A2. REACTOR VESSEL (PRESSURIZED WATER REACTOR)

A2.1 Closure Head

- A2.1.1 Dome
- A2.1.2 Head Flange
- A2.1.3 Stud Assembly
- A2.1.4 Vessel Flange Leak Detection Line

A2.2 Control Rod Drive (CRD) Head Penetration

- A2.2.1 Nozzle
- A2.2.2 Pressure Housing
- A2.2.3 Flange Bolting

A2.3 Nozzles

- A2.3.1 Inlet
- A2.3.2 Outlet
- A2.3.3 Safety Injection (on some)

A2.4 Nozzle Safe Ends

- A2.4.1 Inlet
- A2.4.2 Outlet
- A2.4.3 Safety Injection (on some)

A2.5 Shell

- A2.5.1 Upper (Nozzle) Shell
- A2.5.2 Intermediate and Lower Shell
- A2.5.3 Vessel Flange
- A2.5.4 Bottom Head

A2.6 Core Support Pads/Core Guide Lugs

A2.7 Penetrations

- A2.7.1 Instrument Tubes (Bottom Head)
- A2.7.2 Head Vent Pipe (Top Head)
- A2.7.3 Instrument Tubes (Top Head)

A2.8 Pressure Vessel Support

- A2.8.1 Skirt Support
- A2.8.2 Cantilever/Column Support
- A2.8.3 Neutron Shield Tank

A2. REACTOR VESSEL (PRESSURIZED WATER REACTOR)

Systems, Structures, and Components

This section comprises the pressurized water reactor (PWR) vessel pressure boundary and consists of the vessel shell and flanges, the top closure head and bottom head, the control rod drive (CRD) mechanism housings, nozzles (including safe ends) for reactor coolant inlet and outlet lines and safety injection, and penetrations through either the closure head or bottom head domes for instrumentation and leakage monitoring tubes. Attachments to the vessel such as core support pads, as well as pressure vessel support and attachment welds, are also included in the table. Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all systems, structures, and components that comprise the reactor coolant system are governed by Group A Quality Standards.

System Interfaces

The systems that interface with the PWR reactor vessel include the reactor vessel internals (IV.B2, IV.B3, and IV.B4, respectively, for Westinghouse, Combustion Engineering, and Babcox and Wilcox designs), the reactor coolant system and connected lines (IV.C2), and the emergency core cooling system (V.D1).

IV Reactor Vessel, Internals, and Reactor Coolant System
A2. Reactor Vessel (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
A2.1-a A2.1.1 A2.1.2 A2.1.3	Closure head Dome Head flange Stud assembly (external surfaces)	Dome and flange: SA302-Gr B, SA533-Gr B; stud assembly: SA540-Gr. B23/24, SA320-Gr. L43 (alloy 4340)	Air, leaking chemically treated borated water or steam up to 340°C (644°F)	Loss of material/ Boric acid corrosion of external surfaces	Chapter XI.M10, "Boric Acid Corrosion"	No
A2.1-b A2.1.1	Closure head Dome	SA302-Gr B, SA533-Gr B, SA508-64 class 2 with stainless steel cladding	Chemically treated borated water or steam up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii), and for addressing environmental effects on fatigue. See Chapter X.M1 of this report for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes, TLAA

IV Reactor Vessel, Internals, and Reactor Coolant System
A2. Reactor Vessel (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
A2.1-c A2.1.3	Closure head Stud assembly	SA540-Gr. B23/ 24, SA320-Gr. L43 (alloy 4340), SA193-6 maximum tensile strength <1172 MPa (<170 Ksi)	Air, leaking chemically treated borated water or steam up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking	Chapter XI.M3, "Reactor Head Closure Studs"	No
A2.1-d A2.1.3	Closure head Stud assembly	SA540-Gr. B23/ 24, SA320-Gr. L43 (alloy 4340), SA193-6 maximum tensile strength <1172 MPa (<170 Ksi)	Air, leaking chemically treated borated water or steam up to 340°C (644°F)	Loss of material/ Wear	Chapter XI.M3, "Reactor Head Closure Studs"	No

IV Reactor Vessel, Internals, and Reactor Coolant System
A2. Reactor Vessel (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
A2.1-e A2.1.3	Closure head Stud assembly	SA540-B23 and B24, SA320-L43, SA193-6	Air, leaking chemically treated borated water or steam up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes TLAA
A2.1-f A2.1.4	Closure head Vessel flange leak detection line	Stainless steel	Leaking chemically treated borated water or steam up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking	A plant-specific aging management program is to be evaluated because existing programs may not be capable of mitigating or detecting crack initiation and growth due to SCC in the vessel flange leak detection line.	Yes, plant specific
A2.2-a A2.2.1	Control rod drive head penetration Nozzle	SB-166, SB-167 (alloy 600)	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Primary water stress corrosion cracking	Chapter XI.M11, "Ni-alloy Nozzles and Penetrations," and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
A2.2-b A2.2.2	Control rod drive head penetration Pressure housing	Type 403 and 316 stainless steel; type 304 stainless steel or cast austenitic stainless steel CF-8; SA 508 class 2 with alloy 82/182 cladding	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
A2.2-c A2.2.1 A2.2.2	Control rod drive head penetration Nozzle Pressure housing	Type 403 and 316 stainless steel; type 304 stainless steel or cast austenitic stainless steel CF-8	Chemically treated borated water up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii), and for addressing environmental effects on fatigue. See Chapter X.M1 of this report for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes, TLAA
A2.2-d A2.2.2	Control rod drive head penetration Pressure housing	Cast austenitic stainless steel CF-8	Chemically treated borated water up to 340°C (644°F)	Loss of fracture toughness/ Thermal aging embrittlement	Chapter XI.M12 "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)"	No
A2.2-e A2.2.3	Control rod drive head penetration Flange bolting	Stainless steel (SA 453)	Air, leaking chemically treated borated water or steam up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking	Chapter XI.M18, "Bolting Integrity"	No
A2.2-f A2.2.3	Control rod drive head penetration Flange bolting	Stainless steel (SA 453)	Air with metal temperature up to 340°C (644°F)	Loss of material/ Wear	Chapter XI.M18, "Bolting Integrity"	No
A2.2-g A2.2.3	Control rod drive head penetration Flange bolting	Stainless steel (SA 453)	Air with metal temperature up to 340°C (644°F)	Loss of preload/ Stress relaxation	Chapter XI.M18, "Bolting Integrity"	No

IV Reactor Vessel, Internals, and Reactor Coolant System
A2. Reactor Vessel (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
A2.3-a A2.3.1 A2.3.2 A2.3.3	Nozzles Inlet Outlet Safety injection	SA336, SA508 with stainless steel cladding	Chemically treated borated water up to 340°C (644°F) neutron fluence greater than 10^{17} n/cm^2 (E >1 MeV)	Loss of fracture toughness/ Neutron irradiation embrittlement	Neutron irradiation embrittlement is a time-limited aging analysis (TLAA) to be evaluated for the period of license renewal for all ferritic materials that have a neutron fluence greater than 10^{17} n/cm^2 (E >1 MeV) at the end of the license renewal term. The TLAA is to evaluate the impact of neutron embrittlement on: (a) the RT_{PTS} value based on the requirements in 10 CFR 50.61, (b) the adjusted reference temperature, the plant's pressure-temperature limits, (c) the Charpy upper shelf energy, and (d) the equivalent margins analyses performed in accordance with 10 CFR 50, Appendix G. The applicant may choose to demonstrate that the materials in the inlet, outlet, and safety injection nozzles are not controlling for the TLAA evaluations.	Yes, TLAA
A2.3-b A2.3.1 A2.3.2 A2.3.3	Nozzles Inlet Outlet Safety injection	SA336, SA508 with stainless steel cladding	Chemically treated borated water up to 340°C (644°F) neutron fluence greater than 10^{17} n/cm^2 (E >1 MeV)	Loss of fracture toughness/ Neutron irradiation embrittlement	Chapter XI.M31, "Reactor Vessel Surveillance"	Yes, plant specific

IV Reactor Vessel, Internals, and Reactor Coolant System
A2. Reactor Vessel (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
A2.3-c A2.3.1 A2.3.2 A2.3.3	Nozzles Inlet Outlet Safety injection	SA336, SA508 with stainless steel cladding	Chemically treated borated water up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii), and for addressing environmental effects on fatigue. See Chapter X.M1 of this report for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes, TLAA
A2.4-a A2.4.1 A2.4.2 A2.4.3	Nozzle safe ends Inlet Outlet Safety injection	Stainless steel, cast austenitic stainless steel (NiCrFe buttering, and stainless steel or NiCrFe weld)	Chemically treated borated water up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii), and for addressing environmental effects on fatigue. See Chapter X.M1 of this report for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes, TLAA

