heating systems, and other interfacing systems required for the operation of the above systems is completed and these systems are available as needed to support the specified testing and system configurations. RN-14-110

General Test Acceptance Criteria and Methods

The health physics and hot machine shop HVAC system performance is observed and recorded during a series of individual component and integrated system testing. The following testing verifies that the system functions as described in subsection 9.4.11 and appropriate design specifications:

a) Proper function of the fans, filters, heaters, coolers, and dampers is verified.

b) Proper operation of instrumentation, controls, actuation signals, and alarms and interlocks is verified. This testing includes the following:

- Radiation detectors and alarms
- Air handling unit and fan flows, controls, and alarms
- Air temperatures, alarms, and controls
- Differential air pressure and alarms
- Damper open, close and modulate control

This testing includes operation from the main control room.

c) The health physics and hot machine shop HVAC system is verified to maintain the access control area and hot machine shop at a slightly negative pressure with respect to outdoors and clean areas of the annex building to prevent unmonitored releases of radioactive contaminants.

14.2.9.4.20 Radwaste Building HVAC System

Purpose

The purpose of the radwaste building HVAC system testing is to verify that the as installed system properly performs the normal air conditioning and ventilation functions, as described in subsection 9.4.8, as required for personnel and equipment in serviced areas; and provides the proper filtration of air from potentially contaminated areas.

Prerequisites

The component testing of the radwaste building HVAC system has been successfully completed. The required preoperational testing of the central chilled water and hot water heating systems, the ac electrical power and distribution systems, and other interfacing systems required for the operation of the above systems is completed and these systems are available as needed to support the specified testing and system configurations. RN-14-110

General Test Acceptance Criteria and Methods

The radwaste building HVAC system performance is observed and recorded during a series of individual component and integrated system testing. The following testing verifies that the system functions as described in subsection 9.4.8 and appropriate design specifications:

a) Proper function of the fans, filters, heaters, coolers, and dampers is verified.

b) Proper operation of instrumentation, controls, actuation signals, and alarms and interlocks is verified. This testing includes the following:

- Radiation detectors and alarms
- Air handling unit and fan flows, controls, and alarms
- Air temperatures, alarms, and controls
- Differential air pressures and alarms
- Damper open, close and modulate control in response to monitored parameters

This testing includes operation from the main control room.

c) The radwaste building is verified to be maintained at a slightly negative pressure with respect to outdoors to prevent unmonitored releases of radioactive contaminants.

14.2.9.4.21 Main, Unit Auxiliary and Reserve Auxiliary Transformer Test

Purpose

The purpose of the main, unit auxiliary and reserve auxiliary transformer testing is to demonstrate the energization of the transformers and the proper operation of associated protective relaying, alarms, and control devices.

Prerequisites

The component tests for the individual components associated with the main, unit auxiliary and reserve auxiliary transformers have been completed. The required test instrumentation is properly calibrated and operational. Additionally, the plant offsite grid connection is complete and available. RN-14-110

General Test Methods and Acceptance Criteria

The following tests demonstrate proper energization of the main, unit auxiliary and reserve auxiliary transformers and proper operation of protective relaying, alarms, and control devices associated with the transformers:

a) Energize the unit auxiliary transformers. Verify phase rotation. Verify phase voltages are within design limits.

b) Energize the reserve auxiliary transformers. Verify phase rotation. Verify phase voltages are within design limits.

c) Simulate fault conditions and verify alarms and operation of protective relaying circuits.

14.2.9.4.22 Storm Drains

Purpose

Storm drain system testing verifies that the drains prevent plant flooding by diverting storm water away from the plant, as described in Section 2.4.

Prerequisites

Construction of the storm drain system is completed, and the system is operational.

General Test Methods and Acceptance Criteria

The storm drain system is visually inspected to verify the flow path is unobstructed. The system is observed under simulated or actual precipitation events to verify that the runoff from roof drains and the plant site and adjacent areas does not result in unacceptable soil erosion adjacent to, or flooding of, Seismic Category I structures.

14.2.9.4.23 Off-site AC Power Systems

Purpose

Off-site alternating current (ac) power system testing demonstrates the energization and proper operation of the as-installed switchyard components, as described in Section 8.2.

Prerequisites

Component testing of plant off-site ac power systems, supporting systems, and components is completed. The components are operational and the switchyard equipment is ready to be energized. The required test instrumentation is properly calibrated and operational. The off-site grid connection is complete and available. RN-14-110

General Test Methods and Acceptance Criteria

The plant off-site ac power system components undergo a series of individual component and integrated system tests to verify that the off-site ac power system performs in accordance with the associated component design specifications. The individual component and integrated tests include:

a. Availability of ac and direct current (dc) power to the switchyard equipment is verified.

b. Operation of high voltage (HV) circuit breakers is verified.

c. Operation of HV disconnect switches and ground switches is verified.

d. Operation of substation transformers is verified.

e. Operation of current transformers, voltage transformers, and protective relays is verified.

f. Operation of switchyard equipment controls, metering, interlocks, and alarms that affect plant off-site ac power system performance is verified.

g. Design limits of switchyard voltages and stability are verified.

h. Under simulated fault conditions, proper function of alarms and protective relaying circuits is verified.

i. Operation of instrumentation and control alarms used to monitor switchyard equipment status.

j. Proper operation and load carrying capability of breakers, switchgear, transformers, and cables, and verification of these items by a non-testing means such as a QC nameplate check of as built equipment where testing would not be practical or feasible.

k. Verification of proper operation of the automatic transfer capability of the preferred power supply to the maintenance power supply through the reserve auxiliary transformer.

l. Switchyard interface agreement and protocols are verified.

The test results confirm that the off-site ac power systems meet the technical and operational requirements described in Section 8.2.

14.2.9.4.24 Raw Water System

Purpose

Raw water system testing verifies that the as-installed components supply raw water to the circulating water cooling tower basin, service water system cooling tower basin, fire protection water storage tanks, and other systems, as described in subsection 9.2.11.

Prerequisites

Component testing of the raw water system is completed. The components are operational and the storage tanks and cooling tower basins are able to accept water. Required support systems, electrical power supplies, and control circuits are operational. RN-14-110

General Test Methods and Acceptance Criteria

The raw water system component and integrated system performance is observed to verify that the system functions, as described in subsection 9.2.11 and in appropriate design specifications. The individual component and integrated system tests include:

a. Operation of the system pumps, traveling screens, automatic strainers, and valves is verified.

b. Operation of the system instrumentation, controls, actuation signals, alarms, and interlocks is verified.

c. Operation of heat tracing on system piping is verified.

14.2.9.4.25 Sanitary Drainage System

Purpose

Sanitary drainage system testing verifies that the as-installed components properly collect and discharge sanitary waste, as described in subsection 9.2.6.

Prerequisites

Component testing of the sanitary drainage system is completed. Required support systems, electrical power supplies, and control circuits are operational. RN-14-110

General Test Methods and Acceptance Criteria

The sanitary drainage system component and integrated system performance is observed to verify that the system functions, as described in subsection 9.2.6.2.1 and in appropriate design specifications. The individual component and integrated system tests include:

a. Operation of lift stations and valves is verified.

b. Operation of the system instrumentation, controls, actuation signals, and interlocks is verified.

14.2.9.4.26 Fire Brigade Support Equipment

Purpose

Fire brigade support equipment testing verifies that the equipment operates and is available when needed to perform the fire brigade functions, as described in Section 9.5.

Prerequisites

Equipment is ready and available for testing.

General Test Methods and Acceptance Criteria

The fire brigade support equipment undergoes a series of inspections to verify availability and operability. Equipment is available for selection and use, based on the hazard. Fire brigade support equipment tests include:

a. Location of portable extinguishers is verified; portable extinguishers are verified fully charged.

b. Operation of portable ventilation equipment is verified.

c. Operation of portable communication equipment is verified.

d. Operation of portable lighting is verified.

e. Operation of self-contained breathing apparatus and face masks is verified.

f. Operation of keys to open locked fire area doors is verified.

g. Turnout gear functionality and availability is verified.

h. Compatibility of threads for hydrants, hose couplings, and standpipe risers with the local fire department equipment is verified, or alternatively, an adequate supply of readily available hose thread adaptors is verified.

14.2.9.4.27 Portable Personnel Monitors and Radiation Survey Instruments

Purpose

Portable personnel monitors and radiation survey instruments testing verifies that the devices operate in accordance with their intended function in support of the radiation protection program, as described in Chapter 12.

Prerequisites

Portable personnel monitors, radiation survey instruments, and appropriate certified test sources are on site.

General Test Method and Acceptance Criteria

The portable personnel monitors and radiation survey instruments are source checked, tested, maintained, and calibrated in accordance with the manufacturers' recommendations. The portable monitors and instruments tests include:

a. Proper function of the monitors and instruments to respond to radiation is verified, as required.

b. Proper operation of instrumentation controls, battery, and alarms, if applicable.

14.2.10 Startup Test Procedures

The startup testing program is based on increasing power in discrete steps. Major testing is performed at discrete power levels as described in subsection 14.2.7. The first tests during Power Ascension Testing that verify movements and expansion of equipment are in accordance with design, and are conducted at a power level as low as practical (approximately 5 percent).

The governing Power Ascension Test Plan requires the following operations to be performed at appropriate steps in the power-ascension test phase:

- Conduct any tests that are scheduled at the test condition or power plateau.
- Confirm core performance parameters (core power distribution) are within expectations.
- Determine reactor power by heat balance, calibrate nuclear instruments accordingly, and confirm the existence of adequate instrumentation overlap between the startup range and power range detectors.
- Reset high-flux trips just prior to ascending to the next level to a value no greater than 20 percent beyond the power of the next level unless Technical Specification limits are more restrictive.
- Perform general surveys of plant systems and equipment to confirm that they are operating within expected values.
- Check for unexpected radioactivity in process systems and effluents.
- Perform reactor coolant leak checks.
- Review the completed testing program at each plateau; perform preliminary evaluations, including extrapolation core performance parameters for the next power level; and obtain the required management approvals before ascending to the next power level or test condition.

Upon completion of a given test, a preliminary evaluation is performed that confirms acceptability for continued testing. Smaller transient changes are performed initially, gradually increasing to larger transient changes. Test results at lower powers are extrapolated to higher power levels to determine acceptability of performing the test at higher powers. This extrapolation is included in the analysis section of the lower power procedure.

Surveillance test procedures may be used to document portions of tests, and ITP tests or portions of tests may be used to satisfy Technical Specifications surveillance requirements in accordance with administrative procedures. At Startup Test Program completion, a plant capacity warranty test is performed to satisfy the contract warranty and to confirm safe and stable plant operation.

Those tests comprising the startup test phase are discussed in this subsection. For each test a general description is provided for test objective, test prerequisites, test description, and test performance criteria, where applicable. In describing a test, the operating and safety-related characteristics of the plant to be tested and evaluated are identified.

Where applicable, the relevant performance criteria for the test are discussed. Some of the criteria relate to the value of process variables assigned in the design or analysis of the plant, component systems, and associated equipment. Other criteria may be associated with expectations relating to the performance of systems.

The specifics of the startup tests relating to test methodology, plant prerequisites, initial conditions, performance criteria, and analysis techniques are discussed in Section 14.4 in the form of plant, system and component performance and testing procedures.

14.2.10.1 Initial Fuel Loading and Precritical Tests

Tests performed after preoperational testing is complete but prior to initial criticality are described in this section. These tests include those performed prior to core load to verify the plant is ready for core loading, the loading of the core and the tests performed under hot conditions after the core has been loaded but prior to initial criticality.

Tests to be performed prior to and during initial core loading are described in Subsections 14.2.10.1.1 through 14.2.10.1.5. These tests verify the systems necessary to monitor the fuel loading process are operational and that the core loading is conducted properly.

After core load, tests are performed at hot conditions to bring the plant to a final state of readiness prior to initial criticality.

14.2.10.1.1 Fuel Loading Prerequisites and Periodic Checks

Objectives

- Specify the prerequisites for initial fuel loading, including the status of required systems, plant conditions, and special equipment
- Provide a checklist for periodic verification that the conditions required for fuel loading are being maintained

Prerequisites

- Plant systems required for initial fuel loading have been satisfactorily tested and turned over to the plant operating staff, and are in the status specified
- Plant conditions required for initial core loading are as specified
- Special equipment required for initial fuel loading is available and operable

Test Method

- Prior to the beginning of fuel loading, verify and document the required status of test prerequisites
- Throughout fuel loading, verify through periodic checks that conditions required for safe fuel loading are being maintained

Performance Criterion

The required status of prerequisites for initial fuel loading is verified and documented prior to fuel loading and maintained throughout the loading process.

14.2.10.1.2 Reactor Systems Sampling for Fuel Loading

Objective

- Verify that the dissolved boron concentration in the reactor coolant system and directly connected portions of associated auxiliary systems is uniform and equals or exceeds the value required by the plant Technical Specifications for fuel loading.

Prerequisites

- Plant Technical Specifications for fuel loading are complete and verified

- Boric acid storage tanks, transfer pumps, and associated piping and equipment are filled and operable

- The reactor vessel is filled with borated water to a level approximately equal to the centerline of the outlet nozzles

- The water in the reactor vessel and reactor coolant system piping, including all directly connected auxiliary systems, is borated to a value that equals or exceeds the value specified in the plant Technical Specifications for fuel loading, and that water is circulating through the normal residual heat removal system at a rate that provides reasonable assurance of a uniform concentration.

Test Method

- Obtain and analyze samples from at least one representative point in each auxiliary system and at four equidistant depths in the reactor vessel for boron concentration

- Periodically repeat sampling until the performance criteria are met

Performance Criteria

- The minimum boron concentration of all samples equals or exceeds the value specified in the plant Technical Specifications for fuel loading. If the minimum boron concentration criteria is not met, the chemical and volume control system is used to increase the boron concentration to above the specified limit.

- The boron concentrations of the samples obtained in the reactor vessel and operating residual heat removal loop are within the specified range of each other. The normal residual heat removal system continues to operate until a uniform concentration is established.

14.2.10.1.3 Fuel Loading Instrumentation and Neutron Source Requirements

Objectives

- Verify alignment, calibration, and neutron response of the temporary core loading instrumentation prior to the start of fuel loading

- Verify the neutron response of the nuclear instrumentation system source range channels prior to the start of fuel loading

- Verify the neutron response of the temporary and nuclear instrumentation system source range instrumentation prior to resumption of fuel loading following any delay of 8 hours or more

Prerequisites

- The following special equipment is available:

  - The temporary core loading package consisting of three complete counting channels, including preshipment alignment and calibration data

  - A portable neutron source with sufficient strength to verify detector response

- Preoperational testing of the nuclear instrumentation system source range channels is completed

Test Method

- Prior to the start of fuel loading, verify the response of temporary and nuclear instrumentation system source range channels to neutrons by using a portable neutron source
- Verify proper alignment and calibration of the temporary channels by comparing the neutron response data to the data obtained during preshipment testing
- Prior to resumption of fuel loading following a delay of 8 hours or more, verify proper operation of the temporary and nuclear instrumentation system source range channels by performing a neutron response check (using the portable neutron source or by moving a fuel assembly containing a primary neutron source) or by statistical analysis of the count rate data

Performance Criterion

Equipment used for neutron monitoring during fuel loading is operating correctly and is responsive to changes in neutron flux levels. Minimum count rates of 1/2 counts per second, attributable to core neutrons, are required on at least two of the available pulse-type nuclear channels at all times following installation of the initial nucleus of fuel assemblies (approximately eight fuel assemblies, one of which contains a neutron source), which permits meaningful inverse count-rate monitoring.

14.2.10.1.4 Inverse Count Rate Ratio Monitoring for Fuel Loading

Objective

Verify the neutron monitoring data obtained during initial fuel loading is consistent with calculations showing the predicted response and, for plants subsequent to the first plant, with data obtained during a previous similar fuel loading.

Prerequisites

- Temporary and plant source range nuclear instrumentation has been operational for a minimum of 60 minutes to allow the equipment to attain stable operating conditions
- The plant is prepared for initial fuel loading
- Neutron monitoring data from a previous similar initial fuel loading or calculations showing the predicted response of monitoring channels are available for evaluating monitoring data

Test Method

- Prior to inserting the first fuel assembly into the reactor vessel, obtain background count rates for each temporary and plant source range channel
- During the insertion of each fuel assembly, continuously observe the response of at least one channel for unexpected changes in count rate
- Construct a plot of inverse count rate ratio, versus fuel loading step number, from monitoring data obtained after each fuel assembly is loaded and used to assess the safety with which fuel loading may continue

Performance Criterion

Monitoring data are consistent with calculations showing the predicted response and, for plants subsequent to the first plant, with data obtained during a previous similar fuel loading. Each subsequent fuel addition will be accompanied by detailed neutron count rate monitoring to determine that the just loaded fuel assembly does not excessively increase the count rate and that the extrapolated ICRR is behaving as expected and not decreasing for unexplained reasons.

14.2.10.1.5 Initial Fuel Loading

Objectives

- Establish the conditions under which the initial fuel loading is to be accomplished
- Accomplish initial fuel loading in a safe manner

Prerequisites

- The nuclear design of the initial reactor core specifying the final core configuration of fuel assemblies and inserts is completed.
- Preoperational testing is completed on systems specified as required for initial fuel loading.
- Preoperational testing is completed on required fuel handling tools. Tools are available, operational, and calibrated, including indexing of the manipulator crane with a dummy fuel element.
- Containment integrity is established.
- The reactor vessel is filled with water to a level approximately equal to the center of the vessel outlet nozzles. The water is being circulated at a rate to provide uniform mixing.
- The boron concentration in the reactor coolant equals or exceeds the concentration required by the plant Technical Specifications for refueling. Core moderator chemistry conditions (particularly boron concentration) are prescribed in the core loading procedure document and are verified periodically by chemical analysis of moderator samples taken prior to and periodically during core loading operations.
- Sources of unborated water to the reactor coolant are isolated.
- Temporary and plant source range channels are operable as required to monitor changes in core reactivity.
- A surveillance program verifies that the conditions for fuel loading are maintained throughout the fuel loading program.
- Auxiliary system status is in accordance with Technical Specification requirements.
- The overall process of initial fuel loading will be supervised by a licensed senior reactor operator with no other concurrent duties.

Test Method

- Place fuel assemblies, together with inserted components (control rods, burnable poison elements, primary and secondary neutron sources), in the reactor vessel one at a time according to an established and approved sequence
- During and following the insertion of each fuel assembly and until the last fuel assembly has been loaded, the response of the neutron detectors is observed and compared to previous fuel loading data, or calculations, to verify that the observed changes in response are as expected

- Check sheets are completed at prescribed intervals verifying that the conditions required for initial fuel loading are being maintained

- Fuel assemblies, together with inserted components (control rod assemblies, burnable poison inserts, source spider, or thimble plugging devices) are placed in the reactor vessel one at a time according to a previously established and approved sequence, which was developed to provide reliable core monitoring with minimum possibility of core mechanical damage. The core loading procedure documents include detailed tabular check sheets that prescribe and verify the successive movements of each fuel assembly and its specified inserts from its initial position in the storage racks to its final position and orientation in the core. Multiple checks are made of component serial numbers and types at successive transfer points to guard against possible inadvertent exchanges or substitutions of components, and fuel assembly status boards are maintained throughout the core loading operation. The results of each loading step will be reviewed and evaluated before the next prescribed step is started.

- The criteria for safe loading require that loading operations stop immediately if:

  – An unanticipated increase in the neutron count rate by a factor of two occurs in all responding nuclear channels during any single loading step after the initial nucleus of fuel assemblies is loaded.

  – An unanticipated increase in the count rate by a factor of five occurs on any individual responding nuclear channel during any single loading step after the initial nucleus of fuel assemblies is loading.

  – A decrease in boron concentration greater than 20 ppm is determined from two successive samples of reactor coolant system water until the decrease is explained.

Performance Criteria

All fuel assemblies have been loaded into the vessel in the correct location and orientation consistent with the prespecified configuration for the initial reactor core. All fuel loading steps are documented, including the final core configuration.

14.2.10.1.6 Post-Fuel Loading Precritical Test Sequence

Objective

Specify the sequence of events constituting the precritical test program.

Prerequisites

- Plant system conditions are established as required by the individual test instructions within the precritical test sequence, as described in Subsections 14.2.10.1.7 through 14.2.10.1.20.

- The systems, structures, and components required by Technical Specifications shall be operable as required for the specified plant operational mode prior to initiation of precritical testing. Preoperational and precritical tests shall be completed to confirm the operability of required plant safety systems to support precritical testing prior to the initiation of the precritical tests.

Test Method

The instructions establish the sequence for required testing after core loading, until the plant has completed precritical testing.

Performance Criteria

Performance criteria are contained in the various individual tests conducted during this time (Subsections 14.2.10.1.7 through 14.2.10.1.20). RN-12-057

14.2.10.1.7 Incore Instrumentation System Precritical Verification

Objectives

- Verify that the incore instrumentation thimbles have been installed correctly following initial fuel loading
- Verify proper operation of the incore thermocouples prior to plant heatup

Prerequisites

- Initial fuel loading has been completed, all incore instrumentation thimble assemblies have been installed, and all mechanical and electrical connections have been completed.
- The plant is at ambient temperature and pressure prior to heatup for initial criticality.
- Incore instrumentation system signal processing software has been installed and is operational.

Test Method

- With the plant at ambient conditions following initial fuel loading and prior to heatup for initial criticality, make electrical continuity checks at the incore instrumentation system panel to verify proper installation and connection of the incore sensor strings.
- Obtain incore thermocouple data and compare with the measured reactor coolant system temperature to verify proper operation of the incore thermocouples and signal processing.

Performance Criteria

- Prior to plant heatup, proper connections to the incore instrumentation thimbles are verified and outputs from the incore thermocouple system are consistent with existing plant conditions, and are consistent with design requirements specified in subsection 4.4.6 and Section 7.5 and applicable design specification.
- Data required for calibration of other plant instrumentation are obtained.

