Response to

Request for Additional Information No. 68 (987, 1008), Revision 0

9/24/2008

U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 14.02 - Initial Plant Test Program - Design Certification and New License Applicants
Application Section: 14.2
CQVP Branch

Question 14.02-19:

In RAI 14.02-3, the staff requested that AREVA describe the administrative controls with respect to the transition from initial criticality to low power physics testing in FSAR Section 14.2.1.1. AREVA's response to RAI 14.02-03 stated that the transition from initial criticality to low power physics testing is completed after verifying that the technical specification SR 3.1.2.1 requirement of 1000 pcm is met; however, this statement is not specifically repeated in the FSAR as being the transition from initial criticality to low power physics testing. The NRC staff requests that AREVA add a statement to clarify that verification of technical specification SR 3.1.2.1 is the transition point from initial criticality to low power physics testing.

Also, in its recent revision to Section 14.2.1.1.4 of the U.S. EPR FSAR, AREVA cites that the initial criticality and low power physics testing follow the guidelines described in ANSI 19.6.1. The staff notes that the NRC staff has not endorsed the use of ANSI 19.6.1 for initial test programs. Also, the scope of the ANSI standard is limited and does not cover all of the needed areas for the administrative controls of the initial criticality and low power physics testing. Therefore, the NRC staff requests that AREVA remove the reference to ANSI 19.6.1 in Section 14.2.1.1.4 of the U.S. EPR FSAR or provide justification for its inclusion in a manner consistent with RG 1.68.

Response to Question 14.02-19:

Additional information has been added to U.S. EPR FSAR Tier 2, Section 14.2.1.1.4, Step 6 to clarify that Technical Specification surveillance requirement 3.1.2.1 is the transition from initial criticality to low power physics testing.

The reference to ANSI 19.6.1, “Reload Startup Physics Test for Pressurized Water Reactors,” has been removed from U.S. EPR FSAR Tier 2, Section 14.2.1.1.4, and additional clarification will be added to U.S. EPR FSAR Tier 2, Section 14.2.1.1.4, Step 7. References to ANSI 19.6.1 have also been removed from U.S. EPR FSAR Tier 2, Section 14.2.12 (Tests #190, #191, #192, #218), and U.S. EPR FSAR Tier 2, Section 14.2.13.

FSAR Impact:

U.S. EPR FSAR Tier 2, Sections 14.2.1.1.4, 14.2.12, and 14.2.13 will be revised as described in the response and indicated on the enclosed markup.

Question 14.02-20:

In RAI 14.02-04, the staff requested that AREVA revise Section 14.2.1.1.1, "Construction Activities," of the FSAR to include initial instrument calibration and functional test of components as recommended by Appendix A.1 of RG 1.68. The AREVA response to RAI 14.02-04 revised Section 14.2.1.1.2, "Phase I - Preoperational Testing," to include instrument calibrations and functional tests as objectives of preoperational testing. The revision of the FSAR is inconsistent with RG 1.68 guidance that preoperational tests should not proceed until the designated construction tests and inspections, which consist of initial instrument calibration and functional test of components, have been satisfactorily completed. The NRC staff requests that AREVA confirm that initial instrument calibration and functional test of components are completed in construction activities and revise Section 14.2.1.1.1 of the AREVA U.S. EPR FSAR accordingly.

Response to Question 14.02-20:

U.S. EPR FSAR Tier 2, Section 14.2.1.1.1 lists activities required to confirm that construction is satisfactorily complete and systems are ready for preoperational testing. Functional testing has been specifically mentioned in the text of U.S. EPR FSAR Tier 2, Section 14.2.1.1.1, and instrument calibrations have been added to the list of construction activities in U.S. EPR FSAR Tier 2, Section 14.2.1.1.1.

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 14.2.1.1.1 will be revised as described in the response and indicated on the enclosed markup.

Question 14.02-21:

In RAI 14.02-07, the NRC staff requested that AREVA provide additional information addressing the objectives of initial criticality and low-power physics testing in accordance with the guidance of Appendix A.3 and Appendix A.4 of RG 1.68. The AREVA response revised the U.S. EPR FSAR Section 14.2.1.1.4, "Phase III - Initial Criticality and Low Power Physics Testing," to provide the objectives stated in Appendix A.3 of RG 1.68, however, the objectives in Appendix A.4 were not included. Specifically, the objective A.4(a), "confirm the design and, to the extent practical, validate the analytical models and verify the correctness or conservatism of assumptions used in the safety analyses for the facility," and (b) "confirm the operability of plant systems and design features that could not be completely tested during the preoperational test phase because of the lack of an adequate heat source for the reactor coolant and main steam systems," are not discussed in the objectives of 14.2.1.1.4 of the U.S. EPR. The NRC staff requests that AREVA revise section 14.2.1.1.4 to include the objectives listed in Appendix A.4 of the RG.

Response to Question 14.02-21:

Additional information has been added to U.S. EPR FSAR Tier 2, Section 14.2.1.1.4 to address the objectives of Regulatory Guide 1.68, Appendix A.4, objectives (a) and (b). Also, the control rod or poison removal information in U.S. EPR FSAR Tier 2, Section 14.2.1.1.4 has been split into separate bullets for clarification.

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 14.2.1.1.4 will be revised as described in the response and indicated on the enclosed markup.

Question 14.02-22:

In RAI 14.02-08, the NRC staff requested that AREVA provide additional information addressing the objectives of power ascension testing in accordance with Appendix A.5 of RG 1.68. AREVA's response revised Section 14.2.1.1.5, "Phase IV - Power Ascension Testing," to include objectives as stated in Appendix A.5 of RG 1.68, however, the revision does not include all of the objectives. Specifically, Section 14.2.1.1.5 does not address the following objectives: (1) tests and acceptance criteria should be prescribed to demonstrate the ability of major or principal plant control systems to automatically control process variables within design limits. Such tests are expected to provide assurance that the facility's integrated dynamic response is in accordance with design for plant events such as reactor scram, turbine trip, reactor coolant pump trip, and loss of feedwater heaters or pumps; (2) testing should be sufficiently comprehensive to establish that the facility can operate in all operating modes for which it has been designed; however, tests should not be conducted, or operating modes or plant configurations established, if they have not been analyzed or if they fall outside the range of assumptions used in analyzing postulated accidents in the facility's FSAR; and, (3)appropriate consideration should be given to testing at the extremes of possible operating modes for facility systems. Testing under simulated conditions of maximum and minimum equipment availability within systems should be accomplished if the facility is intended to be operated in these modes.