IV Reactor Vessel, Internals, and Reactor Coolant System
A2. Reactor Vessel (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
A2.4-b A2.4.1 A2.4.2 A2.4.3	Nozzle safe ends Inlet Outlet Safety injection	Stainless steel, cast austenitic stainless steel (NiCrFe buttering, and stainless steel or NiCrFe weld)	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking, primary water stress corrosion cracking	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
A2.5-a A2.5.1 A2.5.2	Vessel shell Upper shell Intermediate and lower shell (including beltline welds)	SA302-Gr B, SA533-Gr B, SA336, SA508-Cl 2 or Cl 3 with type 308 or 309 cladding	Chemically treated borated water up to 340°C (644°F) neutron fluence greater than 10^{17} n/cm^2 (E >1 MeV)	Loss of fracture toughness/ Neutron irradiation embrittlement	Neutron irradiation embrittlement is a time-limited aging analysis (TLAA) to be evaluated for the period of license renewal for all ferritic materials that have a neutron fluence of greater than 10^{17} n/cm^2 (E >1 MeV) at the end of the license renewal term. The TLAA is to evaluate the impact of neutron embrittlement on: (a) the RT_{PTS} value based on the requirements in 10 CFR 50.61, (b) the adjusted reference temperature, the plant's pressure temperature limits, (c) the Charpy upper shelf energy, and (d) the equivalent margins analyses performed in accordance with 10 CFR 50, Appendix G. See the Standard Review Plan, Section 4.2 "Reactor Vessel Neutron Embrittlement" for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes TLAA

IV **Reactor Vessel, Internals, and Reactor Coolant System**
A2. Reactor Vessel (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
A2.5-b A2.5.1 A2.5.2	Vessel shell Upper shell Intermediate and lower shell (including beltline welds)	SA508-Cl 2 forgings clad using a high-heat-input welding process	Chemically treated borated water up to 340°C (644°F) neutron fluence greater than 10^{17} n/cm^2 (E >1 MeV)	Crack growth/ Cyclic loading	Growth of intergranular separations (underclad cracks) in low-alloy steel forging heat affected zone under austenitic stainless steel cladding is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation for all the SA 508-Cl 2 forgings where the cladding was deposited with a high heat input welding process. The methodology for evaluating an underclad flaw is in accordance with the current well-established flaw evaluation procedure and criterion in the ASME Section XI Code. See the Standard Review Plan, Section 4.7, "Other Plant-Specific Time-Limited Aging Analysis," for generic guidance for meeting the requirements of 10 CFR 54.21(c).	Yes TLAA
A2.5-c A2.5.1 A2.5.2	Vessel shell Upper shell, Intermediate and lower shell (including beltline welds)	SA302-Gr B, SA533-Gr B, SA336, SA508-Cl 2 or Cl 3 with type 308 or 309 cladding	Chemically treated borated water up to 340°C (644°F) neutron fluence greater than 10^{17} n/cm^2 (E >1 MeV)	Loss of fracture toughness/ neutron irradiation embrittlement	Chapter XI.M31, "Reactor Vessel Surveillance"	Yes, plant specific

IV **Reactor Vessel, Internals, and Reactor Coolant System**
A2. Reactor Vessel (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
A2.5-d A2.5.1 A2.5.2 A2.5.3 A2.5.4	Vessel shell Upper (nozzle) shell Intermediate and lower shell Vessel flange Bottom head	SA302-Gr B, SA533-Gr B, SA336, SA508 with stainless-steel cladding	Chemically treated borated water up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii), and for addressing environmental effects on fatigue. See Chapter X.M1 of this report for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes, TLAA
A2.5-e A2.5.3	Vessel shell Vessel flange (external surface)	SA336, SA508	Air, leaking chemically treated borated water or steam up to 340°C (644°F)	Loss of material/ Boric acid corrosion of external surfaces	Chapter XI.M10, "Boric Acid Corrosion"	No
A2.5-f A2.5.3	Vessel shell Vessel flange	SA336, SA508	Chemically treated borated water or steam up to 340°C (644°F)	Loss of material/ Wear	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components	No
A2.6-a	Core support pads/core guide lugs	SB-166, SB-168, (alloy 600)	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Primary water stress corrosion cracking	A plant-specific aging management program is to be evaluated. The applicant is to provide a plant-specific AMP or participate in industry programs to determine appropriate AMP.	Yes, plant specific
A2.7-a A2.7.1	Penetrations Instrument tubes (bottom head)	SB-166, SB-167, (alloy 600)	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Primary water stress corrosion cracking	A plant-specific aging management program is to be evaluated. The applicant is to provide a plant-specific AMP or participate in industry programs to determine appropriate AMP.	Yes, plant specific

IV Reactor Vessel, Internals, and Reactor Coolant System

A2. Reactor Vessel (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
A2.7-b A2.7.2 A2.7.3	Penetrations Head vent pipe(top head) Instrument tubes (top head)	SB-166, SB-167, (alloy 600)	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ primary water stress corrosion cracking	Chapter XI.M11, "Ni-alloy Nozzles and Penetrations," and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
A2.8-a A2.8.1	Pressure vessel support Skirt support	SA302-Gr B, SA533-Gr B, SA516-Gr70, SA 36	Air	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
A2.8-b A2.8.1 A2.8.2 A2.8.3	Pressure vessel support Skirt support Cantilever/ column support Neutron shield tank,	SA302-Gr B, SA533-Gr B, SA516-Gr70, SA 36	Air, leaking chemically treated borated water	Loss of material/ Boric acid corrosion of external surfaces	Chapter XI.M10, "Boric Acid Corrosion"	No

B1. REACTOR VESSEL INTERNALS (BOILING WATER REACTOR)

B1.1 Core Shroud and Core Plate

- B1.1.1 Core Shroud (Upper, Central, Lower)
- B1.1.2 Core Plate
- B1.1.3 Core Plate Bolts
- B1.1.4 Access Hole Cover
- B1.1.5 Shroud Support Structure
- B1.1.6 LPCI Coupling

B1.2 Top Guide

B1.3 Core Spray Lines and Spargers

- B1.3.1 Core Spray Lines (Headers)
- B1.3.2 Spray Ring
- B1.3.3 Spray Nozzles
- B1.3.4 Thermal Sleeve

B1.4 Jet Pump Assemblies

- B1.4.1 Thermal Sleeve
- B1.4.2 Inlet Header
- B1.4.3 Riser Brace Arm
- B1.4.4 Holddown Beams
- B1.4.5 Inlet Elbow
- B1.4.6 Mixing Assembly
- B1.4.7 Diffuser
- B1.4.8 Castings
- B1.4.9 Jet Pump Sensing Line

B1.5 Fuel Supports and Control Rod Drive (CRD) Assemblies

- B1.5.1 Orificed Fuel Support
- B1.5.2 CRD Housing

B1.6 Instrumentation

- B1.6.1 Intermediate Range Monitor (IRM) Dry Tubes
- B1.6.2 Low Power Range Monitor (LPRM) Dry Tubes
- B1.6.3 Source Range Monitor (SRM) Dry Tubes
- B1.6.4 Incore Neutron Flux Monitor Guide Tubes

B1. REACTOR VESSEL INTERNALS (BOILING WATER REACTOR)

Systems, Structures, and Components

This section comprises the boiling water reactor (BWR) vessel internals and consists of the core shroud and core plate, the top guide, feedwater spargers, core spray lines and spargers, jet pump assemblies, fuel supports and control rod drive (CRD), and instrument housings, such as the intermediate range monitor (IRM) dry tubes, the low power range monitor (LPRM) dry tubes, and the source range monitor (SRM) dry tubes. Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all structures and components that comprise the reactor vessel are governed by Group A or B Quality Standards.

The steam separator and dryer assemblies are not part of the pressure boundary and are removed during each outage, and they are covered by the plant maintenance program.

System Interfaces

The systems that interface with the reactor vessel internals include the reactor pressure vessel (IV.A1) and the reactor coolant pressure boundary (IV.C1).

