Chapter 5 Reactor Coolant System and Connected Systems

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 Compliance with Codes and Code Cases

5.2.1.1 Compliance with 10 CFR 50.55a

Add the following at the end of this section.

STD SUP 5.2-2

As described in Subsection 5.2.4, preservice and inservice inspection of the reactor coolant pressure boundary is conducted in accordance with the applicable edition and addenda of the ASME Boiler and Pressure Vessel Code, Section XI, required by 10 CFR 50.55a. As described in DCD Section 3.9.6 for pumps and valves, and in DCD Section 3.9.3.7.1 for dynamic restraints, preservice and inservice testing of the reactor coolant pressure boundary components is in accordance with the edition and addenda of the ASME OM Code required by 10 CFR 50.55a.

5.2.4 Preservice and Inservice Inspection and Testing of Reactor Coolant Pressure Boundary

Replace the second sentence in the second paragraph with the following.

STD COL 5.2-1-A

All Class 1 austenitic or dissimilar metal welds are included in the referenced certified design.

Replace the second sentence and subsequent parenthetical sentence in the fourth paragraph with the following.

STD COL 5.2-1-A

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.

5.2.4.2 Accessibility

Replace the last sentence in the second paragraph with the following.

STD COL 5.2-3-A

During the construction phase of the project, anomalies and construction issues are addressed using change control procedures. Procedures require that changes to approved design documents, including field changes and modifications, are subject to the same review and approval process as the original design. Accessibility and inspectability are key components of the design process. Control of accessibility for inspectability and testing during licensee design activities affecting Class I components is provided via procedures for design control and plant modifications.

Ultrasonic techniques (UT) will be the preferred NDE method for all PSI and ISI volumetric examinations; radiographic techniques (RT) will be used as a last resort only if UT cannot achieve the necessary coverage. The same NDE method used during PSI will be used for ISI to the extent possible to assure a baseline point of reference. If a different NDE method is used for ISI than was used for PSI, equivalent coverage will be achieved as required by code.

5.2.4.3.4 Qualification of Personnel and Examination Systems for Ultrasonic Examination

Add the following at the end of the paragraph.

STD COL 5.2-1-A

Certification of NDE personnel shall be in accordance with ASME Section XI, IWA-2300, as modified by 10 CFR 50.55a(b)(2)(xviii).

5.2.4.6 System Leakage and Hydrostatic Pressure Tests

Revise the second sentence of the first paragraph as follows.

STD COL 5.2-1-A

Regardless of which test method is chosen, system leakage and hydrostatic pressure tests will meet all requirements of ASME Code Section XI, IWA-5000 and IWB-5000 for Class I components, including the limitation of 10 CFR 50.55a(b)(2)(xxvi).

Add the following paragraph at the end of this section.

STD SUP 5.2-1

System pressure tests and correlated technical specification requirements are provided in the plant Technical Specifications 3.4.4, “RCS Pressure and Temperature (P/T) Limits,” and 3.10.1, “Inservice Leak and Hydrostatic Testing Operation.”

5.2.4.11 COL Information for Preservice and Inservice Inspection and Testing of Reactor Coolant Pressure Boundary

Replace the first sentence of the first paragraph with the following and delete the last sentence.

STD COL 5.2-1-A

DCD Section 5.2.4 fully describes the Preservice and Inservice Inspection and Testing Programs for the RCPB. The implementation milestones for the Preservice and Inservice Inspection and Testing Programs are provided in Section 13.4.

5.2.5 Reactor Coolant Pressure Boundary Leakage Detection

STD COL 5.2-2-A

Delete the parenthetical statement in the first sentence of the first paragraph.

Replace DCD Section 5.2.5.9 with the following.

STD COL 5.2-2-A

5.2.5.9 Leak Detection Monitoring

Operators are provided with procedures for detecting, monitoring, recording, trending, and determining the sources of reactor coolant pressure boundary leakage. Examples of parameters that are monitored are sump pump run time, sump level, condensate transfer rate, and process chemistry/radioactivity.

The procedures are used for converting different parameter indications for identified and unidentified leakage into common leak rate equivalents (volumetric or mass flow) and leak rate rate-of-change values, including indications from: 1) the drywell floor drain high conductivity water sump monitoring system, 2) the drywell air coolers condensate flow monitoring system, and 3) the drywell fission product monitoring system.