14.2.10.1.8 Resistance Temperature Detectors-Incore Thermocouple Cross Calibration

Objectives

- Verify calibration coefficients for the resistance temperature detectors installed in the reactor coolant system.
- Determine calibration coefficients for resistance temperature detectors replaced in the reactor coolant system following hot functional testing as required.
- Determine calibration coefficients for the incore thermocouples that are part of the incore instrumentation system.

Prerequisites

- Initial fuel loading has been completed and the reactor coolant system is filled and vented prior to heatup for initial criticality.
- Reactor coolant system resistance temperature detectors that were replaced as a result of preoperational testing are operational, and an initial alignment has been completed according to the manufacturer's calibration data.
- The incore instrumentation system, including signal processing software, has been installed and is operational, and the preoperational testing has been completed.
- Instrumentation and data collection equipment is operational and available for logging plant data.

Test Method

- With the reactor coolant system at ambient temperature, and at isothermal conditions at specified temperature plateaus during heatup to normal operating temperature, measure the resistance of each resistance temperature detector installed in the reactor coolant system and the output from each installed incore thermocouple, along with supplemental plant data.
- Using the calibration coefficients determined during hot functional testing and the manufacturer's resistance versus temperature calibration data for the replaced resistance temperature detectors, determine the best-estimate temperature of each temperature plateau from the average of the derived resistance temperature detectors temperatures.
- On an iterative basis, recompute the best-estimate plateau temperature after removing from the average calculation the data from resistance temperature detectors whose temperature differs from the average by a predetermined amount.
- Verify or recompute calibration coefficients for each resistance temperature detector, as required, based on the final plateau average temperatures.
- Compute calibration coefficients for each incore thermocouple based on the final plateau average temperatures and supplemental data obtained during heatup.

Performance Criteria

- For each resistance temperature detector, the adequacy of the final calibration coefficients is verified when the temperature derived from the resistance temperature detector resistance agrees with the plateau average temperatures within predetermined limits as described in Sections 7.2 and 7.3.
- For each incore thermocouple, the adequacy of the final calibration coefficients is verified when the temperature derived from the thermocouple output agrees with the plateau average temperatures within predetermined limits, as described in subsection 4.4.6, Section 7.2 (Table 7.2-1) and Section 7.3 (Table 7.3-4).

14.2.10.1.9 Nuclear Instrumentation System Precritical Verification

Objective

Establish and determine voltage settings, trip settings, operational settings, alarm settings, and overlap of channels on source range instrumentation prior to initial criticality.

Prerequisite

The nuclear instrumentation system is aligned according to the design requirements.

Test Method

- Calibrate, test, and verify functions using permanently installed controls and adjustment mechanisms.

- Set operational modes of the source range channels for their proper functions, in accordance with the test instructions.

Performance Criterion

The nuclear instrumentation system operates in accordance with the design basis functional requirements, as discussed in subsection 4.4.6.

14.2.10.1.10 Setpoint Precritical Verification

Objectives

- Prior to initial criticality, verify that initial values of instrumentation setpoints assumed in the design, operation, and safety analysis of the nuclear steam supply system have been installed correctly, and identify which of these are expected to be readjusted based on the results of startup testing and initial operations.

- Prior to initial criticality, document final values of instrumentation setpoints assumed in the design, operation, and safety analysis of the plant and as modified by initial startup testing, operations, or reanalysis to serve as a basis for future plant operations.

Prerequisites

- Initial alignment and calibration of plant instrumentation has been completed, and initial set points are installed per applicable design documentation.

- Preoperational and startup testing of affected plant instrumentation has been completed, and test results are documented.

Test Method

- Review applicable design documentation and generate a list of the instrumentation setpoints assumed in the design, operation, and safety analysis of the plant. Identify setpoints expected to be modified based on the results of initial startup tests and operations.

- Prior to initial criticality, the results of preoperational and startup tests, as applicable, are reviewed to verify that initial setpoints have been installed correctly. Document the results of this review for future use.

- Prior to initial criticality, summarize and document the setpoint values for future plant operations.

Performance Criterion

Prior to initial criticality, installed setpoint values are verified to be consistent with Technical Specifications.

14.2.10.1.11 Rod Control System

Objective

Demonstrate and document that the rod control system performs the required control and indication functions just prior to initial criticality.

Prerequisites

- The reactor coolant system is at no-load operating temperature and pressure
- The nuclear instrumentation system source range channels are aligned and operable

Test Method

- With the reactor at no-load temperature and pressure, just prior to initial criticality, verify the operation of the rod control system in various modes including tests of control rod block and inhibit functions.
- Verify the operation of status lights, alarms, and indicators

Performance Criteria

- The performance of the rod control system as described in subsection 7.7.1.2.
- The rod control system withdraws and inserts each rod bank
- The rod position and indication system tracks each rod bank as it is being moved
- The control banks overlap system starts and stops rod movement at the designated bank positions

14.2.10.1.12 Rod Position Indication System

Objective

Verify that the rod position indication system satisfactorily performs required indication and alarm functions for each individual rod and that each rod operates satisfactorily over its entire range of travel.

Prerequisites

- The reactor coolant system is at no-load operating temperature and pressure
- At least one reactor coolant pump is in service, with reactor coolant boron concentration not less than specified in the Technical Specifications for refueling shutdown

Test Method

Individually withdraw rod banks from the core and reinsert them, according to the test procedure. Record rod position sensor output voltages, and rod position readouts and group step counters in the main control room.

Performance Criterion

The rod position indication system performs the required indication and alarm functions as discussed in subsection 7.7.1.3, and each rod operates over its entire range of travel.

14.2.10.1.13 Control Rod Drive Mechanisms

Objectives

- Demonstrate operation of each control rod drive mechanism under both cold and hot standby conditions
- Provide verification of slave cycler timing

Prerequisites

- The reactor coolant system is filled and vented at cold shutdown
- Rods are fully inserted
- Nuclear instrumentation channels are available
- A fast-speed oscillograph, or equivalent, to monitor test parameters is available

Test Method

- With the reactor core installed and the reactor in the cold shutdown condition, confirm that the slave cycler devices supply operating signals to the control rod drive mechanism stepping magnet coils.
- Verify operation of all control rod drive mechanisms under both cold and hot standby conditions. Record the control rod drive mechanism magnet coil currents.

Performance Criterion

The control rod drive mechanisms conform to the requirements for proper mechanism operation and timing including control rod withdrawal and insertion speeds as described in the applicable design specifications.

14.2.10.1.14 Rod Drop Time Measurement

Objectives

- Determine the rod drop time of each rod cluster control assembly under cold no-flow and hot full-flow conditions, with the reactor at normal operating temperature and pressure.
- Verify the operability of the control rod deceleration device.

Prerequisites

- Initial core loading is completed
- Source range channels are in operation
- Rods are fully inserted
- Reactor coolant pumps are operational

Test Method

- Withdraw each rod cluster control assembly
- Interrupt the electrical power to the associated control rod drive mechanism
- Measure and record the rod drop time, and verify control rod deceleration
- Perform a minimum of three additional drops for each control rod whose drop time falls outside the two-sigma limit, as determined from the drop times obtained for each test condition

Performance Criteria

- Measured rod drop times are consistent with the design basis functional requirements and the applicable plant Technical Specifications
- The control rod is slowed by the control rod deceleration device during rod drop testing

14.2.10.1.15 Rapid Power Reduction System

Objective

Verify proper operation of the rapid power reduction system prior to power operations.

Prerequisites

- The following systems are operable to the extent necessary to support the test: rod control system, rod position indication system, reactor trip breakers, and reactor protection system.
- The reactor is shut down, the reactor coolant system boron concentration is such that Technical Specifications requirements for shutdown margin will be met with required rod withdrawal, and all control banks are near their fully inserted positions.

Test Method

- Input signals simulating operation at the full power condition to the reactor control and protection system. Close the reactor trip breakers.
- Input signals simulating a rapid loss of load exceeding 50 percent power are input to the rapid power reduction system. Verify the response of the system.
- Demonstrate procedures for returning the plant to power following a partial trip.

Performance Criteria

- Performance of the rapid power reduction system is in accordance with subsection 7.7.1.10.
- In response to the simulated loss of load, gripper power is interrupted to a preselected grouping of control rods, so that rods drop freely into the core.
- Gripper power to only those control rods selected for drop is interrupted.
- Procedures for returning the plant to power operations without a reactor trip are verified.

14.2.10.1.16 Process Instrumentation Alignment

Objective

Align ΔT and T_{avg} process instrumentation under isothermal conditions prior to initial criticality.

Prerequisites

- Reactor coolant pumps are operating
- The reactor coolant system average temperature is at the hot no-load average temperature

Test Method

- Align ΔT and T_{avg} according to test instructions at isothermal conditions prior to criticality

Performance Criterion

The indicated values for reactor coolant system T_{hot}, T_{cold}, T_{avg}, and ΔT under isothermal conditions are within the limits of the applicable design requirements as discussed in Section 7.2 (Table 7.2-1) and Section 7.3 (Table 7.3-4).

14.2.10.1.17 Reactor Coolant System Flow Measurement

Objectives

- Prior to initial criticality, verify that the reactor coolant system flow rate is sufficient to permit operation at power.

Prerequisites

- The core is installed and the plant is at normal operating temperature and pressure.
- Special instrumentation is installed and calibrated for obtaining reactor coolant flow data.

Test Method

- Prior to initial criticality, measure the reactor coolant flow measurement parameters with all four coolant pumps in operation. Estimate the reactor coolant flow rate using these data.

Performance Criterion

The estimated reactor coolant flow rate from data taken prior to initial criticality equals or exceeds 90 percent of the minimum value required by the plant Technical Specifications for full power operation.

14.2.10.1.18 Reactor Coolant System Flow Coastdown

Objectives

- Measure the rate at which reactor coolant loop flow and pump speed changes, subsequent to tripping all reactor coolant pumps.

- Measure the rate at which reactor coolant loop flow and pump speed changes, subsequent to tripping two of four reactor coolant pumps.

Prerequisites

- Required component testing and instrument calibration are complete

- Required electrical power supplies and control circuits are operational

- The reactor core is installed, and the plant is at normal operating temperature and pressure with all reactor coolant pumps running

Test Method

- Record loop flow, pump speeds following the trip of all reactor coolant pumps
- Record loop flows, pump speeds following the trip of two of four reactor coolant pumps

Performance Criterion

The loop flows and pump speed data are obtained for verification of the loss of flow analyses in Subsections 15.3.1 and 15.3.2.

14.2.10.1.19 Pressurizer Spray Capability and Continuous Spray Flow Verification

Objectives

- Establish the optimum continuous spray flow rate
- Determine the effectiveness of the normal control spray

Prerequisites

- The reactor coolant system is at no-load operating temperature and pressure.
- All reactor coolant pumps are operating.

Test Method

- While maintaining constant pressurizer level, adjust spray bypass valves until a minimum flow is achieved that maintains the temperature difference between the spray line and the pressurizer within acceptable limits.

- With the pressurizer heaters de-energized, fully open both spray valves, and record the time to lower the pressurizer pressure a specified amount.

Performance Criteria

- The spray bypass valves are throttled so that the minimum flow necessary to keep the spray line warm is achieved.

- The pressurizer pressure response to the opening of the pressurizer spray valves is within design basis functional limits as specified in subsection 7.7.1.6 and the appropriate pressure control system design specification documentation.

14.2.10.1.20 Feedwater Valve Stroke Test

Objective

Verify proper operation of the main and startup feedwater control valves prior to the start of power operations.

Prerequisites

- Preoperational testing of the feedwater control systems has been completed
- Main and startup feedwater pumps are off
- Initial fuel loading has been completed prior to initial criticality.

Test Method

For each main and startup feedwater flow control valve, the following tests are performed:

- Using simulated signals for several valve demand positions covering the range from fully closed to fully open, verify the actual valve position to be consistent with the demand signal.

- For selected valve position changes, measure the time required from the initiation of the demand signal until the valve reaches the final position. Typical demands changes are the following: fully closed to fully open, fully open to fully closed, 25 percent open to 75 percent open, and 75 percent open to 25 percent open.

Performance Criteria

The main and startup feedwater valves operate as described in subsection 7.7.1.8 and appropriate design specifications including:

- The differences between the measured actual and demand valve positions, over the range of travel, are less than prespecified tolerances.
- The time between the initiation of the demand signal and the final valve position for each of the demand changes is within specified ranges as discussed in applicable design specifications.
- For demand changes to intermediate valve positions, the amount of overshoot is less than specified limits as discussed in applicable design specifications.

14.2.10.2 Initial Criticality Tests

Initial criticality testing is described in this section. Following completion of the core loading and precriticality testing, the plant is brought to initial criticality, according to the test procedures in Subsection 14.2.10.2.1.

14.2.10.2.1 Initial Criticality Test Sequence

Objective

Define the sequence of tests and operations to bring the core to initial criticality.

Prerequisite

Plant system conditions are established as required by the individual test instructions within this sequence.

Test Method

An individual test instruction will establish the plant conditions required for initial criticality.

Performance Criteria

Relevant performance criteria are provided in each of the test procedure abstracts.

14.2.10.2.2 Initial Criticality

Objective

Achieve initial criticality in a controlled manner.

Prerequisites

- The nuclear instrumentation is verified to be operating properly (See 14.2.10.2.3)
- The reactor coolant system temperature and pressure are stable at the normal hot no-load values
- Control rod banks are inserted, and shutdown rod banks are withdrawn
- The reactor coolant system boron concentration is sufficiently high so the reactor is shut down by at least 1000 pcm with all banks withdrawn

Test Method

- Accomplish initial criticality by the controlled withdrawal of the rods using the same rod withdrawal sequence used for normal plant startup, followed by the dilution of the reactor coolant system boron concentration.
- At preselected points during rod withdrawal and/or boron dilution, gather data to plot the inverse count rate ratio to monitor the approach to critical evolution for reactivity monitoring.
- As criticality is approached, slow or stop dilution rate to allow criticality to occur during mixing or by withdrawal of rods that have been slightly inserted for control.

Performance Criterion

The reactor is critical.

14.2.10.2.3 Nuclear Instrumentation System Verification

Objective

Establish and determine voltage settings, trip settings, operational settings, alarm settings, and overlap of channels on source and intermediate range instrumentation, from prior to initial criticality and during initial criticality.

Prerequisite

The nuclear instrumentation system is aligned according to the design requirements.

Test Method

- Calibrate, test, and verify functions using permanently installed controls and adjustment mechanisms.
- Set operational modes of the source and intermediate range channels for their proper functions, in accordance with the test instructions.

Performance Criteria

- The nuclear instrumentation system operates in accordance with the design basis functional requirements, as discussed in subsection 4.4.6.
- The nuclear instrumentation system demonstrates an overlap of indication between the source and intermediate range instrumentation.
- The nuclear instrumentation minimum neutron count rate and noise to signal ratio are within appropriate design specifications.

14.2.10.2.4 Post-Critical Reactivity Computer Checkout

Objective

Demonstrate proper operation of the reactivity computer through a dynamic test using neutron flux signals.

Prerequisites

- The reactor is critical with the neutron flux level within the range for low-power physics testing

- The reactor coolant system temperature and pressure are stable at the normal no-load values
- The neutron flux level and reactor coolant system boron concentration are stable
- The reactivity computer is installed, checked out, and operational, and input flux signals are representative of the core average neutron flux level
- The controlling rod bank is positioned in such a way that the required reactivity insertion can be made by rod motion alone
- The systems, structures, and components required by Technical Specifications shall be operable as required for the specified plant operational mode prior to initiation of precritical, low power physics, and power ascension testing. Verification of proper operation of source-range and intermediate-range excore nuclear instrumentation and associated alarms and protective functions in Startup Test 14.2.10.2.3 shall be completed prior to initiation of this startup test.

Test Method

- By control rod motion, add positive reactivity to the core in accordance with design requirements as discussed in Section 7.7.
- During the resultant increase in flux level, make two independent measurements of core reactivity; one using the reactivity computer, and one using an analysis of the rate of change of flux level (for example, reactor period or doubling time).

Performance Criterion

Each measurement deviation between the two independent sources of reactivity is within design tolerances. Adjustment and recalibration or repair of the reactivity computer may be required if the deviation between the two independent sources of reactivity is not within design tolerances.

14.2.10.3 Low Power Tests

Following successful completion of the initial criticality tests, low power tests are conducted, typically at power levels less than 5 percent, to measure physics characteristics of the reactor system and to verify the operability of the plant systems at low power levels.

14.2.10.3.1 Low-Power Test Sequence

Objective

Define the sequence of tests and operations that constitutes the low-power testing program.

Prerequisite

Plant system conditions are established as required by the individual test instructions within this sequence.

Test Method

Individual test instruction will establish the plant conditions required for and during the low-power testing program following initial criticality.

Performance Criteria

Relevant performance criteria are provided in each of the test procedure abstracts.

14.2.10.3.2 Determination of Physics Testing Range

Objectives

- Determine the reactor power level at which the effects from fuel heating are detectable
- Establish the range of neutron flux in which zero power reactivity measurements are to be performed

Prerequisites

- The reactor is critical, and the neutron flux level is below the expected level of nuclear heating
- The reactor coolant system temperature and pressure are stable at the normal no-load values
- The neutron flux level and reactor coolant system boron concentration are stable
- The reactivity computer is installed, checked out, and operational, and input flux signals are representative of the core average neutron flux level
- The controlling rod bank is positioned in such a way that the required reactivity insertion can be made by rod motion alone

Test Method

- Withdraw the control rod bank and allow the neutron flux level to increase until nuclear heating effects are indicated by the reactivity computer
- Record the reactivity flux level and the corresponding intermediate range channel currents at which nuclear heating occurs
- Multiply the measured reactivity flux level by 0.3 to determine the maximum value for the zero power testing range

Performance Criterion

The zero power testing range is determined.

14.2.10.3.3 Boron Endpoint Determination

Objective

Determine the critical reactor coolant system boron concentration appropriate to an endpoint rod configuration.

Prerequisites

- The reactor is critical, and the neutron flux level is within the range for low-power physics testing
- The reactor coolant system temperature and pressure are stable at the normal no-load values

- The neutron flux level and reactor coolant system boron concentration are stable

- Instrumentation and equipment used to measure and compute reactivity is installed, checked out, and operational, with input flux signals representative of the core average neutron flux level

- The controlling rod bank is positioned in such a way that limited reactivity insertion will be required to achieve the endpoint condition

Test Method

- Move the rods to the desired endpoint configuration without boron concentration adjustment

- Directly measure the just-critical boron concentration by chemical analysis

- Measure and convert the change in reactivity and the reactor coolant temperature difference from program to an equivalent change in boron concentration

- Add the changes to the just-critical boron concentration to yield the endpoint for the given rod configuration

Performance Criterion

The measured value for the boron endpoint is consistent with the design value within design limits as specified in the Technical Specifications.

14.2.10.3.4 Isothermal Temperature Coefficient Measurement

Objectives

- Determine the isothermal temperature coefficient
- Calculate the moderator temperature coefficient

Prerequisites

- The reactor is critical, and the neutron flux level is within the range for low-power physics testing

- The reactor coolant system temperature and pressure are stable at the normal no-load values

- The neutron flux level and reactor coolant system boron concentration are stable

- Instrumentation and equipment used to measure and compute reactivity is installed, checked out, and operational, with input flux signals representative of the core average neutron flux level

- The controlling rod bank is positioned near fully withdrawn

Test Method

- Vary reactor coolant system temperature (heatup/cooldown) while maintaining rods and boron concentration constant

- Monitor reactivity results and determine the isothermal temperature coefficient

- Calculate the moderator temperature coefficient using the isothermal temperature coefficient and design values

Performance Criterion

- The measured value for the moderator temperature coefficient is more negative than the Technical Specification limit

14.2.10.3.5 Bank Worth Measurement

Objective

Validate design calculations of the reactivity worth of the rod cluster control banks.

Prerequisites

- The reactor is critical and the neutron flux level is within the range for low-power physics testing

- The reactor coolant system temperature and pressure are stable at the normal no-load values

- The neutron flux level and reactor coolant system boron concentration are stable

- Instrumentation and equipment used to measure and compute reactivity is installed and operational, with input flux signals representative of the core average neutron flux level

Test Method

- One of the following methods will be used to measure the worth of all of the individual control rod banks:

  – A bank is stepwise inserted into the core from fully withdrawn and the worth is measured using the reactivity computer

  – Exchange bank with another bank measured as above, with the worth determined from the critical positions and the worth of the reference bank

Performance Criteria

- The measured value for the individual bank worth is consistent with the design value within specified limits as discussed in subsection 4.3.2.5.

- The sum of the measured bank worth is consistent with the design value within the assumed uncertainty used in the shutdown margin calculation

14.2.10.3.6 Natural Circulation (First Plant Only)

Objective

Demonstrate that core decay heat can be removed by the steam generators under the conditions of natural circulation (no reactor coolant pumps operating).

Prerequisites

- The reactor is critical, and the neutron flux level is within the range for low-power physics testing

- The neutron flux level and reactor coolant system boron concentration and temperature are stable, and the controlling rod bank is positioned in such a way that an increase in core power level to approximately 3 percent can be achieved by rod motion alone
- Reactor coolant pumps are operating
- The reactivity computer is installed, checked out, and operational, with input flux signals representative of the core average neutron flux level
- Instrumentation and data collection equipment is operational and available for logging plant data
- Special instrumentation is available to measure vessel ΔT with high precision at low-power levels

Test Method

- Because this test is performed at beginning of life when the core fission product density is low, decay heat is simulated by reactor power
- By control rod motion, increase reactor power to approximately 3 percent of full power based on predictions of vessel ΔT at full power
- With reactor coolant pumps running, obtain data for correlating nuclear flux level and loop temperatures with power
- Trip all reactor coolant pumps. Maintain core power at approximately 3 percent by control rod motion while cold leg temperatures remain relatively constant.
- Verify natural circulation by observing the response of the hot leg temperature in each loop. The plant is stable under natural circulation at this power level when hot leg temperature is constant.
- Obtain data characterizing the plant under natural circulation conditions
- Restart reactor coolant pumps only after the reactor is shut down and isothermal conditions are re-established

Performance Criterion

The measured average vessel ΔT under natural circulation conditions is equal to or less than limiting design predictions for the measured reactor power level as specified in the applicable design specifications.