IV **Reactor Vessel, Internals, and Reactor Coolant System**
B1. Reactor Vessel Internals (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B1.1-a B1.1.1	Core shroud and core plate Core shroud (upper, central, lower)	Stainless steel	288°C (550°F) high-purity water	Crack initiation and growth/ Stress corrosion cracking, intergranular stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M9, "BWR Vessel Internals," for core shroud and Chapter XI.M2, "Water Chemistry" for BWR water in BWRVIP-29 (EPRI TR-103515)	No
B1.1-b B1.1.2 B1.1.3	Core shroud and core plate Core plate Core plate bolts (used in early BWRs)	Stainless steel	288°C (550°F) high-purity water	Crack initiation and growth/ stress corrosion cracking, intergranular stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M9, "BWR Vessel Internals," for core plate and Chapter XI.M2, "Water Chemistry" for BWR water in BWRVIP-29 (EPRI TR-103515)	No
B1.1-c B1.1.2	Core shroud and core plate Core plate	Stainless steel	288°C (550°F) high-purity water	Cumulative fatigue damage/ Fatigue	For components for which a fatigue analysis has been performed for the 40-year period, fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA

IV Reactor Vessel, Internals, and Reactor Coolant System
B1. Reactor Vessel Internals (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B1.1-d B1.1.4	Core shroud and core plate Access hole cover (welded covers)	Alloy 600, alloy 182 welds	288°C (550°F) high-purity water	Crack initiation and growth/ Stress corrosion cracking, intergranular stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515) Because cracking initiated in crevice regions is not amenable to visual inspection, for BWRs with a crevice in the access hole covers, an augmented inspection is to include ultrasonic testing (UT) or other demonstrated acceptable inspection of the access hole cover welds.	No
B1.1-e B1.1.4	Core shroud and core plate Access hole cover (mechanical covers)	Alloy 600	288°C (550°F) high-purity water	Crack initiation and growth/ Stress corrosion cracking, intergranular stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515)	No
B1.1-f B1.1.5	Core shroud and core plate Shroud support structure (shroud support cylinder, shroud support plate, shroud support legs)	Alloy 600, alloy 182 welds	288°C (550°F) high-purity water	Crack initiation and growth/ Stress corrosion cracking, intergranular stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M9, "BWR Vessel Internals," for shroud support and Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515)	No

IV **Reactor Vessel, Internals, and Reactor Coolant System**
B1. Reactor Vessel Internals (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B1.1-g B1.1.6	Core shroud and core plate LPCI coupling	Stainless steel	288°C (550°F) high-purity water	Crack initiation and growth/ Stress corrosion cracking, intergranular stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M9, "BWR Vessel Internals," for the LPCI coupling and Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515)	No
B1.2-a	Top guide	Stainless steel	288°C (550°F) high-purity water	Crack initiation and growth/ Stress corrosion cracking, intergranular stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M9, "BWR Vessel Internals," for top guide and Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515)	No
B1.2-b	Top guide	Stainless steel	288°C (550°F) high-purity water	Cumulative fatigue damage/ Fatigue	For components for which a fatigue analysis has been performed for the 40-year period, fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA

IV Reactor Vessel, Internals, and Reactor Coolant System
B1. Reactor Vessel Internals (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B1.3-a B1.3.1 B1.3.2 B1.3.3 B1.3.4	Core spray lines and spargers Core spray lines (headers) Spray rings Spray nozzles Thermal sleeves	Stainless steel	288°C (550°F) high-purity water	Crack initiation and growth/ Stress corrosion cracking, intergranular stress corrosion cracking irradiation-assisted stress corrosion cracking	Chapter XI.M9, "BWR Vessel Internals," for core spray internals and Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515)	No
B1.3-b B1.3.1 B1.3.2 B1.3.3 B1.3.4	Core spray lines and spargers Core spray lines (headers) Spray rings Spray nozzles Thermal sleeves	Stainless steel	288°C (550°F) high-purity water	Cumulative fatigue damage/ Fatigue	For components for which a fatigue analysis has been performed for the 40-year period, fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
B1.4-a B1.4.1 B1.4.2 B1.4.3 B1.4.4 B1.4.5 B1.4.6 B1.4.7 B1.4.8	Jet pump assemblies Thermal sleeve Inlet header Riser brace arm Holddown beams Inlet elbow Mixing assembly Diffuser Castings	Holddown beams: Ni alloy (X-750), castings: cast austenitic stainless steel (CASS), others: stainless steel	288°C (550°F) high-purity water	Crack initiation and growth/ Stress corrosion cracking, intergranular stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M9, "BWR Vessel Internals," for jet pump assembly and Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515)	No

IV **Reactor Vessel, Internals, and Reactor Coolant System**
B1. Reactor Vessel Internals (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B1.4-b B1.4.1 B1.4.2 B1.4.3 B1.4.4 B1.4.5 B1.4.6 B1.4.7 B1.4.8	Jet pump assemblies Thermal sleeve Inlet header Riser brace arm Holddown beams Inlet elbow Mixing assembly Diffuser Castings	Holddown beams: Ni alloy (X-750), others: stainless steel	288°C (550°F) high-purity water	Cumulative fatigue damage/ Fatigue	For components for which a fatigue analysis has been performed for the 40-year period, fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
B1.4-c B1.4.8	Jet pump assemblies Castings	Cast austenitic stainless steel	288°C (550°F) high-purity water	Loss of fracture toughness/ Thermal aging and neutron irradiation embrittlement	Chapter XI.M13, "Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)"	No
B1.4-d B1.4.9	Jet pump assemblies Jet pump sensing line	Stainless steel	288°C (550°F) high-purity water	Crack initiation and growth/ cyclic loading	A plant-specific aging management program is to be evaluated.	Yes, plant specific
B1.5-a B1.5.1	Fuel supports and control rod drive assemblies Orificed fuel support	Cast austenitic stainless steel	288°C (550°F) high-purity water	Loss of fracture toughness/ Thermal aging and neutron irradiation embrittlement	Chapter XI.M13, "Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)"	No
B1.5-b B1.5.1	Fuel supports and control rod drive assemblies Orificed fuel support	Stainless steel, cast austenitic stainless steel	288°C (550°F) high-purity water	Cumulative fatigue damage/ Fatigue	For components for which a fatigue analysis has been performed for the 40-year period, fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA

IV **Reactor Vessel, Internals, and Reactor Coolant System**
B1. Reactor Vessel Internals (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B1.5-c B1.5.2	Fuel supports and control rod drive assemblies Control rod drive housing	Stainless steel	Up to 288°C, (550°F) reactor coolant water	Crack initiation and growth/ Stress corrosion cracking, intergranular stress corrosion cracking	Chapter XI.M9, "BWR Vessel Internals," for lower plenum and Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515)	No
B1.6-a B1.6.1 B1.6.3 B1.6.4	Instrumentation Intermediate range monitor (IRM) dry tubes Source range monitor (SRM) dry tubes Incore neutron flux monitor guide tubes	Stainless steel	288°C (550°F) high-purity water	Crack initiation and growth/ Stress corrosion cracking, intergranular stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI. M9, "BWR Vessel Internals," for lower plenum and Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515)	No
B1.6-b B1.6.1 B1.6.2 B1.6.3 B1.6.4	Instrumentation Intermediate range monitor dry tubes Low power range monitor dry tubes SRM dry tubes Incore neutron flux monitor guide tubes	Stainless steel	288°C (550°F) high-purity water	Cumulative fatigue damage/ Fatigue	For components for which a fatigue analysis has been performed for the 40-year period, fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA

B2. REACTOR VESSEL INTERNALS (PWR) - WESTINGHOUSE

B2.1 Upper Internals Assembly

- B2.1.1 Upper Support Plate
- B2.1.2 Upper Support Column
- B2.1.3 Upper Support Column Bolts
- B2.1.4 Upper Core Plate
- B2.1.5 Upper Core Plate Alignment Pins
- B2.1.6 Fuel Alignment Pins
- B2.1.7 Hold-Down Spring

B2.2 RCCA Guide Tube Assemblies

- B2.2.1 RCCA Guide Tubes
- B2.2.2 RCCA Guide Tube Bolts
- B2.2.3 RCCA Guide Tube Support Pins

B2.3 Core Barrel

- B2.3.1 Core Barrel
- B2.3.2 Core Barrel Flange
- B2.3.3 Core Barrel Outlet Nozzles
- B2.3.4 Thermal Shield

B2.4 Baffle/Former Assembly

- B2.4.1 Baffle and Former Plates
- B2.4.2 Baffle/Former Bolts

B2.5 Lower Internal Assembly

- B2.5.1 Lower Core Plate
- B2.5.2 Fuel Alignment Pins
- B2.5.3 Lower Support Forging or Casting
- B2.5.4 Lower Support Plate Columns
- B2.5.5 Lower Support Plate Column Bolts
- B2.5.6 Radial Support Keys and Clevis Inserts
- B2.5.7 Clevis Insert Bolts

B2.6 Instrumentation Support Structures

- B2.6.1 Flux Thimble Guide Tubes
- B2.6.2 Flux Thimbles

B2. REACTOR VESSEL INTERNALS (PWR) - WESTINGHOUSE

Systems, Structures, and Components

This section comprises the Westinghouse pressurized water reactor (PWR) vessel internals and consists of the upper internals assembly, the rod control cluster assemblies (RCCA) guide tube assemblies, the core barrel, the baffle/former assembly, the lower internal assembly, and the instrumentation support structures. Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all structures and components that comprise the reactor vessel are governed by Group A or B Quality Standards.