The procedures are used to monitor leakage at levels well below Technical Specifications limits and provide guidance for evaluating potential corrective action plans to prevent the plant from exceeding a Technical Specifications limit.

An unidentified leakage rate-of-change alarm provides an early alert to the operators to initiate corrective actions prior to reaching a Technical Specifications limit.

A description of the plant procedures program and implementation milestones are provided in Section 13.5.

5.2.6 COL Information

5.2-1-A Preservice and Inservice Inspection Program Description

STD COL 5.2-1-A This COL Item is addressed in Subsections 5.2.4, 5.2.4.3.4, 5.2.4.6, 5.2.4.11, and 6.6.

5.2-2-A Leak Detection Monitoring

STD COL 5.2-2-A This COL Item is addressed in Subsections 5.2.5 and 5.2.5.9.

5.2-3-A Preservice and Inservice Inspection NDE Accessibility Plan Description

STD COL 5.2-3-A This COL Item is addressed in Subsection 5.2.4.2.

5.3 Reactor Vessel

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

5.3.1.5 Fracture Toughness Compliance with 10 CFR 50, Appendix G

Replace the last sentence in the first paragraph with the following.

STD COL 16.0-1-A 5.6.4-1 The pressure-temperature limit curves are developed in accordance with the Pressure and Temperature Limits Report, as discussed in the Technical Specifications Section 5.6.4 **[START COM 5.03-002]** Prior to fuel load, the pressure-temperature limit curves will be updated to reflect plant-specific material properties, if required. **[END COM 5.03-002]**

5.3.1.8 COL Information for Reactor Vessel Material Surveillance Program

Replace this section with the following.

STD COL 5.3-2-A

The description of the reactor vessel material surveillance program provided in DCD Section 5.3.1.6 is supplemented as follows.

A complete reactor vessel material surveillance program will be developed as described above in accordance with the implementation schedule provided in Section 13.4.

5.3.1.8.1 Locations of Capsules in Core Beltline Region

A total of four irradiation exposure specimen sets containing the required specimens are located near the vessel wall slightly above the core midplane. The irradiation exposure specimen sets are contained in specimen holders that are welded to the inner diameter of the core beltline forging. Each specimen holder houses two specimen containers that form the irradiation exposure set. The elevation and azimuth locations of the exposure specimen sets align with the maximum calculated fluence within the core beltline. Based on the location of the samples relative to the shell forging and their placement at the peak fluence location, the lead factors for the samples will be greater than 1.0. The lead factor for the specimens when placed at the peak location hasbeen estimated to be 1.17.

5.3.1.8.2 Preparation of Capsule Specimens

As stated in DCD Section 5.3.1.6.1, the reactor vessel materials specimens are provided in accordance with the requirements of ASTM E 185 and 10 CFR 50, Appendix H. The surveillance specimen materials are prepared from full thickness samples taken from the actual core beltline forging and from the adjacent forgings and weld materials. The materials include the base metal and weld metal that have the highest adjusted reference temperature at end-of-life. The fabrication or heat treatment history (austenitizing, quench and tempering, and post-weld heat treatment) of the test material is fully representative of the fabrication history of the materials in the beltline of the RPV. The base metal sample blocks from which the specimens are taken are located at least one “T” from any quenched edge of the block, where “T” is the material thickness, and at least 25 mm from a flame cut edge or weld fusion line.

The weld metal sample blocks are fabricated using the same welding procedure and process as the vessel shell weld they represent. The welding materials (electrodes, flux, or gas) are from the same heat and lot as the material used to make the production weld. The welder is qualified to ASME Section IX. The weld must satisfy the same examination and inspection requirements as the production weld. The weld or HAZ samples are taken at least one “T” from any quenched edge of the block, at least 25 mm from a flame cut edge, and at least 13 mm from the root of the weld.

Base Metal Samples

The longitudinal axes of tensile specimens are located 1/4T from the as-quenched vessel surface. The specimens are oriented so that the longitudinal axis is parallel to the forging and normal to the major working direction of the forging.

