CHAPTER 5

REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

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5.1 SUMMARY DESCRIPTION

This section of the referenced DCD is incorporated by reference with no departures or supplements.

5.2 INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

5.2.1.1 Compliance with 10 CFR 50.55a

Add the following text after the second sentence the second paragraph of DCD Subsection 5.2.1.1.

STD COL 5.2-1 If a later Code year/addenda than the Design Certification Code year/addenda is used by the material and/or component supplier, then a code reconciliation is performed. The reconciliation is performed using the methodology set forth in ASME Section XI for the repair and replacement of components. The later Code year/addenda must be authorized in 10 CFR 50.55a or in a specific authorization as provided in 50.55a(a)(3). Similarly, if Code Cases other than those included in DCD Table 5.2-3 are used, a similar review and reconciliation is performed.

5.2.3.2.1 Chemistry of Reactor Coolant

Add the following text to the end of DCD Subsection 5.2.3.2.1.

STD SUP 5.2-1 Monitoring of water chemistry is implemented using the guidance of EPRI TR-1002884 “Pressurized Water Reactor Primary Water Chemistry Guidelines: Volume 1” (Reference 201), Appendix F.

5.2.4 INSERVICE INSPECTION AND TESTING OF CLASS 1 COMPONENTS

Add the following after the first paragraph in DCD Subsection 5.2.4:

STD COL 5.2-2 The initial inservice inspection program incorporates the latest edition and addenda of the ASME Boiler and Pressure Vessel Code approved in 10 CFR 50.55a(b) on the date 12 months before initial fuel load. Inservice examination of components and system pressure tests conducted during successive 120-month inspection intervals must comply with the requirements of the latest edition and addenda of the Code incorporated by reference in 10 CFR 50.55a(b) 12 months before the start of the 120-month inspection interval (or the optional ASME Code cases listed in NRC Regulatory Guide 1.147, that are incorporated by reference in 10 CFR 50.55a(b)), subject to the limitations and modifications listed in 10 CFR 50.55a(b).

5.2.4.1 System Boundary Subject to Inspection

Add the following at the end of DCD Subsection 5.2.4.1:

STD COL 5.2-2 The Class 1 system boundary for both preservice and inservice inspection programs and the system pressure test program includes those items within the Class 1 and Quality Group A (Equipment Class A per DCD Subsection 3.2.2 and DCD Table 3.2-3) boundary. Based on 10 CFR Part 50 and Regulatory Guide 1.26, the Class 1 boundary includes the following:

- Reactor pressure vessel;
- Portions of the Reactor System (RXS);
- Portions of the Chemical and Volume Control System (CVS);
- Portions of the Incore Instrumentation System (IIS);
- Portions of the Passive Core Cooling System (PXS);
- Portions of the Reactor Coolant System (RCS); and
- Portions of the Normal Residual Heat Removal System (RNS).

Those portions of the above systems within the Class 1 boundary are those items that are part of the reactor coolant pressure boundary as defined in Section 5.2.

Exclusions

Portions of the systems within the reactor coolant pressure boundary (RCPB), as defined above, that are excluded from the Class 1 boundary in accordance with 10 CFR Part 50, Section 50.55a, are as follows:

- Those components where, in the event of postulated failure of the component during normal reactor operation, the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system only; or
- Components that are or can be isolated from the reactor coolant system by two valves in series (both closed, both open, or one closed and the

other open). Each open valve is capable of automatic actuation and, assuming the other valve is open, its closure time is such that, in the event of postulated failure of the component during normal reactor operation, each valve remains operable and the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system only.

The description of portions of systems excluded from the RCPB does not address Class 1 components exempt from inservice examinations under ASME Code Section XI rules. The Class 1 components exempt from inservice examinations are defined by ASME Section XI, IWB-1220, except as modified by 10 CFR 50.55a.