14.2.10.3.7 Passive Residual Heat Removal Heat Exchanger (First Plant Only)

Objective

[*Demonstrate the heat removal capability of the passive residual heat removal heat exchanger with the reactor coolant system at prototypic temperatures and natural circulation conditions.*]* Note that this test is performed in conjunction with the reactor coolant system natural circulation test with heat removal via the steam generators described in subsection 14.2.10.3.6.

Prerequisites

As described in subsection 14.2.10.3.6, the following prerequisites have been met in preparation for the natural circulation test with heat removal via the steam generators:

- The reactor is critical and the neutron flux level is within the range for low power physics testing.
- The neutron flux level and reactor coolant system boron concentration and temperature are stable, and the controlling rod bank is positioned in such a way that an increase in core power level to approximately 5 percent can be achieved by rod motion only.
- Reactor coolant pumps are running.
- The reactivity computer is installed, checked out, and operational, with input flux signals representative of the core average neutron flux level.
- Instrumentation and data collection equipment is operational and available for logging plant data.
- Special instrumentation is available to measure the reactor vessel ΔT with high precision at low power levels.
- The passive residual heat removal heat exchanger inlet and outlet temperature instrumentation and heat exchanger flow instrumentation are calibrated and operational.
- The passive residual heat exchanger inlet isolation valve is operational and in its open position, and the heat exchanger outlet isolation valves are operational and in their closed position.
- The startup feedwater system and controls are operating properly to maintain the steam generator secondary side water levels.
- The steam generator steam dump system is operating properly to maintain steam generator pressure so that the reactor coolant system cold leg fluid is at its expected temperature.
- The chemical volume control system auxiliary spray and letdown flow path are operable for controlling the pressurizer pressure and level, respectively after the reactor coolant pumps are shutoff.

Test Method

[*Note that the following test steps are to be performed at the conclusion of the natural circulation test with heat removal via the steam generators.*

- *Verify that the natural circulation test with core power being removed by dumping steam from the steam generators has been completed.*
- *Initiate flow through the passive residual heat removal heat exchanger by slowly opening one of the two parallel heat exchanger outlet isolation valves until it is fully open.*
- *The steam generator steam dump will automatically reduce heat removal by the steam generators in response to passive residual heat exchanger operation. Manual operation of the control rods may be required to maintain core power at approximately 3 percent.*

*NRC Staff approval is required prior to implementing a change in this information.

- *Obtain heat exchanger flow and inlet/outlet temperature data to characterize the heat removal capability of the heat exchanger and heatup of the in-containment refueling water storage tank water with one of two parallel isolation valves open.*

- *Close the open heat exchanger isolation valve to terminate the heat exchanger test. The steam generator steam dump should automatically maintain the reactor coolant system fluid average temperature constant. Note that operation of the passive residual heat exchanger should be terminated before the in-containment refueling water storage tank average water temperature exceeds 150°F.*

- *Shutdown the reactor by inserting the control rods. Restart reactor coolant pumps only after the reactor is shutdown and isothermal conditions are re-established.*]*

Performance Criteria

[*The measured passive residual heat exchanger heat removal rate is equal to or greater than the heat removal rate predicted by the methodology used in the safety analyses at the measured hot leg and in-containment refueling water temperatures.*]*

14.2.10.4 Power Ascension Tests

After low power testing is completed, testing is performed at specified elevated power levels to demonstrate the facility operates in accordance with design during normal steady-state operations, and to the extent practical, during and following anticipated transients. During power ascension, tests are performed to obtain operational data and to demonstrate the operational capabilities of the plant.

14.2.10.4.1 Test Sequence

Objective

Define the sequence of operations, beginning at approximately 5 percent rated thermal power, that constitutes the power ascension testing program.

Prerequisite

Plant system conditions are established, as required, by the individual test instruction within this sequence.

Test Method

Present the sequence of operations and tests, along with instructions, specific plant conditions, and test procedures.

Performance Criteria

Relevant performance criteria are provided in each of the test procedures.

14.2.10.4.2 Incore Instrumentation System

Objectives

- Obtain data for incore thermocouple and flux maps at various power levels during ascension to full power determine flux distributions and verify proper core loading and fuel enrichments.

Prerequisites

- Incore instrumentation system signal processing software is installed and operational

*NRC Staff approval is required prior to implementing a change in this information.

- For incore thermocouple and flux mapping, the plant is at various power levels greater than approximately 20 percent of rated thermal power

Test Method

- With the plant at approximate power levels of 25, 50, 75 and 100 percent of rated thermal power, obtain data from the incore instrumentation system and process to produce incore thermocouple and flux maps. (Actual power levels will be specified in the power ascension program test sequence.)

- Use data from the incore maps to verify that core power distribution is consistent with design predictions and the limits imposed by the plant Technical Specifications, including detection of potential fuel loading errors, and to calibrate other plant instrumentation. Refer to Technical Specifications Section 3.2, Power Distribution Limits.

Performance Criteria

- Core power peaking factors derived from the incore data are consistent with design predictions and the limitations of the plant Technical Specifications

- Data required for calibration of other plant instrumentation are obtained

14.2.10.4.3 Nuclear Instrumentation System

Objective

Establish and determine voltage settings, trip settings, operational settings, alarm settings, and overlap of channels on intermediate range and power range instrumentation from zero power to at or near full rated thermal power.

Prerequisite

The nuclear instrumentation system is aligned according to the design requirements.

Test Method

- Calibrate, test, and verify functions using permanently installed controls and adjustment mechanisms

- Set operational modes of the intermediate range and power range channels for their proper functions, in accordance with the test instructions

Performance Criteria

- The nuclear instrumentation system operates in accordance with the design basis functional requirements as discussed in subsection 4.4.6.

- The nuclear instrumentation system demonstrates an overlap of indication between the intermediate and power range instrumentation.

14.2.10.4.4 Setpoint Verification

Objective

During power ascension, document final values of instrumentation setpoints as modified by initial startup testing, operations, or reanalysis to serve as a basis for future plant operations.

*NRC Staff approval is required prior to implementing a change in this information.

Prerequisites

- Initial alignment and calibration of plant instrumentation have been completed, and initial set points are installed per applicable design documentation
- Preoperational and startup testing of affected plant instrumentation has been completed, and test results are documented
- The results of the precritical verification of the instrument setpoints are completed and documented

Test Method

- Identify setpoints modified based on the results of initial startup tests and operations
- During power ascension testing, readjust specific setpoints noted for readjustment on the data sheets if required. Record final setpoint values.

Performance Criterion

Setpoint changes based on initial startup testing and operations are documented for future reference.

14.2.10.4.5 Startup Adjustments of Reactor Control Systems

Objectives

- Determine the adequacy of the reactor coolant system programmed T_{avg}
- Obtain plant data during power ascension which would provide the basis for any required changes to the T_{avg} program

Prerequisites

- The reactor coolant system is at no-load operating temperature and pressure
- The reactor coolant system temperature is being controlled by the steam dump valves

Test Method

- Obtain system temperature and steam pressure data at steady-state conditions for zero rated thermal power and at hold points during power escalations
- At approximately 75 percent rated thermal power, modify the T_{avg} program as required to achieve design steam generator pressure at full power, based on extrapolation of the data to the full power condition.
- Reevaluate the T_{avg} program as above at approximately 90 and 100 percent rated thermal power making modifications to the T_{avg} program as required.

Performance Criterion

The reactor coolant system T_{avg} program is established such that steam generator pressure at the full rated thermal power condition is within design functional requirements as discussed in Section 5.1.

14.2.10.4.6 Rod Cluster Control Assembly Out of Bank Measurements (First Plant Only)

Objectives

- Demonstrate the sensitivity of the incore and excore instrumentation system to rod cluster control assembly (RCCA) misalignments
- Demonstrate the design conservatism for predicted power distributions with a fully misaligned rod cluster control assembly
- Monitor the power distribution following the recovery of a misaligned rod cluster control assembly

Prerequisites

- The reactor is operating between 30 and 50 percent of full licensed power and has been at that power for a sufficient time to reach xenon equilibrium.
- The reactor power level, reactor coolant system boron concentration, and temperature are stable.
- The control and shutdown banks are positioned as required for the specific measurement, near fully withdrawn for rod cluster control assembly insertion, and at their respective insertion limits for rod cluster control assembly withdrawal.

Test Method

- For the rod cluster control assembly insertion, insert a group of selected rod cluster control assemblies, one at a time, first to the limit of misalignment specified in subsection 15.0.5, then fully inserted, and finally restored to the bank position. Compensate for reactivity changes by dilution and boration as required.
- For the rod cluster control assembly withdrawal, withdraw one or more selected rod cluster control assemblies, one at a time, to the fully withdrawn position. Compensate for reactivity changes by boration and dilution as required.
- Record incore and excore instrumentation signals to determine their response and to determine the power distribution and power peaking factors prior to rod cluster control assembly misalignment, at partial misalignment, at full misalignment, and periodically after restoration to normal.

Performance Criteria

- Measured power distributions and power peaking factors are within Technical Specification limits and are consistent with the predictions.
- The sensitivity of the incore and excore instrumentation to rod cluster control assembly misalignment is demonstrated by examination of the power distribution and power peaking factors measured for each misalignment.

14.2.10.4.7 Axial Flux Difference Instrumentation Calibration

Objectives

- Calibrate the power range nuclear instrumentation signals used as axial flux difference (delta flux) input to the reactor protection system

- Calibrate instrumentation used to display and monitor axial flux difference

Prerequisites

- The reactor is at a power level greater than 50 percent of rated thermal power
- The incore instrumentation system is available for obtaining incore power distribution data
- A preliminary calibration of the axial flux difference indication instrumentation is completed

Test Method

- Using control rod movement, xenon redistribution, or a combination of both, vary the axial power distribution of the core over a specified range of interest. At selected values of indicated axial flux difference, obtain reactor thermal power data along with the outputs from the nuclear instrumentation power range channels and the incore instrumentation system. (For the first plant, a minimum of three data sets will be taken; subsequent cores may require less.)

- Calibrate signals from the nuclear instrumentation power range channels based on incore power distribution and thermal power data.

Performance Criterion

Axial flux difference signals, derived from the nuclear instrumentation power range detectors and input to the reactor protection system, display, and monitoring instrumentation, reflect actual incore power distribution within specified limits, as discussed in subsection 7.7.1.1.

14.2.10.4.8 Primary and Secondary Chemistry

Objective

Verify proper water quality in the reactor coolant system and secondary coolant system.

Prerequisite

The plant is at the steady-state condition at approximately 0, 25, 50, 75, and 100 percent rated thermal power.

Test Method

Analyze samples to determine the chemical and radiochemical concentrations.

Performance Criterion

The chemical and radiochemical control systems maintain the water chemistry within the applicable guidelines as discussed in Subsections 5.2.3.2 and 10.3.5.

14.2.10.4.9 Process Measurement Accuracy Verification

Objectives

- Measure the temperature variation in the reactor coolant loops resulting from non-uniform flow effects such as streaming

- Measure the sensitivity of the excore detectors to variations in control bank position and reactor coolant loop cold leg temperature

Prerequisites

- For the reactor coolant loop temperature measurements:

- Special temperature measuring equipment, including recording and indicating instrumentation, is installed, as required, on the reactor coolant loops hot and cold leg piping

- The reactor is at a stable power level of approximately 0, 50, 75 and 100 percent of rated thermal power

- For the excore detector measurements:

  - The reactor is at a stable power level of approximately 25, 50 and 100 percent of rated thermal power

Test Method

- For the reactor coolant loop temperature measurements, at each power level:

  - Measure reactor power level, using calorimetric data

  - Simultaneously, measure the hot and cold leg temperatures, using normal plant instrumentation and any other required instrumentation

- For the excore detector tests, with the reactor at constant power level:

  - Measure the response of the excore detectors as selected control banks are moved over prescribed ranges of travel

  - Measure excore detector response as the reactor coolant cold leg temperature is varied over a prescribed range

  - Simultaneously, for each of the preceding measurements, obtain calorimetric data to verify reactor power level

Performance Criteria

- Uncertainties in reactor coolant loop temperature measurements resulting from non-uniform flow effects such as streaming are consistent with allowances used in the plant safety analyses.

- Uncertainties in excore detector response resulting from control rod motion and reactor coolant loop cold leg temperature changes are consistent with allowances used in the plant safety analyses.

14.2.10.4.10 Process Instrumentation Alignment at Power Conditions

Objective

Align ΔT and T_{avg} process instrumentation at power conditions.

Prerequisites

- Reactor coolant pumps are operating.
- The reactor system is operating at the required power level.

Test Method

- Align ΔT and T_{avg} according to test instructions at approximately 75 percent rated thermal power. Extrapolate the 75 percent data to determine ΔT and T_{avg} values for the 100 percent plateau.

- At or near 100 percent rated thermal power, check the alignment of the ΔT and T_{avg} channels for agreement with the results of the thermal power measurement.

Performance Criterion

The indicated values for reactor coolant system T_{hot}, T_{cold}, T_{avg}, and ΔT at or near full thermal power are within the limits of the applicable design requirements, as discussed in Section 5.1.

14.2.10.4.11 Reactor Coolant System Flow Measurement at Power Conditions

Objective

At power, verify that the reactor coolant flow equals or exceeds the minimum value required by the plant Technical Specifications.

Prerequisites

- The reactor is at power levels greater than 75 percent and up to and including 100 percent of rated thermal power

- Special instrumentation required for measuring reactor thermal power and reactor coolant inlet and outlet temperatures is installed and calibrated

Test Method

With the reactor at steady-state power greater than 75 percent and up to and including 100 percent of rated thermal power, measure the reactor thermal power and coolant inlet and outlet temperatures. Determine the reactor coolant flow rate using the data in conjunction with hydraulic analysis of differential pressures at different locations in the reactor coolant system.

Performance Criterion

The reactor coolant system flow determined from the measurements at approximately 100 percent rated thermal power equals or exceeds the minimum value required by the plant Technical Specifications.

14.2.10.4.12 Steam Dump Control System

Objective

Verify automatic operation of the T_{avg} steam dump control system, demonstrate controller setpoint adequacy, and obtain final settings from steam pressure control of the condenser dump valves.

Prerequisites

- Steam dump control system is aligned and calibrated to initial settings
- Plant is at no-load temperature and pressure
- Condenser vacuum is established
- Reactor is critical

Test Method

- Increase reactor power to less than 10 percent rated thermal power by rod withdrawal and steam dump to condenser to demonstrate setpoint adequacy
- Increase pressure controller setpoint prior to switching to T_{avg} control, which rapidly modulates open condenser dump valves
- Simulate turbine operating conditions with reactor at power, then simulate a turbine trip resulting in the rapid opening of the steam dump valves

Performance Criteria

- The plant trip controller responds to maintain a stable T_{avg}. After steady-state power is achieved, no divergent oscillations in temperature occur
- The loss of load controller responds properly to maintain a specified stable T_{avg}. After steady-state power is achieved, no divergent oscillations in temperature occur
- The steam header pressure controller responds to maintain a stable pressure at normal no-load pressure

14.2.10.4.13 Steam Generator Level Control System

Objective

Verify the stability of the automatic steam generator level control system by introducing simulated transients at various power levels during escalation to full power.

Prerequisites

- The reactor is critical and stable at various power levels during the power escalation test program. (Typical power levels are 30, 75 and 90 percent of full rated thermal power)
- The steam generator level control system is checked and calibrated
- Steam generator alarm setpoints are set for each generator

Test Method

- At each power level, with the steam generator control system in manual mode, simulate level transients by changing the level setpoint. Verify the steam generator level control response when the control system is returned to automatic control.
- Verify the variable speed features of the main feedwater pumps by manipulating controllers and test input signals.

Performance Criteria

- During recovery from a simulated steam generator level transient, steam generator level control response is consistent with the design for the following: overshoot or undershoot to the new level, time required to achieve the new level, and error between the actual level and control setpoint.
- Feedwater pump discharge pressure oscillations are less than design test limits

- The main feedwater control valves open and stabilize in response to various steam flow conditions in accordance with design requirements discussed in subsection 7.7.1.8.

14.2.10.4.14 Radiation and Effluent Monitoring System

Objectives

- For monitors that:
  - Are used for establishing conformance within the safety limits or limiting conditions for operation that are included in the Technical Specifications, or
  - Are classified as engineered safety features, or are relied on to support operation of the engineered safety features within design limits, or
  - Are assumed to function or for which credit is taken in the accident analysis of the facility, and
  - Are used to process, store, control, or limit the release of radioactive materials
- The objectives are:
  - Verify the calibration of the process and effluent radiation monitor against an acceptable standard
  - Establish baseline activity and background levels
  - Demonstrate that process and effluent radiation monitoring systems respond correctly by performing independent analyses

Prerequisites

- The plant is stable at the desired power level
- The sampling systems for the process and effluent radiation monitoring systems are operable

Test Method

- Perform calibrations with the use of radioactive sources to verify proper operation of the monitors and detectors
- Collect and analyze samples with laboratory instruments, and compare the results from the process and effluent monitor to verify proper monitor operation
- Establish background levels at low power (less than 5 percent rated thermal power)
- Establish background levels and baseline activity levels determined by sampling at 100 percent rated thermal power to monitor the buildup of activity

Performance Criteria

- Radiation monitors are calibrated against radioactive standards
- Baseline activities are established
- Laboratory analyses agree, given sensitivity and energy response, with the process and effluent radiation monitors

14.2.10.4.15 Ventilation Capability

Objective

Verify that heating, ventilation, and air conditioning systems for the containment and areas housing engineered safety features continue to maintain design temperatures.

Prerequisite

The plant is operating at or near the desired power (0, 50, and 100 percent of rated power).

Test Method

- Record temperature readings in specified areas while operating with normal ventilation lineups
- Record temperature readings in specified areas while operating the designed minimum number of heating ventilation and air conditioning components consistent with existing plant conditions
- Record surface concrete temperatures adjacent to the high temperature piping penetrations and at selected locations on the concrete shielding (at 100 percent rated thermal power only)

Performance Criterion

The heating, ventilation and air conditioning systems for the containment and areas housing engineered safeguards features perform as designed in accordance with Subsections 9.4.1 and 9.4.6.

14.2.10.4.16 Biological Shield Survey

Objectives

- Document the radiation levels in accessible locations of the plant outside of the biological shield while at power
- Obtain baseline radiation levels for comparison with future measurements of level buildup with operation

Prerequisites

- Radiation survey instruments are calibrated
- Background radiation levels are measured in designated locations prior to initial criticality
- The plant is stable at the applicable power level

Test Method

Measure gamma and neutron radiation dose rates at designated locations at approximately 25, 50, 75, and 100 percent rated thermal power.

Performance Criterion

Radiation levels are acceptable for full-power operation and consistent with design expectations.

14.2.10.4.17 Thermal Power Measurement and Statepoint Data Collection

Objective

Obtain thermal power measurement and statepoint data at selected power levels during the power ascension testing program, typically at 25, 50, 75, and 100 percent of rated thermal power.

Prerequisites

- The following equipment is installed and is operational: sensors for measuring steam generator feedwater temperature, differential pressure measuring devices for determining feedwater flow to each steam generator, and pressure gauges to measure steam pressure at steam generator outlets.

- The pressurizer pressure and level control system, and the steam generator level control system are in automatic mode.

- Instrumentation and data collection equipment is available for logging supplemental plant data.

- Reactor power is stable at the required level.

Test Method

The required data are obtained using installed plant equipment, special test equipment, and the plant data processing equipment. These data are subsequently used to determine reactor thermal power and assess the performance of the plant.

Performance Criterion

Reactor thermal power is stable at each power level and at the rated level at full power conditions. Operability of the pressurizer pressure and level control systems not previously verified as part of reactor coolant system preoperational testing (subsection 14.2.9.1.1) is demonstrated.

14.2.10.4.18 Dynamic Response

Objectives

Demonstrate during power range testing that the stress analysis for selected systems and components, under transient conditions is within design functional requirements. Portions of systems that meet the selection criteria for subsection 14.2.9.1.7 for dynamic effects testing, but were not tested because system conditions during hot functional testing are not conducive to prototypical systems conditions, are tested.

Prerequisites

- Temporary instrumentation is installed, as required, to monitor the deflections of components under test and the occurrence of water hammer noise and vibration.

- Points are monitored and baseline data are established.

Test Method

- Record deflection measurements during various plant transients.
- Monitor for the occurrence of water hammer noise and vibration.

Performance Criteria

- The movements due to flow-induced loads do not exceed the stress analysis of the monitored points. See subsection 3.9.2.1.1 for the acceptable standard for alternating stress intensity due to vibration.

- Flow-induced movements and loads do not cause malfunctions of plant equipment or instrumentation.

- No effects due to water hammer are detected.

14.2.10.4.19 Reactor Power Control System

Objective

Demonstrate the capability of the reactor power control system to respond to input signals.

Prerequisites

- The reactor is at equilibrium at the power level specified by the startup test program reference document.

- Setpoints and controls for the pressurizer, steam generator steam dump, and feedwater pump are checked and are set to proper values.

Test Method

Vary T_{avg} from the T_{ref} setpoint to verify the transient recovery capabilities of the automatic reactor power control system.

Performance Criterion

T_{avg} returns to the T_{ref} setpoint, within pre-specified limits and without manual intervention.

14.2.10.4.20 Load Swing Test

Objective

Verify nuclear plant transient response, including automatic control system performance, when 10 percent step-load changes are introduced to the turbine-generator at 30, 75, and 100 percent rated thermal power levels.

Prerequisite

The plant is operating in a steady-state condition at the desired thermal power level.