System Interfaces

The systems that interface with the reactor vessel internals include the reactor pressure vessel (IV.A2).

IV **Reactor Vessel, Internals, and Reactor Coolant System**
B2. Reactor Vessel Internals (PWR) – Westinghouse

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B2.1-a B2.1.1 B2.1.4 B2.1.7	Upper internals assembly Upper support plate Upper core plate Hold-down spring	Stainless steel	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M16, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
B2.1-b B2.1.1 B2.1.4 B2.1.7	Upper internals assembly Upper support plate Upper core plate Hold-down spring	Stainless steel	Chemically treated borated water up to 340°C (644°F)	Changes in dimensions/ Void Swelling	A plant-specific aging management program is to be evaluated. The applicant is to provide a plant-specific AMP or participate in industry programs to investigate aging effects and determine appropriate AMP. Otherwise, the applicant is to provide the basis for concluding that void swelling is not an issue for the component.	Yes, plant specific
B2.1-c B2.1.1 B2.1.4 B2.1.7	Upper internals assembly Upper support plate Upper core plate Hold-down spring	Stainless steel	Chemically treated borated water up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	For components for which a fatigue analysis has been performed for the 40-year period, fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
B2.1-d B2.1.7	Upper internals assembly Hold-down spring	Stainless steel	Chemically treated borated water up to 340°C (644°F)	Loss of preload/ Stress relaxation	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and either Chapter XI.M14, "Loose Part Monitoring," or Chapter XI.M15, "Neutron Noise Monitoring"	No

IV Reactor Vessel, Internals, and Reactor Coolant System
B2. Reactor Vessel Internals (PWR) – Westinghouse

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B2.1-e B2.1.2	Upper internals assembly Upper support column	Stainless steel, cast austenitic stainless steel	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M16, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
B2.1-f B2.1.2	Upper internals assembly Upper support column	Stainless steel, cast austenitic stainless steel	Chemically treated borated water up to 340°C (644°F)	Changes in dimensions/ Void swelling	A plant-specific aging management program is to be evaluated. The applicant is to provide a plant-specific AMP or participate in industry programs to investigate aging effects and determine appropriate AMP. Otherwise, the applicant is to provide the basis for concluding that void swelling is not an issue for the component.	Yes, plant specific
B2.1-g B2.1.2	Upper internals assembly Upper support column (only cast austenitic stainless steel portions)	Cast austenitic stainless steel	Chemically treated borated water up to 340°C (644°F) neutron fluence greater than 10^{17} n/cm^2 (E >1 MeV)	Loss of fracture toughness/ Thermal aging and neutron irradiation embrittlement, void swelling	Chapter XI.M13, "Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)"	No
B2.1-h B2.1.2	Upper internals assembly Upper support column	Stainless steel, cast austenitic stainless steel	Chemically treated borated water up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	For components for which a fatigue analysis has been performed for the 40-year period, fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA

IV **Reactor Vessel, Internals, and Reactor Coolant System**
B2. Reactor Vessel Internals (PWR) – Westinghouse

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B2.1-i B2.1.3 B2.1.5 B2.1.6	Upper internals assembly Upper support column bolts Upper core plate alignment pins Fuel alignment pins	Stainless steel, Ni alloy	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M16, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
B2.1-j B2.1.3 B2.1.5 B2.1.6	Upper internals assembly Upper support column bolts Upper core plate alignment pins Fuel alignment pins	Stainless steel, Ni alloy	Chemically treated borated water up to 340°C (644°F)	Changes in dimensions/ Void swelling	A plant-specific aging management program is to be evaluated. The applicant is to provide a plant-specific AMP or participate in industry programs to investigate aging effects and determine appropriate AMP. Otherwise, the applicant is to provide the basis for concluding that void swelling is not an issue for the component.	Yes, plant specific
B2.1-k B2.1.3	Upper internals assembly Upper support column bolts	Stainless steel, Ni alloy	Chemically treated borated water up to 340°C (644°F)	Loss of preload/ Stress relaxation	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M14, "Loose Part Monitoring"	No
B2.1-l B2.1.5	Upper internals assembly Upper core plate alignment pins	Stainless steel, Ni alloy	Chemically treated borated water up to 340°C (644°F)	Loss of material/ Wear	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components	No

IV Reactor Vessel, Internals, and Reactor Coolant System
B2. Reactor Vessel Internals (PWR) – Westinghouse

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B2.1-m B2.1.6	Upper internals assembly Fuel alignment pins	Stainless steel, Ni alloy	Chemically treated borated water up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	For components for which a fatigue analysis has been performed for the 40-year period, fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
B2.2-a B2.2.1	RCCA guide tube assemblies RCCA guide tubes	Stainless steel	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M16, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
B2.2-b B2.2.1	RCCA guide tube assemblies RCCA guide tubes	Stainless steel	Chemically treated borated water up to 340°C (644°F)	Changes in dimensions/ Void swelling	A plant-specific aging management program is to be evaluated. The applicant is to provide a plant-specific AMP or participate in industry programs to investigate aging effects and determine appropriate AMP. Otherwise, the applicant is to provide the basis for concluding that void swelling is not an issue for the component.	Yes, plant specific
B2.2-c B2.2.1	RCCA guide tube assemblies RCCA guide tubes	Stainless steel	Chemically treated borated water up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	For components for which a fatigue analysis has been performed for the 40-year period, fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B2.2-d B2.2.2 B2.2.3	RCCA guide tube assemblies RCCA guide tube bolts RCCA guide tube support pins	Stainless steel, Ni alloy	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M16, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
B2.2-e B2.2.2 B2.2.3	RCCA guide tube assemblies RCCA guide tube bolts, RCCA guide tube support pins	Stainless steel, Ni alloy	Chemically treated borated water up to 340°C (644°F)	Changes in dimensions/ Void swelling	A plant-specific aging management program is to be evaluated. The applicant is to provide a plant-specific AMP or participate in industry programs to investigate aging effects and determine appropriate AMP. Otherwise, the applicant is to provide the basis for concluding that void swelling is not an issue for the component.	Yes, plant specific
B2.2-f B2.2.2 B2.2.3	RCCA guide tube assemblies RCCA guide tube bolts RCCA guide tube support pins	Stainless steel, Ni alloy	Chemically treated borated water up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	For components for which a fatigue analysis has been performed for the 40-year period, fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
B2.3-a B2.3.1 B2.3.2 B2.3.3 B2.3.4	Core barrel Core barrel (CB) CB flange (upper) CB outlet nozzles Thermal shield	Stainless steel	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M16, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No

IV Reactor Vessel, Internals, and Reactor Coolant System
B2. Reactor Vessel Internals (PWR) – Westinghouse