The NRC staff requests that AREVA revise Section 14.2.1.1.5 to address the objectives of power ascension testing in accordance with Appendix A.5 of RG 1.68.

Response to Question 14.02-22:

Additional details have been included in U.S. EPR FSAR Tier 2, Section 14.2.1.1.5 to address the objectives of Regulatory Guide 1.68, Appendix A.5.

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 14.2.1.1.5 will be revised as described in the response and indicated on the enclosed markup.

Question 14.02-23:

In RAI 14.02-14, the NRC staff requested that AREVA include additional information in Section 14.2.10.2, "Initial Criticality," of the U.S. EPR FSAR with respect to the initial criticality criteria in Appendix C.3 of RG 1.68. AREVA's response provided information on the conditions for the safe controlled approach to initial criticality, including: a minimum count rate of two counts per second is met, and a statistical reliability test on each operable source range instrument is performed. RG 1.68 Appendix C.3 states that a neutron count rate of at least 1/2 count per second should be used. The NRC staff requests additional information to justify the use of two counts per second versus the 1/2 counts per second recommended by the regulatory guide. In addition, the NRC staff requests clarification on the process used to perform a statistical reliability test on each operable source range instrument.

Response to Question 14.02-23:

A response to this question will be provided by December 5, 2008.

Question 14.02-24:

In RAI 14.02-12, the NRC staff requested that AREVA revise Section 14.2.10.1, "Initial Fuel Loading," of the FSAR to include the prerequisites for fuel loading from RG 1.68, Appendix C.2.a. AREVA's response revised the Section 14.2.10.1 to include the prerequisites listed in the RAI; however, there are additional prerequisites listed in the RG in Appendix A.2 that are not included in the FSAR revision. Specifically, the objectives are: (1) establish requirements for periodic data-taking; (2) predictions of core reactivity should be prepared in advance to aid in evaluating the measured responses to specified loading increments; (3) establish requirements for the operability of plant systems and components, including reactivity control systems and other systems and components necessary to ensure the safety of plant personnel and the public in the event of errors or malfunctions; and, (4) scram time tests should be sufficient to provide reasonable assurance that the control rods will scram within the required time under plant conditions that bound those under which the control rods might be required to function to achieve plant shutdown (testing should demonstrate control rod scram times at both hot zero power and cold temperature conditions, and with flow and no-flow conditions). The NRC requests that AREVA include all of the prerequisites listed in RG 1.68 in Section 14.2.10.1 of the U.S. EPR FSAR, or justify their exclusion.

In addition, AREVA removed the statement that all ITAAC have been closed from the minimum initial conditions for core load in Section 14.2.10.1. This statement was consistent with the guidance in SRP 1.4.II.4.A.i. The NRC staff requests that AREVA reinstate this statement in Section 14.2.10.1 of the EPR FSAR, or justify its removal.

Response to Question 14.02-24:

Additional details have been included in U.S. EPR FSAR Tier 2, Section 14.2.10.1 to address the objectives of Regulatory Guide 1.68, Appendix A.2.

Inspection, test, analysis, and acceptance criteria (ITAAC) closure is addressed in U.S. EPR FSAR Tier 1 and U.S. EPR FSAR Tier 2, Section 14.3. For example, U.S. EPR FSAR Tier 1, Section 1.2.2 describes ITAAC implementation and explains that ITAAC will be closed prior to fuel load. The statement “All ITAAC have been closed” has been removed from U.S. EPR FSAR Tier 2, Section 14.2.10.1 because it is not specific to the initial testing program.

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 14.2.10.1 will be revised as described in the response and indicated on the enclosed markup.

Question 14.02-25:

Regulatory Guide 1.68, Appendix A.3, “Initial Criticality,” states that “Licensees should conduct the initial approach to criticality in a deliberate and orderly manner using the same rod withdrawal sequences and patterns that will be used during subsequent startups.” Section 14.2.10.2, "Initial Criticality," of the U.S. EPR FSAR does not discuss the use of the same rod withdrawal sequences and patterns in subsequent plant startups. Please revise the U.S. EPR FSAR, Section 14.2.10.2 to address the use of the same rod withdrawal sequences and patterns in subsequent plant startups.

Response to Question 14.02-25:

Additional details have been included in U.S. EPR FSAR Tier 2, Section 14.2.10.2 to address the guidance in Regulatory Guide 1.68, Appendix A.3 on the initial approach to criticality in a deliberate and orderly manner using the same rod withdrawal sequences and patterns that will be used during subsequent startups.

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 14.2.10.2 will be revised as described in the response and indicated on the enclosed markup.

Question 14.02-26:

Regulatory Guide, 1.68.3, "Preoperational Testing of Instrument and Control Air Systems," provides guidance for preoperational testing of I&C air systems. The RG states that as part of the initial preoperational testing program the system and loads should be tested as described in the RG to verify that all components function properly at normal pressures and following possible pressure Increases and that the systems respond as designed to a loss-of-pressure event. Test abstract 178, "Pre-Core Loss of Instrument Air," does not provide for the testing of all of the aspects of the guidance given in the RG. One example is that the test does not include testing of compressors, aftercoolers, oil separator units, air receivers, and pressure-reducing stations to verify proper operation according to system design. The NRC staff requests that AREVA revise test abstract 178 to ensure that all pertinent aspects of RG 1.68.3 are addressed in the test.

Response to Question 14.02-26:

Additional details have been included in U.S. EPR FSAR Tier 2, Section 14.2.12, Test #178 to address Regulatory Guide 1.68.3 guidance on preoperational testing of instrumentation and controls (I&C) air systems.

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 14.2.12, Test #178 will be revised as described in the response and indicated on the enclosed markup.

Question 14.02-27:

Regulatory Guide 1.68, Regulatory Position C.4 and Appendix C, Item 1.f, "Acceptance Criteria," note that the test acceptance criteria should account for measurement errors and uncertainties used in the transient and accident analyses. The U.S. EPR FSAR, Section 14.2.3 does not specifically address this provision. The NRC staff requests that AREVA revise Section 14.2.3 to clarify how test acceptance criteria will account for measurement errors and uncertainties used in the transient and accident analyses.

Response to Question 14.02-27:

Additional details have been included in U.S. EPR FSAR Tier 2, Section 14.2.3 to address Regulatory Guide 1.68, Regulatory Position C.4, and Appendix C, Item 1.f.

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 14.2.3 will be revised as described in the response and indicated on the enclosed markup.