NRC First Revised Order EA-03-009, "Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors," established required inspections of reactor pressure vessel heads and associated penetration nozzles at PWRs in response to primary water stress corrosion cracking. A calculation to determine the susceptibility of the reactor pressure vessel head to primary water stress corrosion cracking is completed for the end of each operating cycle. The calculated value and previous inspection findings are used to determine the appropriate susceptibility category and the associated inspection criteria and inspection frequency for the reactor pressure vessel head and penetration nozzles during each refueling outage. ASME Code Case N-729-1, “Alternative Examination Requirements for Pressurized-Water Reactor (PWR) Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds, “ provides alternative examination requirements in lieu of the requirements in the order. ASME Code Case N-729-1 as modified by the NRC Staff Position on the use of ASME Code Case N-729-1 may be used to perform the inspection the AP1000 reactor pressure vessel head. The staff position on this code case is included as an attachment to NRC letter from John A. Globe, NRC to James H. Riley, NEI dated August 9, 2006. A report is submitted to the NRC detailing the inspection results within 60 days after returning the plant to operation. In addition to the required inspections resulting from the susceptibility category, a visual inspection to identify potential boric acid leaks from pressure-retaining components above the reactor pressure vessel head is performed each refueling outage.

Boric acid corrosion control procedures require inspection of the reactor coolant pressure boundary subject to leakage that can cause boric acid corrosion of the reactor coolant pressure boundary materials. The procedures determine the principal locations where leaks can cause degradation of the primary pressure boundary by boric acid corrosion. Potential paths of the leaking coolant are established. The boric acid corrosion control procedures also contain methods for conducting examinations and performing engineering evaluations to establish the impact on the reactor coolant pressure boundary when leakage is located.

The boric acid corrosion control procedures consist of:

1. Visual inspections of component surfaces that are potentially exposed to borated water leakage.
2. Discovery of leak path and removal of boric acid residue.
3. Assessment of the corrosion.
4. Follow-up inspection for adequacy of corrective actions, as appropriate.

Add the following text at the end of DCD Subsection 5.2.4.1:

STD SUP 5.2-2 The inservice inspection program, along with the boric acid corrosion control procedures, provides guidance for inspecting the integrity of bolting and threaded fasteners.

5.2.4.3 Examination Techniques and Procedures

Add the following at the end of DCD Subsection 5.2.4.3:

5.2.4.3.1 Examination Methods

Ultrasonic Examination of the Reactor Vessel

STD COL 5.2-2 Ultrasonic examination for the RPV is conducted in accordance with the ASME Code, Section XI. The design of the RPV considered the requirements of the ASME Code Section XI with regard to performance of preservice inspection. For the required preservice examinations, the reactor vessel meets the acceptance standards of Section XI, IWB-3510. The RPV shell welds are designed for 100% accessibility for both preservice and inservice inspection. RPV shell welds may be examined from the inside or outside diameter surfaces (or a combination of those techniques) using automated ultrasonic examination equipment. The RPV nozzle-to-shell welds are 100% accessible for preservice inspection but might have limited areas that may not be accessible from the outer surface for inservice examination techniques. If accessibility is limited, an inservice inspection program relief request is prepared and submitted for review approval by the NRC.

Inner radius examinations are performed from the outside of the nozzle using several compound angle transducer wedges to obtain complete coverage of the required examination volume. Alternatively, nozzle inner radius examinations may be performed using enhanced visual techniques, as allowed by 10 CFR 50.55a(b)(2)(xxi).

Visual Examination

Visual examination methods VT-1, VT-2 and VT-3 are conducted in accordance with ASME Section XI, IWA-2210. In addition, VT-2 examinations meet the requirements of IWA-5240.

Where direct visual VT-1 examinations are conducted without the use of mirrors or with other viewing aids, clearance is provided where feasible for the head and shoulders of a man within a working arm's length of the surface to be examined.

Surface Examination

Magnetic particle and liquid penetrant examination techniques are performed in accordance with ASME Section XI, IWA-2221 and IWA-2222, respectively. Direct examination access for magnetic particle (MT) and liquid penetrant (PT) examination is the same as that required for direct visual (VT-1) examination (see Visual Examination), except that additional access is provided as necessary to enable physical contact with the item in order to perform the examination. Remote MT and PT generally are not appropriate as a standard examination process; however, boroscopes and mirrors can be used at close range to improve the angle of vision.

Volumetric Ultrasonic Direct Examination

Volumetric ultrasonic direct examination is performed in accordance with ASME Section XI, IWA-2232, which references mandatory Appendix I.

Alternative Examination Techniques

As provided by ASME Section XI, IWA-2240, alternative examination methods, a combination of methods, or newly developed techniques may be substituted for the methods specified for a given item in this section, provided that they are demonstrated to be equivalent or superior to the specified method. This provision allows for the use of newly developed examination methods, techniques, etc., which may result in improvements in examination reliability and reductions in personnel exposure. In accordance with 10 CFR 50.55a(b)(2)(xix), IWA-2240 as written in the 1997 Addenda of ASME Section XI must be used when applying these provisions.