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B2.3-b B2.3.1 B2.3.2 B2.3.3 B2.3.4	Core barrel Core barrel (CB) CB flange (upper) CB outlet nozzles Thermal shield	Stainless steel	Chemically treated borated water up to 340°C (644°F)	Changes in dimensions/ Void swelling	A plant-specific aging management program is to be evaluated. The applicant is to provide a plant-specific AMP or participate in industry programs to investigate aging effects and determine appropriate AMP. Otherwise, the applicant is to provide the basis for concluding that void swelling is not an issue for the component.	Yes, plant specific
B2.3-c B2.3.1 B2.3.2 B2.3.3 B2.3.4	Core barrel Core barrel (CB) CB flange (upper) CB outlet nozzles Thermal shield	Stainless steel	Chemically treated borated water up to 340°C (644°F) neutron fluence greater than 10^{17} n/cm^2 (E>1 MeV)	Loss of fracture toughness/ Neutron irradiation embrittlement, void selling	Chapter XI.M16, "PWR Vessel Internals"	No
B2.3-d B2.3.1 B2.3.2 B2.3.3 B2.3.4	Core barrel Core barrel (CB) CB flange (upper) CB outlet nozzles Thermal shield	Stainless steel	Chemically treated borated water up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	For components for which a fatigue analysis has been performed for the 40-year period, fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
B2.4-a B2.4.1	Baffle/former assembly Baffle and former plates	Stainless steel	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M16, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No

IV **Reactor Vessel, Internals, and Reactor Coolant System**
B2. Reactor Vessel Internals (PWR) – Westinghouse

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B2.4-b B2.4.1	Baffle/former assembly Baffle and former plates	Stainless steel	Chemically treated borated water up to 340°C (644°F)	Changes in dimensions/ Void swelling	A plant-specific aging management program is to be evaluated. The applicant is to provide a plant-specific AMP or participate in industry programs to investigate aging effects and determine appropriate AMP. Otherwise, the applicant is to provide the basis for concluding that void swelling is not an issue for the component or is to provide an AMP. The applicant is to address the loss of ductility associated with swelling.	Yes, plant specific
B2.4-c B2.4.2	Baffle/former assembly Baffle/former bolts	Stainless steel (type 347 and cold-worked type 316)	Chemically treated borated water up to 340°C (644°F) and high fluence (>10 dpa or 7 x 10^{21} n/cm^2 E >1 MeV)	Crack initiation and growth/ Stress corrosion cracking, irradiation-assisted stress corrosion cracking	A plant-specific aging management program is to be evaluated. Historically, the VT-3 visual examinations have not identified baffle/former bolt cracking because cracking occurs at the juncture of the bolt head and shank, which is not accessible for visual inspection. However, recent UT examinations of the baffle/former bolts have identified cracking in several plants. The industry is currently addressing the issue of baffle bolt cracking in the PWR Materials Reliability Project, Issues Task Group (ITG) activities to determine, develop, and implement the necessary steps and plans to manage the applicable aging effects on a plant-specific basis.	Yes, plant specific

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B2.4-d B2.4.2	Baffle/former assembly Baffle/former bolts	Stainless steel (type 347 and cold-worked type 316)	Chemically treated borated water up to 340°C (644°F) and high fluence	Changes in dimensions/ Void swelling	A plant-specific aging management program is to be evaluated. The applicant is to provide a plant-specific AMP or participate in industry programs to investigate aging effects and determine appropriate AMP. Otherwise, the applicant is to provide the basis for concluding that void swelling is not an issue for the component.	Yes, plant specific
B2.4-e B2.4.1	Baffle/former assembly Baffle and former plates	Stainless steel	Chemically treated borated water up to 340°C fluence $>10^{17}$ n/cm^2 (E >1 MeV)	Loss of fracture toughness/ Neutron irradiation embrittlement, void swelling	Chapter XI.M16, "PWR Vessel Internals"	No
B2.4-f B2.4.2	Baffle/former assembly Baffle/former bolts	Stainless steel (type 347 and cold-worked type 316)	Treated borated water up to 340°C fluence $>10^{17}$ n/cm^2 (E >1 MeV)	Loss of fracture toughness/ Neutron irradiation embrittlement	A plant-specific aging management program is to be evaluated.	Yes, plant specific
B2.4-g B2.4.1 B2.4.2	Baffle/former assembly Baffle and former plates Baffle/former bolts	Stainless steel, Ni alloy (bolts)	Chemically treated borated water up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	For components for which a fatigue analysis has been performed for the 40-year period, fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA

IV Reactor Vessel, Internals, and Reactor Coolant System
B2. Reactor Vessel Internals (PWR) – Westinghouse

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B2.4-h B2.4.2	Baffle/former assembly Baffle/former bolts	Stainless steel, Ni alloy	Chemically treated borated water up to 340°C (644°F)	Loss of preload/ Stress relaxation	A plant-specific aging management program is to be evaluated. Visual inspection (VT-3) is to be augmented to detect relevant conditions of stress relaxation because only the heads of the baffle/former bolts are visible, and a plant-specific aging management program is thus required.	Yes, plant specific
B2.5-a B2.5.1 B2.5.6	Lower internal assembly Lower core plate Radial keys and clevis inserts	Stainless steel	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M16, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
B2.5-b B2.5.1 B2.5.6	Lower internal assembly Lower core plate Radial keys and clevis inserts	Stainless steel	Chemically treated borated water up to 340°C (644°F)	Changes in dimensions/ Void swelling	A plant-specific aging management program is to be evaluated. The applicant is to provide a plant-specific AMP or participate in industry programs to investigate aging effects and determine appropriate AMP. Otherwise, the applicant is to provide the basis for concluding that void swelling is not an issue for the component.	Yes, plant specific
B2.5-c B2.5.1	Lower internal assembly Lower core plate	Stainless steel	Treated borated water up to 340°C fluence $>10^{17}$ n/cm^2 (E >1 MeV)	Loss of fracture toughness/ Neutron irradiation embrittlement, void swelling	Chapter XI.M16, "PWR Vessel Internals"	No

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B2.5-d B2.5.1 B2.5.4	Lower internal assembly Lower core plate Lower support plate columns	Stainless steel	Chemically treated borated water up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	For components for which a fatigue analysis has been performed for the 40-year period, fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
B2.5-e B2.5.2 B2.5.5 B2.5.7	Lower internal assembly Fuel alignment pins Lower support plate column bolts Clevis insert bolts	Stainless steel, Ni alloy	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M16, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry" for PWR primary water in EPRI TR-105714	No
B2.5-f B2.5.2 B2.5.5 B2.5.7	Lower internal assembly Fuel alignment pins Lower support plate column bolts Clevis insert bolts	Stainless steel, Ni alloy	Chemically treated borated water up to 340°C (644°F)	Changes in dimensions/ Void swelling	A plant-specific aging management program is to be evaluated. The applicant is to provide a plant-specific AMP or participate in industry programs to investigate aging effects and determine appropriate AMP. Otherwise, the applicant is to provide the basis for concluding that void swelling is not an issue for the component.	Yes, plant specific
B2.5-g B2.5.2 B2.5.5 B2.5.7	Lower internal assembly Fuel alignment pins Lower support plate column bolts Clevis insert bolts	Stainless steel, Ni alloy	Treated borated water up to 340°C fluence $>10^{17}$ n/cm^2 (E >1 MeV)	Loss of fracture toughness/ Neutron irradiation embrittlement, void swelling	Chapter XI.M16, "PWR Vessel Internals"	No

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B2.5-h B2.5.5	Lower internal assembly Lower support plate column bolts	Stainless steel, Ni alloy	Chemically treated borated water up to 340°C (644°F)	Loss of preload/ Stress relaxation	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M14, "Loose Part Monitoring"	No
B2.5-i B2.5.7	Lower internal assembly Clevis insert bolts	Stainless steel, Ni alloy	Chemically treated borated water up to 340°C (644°F)	Loss of preload/ Stress relaxation	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and either Chapter XI.M14, "Loose Part Monitoring," or Chapter XI.M15, "Neutron Noise Monitoring"	No
B2.5-j B2.5.2 B2.5.5	Lower internal assembly Fuel alignment pins Lower support plate column bolts	Stainless steel, Ni alloy	Chemically treated borated water up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	For components for which a fatigue analysis has been performed for the 40-year period, fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
B2.5-k B2.5.3 B2.5.4	Lower internal assembly Lower support forging or casting Lower support plate columns	Stainless steel, cast austenitic stainless steel	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M16, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
B2.5-l B2.5.3 B2.5.4	Lower internal assembly Lower support forging or casting Lower support plate columns	Stainless steel, cast austenitic stainless steel	Chemically treated borated water up to 340°C (644°F)	Changes in dimensions/ Void swelling	A plant-specific aging management program is to be evaluated. The applicant is to provide a plant-specific AMP or participate in industry programs to investigate aging effects and determine appropriate AMP. Otherwise, the applicant is to provide the basis for concluding that void swelling is not an issue for the component.	Yes, plant specific

IV **Reactor Vessel, Internals, and Reactor Coolant System**
B2. Reactor Vessel Internals (PWR) – Westinghouse