Question 14.02-28:

SRP 14.2.II.5 states that 10 CFR 52.47(b)(1) referring to ITAAC is applicable to design certifications. FSAR Section 14.1, "Specific Information to be addressed for the Initial Plant Test Program," states the regulations that the Initial Test Program addresses; however, the section does not include 10 CFR 52.47(b)(1). The NRC staff requests that AREVA add 10 CFR 52.47(b)(1) to the U.S. EPR FSAR, Section 14.1 since it is applicable to design certification applications.

Response to Question 14.02-28:

The regulations listed in U.S. EPR FSAR Tier 2, Section 14.1 are those specifically addressed in U.S. EPR FSAR Tier 2, Section 14.2. Although 10 CFR 52.47(b)(1) for inspection, test, analysis, and acceptance criteria (ITAAC) is applicable to U.S. EPR FSAR Tier 2, Section 14.2, the ITAAC are specifically addressed in U.S. EPR FSAR Tier 1. The relationship between U.S. EPR FSAR Tier 1, U.S. EPR FSAR Tier 2 (specifically Section 14.3), and 10 CFR 52.47(b)(1) is described in U.S. EPR FSAR Tier 2, Section 14.3.

10 CFR 52.47(b)(1) is addressed in U.S. EPR FSAR Tier 1 and U.S. EPR FSAR Tier 2, Section 14.3, and therefore is not listed in U.S. EPR FSAR Tier 2, Section 14.1.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 14.02-29:

Regulatory Guide 1.68, Section C.3, Scope, Conditions, and Length of Testing, states, “The testing of SSCs should include, to the extent practical, simulation of the effects of control system and equipment failures or malfunctions that could reasonably be expected to occur during the plant’s lifetime.” The U.S. EPR FSAR, Section 14.2.1.1, "Summary of the Startup Test Program," states that the test program will demonstrate that SSC operate and comply with design requirements, however, it does not include the statement that simulation of the effects of control system and equipment failures or malfunctions that could reasonably be expected to occur during the plant’s lifetime will be part of the testing.

The NRC staff requests that AREVA expand the scope of Section 14.2.1.1 of the U.S. EPR FSAR to include a statement to the effect that testing of SSCs will include, to the extent practical, simulation of the effects of control system and equipment failures or malfunctions that could reasonably be expected to occur during the plant’s lifetime.

Response to Question 14.02-29:

Additional details have been included in U.S. EPR FSAR Tier 2, Section 14.2.1.1 to address Regulatory Guide 1.68, Section C.3. The added text states that the testing of structures, systems, and components (SSC) should include, to the extent practical, simulation of the effects of control system and equipment failures or malfunctions that could reasonably be expected to occur during the plant’s lifetime.

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 14.2.1.1 will be revised as described in the response and indicated on the enclosed markup.

Question 14.02-30:

The title of Regulatory Guide 1.9, Revision 4, listed in Section 14.2.7 of the U.S. EPR FSAR is stated as, “Selection, Design, and Qualification of Diesel-Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants,” however, the correct title is “Application and Testing of Safety-Related Diesel Generators in Nuclear Power Plants.” Therefore, the NRC staff requests that AREVA revise Section 14.2.7 accordingly.

Response to Question 14.02-30:

U.S. EPR FSAR Tier 2, Section 14.2.7 will be revised to reflect the correct title of Regulatory Guide 1.9, Revision 4, “Application and Testing of Safety-Related Diesel Generators in Nuclear Power Plants.”

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 14.2.7 will be revised as described in the response and indicated on the enclosed markup.

U.S. EPR Final Safety Analysis Report Markups

14.2 Initial Plant Test Program

14.2.1 Summary of Test Program and Objectives

14.2.1.1 Summary of the Startup Test Program

The startup test program includes testing activities that commence with the completion of construction and installation and end with the completion of the power ascension testing. Testing is performed on SSC that:

- Are used 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.
- Are used for the safe shutdown and cooldown of the reactor under infrequent or moderately frequent transient events, postulated accident conditions, and for maintaining the reactor in a safe condition for an extended shutdown period following such conditions.
- Are used for establishing conformance with safety limits or limiting conditions for operation that shall be included in the Technical Specifications.
- Are classified as engineered safety features (ESF) or used to support or establish that the operations of ESF are within design limits.
- Are assumed to function or that are credited in the accident analysis as described throughout this FSAR.
- Are used to process, store, control, measure, or limit the release of radioactive materials.
- Are used in the special low-power testing program that is conducted at power levels no greater than five percent to provide meaningful technical information in addition to that obtained in the normal startup test program required for the resolution of Three Mile Island (TMI) action plan item 1.G.1.
- Are identified as a significant risk in the facility based on a specific probabilistic risk assessment.
- Are used to mitigate severe accidents that are beyond the U.S. EPR design basis.

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This test program demonstrates that SSC operate and comply with design requirements and meet the requirements of 10 CFR 50, Appendix B, Criterion XI. The startup test program results confirm that performance levels meet the functional safety requirements and verify the adequacy of SSC design and the functionality of systems over their operating ranges. The testing of SSC should include, to the extent practical, simulation of the effects of control system and equipment failures or malfunctions that could reasonably be expected to occur during the plant's lifetime. It also helps to establish baseline performance data and serves to verify that normal operating and emergency procedures achieve their intended purposes. The data collected during the performance of testing shall be categorized as acceptance criteria or baseline data. Acceptance criteria data has clearly defined criteria (minimum or maximum allowable values) that are used to determine if the system or component is capable of meeting the design basis assumptions in the accident analyses. Baseline data is not used to determine if a component or system can meet a design basis assumption. Baseline data is collected for trending purposes and does not have established acceptance criteria, it shall be clearly denoted that the data is recorded for baseline purposes.

The startup test program begins at the end of construction activities and consists of the following phases:

- Phase I - preoperational testing program.
- Phase II - initial fuel loading and precritical testing.
- Phase III - initial criticality and low power physics testing.
- Phase IV - power ascension testing.

14.2.1.1.1 Construction Activities

Construction activities consist of tests and inspections required to confirm that construction is complete and that systems are ready for preoperational testing.

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Construction activities that verify ~~that~~ the construction quality associated with SSC are satisfactorily completed prior to turning control and responsibility over to the startup organization. Construction activities consist of ~~preliminary~~functional tests and inspections which include, but are not limited to:

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- Weld inspections and other types of material examinations.
- Hanger and pipe support inspections.
- Flushing and hydro lazing, excluding flushes that require operation of permanent plant equipment.

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- Cleaning interior and exterior surfaces of piping and other components.

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- Circuit integrity and separation checks, excluding tests that require permanent plant circuits to be energized.
- Hydrostatic pressure tests.

