ArevaEPRDCPEm Resource

From:	Pederson Ronda M (AREVA NP INC) [Ronda.Pederson@areva.com]
Sent:	Friday, February 13, 2009 5:11 PM
To:	Getachew Tesfaye
Cc:	BENNETT Kathy A (OFR) (AREVA NP INC); DELANO Karen V (AREVA NP INC); DUNCAN Leslie E (AREVA NP INC)
Subject:	Response to U.S. EPR Design Certification Application RAI No. 143, Supplement 1
Attachments:	RAI 143 Supplement 1 Response US EPR DC.pdf

Getachew,

AREVA NP Inc. provided responses to 10 of the 11 questions of RAI No. 143 on December 18, 2008. The attached file, "RAI 143 Supplement 1 Response US EPR DC.pdf" provides a technically correct and complete response to the remaining question, as committed.

Appended to this file are affected pages of the U.S. EPR Final Safety Analysis Report in redline-strikeout format which support the response to RAI 143 Question 14.02-64.

The following table indicates the respective page in the response document, "RAI 143 Supplement 1 Response US EPR DC.pdf," that contains AREVA NP's response to the subject question.

Question #	Start Page	End Page
RAI 143 — 14.02-64	2	2

This concludes the formal AREVA NP response to RAI 143, and there are no questions from this RAI for which AREVA NP has not provided responses.

Sincerely,

Ronda Pederson
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Licensing Manager, U.S. EPR Design Certification
AREVA NP Inc.
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From: WELLS Russell D (AREVA US)
Sent: Thursday, December 18, 2008 7:47 PM
To: 'Getachew Tesfaye'
Cc: 'John Rycyna'; Pederson Ronda M (AREVA US); BENNETT Kathy A (OFR) (AREVA US); DELANO Karen V (AREVA US); SLIVA Dana (EXT)
Subject: Response to U.S. EPR Design Certification Application RAI No. 143, FSAR Ch 14

Getachew,

Attached please find AREVA NP Inc.'s response to the subject request for additional information (RAI). The attached file, "RAI 143 Response US EPR DC.pdf" provides technically correct and complete responses to 10 of the 11 questions.

Appended to this file are affected pages of the U.S. EPR Final Safety Analysis Report in redline-strikeout format which support the response to RAI 143 Questions 14.02-61, 14.02-62, 14.02-63, 14.02-65, 14.02-66, 14.02-67, 14.02-68, 14.02-70, and 14.02-71.

The following table indicates the respective pages in the response document, "RAI 143 Response US EPR DC.pdf," that contain AREVA NP's response to the subject questions.

Question #	Start Page	End Page
RAI 143 — 14.02-61	2	2
RAI 143 — 14.02-62	3	3
RAI 143 — 14.02-63	4	4
RAI 143 — 14.02-64	5	5
RAI 143 — 14.02-65	6	6
RAI 143 — 14.02-66	7	7
RAI 143 — 14.02-67	8	8
RAI 143 — 14.02-68	9	9
RAI 143 — 14.02-69	10	10
RAI 143 — 14.02-70	11	11
RAI 143 — 14.02-71	12	12

A complete answer is not provided for 1 of the 11 questions. The schedule for a technically correct and complete response to this question is provided below.

Question #	Response Date
RAI 143 — 14.02-64	February 13, 2009

Sincerely,

(Russ Wells on behalf of)
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From: Getachew Tesfaye [mailto:Getachew.Tesfaye@nrc.gov]
Sent: Tuesday, November 18, 2008 3:43 PM
To: ZZ-DL-A-USEPR-DL
Cc: Dori Votolato; Juan Peralta; Michael Miernicki; Joseph Colaccino; John Rycyna
Subject: U.S. EPR Design Certification Application RAI No. 143 (1386), FSARCh. 14

Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on November 14, 2008, and on November 18, 2008, you informed us that the RAI is clear and no further clarification is needed. As a result, no change is made to the draft RAI. The schedule we have established for review of your application assumes technically correct and complete responses within 30 days of receipt of RAIs. For any RAIs that cannot be answered within 30 days, it is expected that a date for receipt of this information will be provided to the staff within the 30 day period so that the staff can assess how this information will impact the published schedule.