Test Method

Change the turbine-generator output as rapidly as possible to achieve a step 10 percent load increase or decrease. Monitor and record plant parameters of reactor power, reactor coolant system temperature, pressurizer pressure and level, and steam generator pressure and level during the load transients. Core power should not exceed 100-percent power as indicated by the excore nuclear instrumentation.

Performance Criterion

The primary and secondary control systems, with no manual intervention, maintain reactor power, reactor coolant system temperatures, pressurizer pressure and level, and steam generator levels and pressures within acceptable ranges during and following the transient. Control system response is reviewed and compared to the control system setpoint and performance analysis, and adjustments to the control systems are made, if necessary, prior to proceeding to the next power plateau.

14.2.10.4.21 100 Percent Load Rejection

Objective

Demonstrate the ability of the AP1000 plant to accept a 100 percent load rejection from full power.

Prerequisites

- The plant is operating at a stable power level of approximately 100 percent rated thermal power. Reactor and turbine control systems are in the automatic mode of operation. Plant temperatures, pressures, levels, and flow rates are within their normal range for full-power operation.
- Startup testing of the reactor and turbine control and protection systems is completed, and final setpoints are installed according to applicable plant technical manuals.
- The incore instrumentation system, including signal processing software, is operational, and all preoperational and startup testing is completed.
- Instrumentation and data collection equipment is operational and available for logging plant data.
- Special test instrumentation is installed and operational as required to augment normal data logging ability.

Test Method

- With the plant at nominal full-power steady-state conditions, to effect a rejection of 100 percent load, manually place the main step-up transformer high side breaker in the trip position.
- Prior to the load rejection, and until the plant stabilizes at the lower power level, record key plant parameters using the plant computer and special test instrumentation. The key plant parameters include plant temperatures, pressures, levels and flow rates for the primary and secondary systems.

Performance Criteria

- The plant is capable of accepting a 100 percent load rejection from full rated thermal power without reactor trip or operation of the steam generator relief valves or pressurizer safety valves.
- The turbine speed does not exceed 108% of rated speed.
- The turbine is capable of continued stable operation at the minimum house loads.

14.2.10.4.22 Load Follow Demonstration (First Plant Only)

Objective

- Demonstrate the ability of the AP1000 plant to follow a design basis daily load follow cycle.
- Demonstrate the ability of the plant to respond to grid frequency changes while in the load follow cycle.

Prerequisites

- The plant is operating at a stable power level of approximately 100 percent power and has been at that power for a sufficient length of time to have reached an equilibrium xenon condition.

- Startup testing of the reactor and turbine control and protection systems are completed, and final setpoints are installed.

- The incore instrumentation system, including signal processing software, is operational. All preoperational and startup testing is completed.

- Instrumentation and data collection equipment is operational and available for logging plant data.

Test Method

- Prior to any load reduction, obtain thermal power measurement and statepoint data along with incore power distribution maps to serve as the reference plant condition.

- Using normal plant procedures, reduce turbine load at a rate such that a reactor thermal power level of approximately 50 percent is achieved linearly in 2 hours.

- After remaining at 50 percent rated thermal power for more than 2 hours but less than 10 hours, increase turbine load at a rate such that a reactor power level of approximately 100 percent rated thermal power is achieved linearly in 2 hours.

- At selected times during the power decrease, while at reduced power, during the power increase, and after reaching approximately full rated thermal power, obtain data from both incore and excore instrumentation to monitor plant performance.

- While within the load-follow maneuver, demonstrate the ability to respond to grid frequency changes by increasing and decreasing load by as much as 10 percent, at a rate of 2 percent per minute.

Performance Criteria

- Core power distribution limits, as specified in the plant Technical Specifications, are not exceeded when the plant power is varied according to the design basis load-follow cycle, or while in the cycle, responding to load changes simulating grid frequency changes.

- Load follow maneuvers, including response to grid frequency changes, can be accomplished without changes to the reactor coolant boron concentration.

14.2.10.4.23 Hot Full Power Boron Endpoint

Objective

Measure the reactor coolant system critical boron concentration at beginning of cycle life for the all rods out, hot full power, xenon equilibrium condition.

Prerequisites

- The reactor is operating at approximately 100 percent of full licensed power and has been at that power for a sufficient time to reach xenon equilibrium.

- The reactor power level and reactor coolant system boron concentration and temperature are stable, and control and shutdown rod banks are in the near fully withdrawn position.
- Current core burnup data are available.

Test Method

- During the power ascension test program, and, as soon as practicable after achieving xenon equilibrium at full licensed power, obtain and analyze samples of reactor coolant for dissolved boron content.
- Using plant calorimetric and statepoint data obtained at the same time as coolant sampling, correct the measured boron concentration, as required, for control rod insertion, xenon nonequilibrium, and any difference between T_{avg} and T_{ref}.
- The resultant boron value, corresponding to the measured critical boron concentration for all rods out, hot full power, and xenon equilibrium, is compared with design predictions for the current accumulated core burnup (Figure 4.3-3).
- As permitted by the plant Technical Specifications, use the corrected measured boron concentration to renormalize the predicted curve of boron concentration as a function of core burnup.

Performance Criterion

The reactivity equivalent of the difference between measured and predicted boron concentrations (Table 4.3-2) is less than the design limit shown in subsection 4.3.3.3.

14.2.10.4.24 Plant Trip from 100 Percent Power

Objectives

- Verify the ability of the plant automatic control systems to sustain a trip from 100 percent rated thermal power and bring the plant to stable conditions following the transient.
- Assess the dynamic response of the plant for the event that subjects the turbine to its maximum credible overspeed condition.
- Determine the overall response time of the hot leg resistance temperature detector.
- Optimize the control systems setpoints, if necessary.

Prerequisite

The plant is operating in a steady-state condition at full rated thermal power.

Test Method

- Trip the plant by opening the main generator breaker.
- Monitor and record selected plant parameters.
- If necessary, adjust the control systems setpoints to obtain optimal response.

Performance Criteria

- Following the opening of the main generator breaker while at 100 percent rated thermal power, primary and secondary control systems and operator actions can stabilize reactor coolant system temperature, pressurizer pressure and level, and steam generator levels to no-load operating temperature and pressure.

- The steam dump control system operates to prevent opening of primary and secondary safety valves.

- The hot leg resistance temperature detector (RTD) time responses are verified to be less than or equal to values used in the safety analysis.

- The turbine speed does not exceed 108% of rated speed.

14.2.10.4.25 Thermal Expansion

Objective

Demonstrate that essential nuclear steam supply system and balance-of-plant components can expand without obstruction and that the expansion is in accordance with design. Also, during cooldown, the components return to their approximate baseline cold position. Testing is conducted to resolve discrepancies from hot functional testing as in subsection 14.2.9.1.1, and to test modifications made since hot functional testing was completed. Portions of systems that meet the selection criteria for subsection 14.2.9.1.7 for thermal dynamic testing, but were not tested because system conditions during hot functional testing are not conducive to prototypical system conditions are tested.

Prerequisite

Temporary instrumentation is installed, as required, to monitor the deflections for the components under test.

Test Method

For the components tested, the following apply:

- During plant heatup and cooldown, record deflection data.
- Verify support movements by recording hot and cold positions.

Performance Criteria

Thermal expansion testing is performed in accordance with subsection 3.9.2.1.2. For the components tested, the following apply: RN-15-022

- There is no evidence of blocking of the thermal expansion of piping or component, other than by installed supports, restraints, and hangers.

- Spring hanger movements must remain within the hot and cold setpoints and supports must not become fully retracted or extended.

- Piping and components return to their approximate baseline cold position.

14.2.10.4.26 Loss of Offsite Power

Objective

Demonstrate plant response following a plant trip with no offsite power available.

Prerequisites

- The plant is at minimum power level supplying normal house loads through the unit auxiliary transformers.
- The unit is disconnected from the electrical grid.

Test Method

- The turbine is tripped and the generator output breaker opens, removing ac power from the unit auxiliary transformers.

Performance Criteria

- The reactor trips.
- Both standby diesel generators start and pick up the required loads in the proper sequence.
- Class 1E dc and non-1E dc loads are uninterrupted and are provided by the battery subsystems.
- The primary plant is placed in a stable condition.

14.2.10.4.27 Feedwater Heater Loss and Out of Service Test

Objective

Demonstrate the plant response to the loss of one of the feedwater heaters during power operation due to single failure or operator error. Demonstrate the plant response to a pair of feedwater heaters taken out of service during power operation. Verify the ability of operators to manually reduce steam flow and place a pair of feedwater heaters out of service while maintaining reactor power operation.

Prerequisites

The plant is operating in a steady-state condition at the rated thermal powers described.

Test Method

LOSS OF FEEDWATER HEATER

- With the plant operating at 50% power, isolate the extraction steam supply to one of the main feedwater heaters.
- With the plant operating at 90% power, isolate the extraction steam supply to one of the main feedwater heaters.

FEEDWATER OUT OF SERVICE TEST

- The operators calculate the appropriate steam flow reduction which will maintain the plant at the desired thermal load after the heaters have been taken out of service.

- Reduce steam flow by the appropriate amount and allow plant conditions to reach a new steady-state (approximately 10 minutes).
- Take a pair of feedwater heaters out of service.

Performance Criteria

The plant control systems properly respond to the loss of a main feedwater heater, without reactor or turbine trip.

The operator successfully removes a pair of feedwater heaters from service without causing a reactor trip.

14.2.10.4.28 Remote Shutdown Workstation

Objective

Demonstrate the ability of the operators to conduct a remote shutdown of the plant during a simulated main control room evacuation.

Prerequisites

Approved operation procedures for performing a remote shutdown is available. Communication exists between the control room and the remote shutdown room. Procedures for transferring control back to the main control room are available if an emergency or unsafe condition develops during the testing that cannot be managed by the shutdown crew.

The plant is operating in a steady-state condition at 10-20 percent of power.

Test Method

- Using the appropriate operating procedures, the operators transfer control of the plant from the main control room to the remote shutdown workstation.
- From the remote shutdown workstation, the operators bring the plant to hot standby, and maintain hot standby conditions for at least 30 minutes.
- From the remote shutdown workstation, the operators lower the reactor coolant system pressure and temperature to the appropriate conditions, and place the normal residual heat removal system into service. The normal residual heat removal system, in conjunction with the component cooling water system and service water system are used to cool the plant at least 50°F without exceeding prescribed cooldown limits.

Performance Criteria

The operators successfully demonstrate the ability transfer control of the plant to the remote shutdown workstation, shut down the reactor, maintain hot standby, and then demonstrate the ability to transition to cold shutdown conditions, while performing these operations from the remote shutdown workstation.

14.2.10.4.29 Cooling Tower(s)

Objectives

- Verify proper cooling tower(s) function. Provide thermal acceptance testing of the cooling tower’s heat removal capabilities.

Prerequisites

- The cooling tower(s) is structurally complete and in good operating condition.
- Circulating water system testing is complete.
- Required support systems, electrical power supplies, and control circuits are operational.

Test Method

Thermal performance of the cooling towers is tested and verified using established industry test standards.

Performance Criteria

The cooling tower(s) perform as described in subsection 10.4.5 and in appropriate design specifications.

14.3 Certified Design Material

This section provides the selection criteria and processes used to develop the AP1000 Certified Design Material (CDM). This document provides the principal design bases and design characteristics that are certified by the 10 CFR Part 52 rulemaking process and included in the design certification rule.

The top-level design information in the Certified Design Material is extracted directly from the AP1000 design information. Limiting the certified design contents to top-level information reflects the tiered approach to design certification endorsed by the U.S. Nuclear Regulatory Commission (see References 1 through 5).

The objective of this section is to define the bases and methods that were used to develop the Certified Design Material for the AP1000. This section contains no new technical information regarding the AP1000 design.

The AP1000 Certified Design Material consists of the following:

- An introduction section which defines terms used in the Certified Design Material and lists general provisions that are applicable to all Certified Design Material entries. Also included is a list of acronyms and legends used in the Certified Design Material. (Because this material is self-explanatory, it is not discussed in this section.)

- Design descriptions for selected systems that are within the scope of the AP1000 design certification, and the applicable portions of those selected systems that are only partially within the scope of the AP1000 design certification. The Certified Design Material design descriptions delineate the principal design bases and principal design characteristics that are referenced in the design certification rule. The design descriptions are accompanied by the inspections, tests, analyses, and acceptance criteria (ITAAC) required by 10 CFR 52.47(a)(1)(vi) to be part of the design certification application. The ITAAC define verification activities that are to be performed for a facility with the objective of confirming that the plant is built and will operate in accordance with the design certification. Completion of these certified design ITAAC, together with the Combined License applicant's ITAAC for the site-specific portions of the plant, will be the basis for NRC authorization to load fuel per the provisions of 10 CFR Part 52.103.

- Design descriptions and their associated ITAAC for design and construction activities that are applicable to more than one system. Design-related processes have been included in the Certified Design Material for:

  - Aspects of the AP1000 design likely to undergo rapid, beneficial technological developments in the lifetime of the design certification. Certifying the design processes associated with these areas of the design, rather than specific design details, permits future license applicants referencing the AP1000 design certification to take advantage of the best technology available at the time of combined license application and facility construction.

  - Aspects of the design dependant upon characteristics of as-procured, as-installed systems, structures, and components. These characteristics are not available at the time of certification and, therefore, cannot be used to develop and certify design details.

  - Aspects of the seismic, structural and piping design for which detailed design has not been developed. These details are not available at the time of certification and, therefore, cannot be used to certify design details. Certifying the design processes associated with these design details provides the basis for future license applicants referencing the AP1000 design certification to establish and implement seismic, structural and piping design details as part of the COL application process.

- Interface requirements as defined by 10 CFR Part 52.47(a)(1)(vii). Interface requirements are defined as those which must be met by the site-specific portions of the complete nuclear power plant that are not within the scope of the certified design. These requirements define characteristics of the site-specific features that must be provided for the certified design to comply with certification commitments. AP1000 has no interfaces meeting this definition. The Certified Design Material does not include ITAAC or a requirement for COL developed ITAAC for interface requirements.

- Site parameters used as the basis for AP1000 design presented in the Tier 2 Material. These parameters represent a bounding envelope of site conditions for any license application referencing the AP1000 design certification. No ITAAC are necessary for the site parameters entries because compliance with site parameters will be verified as part of issuance of a license for a plant that references the AP1000 design certification.

The following is a description of the criteria and methods used to select specific technical entries for the Certified Design Material. The structure of the description is based on the Certified Design Material report structure.

The criteria and methods discussed in the following sections are guidelines only. For some matters, the contents of the Certified Design Material may not directly correspond to these guidelines because special considerations related to the matters may warrant a different approach. For such matters, a case-by-case determination is made regarding how or whether the matters should be addressed in the Certified Design Material. These determinations are based upon the principles inherent in Part 52.

14.3.1 CDM Section 1.0, Introduction

This section provides definitions, general provisions, a figure legend, and a list of acronyms used in the AP1000 Certified Design Material.

Selection Criteria – Section 1.1 is used to define terms used throughout the Certified Design Material. Selection of entries is based on a judgment that a particular word/phrase merits definition – with particular emphasis on terms associated with implementation of the ITAAC. Section 1.2 contains a mixture of provisions that is selected on the basis that the provision is necessary to either define technical requirements applicable to multiple systems in the Certified Design Material or to provide clarification and guidance for future users of the Certified Design Material.

Selection Methodology – Entries in the Definition section are made on the basis of a self-evident need for a term to be defined. These terms are accumulated during the preparation and review of the Certified Design Material. Entries in the General Provisions section also are developed as part of the Certified Design Material selection and review process. Each entry has a unique background, but the overall intent is to state the broad guidelines and interpretations that are used to prepare Certified Design Material for the AP1000.

14.3.2 CDM Section 2.0, System Based Design Descriptions and ITAAC

This section of the Certified Design Material has the design description and ITAAC material for the selected AP1000 systems. The intent of this list of AP1000 systems is to define at the Certified Design Material level the full scope of the certified design.

14.3.2.1 Design Descriptions

The certified design descriptions for selected AP1000 systems address the top-level design features and performance standards that pertain to the safety of the plant and include descriptive text and supporting figures. The intent of the Certified Design Material design descriptions is to define the AP1000 design characteristics referenced in the design certification rule as a result of the certification provisions of 10 CFR Part 52.

Selection Criteria – The following criteria are considered in determining the information included in the certified design descriptions:

- The information in the certified design descriptions is selected from the technical information presented in the Tier 2 Material. This reflects the approach that the Certified Design Material contains top-level design information and is based on the NRC directive in Reference 2 that there "be less detail in a certification than in an application for certification." In this context, the certification is the Certified Design Material and the application for certification includes the Tier 2 Material.

- The certified design descriptions contain only the information from the Tier 2 Material that is most significant to safety. The Tier 2 Material contains a wide spectrum of information on various aspects of the AP1000 design. Not all of this information is included in the certified design descriptions. This selection criterion reflects the NRC directive in Reference 2 that the certified design should "encompass roughly the same design features that Section 50.59 prohibits changing without prior NRC approval." In determining those structures, systems, or components for which certified design descriptions and ITAAC must be prepared, the following questions are considered for each structure, system, or component:

  – Are there any features or functions classified as Class A, B, or C?

  – Are there any defense–in–depth features or functions provided?

  – For nonsafety–related systems, are there any features or functions credited for mitigation of design basis events?

  – For nonsafety–related systems, are there any features or functions that have been identified in Section 16.3 as candidates for additional regulatory oversight?

  If the answer to the first question is yes, then a certified design description and ITAAC are prepared using the safety function stated in the Tier 2 Material and the parameters from the safety analysis.

  If the answer to either of the next two questions is yes, then a certified design description and ITAAC are prepared using the functions stated in the Tier 2 Material and the parameters from the system design calculations.

  If the answer to the last question is yes and the feature or function is not a programmatic requirement related to operations, maintenance or other programs, then a certified design description and ITAAC are prepared using the functions stated in the Tier 2 Material and the parameters from system design calculations.

  In addition, the following questions were considered for each structure, system, or component not already selected for ITAAC using the above selection criteria:

– Are any features or functions necessary to satisfy the NRC's regulations in Parts 20, 50, 52, 73 and 100?

– Are there any features or functions that represent an important assumption for probabilistic risk assessment?

– Are any features or functions important in preventing or mitigating severe accidents?

– Are there any features or functions that have a significant impact on the safety and operation of the plant?

– Are any features or functions the subject of a provision in the Technical Specifications?

If the answer to any of the above questions is yes, then a design description and ITAAC are prepared using the appropriate functions stated in the Tier 2 material and the parameters from the system design calculations.

A summary of the AP1000 structures, systems, or components considered for selection is given in Table 14.3-1.

- In general, safety-related and defense-in-depth features and functions of structures, systems, and components are discussed in the certified design descriptions. Structures, systems, and components that are not classified as safety-related or defense-in-depth are discussed in the certified design descriptions to the extent that they have features or functions that mitigate a design basis event.

- The certified design descriptions for structures, systems, and components are limited to a discussion of design features and functions. The design bases of structures, systems, and components, and explanations of their importance to safety, are provided in the Tier 2 Material and are not included in the certified design descriptions. The Certified Design Material design descriptions define the certified design. Justification that the design meets regulatory requirements is presented in the Tier 2 Material.

- The certified design descriptions focus on the physical characteristics of the facility. The certified design descriptions do not contain programmatic requirements related to operating conditions or to operations, maintenance, or other programs. These matters are controlled by other means such as the technical specifications.

- The certified design descriptions in Section 2.0 of the Certified Design Material discuss the functional arrangement and performance characteristics that the structures, systems, and components should have after construction is completed. In general, the certified design descriptions do not address the processes that will be used for designing and constructing a plant that references the AP1000 design certification. This is acceptable because the safety-function of a structure, system, or component is dependent upon its final as-built condition and not the processes used to achieve that condition. Exceptions to this criterion are the selected design and qualification processes defined in the instrumentation and control portions and piping portions of Section 2 and the piping, seismic, structural and human factors portion of Section 3.

  The programmatic aspects of the design and construction processes (training, qualification of welders, and the like) are part of the licensee's programs and are subject to commitments made at the time of combined license issuance. Consequently, these issues are not addressed in the AP1000 Certified Design Material.

- The certified design descriptions address fixed design features expected to be in place for the lifetime of the facility. Portable equipment and replaceable items are controlled through operational related programs.

- The certified AP1000 design descriptions do not discuss component types (for example, valve and instrument types), component internals, or component manufacturers. This approach is based on the premise that the safety function of a particular design element can be performed by a variety of component types from different manufacturers.

- The certified design descriptions do not contain proprietary information.

- For the applicant or licensee of a plant that references the AP1000 design certification to take advantage of improvements in technology, the certified design descriptions in general do not prescribe design features that are the subject of rapidly evolving technology.

- The Certified Design Material design description is intended to be self-contained and does not make direct reference to the Tier 2 Material, industrial standards, regulatory requirements, or other documents. (There are some exceptions involving the ASME Code and the Code of Federal Regulations.) If these sources contain technical information of sufficient safety significance to warrant Certified Design Material treatment, the information is extracted from the source and included directly in the appropriate system design description.

  This approach is appropriate because it is unambiguous and it avoids potential questions regarding how much of a referenced document is encompassed in, and becomes part of, the Certified Design Material.

- Selection of the technical terminology to be used in the Certified Design Material is guided by the principle that the terminology should be as consistent as possible with that used in the Tier 2 Material and the body of regulatory requirements and industrial standards applicable to the nuclear industry. This approach is intended to minimize problems in interpreting Certified Design Material commitments.

A review of those sections of the AP1000 Tier 2 Material that document plant safety evaluations was conducted. Specifically, reviews were conducted of the following chapters of the AP1000 Tier 2 Material; the flooding analysis in Chapter 5 the analysis of overpressure protection in Chapter 5, containment analysis in Chapter 6, the core cooling analysis in Chapters 6 and 15, the analysis of fire protection in Chapter 9, the safety analysis of transients in Chapter 15, the analysis of anticipated transients without scram (ATWS) in Chapters 7 and 15, the radiological analysis in Chapter 15, the resolution of unresolved or generic safety issues and Three Mile Island issues in Chapter 1, and the PRA and severe accident information in Chapter 19. These reviews were important in identifying safety-related system design information warranting consideration in the design descriptions and the accompanying design commitments.