5.2.4.3.2 Qualification of Personnel and Examination Systems for Ultrasonic Examination

Personnel performing examinations shall be qualified in accordance with ASME Section XI, Appendix VII. Qualification to ASME Section XI, Appendix VIII, is in compliance with the provisions of 10 CFR 50.55a.

5.2.4.4 Inspection Intervals

Add the following after the second sentence of the first paragraph of DCD Subsection 5.2.4.4:

STD COL 5.2-2 Because 10 CFR 50.55a(g)(4) requires 120-month inspection intervals, Inspection Program B of IWB-2400 must be chosen. The inspection interval is divided into three periods. Period one comprises the first three years of the interval, period two comprises the next four years of the interval, and period three comprises the remaining three years of the inspection interval. Each period can be extended for up to one year to enable an inspection to coincide with a plant outage. The adjustment of period end dates shall not alter the rules and requirements for scheduling inspection intervals.

5.2.4.5 Examination Categories and Requirements

Add the following after the first sentence of DCD Subsection 5.2.4.5 :

STD COL 5.2-2 Class 1 piping supports will be examined in accordance with ASME Section XI, IWF-2500.

Preservice examinations required by design specification and preservice documentation are in accordance with ASME Section III, NB-5280. Components exempt from preservice examination are described in ASME Section III, NB-5283.

Add the following after the last sentence of DCD Subsection 5.2.4.5:

The preservice examination is performed once in accordance with ASME XI, IWB-2200, on all of the items selected for inservice examination, with the exception of the examinations specifically excluded by ASME Section XI from preservice requirements, such as VT-3 examination of valve body and pump casing internal surfaces (B-L-2 and B-M-2 examination categories, respectively) and the visual VT-2 examinations for category B-P.

5.2.4.6 Evaluation of Examination Results

Add the following at the end of DCD Subsection 5.2.4.6:

STD COL 5.2-2 Components containing flaws or relevant conditions and accepted for continued service in accordance with the requirements of IWB-3132.4 or IWB-3142.4 are subjected to successive period examinations in accordance with the requirements of IWB-2420. Examinations that reveal flaws or relevant conditions exceeding Table IWB-3410-1 acceptance standards are extended to include additional examinations in accordance with the requirements of IWB-2430.

STD COL 5.2-2 Add Subsections 5.2.4.8, 5.2.4.9, and 5.2.4.10 after the last paragraph of DCD Subsection 5.2.4.7:

5.2.4.8 Relief Requests

The specific areas where the applicable ASME Code requirements cannot be met are identified after the initial examinations are performed. Should relief requests be required, they will be developed through the regulatory process and submitted to the NRC for approval in accordance with 10 CFR 50.55a(a)(3) or 50.55a(g)(5). The relief requests include appropriate justifications and proposed alternative inspection methods.

5.2.4.9 Preservice Inspection of Class 1 Components

Preservice examinations required by design specification and preservice documentation are in accordance with ASME Section III, NB-5281. Volumetric and surface examinations are performed as specified in ASME Section III, NB-5282. Components described in ASME Section III, NB-5283 are exempt from preservice examination.

5.2.4.10 Program Implementation

The milestones for preservice and inservice inspection program implementation are identified in Table 13.4-201.

5.2.6 COMBINED LICENSE INFORMATION ITEMS

5.2.6.1 ASME Code and Addenda

STD COL 5.2-1 This COL Item is addressed in Subsection 5.2.1.1.

5.2.6.2 Plant-Specific Inspection Program

STD COL 5.2-2 This COL Item is addressed in Subsections 5.2.4, 5.2.4.1, 5.2.4.3.1, 5.2.4.3.2, 5.2.4.4, 5.2.4.5, 5.2.4.6, 5.2.4.8, 5.2.4.9, and 5.2.4.10.

5.2.7 REFERENCES

Add the following information at the end of DCD Subsection 5.2.7.

201. EPRI, "Pressurized Water Reactor Primary Water Chemistry Guidelines: Volume 1," EPRI TR-1002884, Revision 5, October 2003.

5.3 REACTOR VESSEL

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

5.3.2.6 Material Surveillance

Add the following information between the first and second paragraphs of DCD Subsection 5.3.2.6.

STD COL 5.3-2 Reactor materials do not begin to be affected by neutron fluence until the reactor begins critical operation. Table 13.4-201 provides milestones for reactor vessel material surveillance program implementation.