Thanks,
Getachew Tesfaye
Sr. Project Manager
NRO/DNRL/NARP
(301) 415-3361

Hearing Identifier: AREVA_EPR_DC_RAIs
Email Number: 218

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Response to

Request for Additional Information No. 143, Supplement 1 (1386), Revision 0

11/18/2008

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: SRP 14.2

QUESTIONS for Quality and Vendor Branch 1 (AP1000/EPR Projects) (CQVP)

Question 14.02-64:

Regulatory Guide 1.68 Appendix A.1(3) states that for vibration tests Regulatory Guide (RG)1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing," should be used as guidance. However, Table 1.9-2, "U.S. EPR Conformance with Regulatory Guides," states that the U.S. EPR FSAR takes exception to RG 1.20 for Sections 14.2.12, "Individual Test Descriptions," of the startup testing. The staff requests that AREVA revise Table 1.9-2 to include use of RG 1.20 during startup vibration tests or provide justification for the exception to RG 1.20.

Response to Question 14.02-64:

The U.S. EPR conforms to Regulatory Guide (RG) 1.20 for the reactor internals and the piping and components in the reactor coolant, main steam, condensate, and feedwater systems, with the following exceptions:

- The vibration and stress analysis program and the vibration and stress measurement program requirements of RG 1.20 for the steam generator internals. Justification for this exception is described in U.S. EPR FSAR Tier 2, Section 3.9.2.4.

- The vibration and stress measurement program requirements of RG 1.20 for the condensate system instrumentation. Hand-held vibration monitors are used during start-up testing to measure piping acceleration at discrete locations in the U.S. EPR's condensate system. This identifies high vibration locations and is an effective method for determining the severity of piping vibrations. Hand-held vibration instrumentation is suitable for specific frequency ranges, and the frequency range of the instruments used is determined as part of the test program.

The above exceptions to RG 1.20 will be added to U.S. EPR FSAR, Tier 2, Table 1.9-2, Section 3.9.2.4.1, and Section 14.2.7. Additional details on measuring vibration levels will be added to U.S. EPR FSAR, Tier 2, Section 14.2.12 (Tests #066 and #164).

U.S. EPR FSAR, Tier 2, Section 14.2 was reviewed for consistency:

- Section 14.2.7 was reviewed to verify the applicable Regulatory Guides in RG 1.206, Section C.I.14.2.7 are included. RG 1.136 will be added to U.S. EPR FSAR, Tier 2, Section 14.2.7 and Test #029.

- U.S. EPR FSAR Tier 2, Table 1.8-2, Item 14.2-2 will be revised to match the wording in U.S. EPR FSAR Tier 2, Section 14.2.11, which was revised in RAI 16, Question 14.02-15 and Question 14.02-16.

FSAR Impact:

U.S. EPR FSAR, Tier 2, Table 1.8-2, Table 1.9-2, Section 3.9.2.4.1, Section 14.2.7, and Section 14.2.12 (Tests #029, #066, and #164) will be revised as described in the response and indicated on the enclosed markup.

U.S. EPR Final Safety Analysis Report Markups

Table 1.8-2—U.S. EPR Combined License Information Items
Sheet 36 of 43

Item No.	Description	Section	Action Required by COL Applicant	Action Required by COL Holder
14.2-1	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.	14.2.2	Y	
14.2-2	A COL applicant that references the U.S. EPR certified design will develop a test program that considers the following ~~five~~seven guidance components: 1. The applicant should allow at least ~~9~~nine months to conduct preoperational testing. 2. The applicant should allow at least ~~3~~three months to conduct startup testing, including fuel loading, low power tests, and power ascension tests. 3. Plant safety will not be dependent on the performance of untested SSC during any phase of the startup test program. 4. Surveillance test requirements will be completed in accordance with plant Technical Specification requirements for SSC operability before changing plant modes. 5. Overlapping test program schedules (for multi-unit sites) should not result in significant divisions of responsibilities or dilutions of the staff provided to implement the test program. ~~4~~6. The sequential schedule for individual startup tests should establish, insofar as practicable, that test requirements should be completed prior to exceeding 25 percent power for ~~SSCs~~SSC that are relied upon to prevent, limit, or mitigate the consequences of postulated accidents. ~~5~~7. Approved test procedures should be in a form suitable for review by regulatory 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. 14.02-64	14.2.11		Y
14.2-3	A COL Applicant that references the US EPR design certification will provide site-specific information for review and approval of test procedures.	14.2.3	Y	