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B2.5-m B2.5.3 B2.5.4	Lower internal assembly Lower support forging or casting Lower support plate columns	Cast austenitic stainless steel	Chemically treated borated water up to 340°C (644°F) fluence $>10^{17}$ n/cm^2 (E >1 MeV)	Loss of fracture toughness/ Thermal aging and neutron irradiation embrittlement, void swelling	Chapter XI.M13, "Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)"	No
B2.5-n B2.5.3 B2.5.4	Lower internal assembly Lower support forging or casting Lower support plate columns	Stainless steel	Chemically treated borated water up to 340°C (644°F) fluence $>10^{17}$ n/cm^2 (E >1 MeV)	Loss of fracture toughness/ Neutron irradiation embrittlement, void swelling	Chapter XI.M16, "PWR Vessel Internals"	No
B2.5-o B2.5.6	Lower internal assembly Radial keys and clevis Inserts	Stainless steel	Chemically treated borated water up to 340°C (644°F)	Loss of material/ Wear	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components	No
B2.5-p B2.5.6 B2.5.7	Lower internal assembly Radial keys and clevis inserts Clevis insert bolts	Stainless steel	Chemically treated borated water up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	For components for which a fatigue analysis has been performed for the 40-year period, fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA

IV **Reactor Vessel, Internals, and Reactor Coolant System**
B2. Reactor Vessel Internals (PWR) – Westinghouse

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B2.6-a B2.6.1	Instrumentation support structures Flux thimble guide tubes	Stainless steel	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M16, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
B2.6-b B2.6.1	Instrumentation support structures Flux thimble guide tubes	Stainless steel	Chemically treated borated water up to 340°C (644°F)	Changes in dimensions/ Void swelling	A plant-specific aging management program is to be evaluated. The applicant is to provide a plant-specific AMP or participate in industry programs to investigate aging effects and determine appropriate AMP. Otherwise, the applicant is to provide the basis for concluding that void swelling is not an issue for the component.	Yes, plant specific

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B2.6-c B2.6.2	Instrumentation support structures Flux thimble	Stainless steel	Chemically treated borated water up to 340°C (644°F)	Loss of material/ Wear	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and recommendations of NRC I&E Bulletin 88-09 "Thimble Tube Thinning in Westinghouse Reactors," described bellow: In response to I&E Bulletin 88-09, an inspection program, with technical justification, is to be established and is to include (a) an appropriate thimble tube wear acceptance criterion, e.g., percent through-wall loss, and includes allowances for inspection methodology and wear scar geometry uncertainty, (b) an appropriate inspection frequency, e.g., every refueling outage, and (c) inspection methodology such as eddy current technique that is capable of adequately detecting wear of the thimble tubes. In addition, corrective actions include isolation or replacement if a thimble tube fails to meet the above acceptance criteria. Inspection schedule is in accordance with the guidelines of I&E Bulletin 88-09.	No

B3. REACTOR VESSEL INTERNALS (PWR) - COMBUSTION ENGINEERING

B3.1 Upper Internals Assembly

- B3.1.1 Upper Guide Structure Support Plate
- B3.1.2 Fuel Alignment Plate
- B3.1.3 Fuel Alignment Plate Guide Lugs and Guide Lug Inserts
- B3.1.4 Hold-Down Ring

B3.2 Control Element Assembly (CEA) Shroud Assemblies

- B3.2.1 CEA Shrouds
- B3.2.2 CEA Shrouds Bolts
- B3.2.3 CEA Shrouds Extension Shaft Guides

B3.3 Core Support Barrel

- B3.3.1 Core Support Barrel
- B3.3.2 Core Support Barrel Upper Flange
- B3.3.3 Core Support Barrel Alignment Keys

B3.4 Core Shroud Assembly

- B3.4.1 Core Shroud Assembly
- B3.4.2 Core Shroud Assembly Bolts
- B3.4.3 Core Shroud Tie Rods

B3.5 Lower Internal Assembly

- B3.5.1 Core Support Plate
- B3.5.2 Fuel Alignment Pins
- B3.5.3 Lower Support Structure Beam Assemblies
- B3.5.4 Core Support Column
- B3.5.5 Core Support Column Bolts
- B3.5.6 Core Support Barrel Snubber Assemblies

B3. REACTOR VESSEL INTERNALS (PWR) - COMBUSTION ENGINEERING

Systems, Structures, and Components

This section comprises the Combustion Engineering pressurized water reactor (PWR) vessel internals and consists of the upper internals assembly, the CEA shroud assemblies, the core support barrel, the core shroud assembly, and the lower internal assembly. Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all structures and components that comprise the reactor vessel are governed by Group A or B Quality Standards.

System Interfaces

The systems that interface with the reactor vessel internals include the reactor pressure vessel (IV.A2).

IV Reactor Vessel, Internals, and Reactor Coolant System
B3. Reactor Vessel Internals (PWR) – Combustion Engineering

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B3.1-a B3.1.1 B3.1.2 B3.1.3	Upper Internals Assembly Upper guide structure support plate Fuel alignment plate Fuel alignment plate guide lugs and guide lug inserts	Stainless steel	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking , irradiation-assisted stress corrosion cracking	Chapter XI.M16 "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
B3.1-b B3.1.1 B3.1.2 B3.1.3	Upper Internals Assembly Upper guide structure support plate Fuel alignment plate Fuel alignment plate guide lugs and guide lug inserts	Stainless steel	Chemically treated borated water up to 340°C (644°F)	Changes in dimensions/ Void swelling	A plant-specific aging management program is to be evaluated. The applicant is to provide a plant-specific AMP or participate in industry programs to investigate aging effects and determine appropriate AMP. Otherwise, the applicant is to provide the basis for concluding that void swelling is not an issue for the component.	Yes, plant specific
B3.1-c B3.1.2 B3.1.3 B3.1.4	Upper Internals Assembly Fuel alignment plate Fuel alignment plate guide lugs and their lugs Hold-down ring	Stainless steel	Chemically treated borated water up to 340°C (644°F)	Loss of material/ Wear	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components	No
B3.2-a B3.2.1	CEA Shroud Assemblies CEA shroud	Stainless steel cast austenitic stainless steel	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M16, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No

IV Reactor Vessel, Internals, and Reactor Coolant System
B3. Reactor Vessel Internals (PWR) – Combustion Engineering

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B3.2-b B3.2.2	CEA Shroud Assemblies CEA shrouds bolts	Stainless steel, Ni alloy	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M16, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
B3.2-c B3.2.1 B3.2.2	CEA shroud assemblies CEA shroud CEA shrouds bolts	Stainless steel, cast austenitic stainless steel, Ni alloy	Chemically treated borated water up to 340°C (644°F)	Changes in dimensions/ Void swelling	A plant-specific aging management program is to be evaluated. The applicant is to provide a plant-specific AMP or participate in industry programs to investigate aging effects and determine appropriate AMP. Otherwise, the applicant is to provide the basis for concluding that void swelling is not an issue for the component.	Yes, plant specific
B3.2-d B3.2.3	CEA shroud assemblies CEA shroud extension shaft guides	Stainless steel	Chemically treated borated water up to 340°C (644°F)	Loss of material/ Wear	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components	No
B3.2-e B3.2.1	CEA shroud assemblies CEA shroud	Cast austenitic stainless steel	Chemically treated borated water up to 340°C (644°F) neutron fluence of greater than 10^{17} n/cm^2 (E>1 MeV)	Loss of fracture toughness/ Thermal aging and neutron irradiation embrittlement, void swelling	Chapter XI.M13, "Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)"	No

IV Reactor Vessel, Internals, and Reactor Coolant System
B3. Reactor Vessel Internals (PWR) – Combustion Engineering

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B3.2-f B3.2.1 B3.2.2	CEA shroud assemblies CEA shroud CEA shrouds bolts	Stainless steel, cast austenitic stainless steel , Ni alloy	Chemically treated borated water up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	For components for which a fatigue analysis has been performed for the 40-year period, fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes, TLAA
B3.2-g B3.2.2	CEA shroud assemblies CEA shrouds bolts	Stainless steel, Ni alloy	Chemically treated borated water up to 340°C (644°F)	Loss of preload/ Stress relaxation	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M14, "Loose Part Monitoring"	No
B3.3-a B3.3.1 B3.3.2	Core support barrel Core support barrel Core support barrel upper flange	Stainless steel	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M16, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
B3.3-b B3.3.1 B3.3.2	Core support barrel Core support barrel Core support barrel upper flange	Stainless steel	Chemically treated borated water up to 340°C (644°F)	Changes in dimensions/ Void swelling	A plant-specific aging management program is to be evaluated. The applicant is to provide a plant-specific AMP or participate in industry programs to investigate aging effects and determine appropriate AMP. Otherwise, the applicant is to provide the basis for concluding that void swelling is not an issue for the component.	Yes, plant specific