Selection Methodology – The Certified Design Material uses a system report structure. The certified design description entry for any system is based on review of the multiple sources having technical information related to that system. Using the selection criteria listed, design description material is developed for each system by reviewing the Tier 2 Material, safety analysis, test programs, and design documents relating to that system.

Application of the criteria listed results in a graded treatment of the systems. This leads to variation in the scope of the design description entries. The following lists the types of AP1000 systems and is a summary of this graded treatment:

System Type	Scope of Design Description
Systems with safety-related functions that contribute to plant performance during design basis accidents	Major safety-related features and performance characteristics
Systems with defense-in-depth functions that contribute to plant performance during design basis accidents	Major defense-in-depth features and performance characteristics
Nonsafety-related systems potentially impacting safety	Brief discussion of design features that prevent or mitigate the potential safety concern
Nonsafety-related systems with no relationship to safety	No discussion

For safety-related systems, application of this criteria results in design description entries that include the following information, as applicable:

- System name and scope
- System purpose
- Summary of the system's safety-significant components (usually shown by a figure)
- Equipment seismic and ASME classifications
- Piping ASME classification and Leak-Before-Break criteria
- Type of electrical power provided for the system
- System's important instruments, controls, and alarms to the extent located in the main control room or remote shutdown workstation
- Equipment to be qualified for harsh environments
- Motor-operated valves within the system that have an active safety-related function
- Other features or functions that are significant to safety

The certified design descriptions for nonsafety-related systems include the information listed to the extent that the information is relevant to the system and is significant to safety. Since much of this information is not relevant to nonsafety-related systems, the certified design descriptions for nonsafety-related systems are less extensive than the descriptions for safety-related systems.

14.3.2.2 Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)

A table of ITAAC entries is provided for each system that has design description entries. The intent of these ITAAC is to define activities that will be undertaken to verify the as-built system conforms with the design features and characteristics defined in the design description. ITAAC are provided in tables with the following three-column format:

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria

Each design commitment in the left-hand column of the ITAAC tables has an associated inspections, tests, or analyses (ITA) requirement specified in the middle column. The acceptance criteria for the ITA are defined in the right-hand column.

Design Acceptance Criteria (DAC)/ITAAC closure is outlined in Appendix 14A.

Selection Criteria – The following are considered when determining what information is included in the Certified Design Material ITAAC entries:

- The scope and content of the ITAAC correspond to the scope and content of the certified design descriptions. There are no ITAAC for aspects of the design not addressed in the design description. This is appropriate because the objective of the ITAAC design certification entries is to verify that the as-built facility has the design features and performance characteristics defined in the Certified Design Material descriptions.

  Each AP1000 system with a design description has an ITAAC table. This reflects the assessment that a design feature meriting a Certified Design Material description also merits an ITAAC entry to verify that the feature has been included in the as-built facility.

- One inspection, test, or analysis may verify one or more provisions in the certified design description. An ITAAC that specifies a system functional test or an inspection may verify a number of provisions in the design description. There is not necessarily a one-to-one correspondence between the ITAAC and the design descriptions.

- As required by 10 CFR 52.103, the inspections, tests, and analyses must be completed (and the acceptance criteria satisfied) prior to fuel loading. Therefore, the ITAAC do not include any inspections, tests, or analyses that are dependent upon conditions that exist only after fuel load.

- Because the design descriptions are limited to fixed design features expected to be in place for the lifetime of the facility, the ITAAC are limited to a verification of fixtures in the plant. There are no ITAAC for nuclear fuel, fuel channels, and control rods because they are changed by a licensee.

- The ITAAC verify the as-built configuration and performance characteristics of structures, systems, and components as identified in the Certified Design Material design descriptions.

Selection Methodology – Using the selection criteria, ITAAC table entries are developed for each selected system. This is achieved by evaluating the design features and performance characteristics defined in the Certified Design Material design description and preparing an ITAAC table entry for the design description criteria that satisfied the selection criteria. There is a close correlation between the left-hand column of the ITAAC table and the corresponding design description entries.

The ITAAC table is completed by selecting the method to be used for verification (either a test, an inspection, or an analysis [ITA]) and the acceptance criteria for the as-built feature.

The selection of the ITAs is guided by the following:

ITA Approach	Application
Inspection	To be used when verification can be accomplished by visual observations, physical examinations, review of records based on visual observations, or physical examinations that compare the as-built structure, system, or component condition to one or more design description commitments.
Test	To be used when verification can be accomplished by the actuation or operation, or establishment of specified conditions, to evaluate the performance or integrity of the as-built structures, systems, or components. The type of tests identified in the ITAAC tables includes activities such as factory testing, special test facility programs, and laboratory testing.
Analysis	To be used when verification can be accomplished by calculation, mathematical computation, or engineering or technical evaluations of the as-built structures, systems, or components.

The proposed verification activity is identified in the middle column of the ITAAC table. Where appropriate, the Tier 2 Material provides details regarding implementation of the verification activity. This Tier 2 Material is not referenced in the Certified Design Material and is not part of the Certified Design Material; Tier 2 Material is considered as providing one of potentially several acceptable methods for completing the ITA.

Selection of acceptance criteria is dependent upon the design characteristic being verified by the ITAAC table entry: in most cases, the appropriate acceptance criteria is self-evident and is based upon the Certified Design Material design description. For many of the AP1000 ITAAC, the acceptance criteria is a statement that the as-built facility has the design feature or performance characteristic identified in the design description. A guiding principle for acceptance criteria preparation is the recognition that the criteria should be objective and unambiguous. The use of objective and unambiguous terms for the acceptance criteria will minimize opportunities for multiple, subjective (and potentially conflicting) interpretations as to whether an acceptance criteria has, or has not, been met. In some cases, the ITAAC acceptance criteria contain numerical parameters from the Tier 2 Material that are not specifically identified in the Certified Design Material design description or the design commitment column of the ITAAC table. This is acceptable because the design description defines the important design feature/performance that merits Certified Design Material treatment. The acceptance criterion defines a measurement standard for determining if the as-built facility is in compliance with the Certified Design Material design description commitment. Where appropriate, the Tier 2 Material identifies criteria applicable to the same design feature or function that is the subject of more general acceptance criteria in the ITAAC table.

For numerical acceptance criteria, ranges and/or tolerances are included. This is necessary and acceptable because of the following:

- Specification of a single-value acceptance criteria is impractical because trivial deviations will represent unnecessary noncompliances.
- Tolerances recognize that legitimate site variations can occur in complex construction projects.
- Minor variations in plant parameters within the tolerance bounds have no impact on plant safety.

14.3.2.3 Site-Specific ITAAC (SS-ITAAC)

A table of inspections, tests, analyses, and acceptance criteria (ITAAC) entries is provided for each site-specific system described in this FSAR that meets the selection criteria, and that is not included in the certified design. The intent of these ITAAC is to define activities that are undertaken to verify the as-built system conforms with the design features and characteristics defined in the system design description. ITAAC are provided in tables with the following three-column format:

Design Commitment	Inspection, Tests, Analyses	Acceptance Criteria

Each design commitment in the left-hand column of the ITAAC tables has associated inspections, tests, or analyses (ITA) requirements specified in the middle column. The acceptance criteria for the ITA are defined in the right-hand column.

SS-ITAAC do not address ancillary buildings and structures on the site, such as administrative buildings, parking lots, warehouses, training facilities, etc.

Selection Criteria — The following are considered when determining what information is included in the SS-ITAAC:

- In determining those structures, systems, or components for which ITAAC must be prepared, the following questions are considered for each structure, system, or component:

  – Are any features or functions classified as Class A, B, or C?

  – Are any defense-in-depth features or functions provided?

  – For nonsafety-related systems, are any features or functions credited for mitigation of design basis events?

  – For nonsafety-related systems, are there any features or functions that have been identified in Section 16.3 as candidates for additional regulatory oversight?

  If the answer to any of the above questions is yes, then ITAAC are prepared.

- The scope and content of the ITAAC correspond to the scope and content of the site-specific system design description.

- One inspection, test, or analysis may verify one or more provisions in the system design description. An ITAAC that specifies a system functional test or an inspection may verify a number of provisions in the system design description. There is not necessarily a one-to-one correspondence between the ITAAC and the system design descriptions.

- As required by 10 CFR 52.103, the inspections, tests, and analyses are completed (and the acceptance criteria satisfied) prior to initial fuel loading.

- The ITAAC verify the as-built configuration and performance characteristics of structures, systems, and components as identified in the system design descriptions.

Selection Methodology – Using the selection criteria, ITAAC table entries are developed for each selected system. This is achieved by evaluating the design features and performance characteristics defined in the system design descriptions and preparing an ITAAC table entry for each design description criterion that satisfies the selection criteria. A close correlation exists between the left-hand column of the ITAAC and the corresponding design description entries.

The ITAAC table is completed by selecting the method to be used for verification (either a test, an inspection, or an analysis) and the acceptance criteria for the as-built feature.

The approach used to perform the tests, inspections, or analyses is similar to that described in Subsection 14.3.2.2.

14.3.2.3.1 Emergency Planning ITAAC (EP-ITAAC)

EP-ITAAC have been developed to address implementation of elements of the Emergency Plan. Site-specific EP-ITAAC are based on the generic ITAAC provided in Appendix C.II.1-B of Regulatory Guide 1.206. These ITAAC have been tailored to the specific reactor design and emergency planning program requirements.

14.3.2.3.2 Physical Security ITAAC (PS-ITAAC)

Generic PS-ITAAC have been developed in a coordinated effort between the NRC and the Nuclear Energy Institute (NEI). These generic ITAAC have been tailored to the AP1000 design and site-specific security requirements.

14.3.2.3.3 Other Site-Specific Systems

One additional site-specific system has been determined to meet the ITAAC selection criteria, and ITAAC have been included for the Transmission Switchyard and Offsite Power System (ZBS) as indicated in Table 14.3-1. Systems not meeting the selection criteria are subject to the normal functional testing to verify that newly designed and installed systems, structures, or components perform as designed.

A summary of the AP1000 structures, systems, or components considered for selection is given in Table 14.3-1.

14.3.3 CDM Section 3.0, Non-System Based Design Descriptions and ITAAC

Entries in this section of the Certified Design Material have the same structure as the system material discussed in Subsection 14.3.2; that is, design description text and figures and a table of ITAAC entries. The objective of this Certified Design Material is to address selected design and construction activities which are applicable to more than one system. There are six entries in Section 3.0 of the Certified Design Material: nuclear island buildings, initial test program, emergency response facilities, human factors engineering, Design Reliability Assurance Program, and radiation protection.

14.3.3.1 Pipe Rupture Hazard Analysis ITAAC

A pipe rupture hazard analysis is part of the piping design. The analyses will document that structures, systems, and components (SSCs) which are required to be functional during and following a design basis event have adequate high-energy and moderate-energy pipe break mitigation features. The locations of postulated ruptures and essential targets will be established and required pipe whip restraint and jet shield designs will be included. The as-designed pipe rupture hazards analysis will be based on the as-designed piping analysis and will be in accordance with the criteria outlined in Subsections 3.6.1.3.2 and 3.6.2.5. The evaluation will address environmental and flooding effects of cracks in high and moderate energy piping. The report of the pipe rupture hazard analysis shall conclude that, for each postulated piping failure, the systems, structures, and components that are required to be functional during and following a design basis event are protected.

The as-built reconciliation of the pipe rupture hazards evaluation whip restraint and jet shield design in accordance with the criteria outlined in Subsections 3.6.1.3.2 and 3.6.2.5 are covered in as-built ITAAC identified in DCD Tier 1 to demonstrate that the as-built pipe rupture hazards mitigation features reflect the design, as reconciled. The reconciliation report will be made available for NRC inspection or audit when it has been completed.

The as-designed pipe rupture hazard analysis completed for the first standard AP1000 plant will be available to subsequent standard AP1000 plants under the "one issue, one review, one position" approach for closure.

14.3.3.2 Piping Design ITAAC

The piping design ITAAC consists of the piping analysis for safety-related ASME Code piping. The piping design is completed on a package-by-package basis for applicable systems. In order to support closure of the piping design ITAAC, information consisting of the as-designed piping analysis for piping lines chosen to demonstrate all aspects of the piping design will be made available for NRC review, inspection, and/or audit. This information will consist of a design report referencing the as-designed piping calculation packages, including ASME Section III piping analysis, support evaluations and piping component fatigue analysis for Class I piping. The piping packages to be analyzed are identified in the DCD.

The ASME Code prescribes certain procedures and requirements that are to be followed for completing the piping design. The piping design ITAAC includes a verification of the ASME Code design report to ensure that the appropriate code design requirements for each system's safety class have been implemented.

A reconciliation of the applicable safety-related as-built piping systems is covered in as-built ITAAC identified in DCD Tier 1 to demonstrate that the as-built piping reflects the design, as reconciled. The reconciliation report will be made available for NRC inspection or audit when it has been completed.

The piping design completed for the first standard AP1000 plant will be available to subsequent standard AP1000 plants under the "one issue, one review, one position" approach for closure.

14.3.4 Certified Design Material Section 4.0, Interface Requirements

AP1000 is a plant design incorporating the nuclear island, the annex building and associated equipment, the diesel/generator building and associated equipment, the turbine/generator building, the turbine/generator equipment, and the radwaste facilities. As a result, no interfaces need to be identified between or among these portions of the plant. There are no safety-related interfaces between the AP1000 certified design and other portions of a facility with a combined license under 10 CFR Part 52.

Initial testing of interfacing non-safety systems in portions of the plant outside the scope of design certification is as discussed in Section 14.4. Section 1.8, Table 1.8-1, lists the interfacing systems and structures. Those systems that meet the requirements of 10 CFR 52.47(a)(1)(viii) are tabulated in Subsection 14.4.5.

14.3.5 CDM Section 5.0, Site Parameters

This section of the Certified Design Material defines the site parameters used as a basis for the design defined in the AP1000 certification application. These entries respond to the

10 CFR 52.47(a)(1)(iii) requirement that the design certification documentation include site parameter information. It is intended that applicants referencing the AP1000 design certification demonstrate that these parameters for the selected site are within the certification envelope or provide additional analysis to show acceptability of deviations from the interface envelope.

Site-specific external events that relate to the acceptability of the design (and not to the acceptability of the site) are not considered site parameters and are addressed as interface requirements in the appropriate system entry in Section 4 of the Certified Design Material.

Section 5.0 of the Certified Design Material does not include any ITAAC and is limited to defining the AP1000 site parameters. This is an appropriate approach because compliance of the site with these parameters is demonstrated by a license applicant prior to issuance of the license.

Selection Criteria – Section 2.0, Table 2.0-1, provides the envelope of site design parameters used for the AP1000 design. The corresponding Certified Design Material Section 5.0 is based on using Table 2.0-1. Section 5.0 is limited to a tabular entry; no supporting text material is required.

14.3.6 Initial Test Program

The AP1000 Initial Test Program defines testing activities that will be conducted following completion of construction and construction-related inspections and tests. The Initial Test Program extends through the start of commercial operation of the facility. This program is discussed in Chapter 14.

A summary of the Initial Test Program is included in Certified Design Material Section 3.4. This summary includes an overview of the Initial Test Program structure. This information is included in the Certified Design Material because of the importance of the Initial Test Program defining pre- and post-fuel load testing for the as-built facility. Key pre-fuel load Initial Test Program testing for individual systems is defined in the system ITAAC in Certified Design Material Sections 2 and 3.

No ITAAC entries have been included in the Certified Design Material for the Initial Test Program. This is acceptable because of the following:

- The Initial Test Program activities involve testing with the reactor at various power levels and thus cannot be completed prior to fuel load (Part 52 requires ITAAC to be completed prior to fuel load).

- Testing activities specified as part of the ITAAC in Certified Design Material Sections 2 and 3 must be performed prior to fuel load. Because these ITAAC testing activities address the design features and characteristics of safety significance, additional ITAAC for the Initial Test Program are not necessary to ensure that the as-built plant conforms with the certified design.

14.3.7 Elements of AP1000 Design Material Incorporated into the Certified Design Material

Tables 14.3-2 through 14.3-8 summarize the design material that has been incorporated into the CDM in the areas of 1) Design Basis Accident Analysis, 2) Anticipated Transients Without Scram (ATWS), 3) Fire Protection, 4) Flood Protection, 5) Probabilistic Risk Assessment, 6) Radiological Analysis, and 7) Severe Accident Analysis. PRA assumptions incorporated into these tables encompass elements of the system design and assumptions that were expressly included in Tier 1 due to their importance. Both types of PRA assumptions were included for completeness, but are not distinguished in the tables. CDM falling outside of the seven subject areas are intentionally not incorporated in these tables. However, the referenced AP1000 DCD sections may contain more information than encompassed by these seven subject areas. Each table may also include design information (certified or non-certified) that is not directly related to the particular subject area. Further, these tables are not intended to include all system-specific CDM information that is provided in the AP1000 Tier 2 system descriptions.

14.3.8 Summary

An element of the design certification processes deriving from 10 CFR Part 52 is the selection and documentation of the technical information to be included in the design certification rule as the certified design. The certified design material is a subset of the design information presented in the Tier 2 Material. It includes the following:

- Key, important safety-significant aspects of the design described in the certification application
- Inspections, tests, analyses, and acceptance criteria (ITAAC) that will be used to verify that the as-built facility conforms with the certified design
- Interface requirements and site parameters

The information presented in the AP1000 Certified Design Material is prepared using the selection criteria and methodology described in this section and is intended to satisfy the above Part 52 requirements for design certification. The ITAAC entries in Sections 2.0 and 3.0 confirm that key design performance characteristics and design features are implemented in the as-built facility.

14.3.9 References

1. SECY-90-377, “Requirements for Design Certification under 10 CFR Part 52,” February 15, 1991.

2. 10 CFR, Part 52, “Statements of Consideration,” (54 Federal Register 15372 [1989]).

3. SECY-90-241, “Level of Detail Required for Design Certification under Part 52,” August 31, 1990.

4. SECY-90-377, “Requirements for Design Certification Under 10 CFR Part 52,” November 8, 1990.

5. SECY-91-178, “Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for Design Certifications and Combined Licenses,” June 12, 1991.

Table 14.3-1 (Sheet 1 of 4)
ITAAC Screening Summary

Structure/ System Acronym	Structure/ System Description	Selected for ITAAC
ADS	Automatic Depressurization System	X
ASS	Auxiliary Steam Supply System	X
BDS	Steam Generator Blowdown System	X
CAS	Compressed Air System	X
CCS	Component Cooling Water System	X
CDS	Condensate System	X
CES	Condenser Tube Cleaning System	X
CFS	Turbine Island Chemical Feed System	X
CMS	Condenser Air Removal System	X
CNS	Containment System	X
CPS	Condensate Polishing System	X
CVS	Chemical and Volume Control System	X
CWS	Circulating Water System	X
DAS	Diverse Actuation System	X
DDS	Data Display Processing System	X
DOS	Standby Diesel Fuel Oil System	X
DRS	Storm Drain System	XX
DTS	Demineralized Water Treatment System	X
DWS	Demineralized Water Transfer and Storage System	X
ECS	Main AC Power System	X
EDS	Non Class 1E DC and UPS System	X
EFS	Communication System	X
EGS	Grounding and Lightning Protection System	X

Table 14.3-1 (Sheet 2 of 4)
ITAAC Screening Summary

Structure/ System Acronym	Structure/ System Description	Selected for ITAAC
EHS	Special Process Heat Tracing System	X
ELS	Plant Lighting System	X
EQS	Cathodic Protection System	X
FHS	Fuel Handling System	X
FPS	Fire Protection System	X
FWS	Main and Startup Feedwater System	X
GSS	Gland Seal System	X
HCS	Generator Hydrogen and CO_2 Systems	X
HDS	Heater Drain System	X
HSS	Hydrogen Seal Oil System	X
IDS	Class 1E DC and UPS System	X
IIS	Incore Instrumentation System	X
LOS	Main Turbine and Generator Lube Oil System	X
MES	Meteorological and Environmental Monitoring System	XX
MHS	Mechanical Handling System	X
MSS	Main Steam System	X
MTS	Main Turbine System	X
OCS	Operations and Control Centers	X
OWS	Offsite Water Treatment System[a]	XX
PCS	Passive Containment Cooling System	X
PGS	Plant Gas System	X
PLS	Plant Control System	X
PMS	Protection and Safety Monitoring System	X
PSS	Primary Sampling System	X

Table 14.3-1 (Sheet 3 of 4)
ITAAC Screening Summary

Structure/ System Acronym	Structure/ System Description	Selected for ITAAC
PWS	Potable Water System	X
PXS	Passive Core Cooling System	X
RCS	Reactor Coolant System	X
RDS	Gravity and Roof Drain Collection System	X
RMS	Radiation Monitoring System	X
RNS	Normal Residual Heat Removal System	X
RWS	Raw Water System	XX
RXS	Reactor System	X
SDS	Sanitary Drainage System	X
SES	Plant Security System	X
SFS	Spent Fuel Cooling System	X
SGS	Steam Generator System	X
SJS	Seismic Monitoring System	X
SMS	Special Monitoring System	X
SSS	Secondary Sampling System	X
SWS	Service Water System	X
TCS	Turbine Building Closed Cooling Water System	X
TDS	Turbine Island Vents, Drains and Relief Systems	X
TOS	Main Turbine Control and Diagnostics System	X
TVS	Closed Circuit TV System	XX
VAS	Radiologically Controlled Area Ventilation System	X
VBS	Nuclear Island Nonradioactive Ventilation System	X
VCS	Containment Recirculation Cooling System	X
VES	Main Control Room Emergency Habitability System	X

Table 14.3-1 (Sheet 4 of 4)
ITAAC Screening Summary

Structure/ System Acronym	Structure/ System Description	Selected for ITAAC
VFS	Containment Air Filtration System	X
VHS	Health Physics and Hot Machine Shop HVAC System	X
VLS	Containment Hydrogen Control System	X
VPS	Pump House Building Ventilation System	NA
VRS	Radwaste Building HVAC System	X
VTS	Turbine Island Building Ventilation System	X
VUS	Containment Leak Rate Test System	X
VWS	Central Chilled Water System	X
VXS	Annex/Auxiliary Nonradioactive Ventilation System	X
VYS	Hot Water Heating System	X
VZS	Diesel Generator Building Ventilation System	X
WGS	Gaseous Radwaste System	X
WLS	Liquid Radwaste System	X
WRS	Radioactive Waste Drain System	X
WSS	Solid Radwaste System	X
WWS	Waste Water System	X
YFS	Yard Fire Water System	XX
ZAS	Main Generator System	X
ZBS	Transmission Switchyard and Offsite Power System	XX
ZOS	Onsite Standby Power System	X
ZRS	Offsite Retail Power System	NA
ZVS	Excitation and Voltage Regulation System	X

Legend: X= Selected for ITAAC
X = Selected for ITAAC - title only, no entry for Design Certification

XX = Site-specific system selected for ITAAC – title only, no entry for COLA
XX = Selected for ITAAC
NA = System is not part of VCS design

a) OWS is outside the scope of the standard AP1000 but will be on the VCSNS property.