Table 1.9-2—U.S. EPR Conformance with Regulatory Guides
Sheet 2 of 19

RG / Rev	Description	U.S. EPR Assessment	FSAR Section(s)
1.16, R4	Reporting of Operating Information -- Appendix A Technical Specifications	N/A-COL	N/A
1.20, R3	Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing	Y	3.9.2.3 and 3.9.2.4
			14.2.12
	14.02-64	EXCEPTION (Vibration ~~analysis & startup~~ testing of SG internals and condensate system)	3.9.2.3 and 3.9.2.4
			14.2.12
1.21, R1	Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants	Y	11.5.1
1.22, 02/1972	Periodic Testing of Protection System Actuation Functions	Y	7.1.2.2.2
1.23, R1	Meteorological Monitoring Programs for Nuclear Power Plants	Y	2.3
1.24, 03/1972	Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure	N/A-SUP (Refer to RG 1.145)	N/A
1.25, 03/1972	Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors	N/A-SUP (Refer to RG 1.145 and RG 1.183)	N/A
1.26, R4	Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants	Y	3.2.2
1.27, R2	Ultimate Heat Sink for Nuclear Power Plants	Y	3.5
			9.2.5
1.28, R3	Quality Assurance Program Requirements (Design and Construction)	Y	14.2.6
		Y (Per AREVA Topical Report ANP-10266-A	17.5

flow-induced vibration analyses or startup testing is currently planned for these components.

The vibration of representative trains of piping attached to the RCS as well as main steam and main feedwater lines are measured during initial startup testing. These measurements will be taken at discrete piping locations and also at the other key components (e.g., valves and pumps) installed along the length of pipe. Accelerations will be measured using hand-held devices for both steady-state and transient flow conditions.

In the 1970s and 1980s, the above process was employed to address vibration concerns during startup testing for the current operating plants. Since there are very few instances of excessive pipe vibration while operating these plants at their design power level, this is a proven and reliable method of validation. The majority of the cases in which excessive pipe vibration have been observed have been at the stand off branch lines in the main steam piping system when the plant uprates to a higher power level. Therefore, greater scrutiny of these piping configurations will be stressed during the initial startup testing using the method outlined above. Under certain conditions, some of the stand off branch lines may be instrumented with permanent sensors to monitor their accelerations during the life of the plant.

Excessive vibration or instabilities in piping systems can be difficult to analytically predict for most flow-induced vibration mechanisms. This is due to subtle differences in the thermal hydraulic conditions in the piping that can have a significant effect on the vibration response of these piping components. For these reasons, the critical piping systems (e.g., the main steam and feedwater piping systems) will be instrumented with permanent sensors that will measure the accelerations in each translational direction during the operating life of the plant. The acceptance criteria for the piping and other key components installed along the length of the pipe will be based upon satisfying the appropriate displacement, acceleration, stress, and fatigue limits.

3.9.2.4.1 Exceptions to Regulatory Guide 1.20

The U.S. EPR conforms to Regulatory Guide 1.20 in regards to the reactor internals, piping and components in the reactor coolant, main steam, condensate, and feedwater systems with the following exceptions:

1. Vibration and Stress Analysis Program and Vibration and Stress Measurement Program requirements of RG 1.20 with regards to steam generator internals as discussed above.

2. Vibration and Stress Measurement Program requirements of RG 1.20 with regards to instrumenting the condensate system - Accelerations at discreet locations in condensate system of the U.S. EPR will be measured during start-up testing using

14.02-64

14.02-64

hand-held vibration monitors. Hand held vibration measurement equipment is useful as an indicator of piping vibration, in that the personnel using such equipment can identify the locations of high vibration by moving along the piping. By doing so, effective estimates of the severity of piping vibration may be accomplished. Hand held vibration instrumentation is suitable for certain frequency ranges, and the frequency range of the actual instruments used will be identified as part of the test program. This information will be used in the vibration evaluations.

3.9.2.5 Dynamic System Analysis of the Reactor Internals Under Faulted Conditions

The dynamic model used in the analysis of the RPV internals, the RPV Isolated Model, includes the core, the RPV upper and lower internals, the RPV pressure boundary, the reactor coolant loops (piping and components), the RCS supports, and the Reactor Building internal structure. The dynamic analyses consider the effects of the gaps that exist between the vessel and the core barrel, between the vessel and the upper support assembly, between the vessel and the lower support plate, between fuel assemblies, and between the fuel assemblies and the heavy reflector. See Appendix 3C for a representative diagram of the RPV Isolated Model and additional information regarding the dynamic loading analysis of this model.

Analysis of the RPV internals for blowdown loads resulting from a guillotine break of the safety injection line nozzles on the hot and cold legs is performed using direct step-by-step integration methods. Note that breaks are not considered in the main coolant loop piping (hot and cold legs), pressurizer surge line, and main steam line piping (from the steam generators to the first anchor point location) due to the application of leak-before-break methodology to these lines (see Section 3.6.3). The forcing functions obtained from hydraulic analysis of the safety injection line breaks are defined at points in the RPV internals where changes in cross-section or direction of flow occur, such that differential loads are generated during the blowdown transient. Additional details of the structural analysis of the RPV Isolated Model for LOCA loading are given in Appendix 3C.