IV Reactor Vessel, Internals, and Reactor Coolant System
B3. Reactor Vessel Internals (PWR) – Combustion Engineering

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B3.3-a B3.3.1 B3.3.2	Core support barrel Core support barrel Core support barrel upper flange	Stainless steel	Chemically Treated borated water up to 340°C (644°F) neutron fluence $>10^{17}$ n/cm^2 (E >1 MeV)	Loss of fracture toughness/ Neutron irradiation embrittlement, void swelling	Chapter XI.M16, "PWR Vessel Internals"	No
B3.3-b B3.3.2 B3.3.3	Core support barrel Core support barrel upper flange Core support barrel alignment keys	Stainless steel	Chemically treated borated water up to 340°C (644°F)	Loss of material/ Wear	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components	No
B3.4-a B3.4.1 B3.4.3	Core shroud assembly Core shroud assembly Core shroud tie rods (core support plate attached by welds in later plants)	Stainless steel, cast austenitic stainless steel	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M16, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
B3.4-b B3.4.1 B3.4.3	Core shroud assembly Core shroud assembly Core shroud tie rods (core support plate attached by welds in later plants)	Stainless steel, cast austenitic stainless steel, Ni alloy	Chemically treated borated water up to 340°C (644°F)	Changes in dimensions/ Void swelling	A plant-specific aging management program is to be evaluated. The applicant is to provide a plant-specific AMP or participate in industry programs to investigate aging effects and determine appropriate AMP. Otherwise, the applicant is to provide the basis for concluding that void swelling is not an issue for the component.	Yes, plant specific

IV **Reactor Vessel, Internals, and Reactor Coolant System**
B3. Reactor Vessel Internals (PWR) – Combustion Engineering

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B3.4-c B3.4.1 B3.4.3	Core shroud assembly Core shroud assembly Core shroud tie rods (core support plate attached by welds in later plants)	Stainless steel	Chemically treated borated water up to 340°C fluence $>10^{17}$ n/cm^2 (E >1 MeV)	Loss of fracture toughness/ Neutron irradiation embrittlement, void swelling	Chapter XI.M16, "PWR Vessel Internals"	No
B3.4-d B3.4.1 B3.4.2 B3.4.3	Core shroud assembly Core shroud assembly Core shroud assembly bolts Core shroud tie rods	Stainless steel, Ni alloy (bolts)	Chemically treated borated water up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	For components for which a fatigue analysis has been performed for the 40-year period, fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes, TLAA
B3.4-e B3.4.2	Core shroud assembly Core shroud assembly bolts (later plants are welded)	Stainless steel, Ni alloy	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M16, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No

IV Reactor Vessel, Internals, and Reactor Coolant System
B3. Reactor Vessel Internals (PWR) – Combustion Engineering

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B3.4-f B3.4.2	Core shroud assembly Core shroud assembly bolts (later plants are welded)	Stainless steel, Ni alloy	Chemically treated borated water up to 340°C (644°F)	Changes in dimensions/ Void swelling	A plant-specific aging management program is to be evaluated. The applicant is to provide a plant-specific AMP or participate in industry programs to investigate aging effects and determine appropriate AMP. Otherwise, the applicant is to provide the basis for concluding that void swelling is not an issue for the component.	Yes, plant specific
B3.4-g B3.4.2	Core shroud assembly Core shroud assembly bolts (later plants are welded)	Stainless steel, Ni alloy	Chemically treated borated water up to 340°C fluence $>10^{17}$ n/cm^2 (E >1 MeV)	Loss of fracture toughness/ Neutron irradiation embrittlement, void swelling	Chapter XI.M16, "PWR Vessel Internals"	No
B3.4-h B3.4.2 B3.4.3	Core shroud assembly Core shroud assembly bolts Core shroud tie rods	Stainless steel, Ni alloy	Chemically treated borated water up to 340°C (644°F)	Loss of preload/ Stress relaxation	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M14, "Loose Part Monitoring"	No
B3.5-a B3.5.1 B3.5.3 B3.5.4 B3.5.6	Lower internal assembly Core support plate Lower support structure beam assemblies Core support column Core support barrel snubber assemblies	Stainless steel	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M16, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No

IV Reactor Vessel, Internals, and Reactor Coolant System
B3. Reactor Vessel Internals (PWR) – Combustion Engineering

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B3.5-b B3.5.2 B3.5.5	Lower internal Assembly Fuel alignment pins Core support column bolts	Stainless steel, Ni alloy	Chemically treated borated water up to 340°C (644°F)	Crack Initiation and growth/ Stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M16, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
B3.5-c B3.5.1 B3.5.2 B3.5.3 B3.5.4 B3.5.5 B3.5.6	Lower internal assembly Core support plate Fuel alignment pins Lower support structure beam assemblies Core support column Core support column bolts Core support barrel snubber assemblies	Stainless steel, Ni alloy (pins/ bolts), cast austenitic stainless steel (support column)	Chemically treated borated water up to 340°C (644°F)	Changes in dimensions/ Void swelling	A plant-specific aging management program is to be evaluated. The applicant is to provide a plant-specific AMP or participate in industry programs to investigate aging effects and determine appropriate AMP. Otherwise, the applicant is to provide the basis for concluding that void swelling is not an issue for the component.	Yes, plant specific
B3.5-d B3.5.1 B3.5.2 B3.5.3 B3.5.5 B3.5.6	Lower internal assembly Core support plate Fuel alignment pins Lower support structure beam assemblies Core support column bolts Core support barrel snubber assemblies	Stainless steel, Ni alloy (pins/bolts)	Chemically treated borated water up to 340°C (644°F) neutron fluence $>10^{17}$ n/cm^2 (E >1 MeV)	Loss of fracture toughness/ Neutron irradiation embrittlement, void swelling	Chapter XI.M16, "PWR Vessel Internals"	No
B3.5-e B3.5.2 B3.5.6	Lower internal assembly Fuel alignment pins Core support barrel snubber assemblies	Stainless steel, Ni alloy (pins)	Chemically treated borated water up to 340°C (644°F)	Loss of material/ Wear	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components	No

IV **Reactor Vessel, Internals, and Reactor Coolant System**
B3. Reactor Vessel Internals (PWR) – Combustion Engineering

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B3.5-f B3.5.4	Lower internal assembly Core support column	Cast austenitic stainless steel	Chemically treated borated water up to 340°C (644°F)	Loss of fracture toughness/ Thermal aging and neutron irradiation embrittlement, void swelling	Chapter XI.M13, "Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)"	No
B3.5-g B3.5.1 B3.5.2 B3.5.3 B3.5.4 B3.5.5 B3.5.6	Lower internal assembly Core support plate Fuel alignment pins Lower support structure beam assemblies Core support column Core support column bolts Core support barrel snubber assemblies	Stainless steel, Ni alloy (pins/ bolts), cast austenitic stainless steel (support column)	Chemically treated borated water up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	For components for which a fatigue analysis has been performed for the 40-year period, fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes, TLAA

B4 REACTOR VESSEL INTERNALS (PWR) - BABCOCK AND WILCOX

B4.1 Plenum Cover and Plenum Cylinder

B4.1.1 Plenum Cover Assembly
B4.1.2 Plenum Cylinder
B4.1.3 Reinforcing Plates
B4.1.4 Top Flange-to-Cover Bolts
B4.1.5 Bottom Flange-to-Upper Grid Screws

B4.2 Upper Grid Assembly

B4.2.1 Upper Grid Rib Section
B4.2.2 Upper Grid Ring Forging
B4.2.3 Fuel Assembly Support Pads
B4.2.4 Plenum Rib Pads
B4.2.5 Rib-to-Ring Screws

B4.3 Control Rod Guide Tube (CRGT) Assembly

B4.3.1 CRGT Pipe and Flange
B4.3.2 CRGT Spacer Casting
B4.3.3 CRGT Spacer Screws
B4.3.4 Flange-to-Upper Grid Screws
B4.3.5 CRGT Rod Guide Tubes
B4.3.6 CRGT Rod Guide Sectors