Table 14.3-2 (Sheet 1 of 17)
Design Basis Accident Analysis

Reference		Design Feature	Value
Section	3.10	The protection and safety monitoring system equipment is seismically qualified to meet safe shutdown earthquake levels.	
Section	3.11.3	The design of the protection and safety monitoring system equipment has margin to accommodate a loss of the normal HVAC.	
Section	5.1.2	Safety valves are installed above and connected to the pressurizer to provide overpressure protection for the reactor coolant system.	
Section	5.1.2	The RCS has two hot legs and four cold legs.	
Section	5.1.2	The RCS has two steam generators and four reactor coolant pumps.	
Section	5.1.2	The RCS contains a pressurizer and a surge line connected to one hot leg.	
Section	5.1.3.3	Rotating inertia needed for flow coast-down, is provided.	
Table	5.1-3	Minimum measured flow rate with 10% tube plugging (gpm/ loop)	150,835
Table	5.1-3	Initial rated reactor core thermal power (MWt)	3400
Section	5.2.2	Reactor coolant system and steam system overpressure protection during power operation are provided by the pressurizer safety valves and the steam generator safety valves, in conjunction with the action of the PMS.	
Section	5.2.2.1	Safety valve capacity exists to prevent exceeding 110 percent of system design pressure for the following events: – Loss of electrical load and/or turbine trip – Uncontrolled rod withdrawal at power – Loss of reactor coolant flow – Loss of normal feedwater – Loss of offsite power to the station auxiliaries	
Section	5.2.2.1	Overpressure protection for the steam system is provided by steam generator safety valves	
Section	5.3.2.3	Non-destructive examination (NDE) of the reactor vessel and its appurtenances is conducted in accordance with ASME Code Section III requirements.	

Table 14.3-2 (Sheet 2 of 17) Design Basis Accident Analysis

Reference		Design Feature	Value
Section	5.3.2.5	The initial Charpy V-notch minimum upper shelf fracture energy levels for the reactor vessel beltline base metal transverse direction and welds are 75 foot-pounds, as required by Appendix G of 10 CFR 50.	
Section	5.4.1.2.1	Resistance temperature detectors (RTDs) monitor motor cooling circuit water temperature. These detectors provide indication of anomalous bearing or motor operation. They also provide a system for automatic shutdown in the event of a prolonged loss of component cooling water.	
Section	5.4.1.3.4	It is important to reactor protection that the reactor coolant continues to flow for a time after reactor trip and loss of electrical power. To provide this flow, each reactor coolant pump has a high-inertia rotor.	
Section	5.4.1.3.4	A safety-related pump trip occurs on high bearing water temperature.	
Section	5.4.5.2.3	Power to the pressurizer heaters is blocked when the core makeup tanks are actuated.	
Section	5.4.6	Automatic depressurization system stage 1, 2 and 3 valves are connected to the pressurizer and discharge via the spargers to the in-containment refueling water storage tank.	
Section	5.4.6	Automatic depressurization system stage 4 valves are connected to each hot leg.	
Section	5.4.9.3	In the analysis of overpressure events, the pressurizer safety valves are assumed to actuate at 2500 psia. The safety valve flowrate assumed is based on full flow at 2575 psia, assuming 3 percent accumulation.	
Section	5.4.9.3	The pressurizer safety valves prevent reactor coolant system pressure from exceeding 110% of system design pressure.	
Section	5.4.12	The reactor head vent valves can be operated from the main control room to provide an emergency letdown path.	
Table	5.4-1	Minimum reactor coolant motor/pump moment of inertia sufficient to provide flow coastdown as given in Figure 15.3.2.	
Table	5.4-11	Reactor Coolant System Design Pressure Settings: – Safety valves begin to open (psig)	2485

Table 14.3-2 (Sheet 3 of 17) Design Basis Accident Analysis

Reference		Design Feature	Value
Table	5.4-17	Pressurizer Safety Valves - Design Parameters: – Number – Minimum required relieving capacity per valve (lbm/hr) – Set pressure (psig)	 2 ≥ 750,000 2485 ± 25
Section	6.1.1.4	The exposed surfaces of the excore detectors are made of stainless steel or titanium.	
Table	6.1-2	The exterior of the containment vessel (above plant elevation 135′ 3″) and the interior of the containment vessel (above 7′ above the operating deck) is coated with an inorganic zinc coating.	
Section	6.1.2.1.5	The nonsafety-related coatings used inside containment on walls, floors, ceilings, structural steel which is part of the building structure, and on the polar crane have a minimum dry film density (lb/ft^3).	≥ 100
Figure	6.2.2-1	The passive containment cooling system consists of a water storage tank, cooling water flow discharge path to the containment shell, a water distribution system for the containment shell, and a cooling air flow path.	
Table	6.2.2-1	The minimum duration the PCS cooling water flow is provided from the PCCWST (hours).	≥ 72
Table	6.2.2-1	The water coverage of the containment shell exceeds the amount used in the safety analysis.	
Table	6.2.2-1	The minimum drain flow rate capacity of the upper annulus drain (gpm).	≥ 525
Table	6.2.2-1	The minimum makeup flow rate capability from an external source to the PCS water storage tank (gpm).	≥ 100
Table	6.2.2-1	The minimum makeup flow rate capability from the PCS water storage tank to the spent fuel pit (gpm).	≥ 118
Table	6.2.2-1	The minimum PCS water storage tank volume for makeup to the spent fuel pit (non-coincident with PCS operation) (gallons).	≥ 756,700
Table	6.2.2-1	The minimum long term makeup capability from the PCCAWST to the PCCWST (days).	≥ 4

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Table 14.3-2 (Sheet 4 of 17)
Design Basis Accident Analysis

Reference		Design Feature	Value
Table	6.2.2-1	The minimum long term makeup flow capability from the PCCAWST to the PCCWST (gpm).	≥ 100
Table	6.2.2-1	The minimum long term makeup flow capability from the PCCAWST to the spent fuel pool (gpm).	> 35
Table	6.2.2-2	The first (i.e., tallest) standpipe's elevation above the tank floor (feet).	24.1 ± 0.2
Table	6.2.2-2	The second tallest standpipe's elevation above the tank floor (feet).	20.3 ± 0.2
Table	6.2.2-2	The third tallest standpipe's elevation above the tank floor (feet).	16.8 ± 0.2
Table	6.2.2-1	The passive containment cooling water flow rate at 72 hours (gpm).	≥ 100.7
Table	6.2.2-1	The passive containment cooling water flow rate when the PCCWST water level uncovers the third tallest standpipe (gpm).	≥ 144.2
Table	6.2.2-1	The passive containment cooling water flow rate when the PCCWST water level uncovers the second tallest standpipe (gpm).	≥ 176.3
Table	6.2.2-1	The passive containment cooling water flow rate when the PCCWST water level uncovers the first (i.e., tallest) standpipe (gpm).	≥ 226.6
Table	6.2.2-1	The passive containment cooling water flow rate with water inventory at a level of 27.4 ft + 0.2, -0.0 ft above the tank floor (gpm).	≥ 469.1
Section	6.3	The passive core cooling system provides core decay heat removal during design basis events.	
Section	6.3	The passive core cooling system provides RCS makeup, boration, and safety injection during design basis events.	
Section	6.3	The passive core cooling system provides pH adjustment of water flooding the containment following design bases events.	

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Table 14.3-2 (Sheet 5 of 17)
Design Basis Accident Analysis

Reference	Design Feature	Value
Section 6.3.1.1	The passive core cooling system is designed to provide emergency core cooling during events involving increases and decreases in secondary side heat removal and decreases in reactor coolant system inventory.	
Section 6.3.2.1.1	The heat exchanger consists of a bank of C-tubes, connected to a tubesheet and channel heat arrangement at the top (inlet) and bottom (outlet). The passive exchanger connects to the reactor coolant system through an inlet line from one reactor coolant system hot leg and an outlet line to the associated steam generator cold leg plenum (reactor coolant pump suction).	
Section 6.3.2.1.1	For the passive residual heat removal heat exchanger, the normal water temperature in the inlet line will be hotter than the discharge line.	
Section 6.3.2.1.2	The actuation of the core makeup tanks following a steam line break provides injection of borated water via water recirculation to mitigate the reactivity transient and provide the required shutdown margin.	
Section 6.3.2.2.1	The CMT inlet diffuser has a minimum flow area (in^2).	≥ 165
Section 6.3.2.2.3	The in-containment refueling water storage tank contains one passive residual heat removal heat exchanger.	
Section 6.3.2.2.6	The connection of the sparger branch arms to the sparger hub are submerged below the in-containment refueling water storage tank overflow level (ft).	≤ 11.5
Section 6.3.2.2.6	Automatic depressurization system stage 1, 2 and 3 valves are connected to the pressurizer and discharge via the spargers to the in-containment refueling water storage tank.	
Section 6.3.2.2.7.1	The containment recirculation screens have plates that are located no more than 1 foot above the top of the screens and extend out at least 10 feet in front and at least 7 feet to the side of the screens to prevent coating debris from reaching the screens.	
Section 6.3.2.2.7.1	The type of insulation used on ASME Class 1 lines inside containment and on the reactor vessel, reactor coolant pumps, pressurizer and steam generators is a metal reflective or suitable equivalent insulation.	

Table 14.3-2 (Sheet 6 of 17)
Design Basis Accident Analysis

Reference		Design Feature	Value
Section	6.3.2.2.7.3	The surface materials used in the vicinity of the containment recirculation screens are stainless steel. In the vicinity of the containment recirculation screens includes surfaces located above the bottom of the recirculation screens up to and including the bottom surface of the plate discussed in subsection 6.3.2.2.7.1, and the surfaces 10 feet in front and 7 feet to the sides of the screen face.	
Section	6.3.2.2.7.3	The bottom of the containment recirculation screens are located above the loop compartment floor (ft).	≥ 2
Section	6.3.3.2.1	For a loss of main feedwater event, the passive residual heat removal heat exchanger is actuated. If the core makeup tanks are not initially actuated, they actuate later when passive residual heat exchanger cooling sufficiently reduces pressurizer level.	
Section	6.3.3.2.2	For a feedwater system pipe failure event, the passive residual heat removal heat exchanger and the core makeup tanks are actuated.	
Section	6.3.3.3.1	For a steam generator tube rupture event, the nonsafety-related makeup pumps are automatically actuated when reactor coolant system inventory decreases and a reactor trip occurs, followed by actuation of the startup feedwater pumps. Makeup pumps automatically function to maintain the programmed pressurizer level. The core makeup tanks subsequently actuate on low pressurizer level, if they are not already actuated. Actuation of the core makeup tanks automatically actuates the passive residual heat removal system heat exchanger.	
Section	6.3.6.1	The piping resistances connecting the following PXS components and the RCS are bounded by the resistances assumed in the Chapter 15 safety analysis: – Core makeup tanks – Accumulators – In-containment refueling water storage tank injection – Containment recirculation – Automatic depressurization system valves	
Section	6.3.6.1.3	The bottom of the core makeup tanks are located above the reactor vessel direct vessel injection nozzle centerline (ft).	≥ 7.5

Table 14.3-2 (Sheet 7 of 17)
Design Basis Accident Analysis

Reference		Design Feature	Value	
Section	6.3.6.1.3	The bottom of the in-containment refueling water storage tank is located above the direct vessel injection nozzle centerline (ft).	≥ 3.4	
Section	6.3.6.1.3	The pH baskets are located below plant elevation 107′ 2″.		
Figure	6.3-1	The passive core cooling system has two direct vessel injection lines.		
Table	6.3-2	The passive core cooling system has two core makeup tanks, each with a minimum required volume (ft^3).	2500	
Table	6.3-2	The passive core cooling system has two accumulators, each with a minimum required volume (ft^3)	2,000	
Table	6.3-2	The passive core cooling system has an in-containment refueling water storage tank with a minimum required water volume (ft^3)	73,900	
Section	6.3.2.2.3	The containment floodup volume for a LOCA in PXS room B has a maximum volume (ft^3) (excluding the IRWST) below a containment elevation of 108 feet.	73,500	
Table	6.3-2	Each sparger has a minimum discharge flow area (in^2).	≥ 274	
Table	6.3-2	The passive core cooling system has four pH adjustment baskets with a total minimum required volume (ft^3).	560	RN-15-021
Section	14.2.9.1.3f	The passive residual heat removal heat exchanger minimum natural circulation heat transfer rate (Btu/hr) – With 520°F hot leg and 80°F IRWST – With 420°F hot leg and 80°F IRWST	 ≥ 1.78 E+08 ≥ 1.11 E+08	
Section	6.3.6.1.3	The centerline of the HX's upper channel head is located above the HL centerline (ft).	≥ 26.3	
Section	6.3.7.4.1	The CMT level sensors (PXS-11A/B/C/D, -12A/B/C/D, -13A/B/C/D, and -14A/B/C/D) upper level tap line has a downward slope of ≥ 2.4 degrees from the centerline of the connection to the CMT to the centerline of the connection to the standpipe.	≥ 2.4 degrees	RN-14-073
Figure	6.3-1	The CMT inlet lines (cold leg to high point) have no downward sloping sections.		
Figure	6.3-1	The maximum elevation of the CMT injection lines between the connection to the CMT and the reactor vessel is the connection to the CMTs.		
Figure	6.3-2	The PRHR inlet line (hot leg to high point) has no downward sloping sections.		

Table 14.3-2 (Sheet 8 of 17)
Design Basis Accident Analysis

Reference		Design Feature	Value
Figure	6.3-2	The maximum elevation of the IRWST injection lines (from the connection to the IRWST to the reactor vessel) and the containment recirculation lines (from the containment to the IRWST injection lines) is less than the bottom inside surface of the IRWST.	
Figure	6.3-2	The maximum elevation of the PRHR outlet line (from the PRHR to the SG) is less than the PRHR lower channel head top inside surface.	
Section	7.1.2.10	Isolation devices are used to maintain the electrical independence of divisions and to see that no interaction occurs between nonsafety-related systems and the safety-related system. Isolation devices serve to prevent credible faults in circuit from propagating to another circuit.	
Section	7.1.4.2	The ability of the protection and safety monitoring system to initiate and accomplish protective functions is maintained despite degraded conditions caused by internal events such as fire, flooding, explosions, missiles, electrical faults and pipe whip.	
Section	7.1.2	The flexibility of the protection and safety monitoring system enables physical separation of redundant divisions.	
Section	7.2.2.2.1	The protection and safety monitoring system initiates a reactor trip whenever a condition monitored by the system reaches a preset level.	
Section	7.2.2.2.8	The reactor is tripped by actuating one of two manual reactor trip controls from the main control room.	
Section	7.3.1.2.2	The in-containment refueling water storage tank is aligned for injection upon actuation of the fourth stage automatic depressurization system via the protection and safety monitoring system.	
Section	7.3.1.2.3	The core makeup tanks are aligned for operation on a safeguards actuation signal or on a low-2 pressurizer level signal via the protection and safety monitoring system.	
Section	7.3.1.2.4	The fourth stage valves of the automatic depressurization system receive a signal to open upon the coincidence of a low-2 core makeup tank water level in either core makeup tank and low reactor coolant system pressure following a preset time delay after the third stage depressurization valves receive a signal to open via the protection and safety monitoring system.	

Table 14.3-2 (Sheet 9 of 17)
Design Basis Accident Analysis

Reference	Design Feature	Value
Section 7.3.1.2.4	The first stage valves of the automatic depressurization system open upon receipt of a signal generated from a core makeup tank injection alignment signal coincident with core makeup tank water level less than the Low-1 setpoint in either core makeup tank via the protection and safety monitoring system.	
Section 7.3.1.2.4	The second and third stage valves open on time delays following generation of the first stage actuation signal via the protection and safety monitoring system.	
Section 7.3.1.2.5	The reactor coolant pumps are tripped upon generation of a safeguards actuation signal or upon generation of a low-2 pressurizer water level signal.	
Section 7.3.1.2.7	The passive residual heat removal heat exchanger control valves are opened on low steam generator water level or on a CMT actuation signal via the protection and safety monitoring system.	
Section 7.3.1.2.9	The containment recirculation isolation valves are opened on a safeguards actuation signal in coincidence with low-3 in-containment refueling water storage tank water level via the protection and safety monitoring system.	
Section 7.3.1.2.14	The demineralized water system isolation valves close on a signal from the protection and safety monitoring system derived from either a reactor trip signal, a source range flux doubling signal, or low input voltage to the 1E dc uninterruptible power supply battery chargers.	
Section 7.3.1.2.15	The chemical and volume control system makeup line isolation valves automatically close on a signal from the protection and monitoring system derived from a source range flux doubling, high-2 pressurizer level, high-2 steam generator level signal, a safeguards signal coincident with high-1 pressurizer level, high steam generator water level coincident with P-4 permissive (reactor trip), or high-2 containment radioactivity.	
Section 7.3.2.2.1	The protection and monitoring system automatically generate an actuation signal for an engineered safety feature whenever a monitored condition reaches a preset level.	
Section 7.3.2.2.9	Manual initiation at the system-level exists for the engineered safety features actuation.	

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Table 14.3-2 (Sheet 10 of 17)
Design Basis Accident Analysis

Reference	Design Feature	Value
Section 7.4.3.1	If temporary evacuation of the main control room is required because of some abnormal main control room condition, the operators can establish and maintain safe shutdown conditions for the plant from outside the main control room through the use of controls and monitoring located at the remote shutdown workstation.	
Section 7.4.3.1.1	The remote shutdown workstation equipment is similar to the operator workstations in the main control room and is designed to the same standards. One remote shutdown workstation is provided.	
Section 7.4.3.1.3	The remote shutdown workstation achieves and maintains safe shutdown conditions from full power conditions and maintains safe shutdown conditions thereafter.	
Section 7.5.4	The protection and safety monitoring system provides signal conditioning, communications, and display functions for Category 1 variables and for Category 2 variables that are energized from the Class 1E uninterruptible power supply system.	
Section 7.6.1.1	An interlock is provided for the normally closed motor-operated normal residual heat removal system inner and outer suction isolation valves. Each valve is interlocked so that it cannot be opened unless the reactor coolant system pressure is below a preset pressure.	
Section 8.2.2	Following a turbine trip during power operation, the reverse-power relay will be blocked for a minimum time period (sec).	≥ 15
Section 8.3.2.1.2	The non-Class 1E dc and UPS system (EDS) consists of the electric power supply and distribution equipment that provides dc and uninterruptible ac power to nonsafety-related loads.	

Table 14.3-2 (Sheet 11 of 17)
Design Basis Accident Analysis

Reference	Design Feature	Value
Section 9.1.1.2.1.C	During normal fuel handling operations, a single failure-proof hoist, designed to meet the requirements of NUREG-0554, is the only hoist capable of moving new fuel above the operating floor. Per the design criteria contained in NUREG-0554, drops from a single failure-proof hoist are deemed unlikely and do not require further analysis. The consequences of such a drop are minimal since no safety-related equipment would be impacted and there are no radiological releases with new unirradiated fuel. Because the likelihood of a new fuel assembly being dropped into the new fuel pit and onto the new fuel racks is minimal, it is unnecessary to evaluate drop scenarios for the new fuel storage rack.	
Section 9.1.3.5	The spent fuel pool is designed such that a water level is maintained above the spent fuel assemblies for at least 7 days following a loss of the spent fuel cooling system using only on-site makeup water sources (See Table 9.1-4).	
Section 9.1.3.5	The spent fuel pool cooling system includes safety-related connections to establish safety-related makeup to the spent fuel pool following a design basis event including a seismic event.	
Section 9.1.4.1.1	In the event of a safe shutdown earthquake (SSE), handling equipment cannot fail in such a manner as to prevent required function of seismic Category 1 equipment.	
Section 9.3.6.3.7	The chemical and volume control system contains two redundant safety-related valves to isolate the demineralized water system from the makeup pump suction.	
Section 9.3.6.3.7	The chemical and volume control system contains two safety-related valves to isolate the makeup flow to the reactor coolant system.	
Section 9.3.6.4.5	The chemical and volume control system contains two safety-related valves to isolate the makeup flow to the reactor coolant system.	
Section 9.3.6.4.5.1	The chemical and volume control system contains two redundant safety-related valves to isolate the demineralized water system from the makeup pump suction.	