Analysis of the RPV internals for safe shutdown earthquake (SSE) loading uses direct step-by-step time-history analysis techniques. The SSE analysis of the RPV Isolated Model is described in Appendix 3C.

The response of the RPV internals to SSE loading are combined with their response to the safety injection line breaks by the square-root-of-the-sum-of-the-squares method. Section 3.9.3 provides the faulted load combinations considered in the stress and fatigue analyses of the RPV internals.

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:

- 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. 14.02-64 Exceptions to regulatory guidance are described in Section 3.9.2.4.1.

- 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

Cleanup Systems in Light-Water-Cooled Nuclear Power Plants, Revision 3, June 2001.

- RG 1.68.2 - Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants, Revision 1 July 1978.
- RG 1.68.3 - Preoperational Testing of Instrument and Control Air Systems, Revision 0, April 1982.
- RG 1.72 - Spray Pond Piping Made from Fiberglass-Reinforced Thermosetting Resin, Revision 2, November 1978. This RG is not applicable because the U.S. EPR does not use this type of spray pond piping.
- RG 1.78 - Evaluating the Habitability of a Nuclear Power Plant Control Room during a Postulated Hazardous Chemical Release, Revision 1, December 2001. ~~This RG is not applicable to the initial plant test program for the U.S. EPR.~~
- RG 1.79 - Preoperational Testing of Emergency Core Cooling Systems for Pressurized-Water Reactors, Revision 1, September 1975.
- RG 1.116 - Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems, Revision 0-R, May 1977.
- RG 1.118 - Periodic Testing of Electric Power and Protection Systems, Revision 3, April 1995.
- RG 1.128 - Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants, Revision 2, February 2007.

14.02-64

- RG 1.136 - Design Limits, Loading Combinations, Materials, Construction, and Tesing of Concrete Containments, Revision 3, March 2007.
- RG 1.139 - Guidance for Residual Heat Removal. Revision 0, May 1978.
- RG 1.140 - Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants, Revision 2, June 2001.

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

The design of the U.S. EPR is an evolutionary design. As such, the experience gained from previous successful startups is factored into the initial test program. This information reflects both AREVA NP operating and test experience and industry wide experience concerning pressurized water reactors. A summary will be developed to provide conclusions from this review and the effects on the test program.

The plant operations staff reviews reactor operating and testing experiences at other facilities that are similar in design and capacity prior to the unit starting up. This review is carried out by circulating the following information to startup and

3.7 Verify the test accuracy by supplementary means.

4.0 DATA REQUIRED

4.1 Structural integrity data.

4.1.1 The readings of instrumentation to measure containment building movement shall be recorded at selected pressure levels.

4.1.2 Displacements shall be measured at several points along locations spaced evenly around the containment. The locations shall include the following:

- Springline.
- Top of dome (the top of the concrete).
- Locations with varying stiffness characteristics such as major penetrations.
- Radial and vertical defections of the containment wall adjacent to the equipment hatch opening shall be measured at twelve points around the hatch penetration. The twelve points shall be three locations each at the three, six, nine and twelve o'clock positions around the penetration.

4.2 Integrated leak rate data.

4.2.1 Containment temperature, pressure, and humidity.

4.2.2 Reference vessel temperature and pressure.

4.2.3 Atmospheric pressure and temperature.

4.2.4 Known leakage air flow.

5.0 ACCEPTANCE CRITERIA

5.1 SIT results are satisfactory, in accordance with ASME BPV Code, Article CC-6000 and RG 1.136, as detailed in ~~Section 3.8.1~~Section 3.8.1.7.1. 14.02-64

5.2 Containment vessel shows no signs of structural degradation following the SIT as designed ~~described in~~(refer to Section 3.8.1).

5.3 The ILRT results are satisfactory as specified in Section 6.2.6.

5.3.1 FSAR Table 14.3-1 Item 1-10.

5.3.2 FSAR Table 14.3-2 Item 2-2.

5.4 Containment ~~performs as~~meets design requirements (~~described in~~refer to Section 3.1.5, Section 3.8.1, and Section 6.2.6).

3.6 Demonstrate that operation of the hotwell level control system meet design requirements.

3.7 Verify that operation of designated components, such as protective devices, controls, interlocks, instrumentation, and alarms using actual or simulated inputs function as designed.

3.8 Operate control valves remotely while:

a. Observing each valve operation and position indication.

b. Measuring valve performance data (e.g., thrust, opening and closing times).

3.9 ~~Verify~~Observe response of power-operated valves ~~fail~~ upon loss of motive power ~~as designed~~ (refer to Section 10.4.7 for anticipated response).