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- Instrument calibrations, excluding portions of calibration procedures that require permanent plant circuits to be energized or plant computer conversions from field

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units to engineering units. Construction personnel can use temporary power sources to verify that instruments respond to calibrated input sources.

Specific construction test requirements shall be established in accordance with the site administrative procedures.

14.2.1.1.2 Phase I - Preoperational Testing

Upon the completion of construction and installation testing, preoperational tests are performed to demonstrate that SSC operate in accordance with design bases. Simulated signals or inputs are often used to demonstrate the full range of the system operation when it would be undesirable to create real system conditions. The general objectives of the preoperational test phase are:

- To demonstrate that appropriate acceptance criteria are met for SSC for safety-related SSC, including alarms and indications.
- To provide documentation of the performance and safety of equipment and systems in all operating modes, including degraded modes (e.g., stuck open miniflow valves, open cross connects) for which the systems are designed to remain operational.
- To demonstrate equipment performance throughout the full design operating range.
- To test, as appropriate, manual operation, operation of systems and their components, automatic operation, operation in alternate or secondary modes of control, and operation and verification tests to demonstrate expected operation following a loss of power sources.
- To test the proper functioning of instrumentation and controls, permissive and prohibit interlocks, and equipment protective devices, for which malfunction or premature actuation may shut down or defeat the operation of systems or equipment.
- To provide baseline test and operating data of equipment and subsystems for future reference.
- To operate ~~new~~ equipment for a sufficient period to demonstrate performance so that ~~any~~ design, manufacturing, or installation defects can be detected and corrected.
- To provide the permanent plant operating staff with the maximum opportunity to obtain practical experience in the operation and maintenance of equipment and systems and their associated procedures. Maintenance activities should include, but not be limited to, instrument calibrations, powered valve functional tests, and lubrication programs.

- Systems required for startup or protection of the plant, including the reactor protection system and emergency shutdown system, are operable and in a state of readiness.

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- The control rod or poison removal sequence is accomplished using detailed procedures approved by personnel or groups designated by the COL holder.
- The reactor achieves initial criticality by boron dilution and control rods are withdrawn before dilution begins.
- The control rod insertion limits defined in the Technical Specifications are observed and complied with.
- The reactivity addition sequence is prescribed, and the procedure will require a cautious approach in achieving criticality to prevent passing through criticality in a period shorter than approximately 30 seconds (<1 decade per minute).

A description of the procedures followed during the approach to initial criticality is included in Section 14.2.10.2. Following initial criticality, a series of low-power physics tests are performed to verify selected core design parameters. These tests serve to substantiate the following: ~~safety analysis assumptions and Technical Specifications. They also demonstrate that core characteristics are within the expected limits and provide data for benchmarking the design methodology used for predicting core characteristics later in life.~~

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- Confirm the design and, to the extent practical, validate the analytical models.
- Verify the correctness or conservatism of assumptions used in the safety analyses and Technical Specifications.
- Confirm the operability of plant systems and design features that could not be completely tested during the preoperational test phase because of the lack of an adequate heat source for the reactor coolant and main steam systems.
- Demonstrate that core characteristics are within the expected limits.
- Provide data for benchmarking the design methodology used for predicting core characteristics later in life.

14.02-19

The initial criticality and low-power physics tests (LPPT) ~~follow the guidelines described in ANSI 19.6.1 and~~ as a minimum consist of the following:

1. Withdrawal of the shutdown bank RCCAs.
2. Withdrawal of the control bank RCCAs in sequence and overlap until the final control bank is inserted approximately 50 to 100 pcm.

3. Reduction of the reactor coolant boron concentration (dilution) in a gradual manner until the reactor is just critical or with source range counts increasing gradually.

4. Increasing source range counts slowly to the point of adding heat (POAH) and then reducing the intermediate range indication by one-half to one decade.

5. Determination of adequate overlap of source and intermediate-range neutron instrumentation, and verification that proper operation of associated protective functions and alarms provide plant protection in the low-power range.

6. Verification that the Technical Specification SR 3.1.2.1 requirement of 1000 pcm is met. At this point the Test Coordinator should verify that initial criticality activities have been completed and transition to those activities supporting low power physics testing.

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7. Establish the LPPT band and reduce flux until the reactor is approximately at the lower end of the flux band if the isothermal temperature coefficient is expected to be positive and at the upper end of the band if the isothermal temperature coefficient is expected to be negative.

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8. Measurement of the all rods out boron concentration (boron endpoint) to verify calculational models and accident analysis assumptions.

9. Measurement of the isothermal coefficient which infers the boron and moderator temperature reactivity coefficients over the temperature and boron concentration ranges in which the reactor may initially be taken critical.

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10. Perform a pseudo-rod-ejection test to verify calculational models and accident analysis assumptions.

11. Measurements of control rod and control rod bank reactivity worths to (1) confirm that they are in accordance with design predictions and (2) confirm by analysis that the rod insertion limits will be adequate to confirm a shutdown margin consistent with accident analysis assumptions throughout core life, with the greatest worth control rod stuck out of the core.

~~–~~Abstracts of tests performed during this phase are provided in Section 14.2.12.

14.2.1.1.5 Phase IV - Power Ascension Testing

14.02-22

A series of power ascension tests is conducted to bring the reactor to full power. Testing is performed at various power levels and is intended to demonstrate that the facility operates in accordance with its design bases during steady state conditions and, to the extent practicable, during anticipated transients. ~~To check the analytical models used to predict plant responses to anticipated transients and postulated accidents are bounding, the measured responses are compared to the predicted responses. The predicted responses should be developed using real or expected values of such attributes as beginning of life core reactivity coefficients, flowrates, pressures,~~

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~~temperatures, pump coastdown characteristics, and response times of equipment, as well as the status of the plant.~~To validate the analytical models used to predict plant responses to anticipated transients and postulated accidents, these tests should establish that measured responses are in accordance with predicted responses. The predicted responses should be developed using real or expected values of such attributes as beginning-of-life core reactivity coefficients, flow rates, pressures, temperatures, pump coastdown characteristics, and response times of equipment, as well as the actual status of the plant (not those values or plant conditions assumed for conservative evaluations of postulated accidents).

Tests and acceptance criteria should also be prescribed to demonstrate the ability of major or principal plant control systems to automatically control process variables within design limits. Such tests are expected to provide assurance that the facility's integrated dynamic response is in accordance with design for plant events such as reactor trip, turbine trip, reactor coolant pump trip, and loss of feedwater heaters or feedwater pumps. Testing should be sufficiently comprehensive to establish that the facility can operate in all operating modes for which it has been designed; however, tests should not be conducted, or operating modes or plant configurations established, if they have not been analyzed or if they fall outside the range of assumptions used in analyzing postulated accidents described in the EPR FSAR.