Table 14.3-2 (Sheet 12 of 17)
Design Basis Accident Analysis

Reference	Design Feature	Value
Section 9.3.6.7	The demineralized water system isolation valves close on a signal from the protection and safety monitoring system derived from either a reactor trip signal, a source range flux doubling signal, low input voltage to the 1E dc and uninterruptible power supply battery chargers, or a safety injection signal.	
Section 9.3.6.7	The chemical and volume control system makeup line isolation valves automatically close on a signal from the protection and safety monitoring system derived from a source range flux doubling, high-2 pressurizer level, high steam generator level signal, high steam generator water level coincident with P-4 permissive (reactor trip), high-2 containment radioactivity, or a safeguards signal coincident with high-1 pressurizer level.	RN-15-045
Section 10.1.2	Safety valves are provided on both main steam lines.	
Section 10.2.2.4.3	The flow of the main steam entering the high-pressure turbine is controlled by four stop valves and four governing control valves. The stop valves are closed by actuation of the emergency trip system devices.	
Section 10.3.1.1	The main steam supply system is provided with a main steam isolation valve and associated MSIV bypass valve on each main steam line from its respective steam generator.	
Section 10.3.1.1	A main steam isolation valve (MSIV) on each main steam line prevents the uncontrolled blowdown of more than one steam generator and isolates nonsafety-related portions of the system.	
Section 10.3.1.2	Power-operated atmospheric relief valves are provided to allow controlled cooldown of the steam generator and the reactor coolant system when the condenser is not available.	
Section 10.3.2.1	The main steam supply system includes: – One main steam isolation valve and one main steam isolation valve bypass valve per main steam line. – Main steam safety valves. – Power-operated atmospheric relief valves and upstream isolation valves.	
Section 10.3.2.3.2	In the event that a design basis accident occurs, which results in a large steam line break, the main steam isolation valves with associated main steam isolation bypass valves automatically close.	

Table 14.3-2 (Sheet 13 of 17) Design Basis Accident Analysis

Reference		Design Feature	Value
Figure	10.3.2-1	The steam generator system consists of two main steam, two main feedwater, and two startup feedwater lines.	
Table	10.3.2-2	Design data for main steam supply safety system valves: – Number per main steam line – Minimum relieving capacity per valve at 110% of design pressure (lb/hr)	 6 1,370,000
Table	10.3.2-2	The flow capacity of the steam generator safety valves (lbm/hr) at 110% of design pressure.	≥ 8,240,000
Table	10.3.2-2	The maximum set pressure of the steam generator safety valves (psig).	≤ 1,242
Section	10.4.8.3	The safety-related portions of the steam generator blowdown system are located in the containment and auxiliary buildings and are designed to remain functional after a safe shutdown earthquake.	
Section	10.4.7.1.1	Double valve main feedwater isolation is provided via the main feedwater control valve and main feedwater isolation valve. Both valves close automatically on main feedwater isolation signals, an appropriate engineered safety features isolation signal, within the time established with the Technical Specifications, Section 16.1. The startup feedwater control valve also serves as a containment isolation valve.	
Section	10.4.7.1.1	The condensate and feedwater system provides redundant isolation valves for the main feedwater lines routed into containment.	
Section	10.4.7.1.1	For a main feedwater or main steam line break (MSLB) inside the containment, the condensate and feedwater system is designed to limit high energy fluid to the broken loop.	
Section	10.4.7.1.2	The booster/main feedwater pumps are tripped simultaneously with the feedwater isolation signal to close the main feedwater isolation valves.	
Section	10.4.7.2.1	The main feedwater pumps and booster pumps are tripped with the feedwater isolation signal that closes the main feedwater isolation valves. The same isolation signal closes the isolation valve in the cross connect line between the main feedwater pump discharge header and the startup feedwater pump discharge header.	

Table 14.3-2 (Sheet 14 of 17)
Design Basis Accident Analysis

Reference	Design Feature	Value
Section 10.4.7.2.2	One MFIV is installed in each of the two main feedwater lines outside the containment and downstream of the feedwater control valve. The MFIVs are installed to prevent uncontrolled blowdown from the steam generators in the event of a feedwater pipe rupture. The main feedwater check valve provides backup isolation. In the event of a secondary side pipe rupture inside the containment, the MFIVs limit the quantity of high energy fluid that enters the containment through the broken loop and limit cooldown. The MFCV provides backup isolation to limit cooldown and high energy fluid addition.	
Section 10.4.7.2.2	In the event of a secondary side pipe rupture inside the containment, the main feedwater control valves provide a redundant isolation to the MFIVs to limit the quantity of high energy fluid that enters the containment through the broken loop.	
Section 10.4.7.3	For a main feedwater line break inside the containment or a main steam line break, the MFIVs and the main feedwater control valves automatically close upon receipt of a feedwater isolation signal.	
Section 10.4.7.3	For a steam generator tube rupture event, positive and redundant isolation is provided for the main feedwater (MFIV and MFCV) with isolation signals generated by the protection and safety monitoring system (PMS).	
Section 10.4.8.2.2.7	Blowdown system isolation is actuated on low steam generator water levels. The isolation of steam generator blowdown provides for a continued availability of the steam generator as a heat sink for decay heat removal in conjunction with operation of the passive residual heat removal system and the startup feedwater system.	
Section 10.4.8.3	The safety-related portions of the steam generator blowdown system located in the containment and auxiliary buildings are designed to remain functional after a safe shutdown earthquake.	
Section 10.4.9.1.1	Double valve startup feedwater isolation is provided by the startup feedwater control valve and the startup feedwater isolation valve. Both valves close on a startup feedwater isolation signal, an appropriate engineered safeguards features signal, within the time established within the Technical Specifications, Section 16.1.	

**Table 14.3-2 (Sheet 15 of 17)
Design Basis Accident Analysis**

Reference		Design Feature	Value
Section	10.4.9.1.1	For a steam generator tube rupture event, positive and redundant isolation is provided for the startup feedwater system (startup feedwater isolation valve and startup feedwater control valve), with isolation signals generated by the protection and safety monitoring system.	
Section	10.4.9.2.2	In the event of a steam generator tube rupture, the startup feedwater isolation valve and startup feedwater control valve limit overfill of the steam generator by terminating startup feed flow.	
Section	10.4.9.2.2	In the event of a secondary pipe rupture inside containment, the startup feedwater isolation valve and startup feedwater control valve provide isolation to limit the quantity of high energy fluid that enters the containment.	
Section	10.4.9.2.2	The startup feedwater isolation valve is provided to prevent the uncontrolled blowdown from more than one steam generator in the event of startup feedwater line rupture. The startup feedwater isolation valve provides backup isolation.	
Table	15.0-1	Initial core thermal power (MWt).	3400
Table	15.0-3	Nominal values of pertinent plant parameters used in accident analysis with 10% steam generator tube plugging – Reactor coolant flow (gpm)	296,000
Section	15.1.2.1	Continuous addition of excessive feedwater is prevented by the steam generator high-2 water level signal trip, which closes the feedwater isolation valves and feedwater control valves and trips the turbine, main feedwater pumps and reactor.	
Section	15.1.4.1	For an inadvertent opening of a steam generator relief or safety valve, core makeup tank actuation occurs from one of four sources: – Two out of four low pressurizer pressure signals – Two out of four low-2 pressurizer level signals – Two out of four low T_{cold} signals in any one loop – Two out of four low steam line pressure signals in any one loop	
Section	15.1.4.1	After an inadvertent opening of a steam generator relief or safety valve, redundant isolation of the main feedwater lines closes the feedwater control valves and feedwater isolation valves, and trips the main feedwater pumps.	

Table 14.3-2 (Sheet 16 of 17)
Design Basis Accident Analysis

Reference	Design Feature	Value
Section 15.1.5.1	Following a steam line rupture, core makeup tank actuation occurs from one of five sources: – Two out of four low pressurizer pressure signals – Two out of four high-2 containment pressure signals – Two out of four low steam line pressure signals in any loop – Two out of four low T_{cold} signals in any one loop – Two out of four low-2 pressurizer level signals	
Section 15.1.5.1	After a steam line rupture, redundant isolation of the main feedwater lines closes the feedwater control valves and feedwater isolation valves, and trips the main feedwater pumps.	
Section 15.1.5.2.1	Core makeup tanks and the accumulators are the portions of the passive core cooling system used in mitigating a steam line rupture.	
Section 15.1.6.1	The heat sink for the PRHR heat exchanger is provided by the IRWST, in which the PRHR heat exchanger is submerged.	
Section 15.2.6.2.1	Following a loss of ac power, the PRHR heat exchanger is actuated by the low steam generator water level (wide range).	
Section 15.2.8.2.1	Receipt of a low steam line pressure signal in at least one steam line initiates a steam line isolation signal that closes all main steam line and feed line isolation valves. This signal also gives a safeguards signal that initiates flow of cold borated water from the core makeup tanks to the reactor coolant system.	
Section 15.3.3.2.2	The pressurizer safety valves are fully open at 2575 psia. Their capacity for steam relief is described in Section 5.4.	
Section 15.4.6.2.2	A safety signal from the protection and safety monitoring system automatically isolates the potentially unborated water from the demineralized water transfer and storage system and thereby terminates the dilution.	
Section 15.5.1.1	Following inadvertent operation of the core makeup tanks during power operation, the high-3 pressurizer level signal actuates the PRHR heat exchanger and blocks the pressurizer heaters.	

Table 14.3-2 (Sheet 17 of 17) Design Basis Accident Analysis

Reference		Design Feature	Value
Section	15.5.2.1	The pressurizer heaters are blocked, and the main feedwater lines, steam lines, and chemical and volume control system are isolated.	
Table	15.6.5-10	ADS Valve Flow Areas (in^2) – ADS Stage 1 Control Valve – ADS Stage 2 Control Valve – ADS Stage 3 Control Valve – ADS Stage 4A Valve – ADS Stage 4B Valve	 ≥ 4.6 ≥ 21 ≥ 21 ≥ 67 ≥ 67
Table	15.6.5-10	ADS Valve Opening Times (sec) – ADS Stage 1 Control Valve – ADS Stage 1 Isolation Valve – ADS Stage 2 Control Valve – ADS Stage 2 Isolation Valve – ADS Stage 3 Control Valve – ADS Stage 3 Isolation Valve	 ≤ 40 ≤ 30 ≤ 100 ≤ 60 ≤ 100 ≤ 60
Section	18.8.3.2	The main control area includes the reactor operator workstations, the supervisor’s workstation, the dedicated safety panel and the wall panel information system.	
Section	18.8.3.2	The human system interface resources available at each workstation are the plant information system displays, the control displays (soft controls), the alarm system support displays, procedure system, and the screen and component selector.	

Note:
The valve closure times reflect the design basis of the AP1000. The applicable Chapter 15 accidents were evaluated for these design basis valve closure times. The results of this evaluation have concluded that there is a small impact on the Chapter 15 analysis and the conclusions remain valid.

Table 14.3-3
Anticipated Transient Without Scram

Reference	Design Feature	Value
Section 7.7.1.11	The diverse actuation system is a nonsafety-related system that provides a diverse backup to the protection and safety monitoring system.	
Section 7.7.1.11	The diverse actuation system trips the reactor control rods and the turbine on low wide range steam generator water level, or on low pressurizer water level, or on high hot leg temperature.	
Section 7.7.1.11	The diverse actuation system initiates passive residual heat removal on low wide range steam generator water level or high hot leg temperature; actuates core makeup tanks and trips the reactor coolant pumps on low pressurizer water level; and isolates selected containment penetrations and starts passive containment cooling on high containment temperature.	
Section 7.7.1.11	The manual actuation function of the diverse actuation system is implemented by wiring the controls located in the main control room directly to the final loads in a way that bypasses the normal path through the control room multiplexers, the protection and safety monitoring system cabinets, and the diverse actuation system logic.	
Section 7.7.1.11	The diverse actuation system uses microprocessor or special purpose logic processor boards different from those used in the protection and safety monitoring system.	
Section 7.7.1.11	The diverse actuation system hardware implementation is different from that of the protection and safety monitoring system.	
Section 7.7.1.11	The operating system and programming language of the diverse actuation system is different from that of the protection and safety monitoring system.	

Table 14.3-4 (Sheet 1 of 2)
Fire Protection

Reference	Design Feature	Value
Section 9A.3.1	Separation is maintained between Class 1E divisions and between Class 1E divisions and non-Class 1E cables in accordance with the fire areas.	
Section 3.4.1.1.2	The AP1000 arrangement provides physical separation of redundant safety-related components and systems from each other and from nonsafety-related components.	
Section 3.8.4.1.1	The conical roof supports the passive containment cooling system tank, which is constructed with a stainless steel liner on reinforced concrete walls.	
Section 7.1.2	The ability of the protection and safety monitoring system to initiate and accomplish protective functions is maintained despite degraded conditions caused by internal events such as fire and flooding.	
Section 7.4.3.1	If temporary evacuation of the main control room is required because of some abnormal main control room condition, the operators can establish and maintain safe shutdown conditions for the plant from outside the main control room through the use of controls and monitoring located at the remote shutdown workstation.	
Section 7.4.3.1.1	The remote shutdown workstation equipment is similar to the operator workstations in the main control room and is designed to the same standards. One remote shutdown workstation is provided.	
Section 7.4.3.1.3	The remote shutdown workstation achieves and maintains safe shutdown conditions from full power conditions and maintains safe shutdown conditions thereafter.	
Section 8.3.2.2	The four divisions of Class 1E battery chargers and Class 1E voltage regulating transformers are independent, located in separate rooms, cannot be interconnected, and their circuits are routed in dedicated, physically separated raceways.	
Section 8.3.2.3	Each safety-related circuit and raceway is given a unique identification number to distinguish between circuits and raceways of different voltage level or separation groups.	

Table 14.3-4 (Sheet 2 of 2)
Fire Protection

Reference	Design Feature	Value
Section 8.3.2.4.2	Cables of one separation group are run in separate raceway and physically separated from cables of other separation groups. Group N raceways are separated from safety-related groups A, B, C, and D. Non-class 1E circuits are electrically isolated by isolation devices, shielding and wiring techniques, physical separation, or an appropriate combination thereof.	
Section 9.5.1.2.1.1	Separation is maintained between redundant safe shutdown components, including equipment, electrical cables, and instrumentation controls, in accordance with the fire areas.	
Section 9.5.1.2.1.5	The standpipe system is supplied with water from the safety-related passive containment cooling system storage tank and normally operates independently of the rest of the fire protection system. The supply line draws water from a portion of the storage tank, using water allocated for fire protection.	
Section 9.5.1.2.1.5	The standpipe system serving areas containing equipment required for safe shutdown following a safe shutdown earthquake is designed and supported so that it can withstand the effects of a safe shutdown earthquake and remain functional.	
Section 9.5.1.2.1.5	The volume of the water in the PCS tank is sufficient to supply two hose streams, each with a flow of 75 gallons per minute, for two hours (gal).	≥ 18,000
Table 9.5.1-2	Each fire pump is rated: – Flow rate (gpm) – Total head (ft)	 ≥ 2000 ≥ 300
Section 18.8.3.2	The human system interface resources available at each workstation are the plant information system displays, the control displays (soft controls), the alarm system support displays, procedure system, and the screen and component selector.	
Section 18.8.3.4	The mission of the remote shutdown workstation is to provide the resources to bring the plant to a safe shutdown condition after an evacuation of the main control room.	
Section 18.12.3	The controls, displays, and alarms listed in Table 18.12.2-1 are retrievable from the remote shutdown workstation.	

Table 14.3-5 (Sheet 1 of 2)
Flood Protection

Reference		Design Feature	Value
Section	Appendix 1-A RG 1.143 Section C.1.1.3 Clarification	The lowest level of the auxiliary building, elevation 66′-6″, contains the components of the radwaste system within a common flood zone with watertight floors and walls. This volume of this enclosed flood zone is sufficient to contain the contents of the radwaste system.	
Table	2.0-201	Plant elevation for maximum flood level (ft).	≤ 100
Section	3.4.1.1.1	The seismic Category I structures below grade are protected against flooding by a waterproofing system.	
Section	3.4.1.1.2	The boundaries between mechanical equipment rooms and the electrical and instrumentation and control equipment rooms of the auxiliary building are designed to prevent flooding of rooms that contain safe shutdown equipment up to the maximum flood level for each room.	
Section	3.4.1.2.2	The boundaries between mechanical equipment rooms inside containment and the electrical and instrumentation and control equipment rooms of the auxiliary building are designed to prevent flooding of rooms that contain safe shutdown equipment up to the maximum flood level for each room.	
Section	3.4.1.2.2	Boundaries exist to prevent flooding between the following rooms which contain safety-related equipment: PXS valve/ accumulator room A, PXS valve/accumulator room B, and chemical and volume control room.	
Section	3.4.1.2.2	The AP1000 arrangement provides physical separation of redundant safety-related components and systems from each other and from nonsafety-related components.	
Section	3.4.1.2.2	The safety-related components available for safety shutdown are located in the auxiliary building and inside containment. No credit is taken for operation of sump pumps to mitigate the consequences of flooding.	
Section	3.4.1.2.2.1	The PXS-A compartment, PXS-B compartment and the chemical and volume control system compartment are physically separated and isolated from each other by structural walls such that flooding in any one of these compartments cannot cause flooding in any of the other compartments at elevations up to the top of these compartments.	

Table 14.3-5 (Sheet 2 of 2)
Flood Protection

Reference	Design Feature	Value
Section 3.6	In the event of a high- or moderate-energy pipe failure within the plant, adequate protection is provided so that essential structures, systems, or components are not impacted by the adverse effects of postulated pipe failure.	
Section 7.1.2	The ability of the protection and safety monitoring system to initiate and accomplish protective functions is maintained despite degraded conditions caused by internal events such as fire and flooding.	

Table 14.3-6 (Sheet 1 of 10)
Probabilistic Risk Assessment

Reference		Design Feature	Value
Table	3.2-2	The Nuclear Island structures include the containment and the shield and auxiliary buildings. These structures are seismic Category I.	
Table	3.2-3	The components identified under Reactor Systems in Table 3.2-3, as ASME Code Section III are designed and constructed in accordance with ASME Code Section III Requirements.	
Table	3.2-3	The Nuclear Island structures include the containment and the Shield and Auxiliary Buildings. These structures are seismic Category I.	
Section	3.4.1.1.2	The boundaries between mechanical equipment rooms and the electrical and instrumentation and control equipment rooms of the auxiliary building are designed to prevent flooding of rooms that contain safe shutdown equipment up to the maximum flood level for each room.	
Section	3.4.1.1.2	The AP1000 arrangement provides physical separation of redundant safety-related components and systems from each other and from nonsafety-related components.	
Section	9A.3.1	Separation is maintained between Class 1E divisions and between Class 1E divisions and non-Class 1E cables in accordance with the fire areas.	
Section	3.4.1.2.2	Boundaries exist to prevent flooding between the following rooms which contain safety-related equipment: PXS valve/accumulator room A, PXS valve/accumulator room B, and chemical and volume control room.	
Section	3.4.1.2.2	The boundaries between mechanical equipment rooms inside containment and the electrical and instrumentation and control equipment rooms of the auxiliary building are designed to prevent flooding of rooms that contain safe shutdown equipment up to the maximum flood level for each room.	
Section	3.4.1.2.2	The safety-related components available for safety shutdown are located in the auxiliary building and inside containment. No credit is taken for operation of sump pumps to mitigate the consequences of flooding.	

Table 14.3-6 (Sheet 2 of 10)
Probabilistic Risk Assessment

Reference	Design Feature	Value
Section 3.4.1.2.2.1	The PXS-A compartment, PXS-B compartment and the chemical and volume control system compartment are physically separated and isolated from each other by structural walls such that flooding in any one of these compartments or in the reactor coolant system compartment cannot cause flooding in any of the other compartments.	
Section 3.11.3	The design of the protection and safety monitoring system equipment has margin to accommodate a loss of the normal HVAC.	
Section 3D.6	RXS equipment in Appendix 3D is seismically qualified.	
Section 5.1.3.7	ADS has four stages. Each stage is arranged into two separate groups of valves and lines. – Stages 1, 2, and 3 discharge from the top of the pressurizer to the IRWST. – Each stage 4 discharges from a hot leg to the RCS loop compartment.	
Section 5.3.1.1	The reactor vessel provides a high integrity pressure boundary to contain the reactor coolant, heat generating reactor core, and fuel fission products. The reactor vessel is the primary boundary for the reactor coolant and the secondary barrier against the release of radioactive fission products.	
Section 5.4.6	ADS has four stages. Each stage is arranged into two separate groups of valves and lines. – Stages 1, 2, and 3 discharge from the top of the pressurizer to the IRWST. – Each stage 4 discharges from a hot leg to the RCS loop compartment.	
Section 5.4.6.2	Each ADS stage 1, 2, and 3 line contains two normally closed motor-operated valves (MOVs).	
Section 5.4.6.2	Each ADS stage 4 line contains a normally open MOV valve and a normally closed squib valve.	
Section 5.4.7	The RNS removes heat from the core and reactor coolant system at reduced RCS pressure and temperature conditions after shutdown.	

Table 14.3-6 (Sheet 3 of 10)
Probabilistic Risk Assessment

Reference	Design Feature	Value
Section 5.4.7	The normal residual heat removal system (RNS) provides a safety-related means of performing the following functions: – Containment isolation for the RNS lines that penetrate the containment – Long-term, post-accident makeup water to the RCS	
Section 5.4.7.1.1	The RNS containment isolation and pressure boundary valves are safety-related. The motor-operated valves are powered by Class 1E dc power.	
Section 5.4.7.1.2.1	The component cooling water system (CCS) provides cooling to the RNS heat exchanger.	
Section 6.2.4	The containment hydrogen control system provides nonsafety-related hydrogen igniters for control of the containment hydrogen concentration for beyond design basis accidents.	
Section 6.2.4.2.3	At least 64 hydrogen igniters are provided.	
Section 6.3.1.1.3	The automatic depressurization system provides a safety-related means of depressurizing the RCS.	
Section 6.3	The in-containment refueling water storage tank subsystem provides a safety-related means of performing the following functions: – Low-pressure safety injection – Core decay heat sink during design basis events – Flooding of the lower containment, the reactor cavity and the loop compartment by draining the IRWST into the containment. – Borated water	
Section 6.3.1	The core makeup tanks provide safety-related means of safety injection of borated water to the RCS.	
Section 6.3.1	Passive residual heat removal (PRHR) provides a safety-related means of removing core decay heat during design basis events.	
Section 6.3.2	The ADS valves are powered from Class 1E dc power.	