3.10 Verify that vibration levels during normal system operation and transients meet design requirements.

14.02-64

4.0 DATA REQUIRED

4.1 Head versus flow performance and pump operating data.

4.2 Valve performance data, where required.

4.3 Valve position indication.

4.4 Position response of valves to loss of motive power.

4.5 Setpoints at which alarms and interlocks occur.

4.6 Setpoints of the hotwell level controls.

4.7 Setpoints of the pumps minimum flow recirculation protection.

5.0 ACCEPTANCE CRITERIA

5.1 The CS operates as designed (refer to ~~described in~~ Section 10.4.7).

5.1.1 CS pumps, including pump seals, perform as designed.

5.1.2 The CS alarms, interlocks, protective devices, and controls (automatic and manual) function as designed.

5.1.3 CS valves perform as designed (e.g., thrust, seat leakage, opening time, closing time, bonnet air in-leakage, failure mode upon loss of motive power).

14.02-64

5.2 Vibration levels meet the requirements described in Section 3.9.2.4.

14.2.12.7.9 Steam Generator Blowdown System (Test #067)

1.0 OBJECTIVE

1.1 To verify the proper operation of the SG blowdown system (SGBS).

2.1 Instrumentation has been calibrated and is functional.

3.0 TEST METHOD

3.1 Record MCR instrumentation steady-state readings as directed by the pre-core HFT controlling document.

4.0 DATA REQUIRED

4.1 Plant conditions at the time instrument readings are recorded.

4.2 Instrument readings.

5.0 ACCEPTANCE CRITERIA

5.1 Similar instrumentation readings shall agree within the accuracy limits of the instrumentation as ~~described in~~designed (refer to Section 7.1.2~~,~~).

14.2.12.13.4 Pre–Core Reactor Internals Vibration Measurements (Test #164)

1.0 OBJECTIVE

1.1 To demonstrate that the reactor internal vibration ~~measurements are~~assessment is performed within design limits.

2.0 PREREQUISITES

14.02-64

2.1 Construction activities have been completed on the reactor vessel.

2.2 The U.S. EPR heavy reflector lower internals have been installed.

2.3 Dummy fuel assemblies have been constructed from ~~actual~~ fuel skeletons with stainless fuel pins or suitable alternate flow restriction devices that have been constructed.

2.4 Dummy fuel assemblies or suitable alternate flow restriction devices have been installed in available core locations.

2.5 If the ITAAC requires vibration data from inside the reactor, install a special instrumented dummy fuel assembly with data cables routed through one of the instrument ports in the reactor head.

3.0 TEST METHOD

3.1 Operate the reactor normally and record operating data during HFT.

3.2 Remove the upper internals, dummy fuel assemblies, and lower internals and place in storage location.

3.3 Inspect upper and lower internals, paying special attention to contact surfaces between internals and reactor vessel or upper and lower internals using the inspection guidance specified in Section 3.9.2.3 and Section 3.9.2.4.

14.02-64

4.0 DATA REQUIRED

4.1 Plant conditions.

4.2 Clearances at the upper to lower internals interfaces.

4.3 Clearances at the upper and lower internals interface with the reactor vessel.

4.4 Record of observed wear marks.

5.0 ACCEPTANCE CRITERIA

14.02-64

5.1 Observed vibration and wear scars are within design limits described in Section 3.9.2.3 and Section 3.9.2.4.

14.2.12.13.5 Pre–Core Reactor Coolant System Expansion Measurements (Test #165)

1.0 OBJECTIVE

1.1 To demonstrate that RCS components are free to expand thermally as-designed during initial plant heatup and return to their baseline cold position after the initial cooldown to ambient temperatures.

2.0 PREREQUISITES

2.1 All construction activities have been completed on the RCS components.

2.2 Initial ambient dimensions have been set on the SG and RCP hydraulic snubbers, upper and lower SG and reactor vessels keys, and RC pump columns.

2.3 Initial ambient dimensions for the SG, reactor vessel and RCP supports have been recorded.

3.0 TEST METHOD

3.1 Check clearances at hydraulic snubber joints, keys and column clevises at 50°F increments during heatup and recorded at least 100°F increments.

3.2 Record SG, reactor vessel and RCP clearances at stabilized HZP (pressure and temperature) conditions.

4.0 DATA REQUIRED

4.1 Plant conditions.

4.2 Clearances at the SG sliding base keys, hydraulic snubber joints, upper keys, and piston setting at hydraulic snubbers.

4.3 Clearance between the reactor vessel upper and lower supports and expansion plates.

4.4 Reactor vessel support temperature.