Appropriate consideration should be given to testing at the extremes of possible operating modes for facility systems. Testing under simulated conditions of maximum and minimum equipment availability within systems should be accomplished if the facility is intended to be operated in these modes (e.g., testing with different reactor coolant pump configurations, single-loop reactor coolant system operation, operation with the minimum allowable number of pumps, heat exchangers, or control valves in the feedwater, condensate, circulating, and other cooling water systems).

The following items illustrate some of the types of performance demonstrations, measurements, and tests that are included in the power ascension test phase.

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1. Determine ~~power coefficients and steady state core performance~~ steady state core performance and power coefficients are within design limits (Test Numbers 190, 191, 192, 206, and 207).

2. Check rod drop times against plant data (Test Number 222).

3. Demonstrate capability and sensitivity to detect a control rod misalignment equal to or less than the Technical Specification limits (Test Number 213).

4. Verify that plant performance is as expected for runback and following a partial trip (Test Number 221).

5. Verify the capability of plant monitoring systems (Test Numbers 193, 197, 204, and 205).

6. Demonstrate the adequacy of design by comparing design values to performance data (Test Numbers 194, 199, 203, 210, 212, 215, and 216).

7. Demonstrate the ability of the plant to withstand transient conditions (Test Numbers 196, 198, 200, 211, 214, 217, 219, and 220).

~~-~~Abstracts of tests performed during power ascension are provided in Section 14.2.12.

14.02-22

A pseudo-rod-ejection test will be performed during initial criticality and LPPT and not during power ascension. AREVA NP has reviewed previous pseudo-rod-ejection tests performed during power ascension and the three-dimensional nodal models used for currently operating plants and the EPR. AREVA NP has determined that the data generated by this type of test would not be beneficial for modeling a rod ejection event.

14.2.2 Organization and Staffing

It is the responsibility of the COL applicant to organize and staff phases of the test program. A COL applicant that references the U.S. EPR certified design will provide site-specific information that describes the organizational units that manage, supervise, or execute any phase of the test program. This description should address the organizational authorities and responsibilities, the degree of participation of each identified organizational unit, and the principal participants. The COL applicant should also describe how, and to what extent, the plant's operating and technical staff participates in each major test phase. This description should include information pertaining to the experience and qualification of supervisory personnel and other principal participants who are responsible for managing, developing, or conducting each test phase. In addition, the COL applicant is responsible for developing a training program for each fundamental group in the organization.

14.2.3 Test Procedures

Detailed procedure guidelines and procedures provided by the appropriate design organization are utilized to develop various system test procedures. Thus, test procedures are based on the requirements of system designers and the applicable RGs.

Each test procedure is prepared using pertinent reference material provided by the appropriate design and vendor organizations, the FSAR, the Technical Specifications and the applicable RGs. A test procedure is prepared for each specific system test to be performed during the test program. Each system test procedure contains, at a minimum, the following major topic areas:

- Test objectives.

- Acceptance criteria.

- References.
- Prerequisites.
- System initial conditions.
- Environmental conditions.
- Special precautions.
- Detailed procedure (including data collection).
- Restoration.
- Documentation of test results.

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Acceptance criteria will be based on generic and site specific safety analyses. Once a safety analysis value has been identified it is necessary to decide the direction to conservatively bias the value for each test. Note that it is not uncommon to bias the safety analysis value one direction for one test and in the opposite direction for another test. The appropriate amount of bias to apply to the safety analysis value is the sum of several parameters. Some of the parameters that should be considered are as follows:

- Instrument uncertainty, including all components from the sensor to the indicator.
- Uncertainty due to sensing line. Is the fluid in the sensing line the same as the process fluid? If the effect is conservative it may be ignored but if it is non-conservative it should be considered.
- Static head correction from the point of interest. For example, from the reactor vessel flange to the instrument location. If this correction is conservative it may be ignored but if it is non-conservative is should be considered.
- Dynamic head correction. If this correction is conservative it may be ignored but if it is non-conservative is should be considered.
- Readability, round off error and instrument indicator should be considered if this correction is conservative it may be ignored but if it is non-conservative is should be considered.
- Engineering design margin is a term that is usually applied to a conservative bias applied to the acceptance criteria to account for component wear and other degradation factors.
- Atmospheric corrections is a term that is used to describe the conservative bias that is applied to the safety analysis values to account for the atmospheric condition differences between the conditions assumed in the accident analysis and present in the field during the test (fluid density, temperature, atmospheric pressure, etc.).

14.2.5 Review, Evaluation, and Approval of Test Results

A COL applicant that references the U.S. EPR design certification will address the site-specific administration procedures for review and approval of test results. Completed procedures and test reports included in the ITAAC shall be routed to the NRC Resident for Commission review. Final review and approval, including ITAAC reviews of overall test phase results for selected milestones or hold-points within test phases shall be completed before beginning the next phase of startup testing.

14.2.6 Test Records

According to applicable regulatory requirements, initial test program results are compiled and maintained in compliance with administrative procedures. Retention periods for test records are based on considerations of their usefulness in documenting plant performance characteristics, and are retained in accordance with RG 1.28, Quality Assurance Program Requirements – Design and Construction, as described in Section 17. Startup test reports will be prepared in accordance with RG 1.16, Reporting of Operating Information – Appendix A Technical Specifications.

14.2.7 Conformance of Test Programs with Regulatory Guides

The primary regulatory guide for the startup test program is RG 1.68, Initial Test Program for Water Cooled Nuclear Power Plants, Revision 3, March 2007. The startup test program will conform to the relevant testing guidance in applicable regulatory guides. The RGs which provide specific guidance related to testing and testing programs are:

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- RG 1.9 - ~~Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants~~Application and Testing of Safety-Related Diesel Generators in Nuclear Power Plants, Revision 4, March 2007.
- RG 1.20 - Comprehensive Vibration Assessment Program for Reactor Internals During Preoperation and Initial Startup Testing, Revision 3, March 2007.
- RG 1.30 - Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment, Revision 0, August 1972.
- RG 1.37 - Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants, Revision 1, March 2007.
- RG 1.41 - Preoperational Testing of Redundant On-Site Electric Power Systems to Verify Proper Load Group Assignments, Revision 0, March1973.
- RG 1.52 - Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere

14.2.9 Trial Use of Plant Operating and Emergency Procedures

The test program schedule is addressed in Section 14.2.11. The schedule for the development of the plant operating and emergency procedures shall allow sufficient time for trial use of these procedures during the initial test program as appropriate and to the extent possible. For example, the Plant Operations staff should take every available opportunity to use the plant procedures as follows:

- Normal operations procedures should be used to perform basic valve alignments for preoperational tests.
- Hot Functional testing should be performed with as many normal operations procedures as practical.
- Emergency operating procedures that require special plant conditions, such as the reactor head removed and the refueling cavity available to receive water, should be performed when those conditions have been created for preoperational testing.
- Technical specification surveillance tests should be performed and surveillance test problems corrected prior to fuel loading.