Add the following subsection after DCD Subsection 5.3.2.6.2.2.

5.3.2.6.3 Report of Test Results

STD COL 5.3-2 A summary technical report for each capsule withdrawn with the test results is submitted, as specified in 10 CFR 50.4, within one year of the date of capsule withdrawal unless an extension is granted by the Director, Office of Nuclear Reactor Regulation.

The report includes the data required by ASTM E185-82, as specified in paragraph III.B.1 of 10 CFR Part 50, Appendix H, and includes the results of the fracture toughness tests conducted on the beltline materials in the irradiated and unirradiated conditions.

If the test results indicate a change in the Technical Specifications is required, either in the pressure-temperature limits or in the operating procedures required to meet the limits, the expected date for submittal of the revised Technical Specification is provided with the report.

Add the following subsection after DCD Subsection 5.3.3.1.

5.3.3.2 Operating Procedures

STD SUP 5.3-1 Plant operating procedures are developed and maintained to prevent exceeding the pressure-temperature limits identified in reactor coolant system pressure and temperature limits report, as required by Technical Specification 5.6.6, during normal and abnormal operating conditions and system tests.

5.3.6 COMBINED LICENSE INFORMATION

5.3.6.1 Pressure-Temperature Limit Curves

Replace the text in DCD Subsection 5.3.6.1 with the following.

STD COL 5.3-1 The pressure-temperature curves shown in DCD Figures 5.3-2 and 5.3-3 are generic curves for AP1000 reactor vessel design, and they are the limiting curves based on copper and nickel material composition. Plant-specific curves will be developed based on material composition of copper and nickel. Use of plant-specific curves will be addressed during procurement and fabrication of the reactor vessel. As noted in the bases to Technical Specification 3.4.14, use of plant-specific curves requires evaluation of the LTOP system. This includes an evaluation of the setpoint pressure for the RNS relief valve to determine if the setpoint pressure needs to be changed based on the plant-specific pressure-temperature curves. The development of the plant-specific curves and evaluation of the setpoint pressure are required prior to fuel load.

5.3.6.2 Reactor Vessel Materials Surveillance Program

STD COL 5.3-2 This COL Item is addressed in Subsection 5.3.2.6.

5.3.6.4 Reactor Vessel Materials Properties Verification

Replace the text in DCD Subsection 5.3.6.4.1 with the following.

5.3.6.4.1 Reactor Vessel Materials Properties Verification

STD COL 5.3-4 The verification of plant-specific belt line material properties consistent with the requirements in DCD Subsection 5.3.3.1 and DCD Tables 5.3-1 and 5.3-3 will be completed prior to fuel load. The verification will include a pressurized thermal shock evaluation based on as procured reactor vessel material data and the projected neutron fluence for the plant design objective of 60 years. This evaluation report will be submitted for NRC staff review.

5.4 COMPONENT AND SUBSYSTEM DESIGN

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

5.4.2.5 Steam Generator Inservice Inspection

Add the following information at the end of DCD Subsection 5.4.2.5.

STD COL 5.4-1 A steam generator tube surveillance program is implemented in accordance with the recommendations and guidance of Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines" (Reference 201). A program for periodic monitoring of degradation of steam generator internals is also implemented in accordance with NEI 97-06. Applicable Electric Power Research Institute (EPRI) Steam Generator Management Program (SGMP) guidelines are followed as described in the NEI 97-06. The Programs are in compliance with applicable sections of ASME Section XI.

NEI 97-06 and the referenced EPRI SGMP guidelines provide recommendations concerning the inspection of tubes, which cover inspection equipment, baseline inspections, tube selection, sampling and frequency of inspection, methods of recording, required actions based on findings, and tube plugging. The minimum requirements for inservice inspection of steam generators, including plugging criteria, are established in Technical Specification 5.5.4.

The tube surveillance and degradation monitoring programs include provisions to maintain the compatibility of steam generator tubing with primary and secondary coolant to limit the steam generators' susceptibility to corrosion. These provisions are in accordance with NEI 97-06.

5.4.15 COMBINED LICENSE INFORMATION ITEMS

STD COL 5.4-1 This COL Item is addressed in Subsection 5.4.2.5.

5.4.16 REFERENCES

Insert the following information at the end of DCD Subsection 5.4.16.

201. Nuclear Energy Institute, "Steam Generator Program Guidelines," NEI 97-06, Revision 2, May 2005.