Charpy V-notch specimens are removed 1/4T from the as-quenched vessel surface. The longitudinal axes of specimens are oriented parallel to the forging surface and normal to the major working direction.

Weld Metal Samples

The longitudinal axes of tensile specimens are located in the approximate center of the weld metal and at least 13 mm from the final weld surface and the root of the weld. The axis is parallel to the plate or forging surface.

The roots of the notch of Charpy V-notch specimens are in the approximate center of the weld metal. The specimens are taken at least 13 mm from the final weld surface and the root of the weld. The notch is perpendicular to the plate or forging surface.

All tensile specimens and Charpy V-notch specimens correspond to the allowable specimen types, as defined in ASTM E 185.

Fracture Toughness Samples

Fracture toughness specimens are provided from the limiting base and weld metals and are consistent with the guidelines in ASTM E 1820 and ASTM E 1921.

5.3.1.8.3 Number and Type of Specimens

The number of specimens in each exposure set satisfies or exceeds the requirements of ASTM E 185. Additional fracture toughness specimens of the limiting materials are included as shown in Table 5.3-201. Four sets of specimens are provided for the 60-year life of the ESBWR. The quantities of specimens per irradiation exposure set are provided in Table 5.3-201.

5.3.1.8.4 Report of Test Results

A summary technical report, including test results, is submitted as specified in 10 CFR 50.4, for the contents of each capsule withdrawn, 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 50, Appendix H, and includes the results of the fracture toughness tests conducted on the beltline materials in the irradiated and unirradiated conditions. **[START COM 5.3.001]** If the test results indicate a change in the Technical Specifications is required, the expected date for submittal of the revised Technical Specification will be provided with the report.**[END COM 5.3.001]**

5.3.3.6 Operating Conditions

Add the following after the first sentence.

STD SUP 5.3-1 Development of plant operating procedures is addressed in Section 13.5. These procedures require compliance with the Technical Specifications. The Technical Specifications (which are developed by the methodology also identified in the Technical Specifications) are intended to ensure that the P-T limits identified in DCD Section 5.3.2 are not exceeded during normal operating conditions and anticipated plant transients.

5.3.4 COL Information

5.3-2-A Materials and Surveillance Capsule

STD COL 5.3-2-A This COL Item is addressed in Subsection 5.3.1.8.

5.4 Component and Subsystem Design

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

5.4.8 Reactor Water Cleanup/Shutdown Cooling System

Add the following paragraph at the end of this section.

STD SUP 5.4-1

Operating procedures provide guidance to prevent severe water hammer caused by mechanisms such as voided lines.

5.4.12 Reactor Coolant System High Point Vents

Add the following paragraph at the end of this section.

STD SUP 5.4-2

A human factors analysis of the control room displays for the RCS vents is included as part of the overall human factors analysis of the control room displays and controls described in DCD Chapter 18. This analysis considers:

- The use of this information by an operator during both normal and abnormal plant conditions;
- Integration into emergency procedures;
- Integration into operator training; and
- Other alarms during an emergency and the need for prioritization of alarms.

5.4.12.1 Operation of RPV Head Vent System

Add the following paragraph at the end of this section.

STD SUP 5.4-3

Operating procedures for the reactor vent system address considerations regarding when venting is needed and when it is not needed, including a variety of initial conditions for which venting may be required. The development of operating procedures is addressed in Section 13.5.

Table 5.3-201 Quantities of Reactor Vessel Material Specimens per Irradiation Exposure Set [STD COL 5.3-2-A]

Material	Specimen Type	No. of Specimen per Irradiation Exposure Set	Comments
Base Metal	Charpy	45	15 samples from each of three forgings in accordance with ASTM E 185-02.
	Tensile	9	3 samples from each of three forgings in accordance with ASTM E 185-02.
	Fracture Toughness	8	Taken from most limiting material in accordance with ASTM E 185-02.
Weld Metal	Charpy	30	15 specimens per weld in accordance with ASTM E 185-02.
	Tensile	6	3 specimens per weld in accordance with ASTM E 185-02.
	Fracture Toughness	8	Taken from most limiting material in accordance with ASTM E 185-02.
HAZ	Charpy	12	In accordance with ASTM E 185-82.