In addition, the COL applicant should identify the specific operator training to be conducted as part of the low-power testing program related to the resolution of TMI Action Plan Item I.G.1, as described in the following reports:

NUREG-0660 - NRC Action Plans Developed as a Result of the TMI-2 Accident, Revision 1, August 1980.

NUREG-0694 - TMI-Related Requirements for New Operating Licenses, June 1980.

NUREG-0737 - Clarification of TMI Action Plan Requirements.

To accomplish these requirements, the emergency procedures will be performed on the plant simulator for procedure validation and operator training.

14.2.10 Initial Fuel Loading and Initial Criticality

Initial fuel loading and initial criticality are performed in a controlled manner during the startup test program. These activities are performed in a controlled and safe manner using the test procedures addressed in Section 14.2.12. Technical Specification requirements are applicable and must be satisfied prior to these operations.

14.2.10.1 Initial Fuel Loading

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Licensees should establish and follow specific safety measures, such as:

1. Establish requirements for periodic data-taking.

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2. Predictions of core reactivity should be prepared in advance to aid in evaluating the measured responses to specified loading increments.

3. Establish requirements for the operability of plant systems and components, including reactivity control systems and other systems and components necessary to ensure the safety of plant personnel and the public in the event of errors or malfunctions.

4. Scram time tests should be sufficient to provide reasonable assurance that the control rods will scram within the required time under plant conditions that bound those under which the control rods might be required to function to achieve plant shutdown (testing should demonstrate control rod scram times at both hot zero power and cold temperature conditions, and with flow and no-flow conditions).

Minimum initial conditions for core load:

- ~~All ITAAC have been closed.~~

- The fuel loading evolution is controlled by the use of approved plant procedures, which establish plant conditions, control access, establish security, control maintenance activities, and provide instructions that pertain to the use of fuel handling equipment.

- The boron concentration and isotopic content in the coolant is verified to be equal to or greater than that required for refueling. It is not anticipated that the refueling cavity will be completely filled. However, the water level in the reactor vessel shall remain above the installed fuel assemblies at times.

- The residual heat removal system (RHRS) provides coolant circulation that verifies adequate boron mixing and a means of controlling water temperature. The in-containment refueling water storage tank (IRWST) is in service and contains borated water at a volume and concentration that complies with the requirements. Applicable administrative controls shall be used to prevent unauthorized alteration of system lineups or change to the boron concentration in the RCS.

- The initial core loading is directly supervised by a senior licensed operator having no other concurrent duties.

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

- The status of all systems required for fuel loading is specified.

- The status of containment is specified.

- The status of the reactor vessel is specified.

- The fuel handling equipment has been verified to be operating correctly by performing preoperational tests prior to handling fuel.

- The status of protection systems, interlocks, alarms, and radiation protection equipment has been verified.

- A minimum of two permanent or temporary neutron detectors are located so that core reactivity changes can be detected and recorded. The neutron detectors shall be calibrated and operable prior to fuel movement.

- Response checks of neutron detectors are required prior to the commencement of fuel loading.

- Continuous area radiation monitoring shall be provided during fuel handling and fuel loading operations. Permanently installed radiation monitors display radiation levels in the main control room (MCR) and shall be monitored by licensed operators.

Fuel assemblies, together with inserted components, are placed in the reactor vessel, one at a time, according to previously established and approved sequences. The initial fuel loading procedure shall include detailed instructions, which prescribe successive movements of each fuel assembly. The procedures allow each fuel assembly movement to be verified prior to proceeding with the next assembly. Multiple checks are made for fuel assembly and inserted component serial numbers to guard against possible inadvertent exchanges or substitutions.

At least two fuel assemblies that contain primary neutron sources shall be placed into the core at appropriate specified points in the initial fuel loading procedure. This will ~~ensure~~provide a neutron population large enough for adequate monitoring of the core. As each fuel assembly is loaded, at least two separate inverse count-rate plots shall be maintained to verify that the extrapolated inverse count-rate ratio (ICRR) behaves as expected. The ICRR plots should also include related plant data (RHR flow, RCS temperature, etc.) that is taken on the same frequency. In addition, nuclear instrumentation shall be monitored to verify that each just-loaded fuel assembly does not excessively increase the count-rate. The results of each loading step shall be reviewed and evaluated before the next sequence fuel assembly is grappled by the manipulator crane.

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Criteria for the safe loading of fuel require that loading operations stop immediately if:

- The neutron count-rate from either temporary nuclear channel unexpectedly doubles during any single loading step, excluding anticipated change due to detector or source movement, or spatial effects such as a fuel assembly coupling source with a detector.

- The neutron count-rate on any individual nuclear channel increases by a pre-established maximum multiplication factor during any single loading step, excluding anticipated changes due to detector or source movement, or spatial effects such as a fuel assembly coupling source with a detector.

- There is a loss of communications between the control room and the senior licensed operator or fuel handling personnel.
- There is less than the required minimum number of operable source-range detectors.
- The extra borating system is inoperable.

A fuel assembly shall not be un-grappled from the refueling machine until stable count-rates have been obtained. In the event that an unexplained increase in count-rate is observed on any nuclear channel, the last fuel assembly loaded shall be withdrawn. Before proceeding, the procedure and loading operation shall be reviewed and evaluated to verify the safe loading of fuel.

Plant procedures shall establish criteria for the following:

- Emergency boron injection.
- Containment evacuation.
- Actions to be followed in the event of fuel damage.
- Actions to be followed or approvals to be obtained before routine loading may resume after one of the above limitations has been reached or invoked.

14.2.10.2 Initial Criticality

Initial criticality is controlled by the use of approved plant procedures which establish required plant conditions and successful completion of prerequisite tests described in Section 14.2.12. Initial criticality is obtained by a specified, controlled and orderly combination of a rod cluster control assembly (RCCA) withdrawal, and a boron concentration reduction. The approach to criticality requires that RCCA groups be withdrawn in sequence with overlap, except for the last regulating group, which shall remain far enough into the core to provide effective control when criticality is achieved. The RCS boron concentration is then reduced to achieve criticality, at which time the regulating group shall be used to maintain criticality.