Table 14.3-6 (Sheet 4 of 10)
Probabilistic Risk Assessment

Reference	Design Feature	Value
Section 6.3.2	There are two CMTs, each with an injection line to the reactor vessel/DVI nozzle. – Each CMT has a pressure balance line from an RCS cold leg. – Each injection line is isolated with a parallel set of air-operated valves (AOVs). – These AOVs open on loss of air. – The injection line for each CMT also has two check valves in series.	
Section 6.3.2	The IRWST subsystem has the following flow paths: – Two (redundant) injection lines from the IRWST to the reactor vessel/DVI nozzle. Each line is isolated with a parallel set of valves; each set with a check valve in series with a squib valve. – Two (redundant) recirculation lines from the containment to the IRWST injection line. Each recirculation line has two paths: one path contains a squib valve and an MOV, the other path contains a squib valve and a check valve. – The two MOV/squib valve lines also provide the capability to flood the reactor cavity.	
Section 6.3.2	There are screens for each IRWST injection line and recirculation line.	
Section 6.3.2	PRHR is actuated by opening redundant, parallel air-operated valves. These air-operated valves open on loss of air.	
Section 6.3.2.2	The passive core cooling system (PXS) is composed of the following: – Accumulator subsystem – Core makeup tank (CMT) subsystem – In-containment refueling water storage tank (IRWST) subsystem – Passive residual heat removal (PRHR) subsystem. – The automatic depressurization system (ADS), which is a subsystem of the reactor coolant system (RCS), also supports passive core cooling functions.	
Section 6.3.2.2.2	There are two accumulators, each with an injection line to the reactor vessel/direct vessel injection (DVI) nozzle. Each injection line has two check valves in series.	

Table 14.3-6 (Sheet 5 of 10)
Probabilistic Risk Assessment

Reference	Design Feature	Value
Section 6.3.2.2.2	The accumulators provide a safety-related means of safety injection of borated water to the RCS.	
Section 6.3.2.2.8.7	The accumulator discharge check valves are of a different type than the CMT discharge check valves.	
Section 6.3.3	IRWST squib valves and MOVs are powered by Class 1E dc power.	
Section 6.3.3	The CMT AOVs are automatically and manually actuated from PMS and DAS.	
Section 6.3.3	The PRHR air-operated valves are automatically actuated and manually actuated from the control room by either PMS or DAS.	
Section 6.3.3	The squib valves and MOVs for injection and recirculation are automatically and manually actuated via PMS, and the squib valves are manually actuated via DAS.	
Section 6.3.3	The squib valves and MOVs for reactor cavity flooding are manually actuated via PMS from the control room. The squib valves are manually actuated via DAS from the control room.	
Section 6.3.7	The positions of the containment recirculation isolation MOVs are indicated in the control room.	
Section 6.3.7	The position of the inlet PRHR valve is indicated in the control room.	
Section 6.3.7.6.1	The ADS first-, second-, and third-stage valve positions are indicated in the control room.	
Section 7.1.1	The diverse actuation system provides a nonsafety-related means of performing the following functions: – Initiates automatic and manual reactor trip – Automatic and manual actuation of selected engineered safety features – Main control room display of selected plant parameters.	
Section 7.1.1	The protection and safety monitoring system provides a safety-related means of performing the following functions: – Automatic and manual reactor trip – Automatic and manual actuation of engineered safety features (ESF).	

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Table 14.3-6 (Sheet 6 of 10) Probabilistic Risk Assessment

Reference	Design Feature	Value
Section 7.1.1	PMS provides for the minimum inventory of fixed position controls and displays in the control room.	
Section 7.1.2	Each PMS division is powered from its respective Class 1E dc division.	
Section 7.1.2	PMS has four divisions of reactor trip and ESF actuation.	
Section 7.1.2.5	PMS has two divisions of safety-related post-accident parameter display.	
Section 7.1.2.9	PMS automatically blocks an attempt to bypass more than one channel of a function that uses 2-out-of-4 logic.	
Section 7.1.2.14	The PMS hardware and software are developed using a planned design process which provides for specific design documentation and reviews during the design requirement, system definition, development, test and installation phases.	
Section 7.1.4.2	The ability of the protection and safety monitoring system to initiate and accomplish protective functions is maintained despite degraded conditions caused by internal events such as fire and flooding.	
Section 7.1.2	The flexibility of the protection and safety monitoring system enables physical separation of redundant divisions.	
Section 7.2.2.2.1	The protection and safety monitoring system initiates a reactor trip whenever a condition monitored by the system reaches a preset level.	
Section 7.4.3 18.12.2	The PMS allows for the transfer of control capability from the main control room to the remote shutdown workstation. The minimum inventory of displays and controls in the remote shutdown workstation is provided.	
Section 7.3.1 8.3.2.1.1	The ADS valves are powered from Class 1E dc power.	
Section 7.7.1.11 7.3.1.2.4	The ADS valves are automatically and manually actuated via the protection and safety monitoring system (PMS), and manually actuated via the diverse actuation system (DAS).	
Section 7.3.1.2.3 7.7.1.11	The CMT AOVs are automatically and manually actuated from PMS and DAS.	
Section 7.3.1.2.2 7.3.1.2.9 7.7.1.11	The squib valves and MOVs for injection and recirculation are automatically and manually actuated via PMS, and the squib valves are manually actuated via DAS.	

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Table 14.3-6 (Sheet 7 of 10)
Probabilistic Risk Assessment

Reference		Design Feature	Value
Figure	7.2-1 (Sheets 16 and 20)	The squib valves and MOVs for reactor cavity flooding are manually actuated via PMS from the control room. The squib valves are manually actuated via DAS from the control room.	
Section	7.3.1.2.7 7.7.1.11	The PRHR air-operated valves are automatically actuated and manually actuated from the control room by either PMS or DAS.	
Section	7.3.1.2.20	The RNS containment isolation MOVs are actuated via PMS.	
Section	7.5.4	PMS has two divisions of safety-related post-accident parameter display.	
Section	7.6.1.1	An interlock is provided for the normally closed motor-operated normal residual heat removal system inner and outer suction isolation valves. Each valve is interlocked so that it cannot be opened unless the reactor coolant system pressure is below a preset pressure.	
Section	7.7.1.11	The diverse actuation system is a nonsafety-related system that provides a diverse backup to the protection and safety monitoring system.	
Section	7.7.1.11	The diverse actuation system trips the reactor control rods and the turbine on low wide range steam generator water level, or on low pressurizer water level, or on high hot leg temperature.	
Section	7.7.1.11	DAS manual initiation functions are implemented in a manner that bypasses the signal processing equipment of the DAS.	
Section	7.7.1.11	The DAS automatic actuation signals are generated in a functionally diverse manner from the PMS signals. Diversity between DAS and PMS is achieved by the use of different architectures, different hardware implementations, and different software, if any. Software diversity between the DAS and PMS will be achieved through the use of different algorithms, logic, program architecture, executable operating system, and executable software/logic.	

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Table 14.3-6 (Sheet 8 of 10)
Probabilistic Risk Assessment

Reference	Design Feature	Value
Section 8.3.1.1.1	On loss of power to a 6900V diesel-backed bus, the associated diesel generator automatically starts and produces ac power. The source circuit breakers and bus load circuit breakers are opened, and the generator is connected to the bus. Each generator has an automatic load sequencer to enable controlled loading on the associated buses.	
Section 8.3.1.1.2.1	Two onsite standby diesel generator units provide power to the selected nonsafety-related ac loads.	
Section 8.3.1.1.4	The main ac power system distributes non-Class 1E power from onsite sources to selected nonsafety-related loads.	
Section 8.3.2.1	The Class 1E dc and uninterruptible power supply (UPS) system (IDS) provides dc and uninterruptible ac power for the safety-related equipment.	
Section 8.3.2.1.1.1	There are four independent, Class 1E 250 Vdc divisions. Divisions A and D are each composed of one battery bank, one switchboard, and one battery charger. Divisions B and C are each composed of two battery banks, two switchboards, and two battery chargers. The first battery bank in the four divisions is designated as the 24-hour battery bank. The second battery bank in Divisions B and C is designated as the 72-hour battery bank.	
Section 8.3.2.1.1.1	Battery chargers are connected to dc switchboard buses. The input ac power for the Class 1E dc battery chargers is supplied from onsite diesel-generator-backed low-voltage ac power supplies.	
Section 8.3.2.1.1.1	The 24-hour battery banks provide power to the loads for a period of 24 hours without recharging. The 72-hour battery banks supply a dc switchboard bus load for a period of 72 hours without recharging.	
Section 8.3.2.1.2	The non-Class 1E dc and UPS system (EDS) consists of the electric power supply and distribution equipment that provides dc and uninterruptible ac power to nonsafety-related loads.	
Section 8.3.2.1.2	EDS load groups 1, 2, 3, and 4 provide 125 Vdc power to the associated inverter units that supply the ac power to the non-Class 1E uninterruptible power supply ac system.	

Table 14.3-6 (Sheet 9 of 10)
Probabilistic Risk Assessment

Reference	Design Feature	Value
Section 8.3.2.1.2	Battery chargers are connected to dc switchboard buses. The input ac power for the non-Class 1E dc battery chargers is supplied from onsite diesel-generator-backed low-voltage ac power supplies.	
Section 8.3.2.1.2	The onsite standby diesel-generator-backed low-voltage ac power supply provides the normal ac power to the battery chargers.	
Section 8.3.2.4.2	Separation is provided between Class 1E divisions, and between Class 1E divisions and non-Class 1E cables.	
Section 9.2.1	The service water system is a nonsafety-related system that transfers heat from the component cooling water heat exchangers to the atmosphere.	
Section 9.2.1.2.1	The SWS is arranged into two trains. Each train includes one pump and one cooling tower cell.	
Section 9.2.2	The component cooling water system is a nonsafety-related system that removes heat from various components and transfers the heat to the service water system (SWS).	
Section 9.2.2.2	The CCS is arranged into two trains. Each train includes one pump and one heat exchanger.	
Section 9.3.6	The CVS provides a nonsafety-related means to perform the following functions: – Makeup water to the RCS during normal plant operation – Boration following a failure of reactor trip – Coolant to the pressurizer auxiliary spray line.	
Section 9.3.6.1.1	The chemical and volume control system (CVS) provides a safety-related means to terminate inadvertent RCS boron dilution.	
Section 9.4.1	The main control room has its own ventilation system and is pressurized. The ventilation system for the remote shutdown room is independent of the ventilation system for the main control room.	
Section 9.5.1.2.1.1	The PMS allows for the transfer of control capability from the main control room to the remote shutdown workstation. The minimum inventory of displays and controls at the remote shutdown workstation is provided.	

Table 14.3-6 (Sheet 10 of 10)
Probabilistic Risk Assessment

Reference	Design Feature	Value
Section 9.5.1.2.1.1	Class 1E divisional cables are routed in their respective divisional raceways.	
Section 9.5.1.2.1.1	Separation is maintained between Class 1E divisions and between Class 1E divisions and non-Class 1E cables in accordance with the fire areas.	
Section 17.4.1	D-RAP provides reasonable assurance that the design of risk-significant SSCs is consistent with their PRA assumptions.	
Section 18.8.3.2	The main control area includes the reactor operator workstations, the supervisor's workstation, the dedicated safety panel and the wall panel information system.	
Section 18.12.2	The minimum inventory of instrumentation includes those displays, controls, and alarms that are used to monitor the status of the critical safety functions and to manually actuate the safety-related systems that achieve the critical safety functions. The minimum inventory resulting from the implementation of the selection criteria is provided in Table 18.12.2-1.	

Table 14.3-7 (Sheet 1 of 3)
Radiological Analysis

Reference		Design Feature	Value
Table	2.0-201	Plant elevation for maximum flood level (ft)	≤ 100
Section	2.3.4	Atmospheric dispersion factors - X/Q (sec/m^3) – Site Boundary X/Q 0 - 2 hour time interval – Low Population Zone Boundary X/Q 0 - 8 hours 8 - 24 hours 24 - 96 hours 96 - 720 hours	 ≤ 5.1 x 10^{-4} ≤ 2.2 x 10^{-4} ≤ 1.6 x 10^{-4} ≤ 1.0 x 10^{-4} ≤ 8.0 x 10^{-5}
Table	6.2.3-1	Containment penetration isolation features are configured as in Table 6.2.3-1	
Table	6.2.3-1	Maximum closure time for remotely operated containment purge valves (seconds)	≤ 10
Table	6.2.3-1	Maximum closure time for all other remotely operated containment isolation valves (seconds)	≤ 60
Section	6.4.2.3	The minimum storage capacity of all storage tanks in the VES (scf)	≥ 327,574
Section	6.4.3.2	The maximum temperature rise in the main control room pressure boundary following a loss on the nuclear island nonradioactive ventilation system over a 72-hour period (°F)	+ 10.8
Section	6.4.4	The maximum temperature in the instrumentation and control rooms and dc equipment rooms following a loss of the nuclear island nonradioactive ventilation system remains over a 72-hour period (°F).	≤ 120
Section	6.4.4	The main control emergency habitability system nominally provides 65 scfm of ventilation air to the main control room from the compressed air storage tanks.	65 ± 5
Section	6.4.4	Sixty-five ± five scfm of ventilation flow is sufficient to pressurize the control room to 1/8th inch water gauge differential pressure (WIC).	1/8th
Figure	6.4-2	The main control room emergency habitability system consists of four sets of emergency air storage tanks and an air delivery system to the main control room.	
Section	6.5.3	The passive heat removal process and the limited leakage from the containment result in offsite doses less than the regulatory guideline limits.	

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Table 14.3-7 (Sheet 2 of 3)
Radiological Analysis

Reference	Design Feature	Value
Section 8.3.1.1.6	Electrical penetrations through the containment can withstand the maximum short-circuit currents available either continuously without exceeding their thermal limit, or at least longer than the field cables of the circuits so that the fault or overload currents are interrupted by the protective devices prior to a potential failure of a penetration.	
Section 9.4.1.1.1	The VBS isolates the HVAC ductwork that penetrates the main control room boundary on high particulate or iodine concentrations in the main control room supply air or on extended loss of ac power to support operation of the main control room emergency habitability system.	
Section 12.3.2.2.1	During reactor operation, the shield building protects personnel occupying adjacent plant structures and yard areas from radiation originating in the reactor vessel and primary loop components. The concrete shield building wall and the reactor vessel and steam generator compartment shield walls reduce radiation levels outside the shield building to less than 0.25 mrem/hr from sources inside containment. The shield building completely surrounds the reactor components.	
Section 12.3.2.2.2	The reactor vessel is shielded by the concrete primary shield and by the concrete secondary shield which also surrounds other primary loop components. The secondary shield is a structural module filled with concrete surrounding the reactor coolant system equipment, including piping, pumps and steam generators. Extensive shielding is provided for areas surrounding the refueling cavity and the fuel transfer canal to limit the radiation levels.	
Section 12.3.2.2.3	Shielding is provided for the liquid radwaste, gaseous radwaste and spent resin handling systems consistent with the maximum postulated activity. Corridors are generally shielded to allow Zone II access, and operator areas for valve modules are generally Zone II or III for access. Shielding is provided to attenuate radiation from normal residual heat removal equipment during shutdown cooling operations to levels consistent with radiation zoning requirements of adjacent areas.	

Table 14.3-7 (Sheet 3 of 3)
Radiological Analysis

Reference	Design Feature	Value
Section 12.3.2.2.4	The concrete shield walls surrounding the spent fuel cask loading and decontamination areas, and the shield walls surrounding the fuel transfer and storage are sufficiently thick to limit radiation levels outside the shield walls in accessible areas to Zone II. The building walls are sufficient to shield external plant areas which are not controlled to Zone I.	
Section 12.3.2.2.5	Shielding is provided as necessary for the waste storage areas in the radwaste building to meet the radiation zone and access requirements.	
Section 12.3.2.2.7	Shielding combined with other engineered safety features is provided to permit access and occupancy of the control room following a postulated loss-of-coolant accident, so that radiation doses are limited to five rem whole body from contributing modes of exposure for the duration of the accident, in accordance with General Design Criteria 19.	
Section 12.3.2.2.9	The spent fuel transfer tube is shielded to within adjacent area radiation limits, is completely enclosed in concrete, and there is no unshielded portion of the spent fuel transfer tube during the refueling operation.	

Table 14.3-8
Severe Accident Analysis

Reference	Design Feature	Value
Section 1.2	The discharge from the IRWST vents located in the roof of the IRWST next to the containment vessel are oriented away from the containment vessel.	
Section 5.3.1.2	There are no penetrations in the reactor vessel below the core.	
Section 5.3.5	The reflective (or suitable equivalent) reactor vessel insulation provides an engineered flow path to allow the ingression of water and venting of steam for externally cooling the vessel. – A flow path exists from the loop compartment to the reactor vessel cavity (ft^2). – A flow path area to vent steam exists between the vessel insulation and the reactor vessel (ft^2).	 ≥ 6 ≥ 12
Section 6.2.4.2.3	The hydrogen ignition subsystem consists of 64 hydrogen igniters strategically distributed throughout the containment.	
Table 6.2.4-3	The minimum surface temperature of the hydrogen igniters (°F).	≥ 1,600
Section 6.3	The ADS provides a safety-related means of depressurizing the RCS.	
Section 6.3	The PXS provides a safety-related means of flooding the reactor cavity by draining the IRWST into the containment.	
Section 7.3.1.2.9	Signals to align the IRWST containment recirculation isolation valves are generated by manual initiation.	
Section 7.7.1.11	Initiation of containment recirculation is a diverse manual function.	

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14.4 Combined License Applicant Responsibilities

14.4.1 Organization and Staffing

The staff, staff responsibilities, authorities, and personnel qualifications for performing the AP1000 initial test program are addressed in Section 14.2. This test organization is responsible for the planning, executing, and documenting of the plant initial testing and related activities that occur between the completion of plant/system/component construction and commencement of plant commercial operation. Transfer and retention of experience and knowledge gained during initial testing for the subsequent commercial operation of the plant is an objective of the test program.

14.4.2 Test Specifications and Procedures

Preoperational and startup test specifications and procedures are available to the NRC in accordance with the requirements of subsection 14.2.3. The controls for development of test specifications and procedures are also described in subsection 14.2.3. RN-14-110

A cross reference list is provided between ITAACs and test procedures and/or sections of test procedures.

14.4.3 Conduct of Initial Test Program RN-15-099

A site-specific startup administrative manual (procedures), which contains the administrative procedures and requirements that govern the activities associated with the plant initial test program, as described in Section 14.2 is provided.

14.4.4 Review and Evaluation of Test Results

Review and evaluation of individual test results, as well as final review of overall test results and selected milestones or hold points are addressed in subsection 14.2.3.2. Test exceptions or results that do not meet acceptance criteria are identified to the affected and responsible design organizations, and corrective actions and retests, as required, are performed.

14.4.5 Interface Requirements

The Test Specifications and acceptance criteria required of structures and systems which are outside the scope of the design certification are addressed in Subsections 14.2.9.4.15, 14.2.9.4.22 through 14.2.9.4.27, 14.2.10.4.29, and in the Physical Security Plan.

14.4.6 First-Plant-Only and Three-Plant-Only Tests

[*The COL holder for the first plant and the first three plants will perform the tests listed in subsection 14.2.5. For subsequent plants, either tests listed in subsection 14.2.5 shall be performed, or the COL applicant shall provide a justification that the results of the first-plant-only tests or first-three-plant tests are applicable to the subsequent plant.*]*

First-plant-only and first-three-plant-only tests either are performed in accordance with subsection 14.2.5 or a justification is provided that the results of the first-plant-only and first-three-plant-only tests are applicable to a subsequent plant. If the tests are not performed, the justification is provided prior to preoperational testing.

*NRC Staff approval is required prior to implementing a change in this information.

Appendix 14A Design Acceptance Criteria/ITAAC Closure Process

Design Acceptance Criteria (DAC) are a set of prescribed limits, parameters, procedures, and attributes upon which the NRC relies, in a limited number of technical areas, in making a final safety determination to support a design certification. DAC is to be objective (measurable, testable, or subject to analysis using pre-approved methods), and must be verified as a part of the ITAAC performed to demonstrate that the as-built facility conforms to the certified design (SECY 92-053).

There are three process options for DAC/ITAAC resolution:

- Resolve through amendment to design certification
- Resolve as part of COL review
- Resolve after COL is issued

In the first two options, the applicant will submit the design information and the NRC will document its review in a safety evaluation. In the third option, the COL holder notifies the NRC of availability of design information and the staff will document its review in an inspection report.

Should the third option be implemented for the first standard AP1000 plant, subsequent COL applicants may reference the first standard plant closure documentation and close the DAC/ITAAC under the concept of “one issue, one review, one position,” identified in NRC guidance.

Additionally, Westinghouse may submit licensing topical reports for NRC review of the material supporting the DAC/ITAAC closure and request that the NRC issue a safety evaluation in conjunction with a closure letter or inspection report concluding that the acceptance criteria of the DAC/ITAAC have been met. Subsequent COL applicants may reference these reports and NRC closure documents in an effort to close the DAC/ITAAC.

For technical areas where DAC/ITAAC applies in the design certification rule, COL applicants will provide an ITAAC and associated closure schedule indicating the approach to be applied. For subsequent COL applicants following the first standard AP1000 plant, the indication could be to reference the existing DAC/ITAAC closure documentation for the first standard plant.

NRC guidance for DAC/ITAAC is provided in Regulatory Guide 1.206, Section C.III.5. Further information on the staff’s position of DAC/ITAAC being used as part of the 10 CFR Part 52 review process is provided in SECY-92-053.