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Core response during RCCA group withdrawal and RCS boric acid concentration reduction shall be monitored in the MCR by observing the change in neutron count-rate as indicated by the permanent nuclear instrumentation. The U.S. EPR plans to use the rod withdrawal sequence and dilution to criticality in subsequent plant startups that require LPPT. For U.S. EPR startups not requiring LPPT, the plan is to dilute to a concentration that corresponds to an estimated critical condition with Control Bank D near the core mid-plane. The same withdrawal sequence and pattern is used with criticality being achieved during Control Bank D withdrawal.

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During reactor startup, the~~The~~ neutron count-rate is plotted as a function of RCCA group position and RCS boron concentration during the approach to criticality. The approach to criticality shall be controlled and specific hold points shall be specified in the procedure. The results of the inverse count-rate monitoring and the indications on installed instrumentation shall be reviewed and evaluated before proceeding to the next prescribed hold point.

The criteria for providing a safe and controlled approach to criticality require that the following conditions are met:

- ~~That h~~High flux trip setpoints ~~be~~are reduced to a value consistent with performance of the next test plateau.

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- Rod drop time tests have been performed to provide reasonable assurance that the control rods will trip within the required time under plant conditions that bound those under which the control rods might be required to function to achieve plant shutdown (rod drop testing should demonstrate control rod drop times at both hot zero power and cold temperature conditions, and with all RCPs operating and no-RCPs operating conditions).
- Technical Specifications required for entry into MODE 2 are met.
- A minimum count rate of two counts per second (cps) is met.
- A signal-to-noise ratio greater than two is met.
- A statistical reliability test on each operable source range instrument is performed.
- A sustained startup rate of one decade per minute is not exceeded.
- ~~That a sustained startup rate of one decade per minute not be exceeded.~~
- ~~That~~ RCCA withdrawal or boron dilution is suspended if unexplainable changes in neutron count-rates are observed.
- ~~That a~~ A minimum of one decade of overlap ~~be~~is observed between the source and intermediate channels of the excore nuclear instruments.

14.2.11 Test Program Schedule

The scheduling of individual tests or test sequences is established so that systems and components that are required to prevent or mitigate the consequences of postulated accidents are tested prior to fuel loading. Tests that require a substantial core power level for proper performance are performed at the lowest power level commensurate with obtaining acceptable test data.

A COL applicant that references the U.S. EPR certified design will develop a test program that considers the following ~~five~~seven guidance components:

3.1 Initiate an SI signal.

4.0 DATA REQUIRED

4.1 The following time dependent data shall be collected at a frequent scan rate:

4.1.1 RCS parameters (temperature and pressure).

4.1.2 SI parameters (e.g., flow, pump discharge pressure, fluid temperature).

4.2 Pressurizer parameters (e.g., pressure, level).

4.3 VCT pressure and level.

4.4 Letdown ~~flowrate~~flow rate.

5.0 ACCEPTANCE CRITERIA

5.1 The safety injection system performs as described in Section 6.3.

14.2.12.13.18 Pre-Core Loss of Instrument Air (Test #178)

1.0 OBJECTIVES

1.1 To demonstrate that a reduction and loss of instrument air pressure causes no adverse operation of active safety-related equipment.

2.0 PREREQUISITES

2.1 Construction activities on items to be tested have been completed.

2.2 Individual valves and equipment are functional.

2.3 The instrument air system is in service at rated pressure with support systems functional to the extent necessary to conduct the test. Pneumatic loads are cut-in to the extent possible at the time test begins.

2.4 A listing of the air-operated active safety-related equipment important to safety which includes the loss of air failed position and the fail safe position of each component has been compiled.

2.5 The CAS test, in conjunction with this test satisfies the requirements of RG 1.68.3, RG C.1-C.11.

2.6 Loss-of-air supply tests shall be conducted on branches of the instrument air system simultaneously, if practicable, or on the largest number of branches of the system that can be adequately managed.

3.0 TEST METHOD

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3.1 Place the valves in the normal operating position, and maintain plant in as close to normal conditions as it practicable and verify proper operation of the following components:

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3.1.1 Compressors.

3.1.2 Aftercoolers.

3.1.3 Oil separator units.

3.1.4 Air receivers.

3.1.5 Dryers.

3.1.6 Pressure controls and compressor unloaders.

3.1.7 Pressure-reducing stations.

3.1.8 Automatic and manual start / stop circuits of standby compressors.

3.1.9 Controls to change operating sequence of units (spread operating time and starting duty).

3.1.10 High and low pressure alarms.

3.1.11 Pressure indicators.

3.1.12 Temperature indicators.

3.1.13 Relief valve settings.

3.2 Where safe to personnel and equipment, conduct a loss of air test on integrated systems by performing the following tests:

3.2.1 Shutoff the instrument air system in a manner that would simulate a sudden air pipe break and verify that the affected components respond as designed.

3.2.2 Repeat Test A, but shut the instrument air system off slowly to simulate a gradual loss of pressure.

3.2.3 Where deemed necessary, depressurize individual components. Note component response.

3.2.4 Return instrument air to the depressurized systems and components. Note responses.

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3.2.5 Verify automatic isolations between safety and non-safety or between plant critical and plant non-critical components function as designed.

3.2.6 Simulate worse case loads by simultaneous operation of components or by creating a false parasitic load that bounds estimates of simultaneous operation of worse case loads.

4.0 DATA REQUIRED

4.1 Response of systems and components to loss of instrument air and subsequent restoration.

5.0 ACCEPTANCE CRITERIA

5.0 ACCEPTANCE CRITERIA

5.1 Performance of the LDS is as described in Section 5.2.5.

5.2 The leak detection alarm setpoints have been adjusted using the baseline data.

14.2.12.15 Phase III: Initial Criticality and Low Power Physics Tests

14.2.12.15.1 Critical Boron Concentration: All Rods Out (Test #190)

1.0 OBJECTIVE

1.1 To measure the HZP critical boron concentration with rods fully withdrawn.

2.0 PREREQUISITES

2.1 The rods are fully withdrawn except for the control bank, which is 50 pcm to 200 pcm inserted.

2.2 Available pressurizer heaters are energized.

2.3 The reactor is taken critical by boron dilution method.

2.4 The reactivity computer is functional.

2.5 Reactor power is below the point of adding heat.

3.0 TEST METHOD

3.1 Verify that just critical reactor is maintained by rod movement until boron concentration is stabilized and boron sample results are recorded.

3.2 Verify that rods are fully withdrawn except for the control bank, which is 50 pcm to 200 pcm inserted. If rod position is no longer within required limits borate or dilute as necessary to restore rod position and return to previous step.

14.02-19 3.3 Measure critical boron concentration in a known reactivity configuration using an approved method ~~described in ANSI/ANS-19.6.1~~.

4.0 DATA REQUIRED

4.1 Critical conditions:

4.1.1 Boron concentration (i.e., RCS and pressurizer).

4.1.2 RCCA positions.

4.1.3 RCS temperature.

4.1.4 Pressurizer pressure.

5.0 ACCEPTANCE CRITERIA

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5.1 The measured critical boron concentration when compared to the predicted boron concentration is within the acceptance criteria ~~specified in ANSI/ANS-19.6.1~~.

14.2.12.15.2 Isothermal Temperature Coefficient (Test #191)

1.0 OBJECTIVE

1.1 To measure the isothermal temperature coefficient (ITC) for the reactor.

2.0 PREREQUISITES

2.1 Available pressurizer heaters are energized.

2.2 The reactor is critical with a stable boron concentration, RCS temperature and pressure.

2.3 The rods are fully withdrawn except for the control bank, which is 50 pcm to 200 pcm inserted.

2.4 The reactivity computer is functional.

2.5 Reactor power is below the point of adding heat.

3.0 TEST METHOD

3.1 Introduce changes in RCS temperature while measuring the resultant changes in reactivity.

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3.2 Measure isothermal temperature coefficient in a known reactivity configuration using an approved method ~~described in ANSI/ANS-19.6.1~~.

4.0 DATA REQUIRED

4.1 Critical conditions:

4.1.1 Pressurizer pressure.

4.1.2 RCCA configuration.

4.1.3 Boron concentration (i.e., RCS and Pressurizer).

4.1.4 Time dependent information:

- Reactivity.
- RCCA position.
- Temperature.

5.0 ACCEPTANCE CRITERIA

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5.1 The measured ITC when compared to the predicted ITC is within the acceptance criteria ~~specified in ANSI/ANS-19.6.1~~.

14.2.12.15.3 Rod Worth (Test #192)

1.0 OBJECTIVE

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1.1 To measure the worth of rod groups. For the U.S. EPR, rod groups shall be measured by one of the approved methods ~~described in ANSI/ANS-19.6.1~~. During initial criticality one of the methods that determine the worth of rod groups shall be used.

2.0 PREREQUISITES

2.1 The reactor is critical.

2.2 All available pressurizer heaters are energized.

2.3 The reactivity computer is operating.

2.4 Reactor power is below the point of adding heat.

3.0 TEST METHOD

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3.1 Measure the control rod worth in a known reactivity configuration using an approved method ~~described in ANSI/ANS-19.6.1~~.

4.0 DATA REQUIRED

4.1 Conditions of the measurement:

4.1.1 RCS temperature.

4.1.2 Pressurizer pressure.

4.1.3 RCCA configuration.

4.1.4 Boron concentration.

4.2 Time dependent information:

4.2.1 Reactivity variation.

4.2.2 RCCA positions.

5.0 ACCEPTANCE CRITERIA

14.02-19

5.1 The measured control rod worths when compared to the predicted rod worths is within the acceptance criteria ~~specified in ANSI/ANS-19.6.1~~.

14.2.12.16 Phase IV: Power Ascension Tests, 5 Percent Power Ascension Plateau

Some of the following tests are performed in more than one plateau. In those instances the test is listed in the first plateau that it is recommended to be performed. The plant instrumentation shall be functional prior to each test.

14.2.12.16.1 Low Power Biological Shield Survey (Test #193)

1.0 OBJECTIVE

14.02-19

1.2 To measure the full-power critical boron concentration with rods fully withdrawn ~~per ANSI/ANS 19.6.1~~. This procedure shall be performed at the following plateau:

1.2.1 ≥98 percent reactor power in accordance with RG 1.68. This test combined with the steady-state core performance test satisfies the reactivity coefficient evaluation requirement of RG 1.68.

2.0 PREREQUISITES

2.1 The reactor is at a high power level with the following:

2.1.1 Equilibrium xenon.

2.1.2 RCCAs are within 10 steps of the rods out position.

2.1.3 AO has been stable (±2 percent) for previous 24 hours.

2.1.4 Measure boron concentration and boron–10 isotopic concentration.

3.0 TEST METHOD

3.1 The reactivity coefficients are determined by updating the three dimensional core model (POWERTRAX) with current plant data.

3.2 The tests.

4.0 DATA REQUIRED

4.1 Reactor thermal power.

4.2 RCCA configuration.

4.3 Boron concentration (ppmB) and isotopic abundance of boron-10.

4.4 Core burnup.

4.5 RCS temperature.

5.0 ACCEPTANCE CRITERIA

5.1 The core designer shall investigate differences greater than 500 pcm between measured values and predications.

5.2 Test criteria specified ~~in ANSI/ANS 19.6.1~~. 14.02-19

14.2.12.21.2 Trip of Generator Main Breaker (Test #219)

1.0 OBJECTIVE

1.1 To demonstrate that the plant responds and is controlled as designed following a full-power turbine trip.

1.2 This procedure shall be performed at the following plateau:

1.2.1 ≥98 percent reactor power in accordance with RG 1.68.

5.1 RCSL and turbine controls remain within analyzed limits and reactor power is stabilized at the lower power for at least 30 minutes following the test initiation without unanticipated operator action.

5.2 Electrical distribution system voltage and frequency measurements can be correlated with the transient load flow analysis.

14.2.13 References

1. ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Facility Components," Class 1, 2, and 3 Components, The American Society of Mechanical Engineers, 2004 (No Addenda).

2. ASME BPV Code, Section III, Division 1, Subsection NE, "Class MC Components," The American Society of Mechanical Engineers, 2004 (No Addenda).

3. ASME PTC-6, "Guidance for Evaluation of Measurement Uncertainty in Performance Tests of Steam Turbines," The American Society of Mechanical Engineers, 2004.

4. ASME PTC-4.1 1964 (R 1991), "Steam Generating Units," The American Society of Mechanical Engineers, 1964 (Reaffirmed 1991).

5. NFPA 72, "National Fire Alarm Code," National Fire Protection Association Standards, 2002.

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6. ~~ANSI/ANS 19.6.1, "Reload Startup Physics Test for Pressurized Water Reactors," American National Standards Institute, 1997.~~

7. ANSI/ACI 349, "Code Requirements for Nuclear Safety Related Concrete Structures," American National Standards Institute, 2006.