B4.4 Core Support Shield Assembly

B4.4.1 Core Support Shield Cylinder (Top and Bottom Flange)
B4.4.2 Core Support Shield-to-Core Barrel Bolts
B4.4.3 Outlet and Vent Valve Nozzles
B4.4.4 Vent Valve Body and Retaining Ring
B4.4.5 Vent Valve Assembly Locking Device

B4.5 Core Barrel Assembly

B4.5.1 Core Barrel Cylinder (Top and Bottom Flange)
B4.5.2 Lower Internals Assembly-to-Core Barrel Bolts
B4.5.3 Core Barrel-to-Thermal Shield Bolts
B4.5.4 Baffle Plates and Formers
B4.5.5 Baffle/Former Bolts and Screws

B4.6 Lower Grid (LG) Assembly

B4.6.1 Lower Grid Rib Section
B4.6.2 Fuel Assembly Support Pads
B4.6.3 Lower Grid Rib-to-Shell Forging Screws
B4.6.4 Lower Grid Flow Distributor Plate
B4.6.5 Orifice Plugs
B4.6.6 Lower Grid and Shell Forgings

B4. REACTOR VESSEL INTERNALS (PWR) - BABCOCK AND WILCOX

Systems, Structures, and Components

This section comprises the Babcock and Wilcox pressurized water reactor (PWR) vessel internals and consists of the plenum cover and plenum cylinder, the upper grid assembly, the control rod guide tube (CRGT) assembly, the core support shield assembly, the core barrel assembly, the lower grid assembly, and the flow distributor assembly. Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all structures and components that comprise the reactor vessel are governed by Group A or B Quality Standards.

System Interfaces

The systems that interface with the reactor vessel internals include the reactor pressure vessel (IV.A2).

IV **Reactor Vessel, Internals, and Reactor Coolant System**
B4. Reactor Vessel Internals (PWR) – Babcock and Wilcox

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B4.1-a B4.1.1 B4.1.2 B4.1.3	Plenum cover and plenum cylinder Plenum cover assembly Plenum cylinder Reinforcing plates	Type 304 stainless steel, plenum cylinder: type 304 forging	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M16, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
B 4.1-b B4.1.4 B4.1.5	Plenum cover and plenum cylinder Top flange-to-cover bolts Bottom flange-to-upper grid screws	Gr. B-8 stainless steel	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M16, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
B4.1-c B4.1.1 B4.1.2 B4.1.3 B4.1.4 B4.1.5	Plenum cover and plenum cylinder Plenum cover assembly Plenum cylinder Reinforcing plates Top flange-to-cover bolts Bottom flange-to-upper grid screws	Type 304 stainless steel, bolts: Gr. B-8 stainless steel	Chemically treated borated water up to 340°C (644°F)	Changes in dimensions/ Void swelling	A plant-specific aging management program is to be evaluated. The applicant is to provide a plant-specific AMP or participate in industry programs to investigate aging effects and determine appropriate AMP. Otherwise, the applicant is to provide the basis for concluding that void swelling is not an issue for the component.	Yes, plant specific
B4.1-d B4.1.1 B4.1.2 B4.1.3 B4.1.4 B4.1.5	Plenum cover and plenum cylinder Plenum cover assembly Plenum cylinder Reinforcing plates Top flange-to-cover bolts Bottom flange-to-upper grid screws	Type 304 stainless steel, plenum cylinder: type 304 forging	Chemically treated borated water up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	For components for which a fatigue analysis has been performed for the 40-year period, fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes, TLAA

IV **Reactor Vessel, Internals, and Reactor Coolant System**
B4. Reactor Vessel Internals (PWR) – Babcock and Wilcox

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B4.2-a B4.2.1 B4.2.2 B4.2.3 B4.2.4	Upper grid assembly Upper grid rib section Upper grid ring forging Fuel assembly support pads Plenum rib pads	Type 304 stainless steel	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M16, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
B4.2-b B4.2.5	Upper grid assembly Rib- to-ring screws	Gr. B-8 stainless steel	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M16, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
B4.2-c B4.2.1 B4.2.2 B4.2.3 B4.2.4 B4.2.5	Upper grid assembly Upper grid rib section Upper grid ring forging Fuel assembly support pads Plenum rib pads Rib-to-ring screws	Type 304 stainless steel, screws: Gr. B-8 stainless steel	Chemically treated borated water up to 340°C (644°F)	Changes in dimensions/ Void swelling	A plant-specific aging management program is to be evaluated. The applicant is to provide a plant-specific AMP or participate in industry programs to investigate aging effects and determine appropriate AMP. Otherwise, the applicant is to provide the basis for concluding that void swelling is not an issue for the component.	Yes, plant specific
B4.2-d B4.2.1 B4.2.2 B4.2.3 B4.2.4 B4.2.5	Upper grid assembly Upper grid rib section Upper grid ring forging Fuel assembly support pads Plenum rib pads Rib-to-ring screws	Type 304 stainless steel	Chemically treated borated water up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	For components for which a fatigue analysis has been performed for the 40-year period, fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes, TLAA

IV **Reactor Vessel, Internals, and Reactor Coolant System**

B4. Reactor Vessel Internals (PWR) – Babcock and Wilcox

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B4.2-e B4.2.1 B4.2.2 B4.2.3 B4.2.4 B4.2.5	Upper grid assembly Upper grid rib section Upper grid ring forging Fuel assembly support pads Plenum rib pads Rib-to-ring screws	Type 304 stainless steel, screws: Gr. B-8 stainless steel	Chemically treated borated water up to 340°C (644°F)	Loss of fracture toughness/ Neutron irradiation embrittlement, void swelling	Chapter XI.M16, "PWR Vessel Internals"	No
B4.2-f B4.2.3 B4.2.4	Upper grid assembly Fuel assembly support pads Plenum rib pads	Type 304 stainless steel	Chemically treated borated water up to 340°C (644°F)	Loss of material/ Wear	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components	No
B4.3-a B4.3.1 B4.3.2 B4.3.5 B4.3.6	Control rod guide tube (CRGT) assembly CRGT pipe and flange CRGT spacer casting CRGT rod guide tubes CRGT rod guide sectors	Pipe and flange: type 304 stainless steel, spacer casting: CF-3M, guide tubes and sectors: type 304L	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI. M16, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
B4.3-b B4.3.3 B4.3.4	Control rod guide tube (CRGT) assembly CRGT spacer screws Flange-to-upper grid screws	Gr. B-8 stainless steel	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M16, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No

IV Reactor Vessel, Internals, and Reactor Coolant System
B4. Reactor Vessel Internals (PWR) – Babcock and Wilcox

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B4.3-c B4.3.1 B4.3.2 B4.3.3 B4.3.4 B4.3.5 B4.3.6	Control rod guide tube (CRGT) assembly CRGT pipe and flange CRGT spacer casting CRGT spacer screws Flange-to-upper grid screws CRGT rod guide tubes CRGT rod guide sectors	Pipe and flange: type 304 stainless steel; spacer casting: CF-3M; guide tubes and sectors: type 304L; screws: Gr. B-8 stainless steel	Chemically treated borated water up to 340°C (644°F)	Changes in dimensions/ Void swelling	A plant-specific aging management program is to be evaluated. The applicant is to provide a plant-specific AMP or participate in industry programs to investigate aging effects and determine appropriate AMP. Otherwise, the applicant is to provide the basis for concluding that void swelling is not an issue for the component.	Yes, plant specific
B4.3-d B4.3.2	Control rod guide tube (CRGT) assembly CRGT spacer casting	Cast austenitic stainless steel CF-3M	Chemically treated borated water up to 340°C (644°F) neutron fluence >10^{17} n/cm^2 (E >1 MeV)	Loss of fracture toughness/ Thermal aging and neutron irradiation embrittlement, void swelling	Chapter XI.M13, "Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)"	No
B4.3-e B4.3.4	Control rod guide tube (CRGT) assembly Flange-to-upper grid screws	Gr. B-8 stainless steel	Chemically treated borated water up to 340°C (644°F)	Loss of preload/ Stress relaxation	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M14, "Loose Part Monitoring"	No

IV Reactor Vessel, Internals, and Reactor Coolant System
B4. Reactor Vessel Internals (PWR) – Babcock and Wilcox

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B4.3-f B4.3.1 B4.3.2 B4.3.3 B4.3.4 B4.3.5 B4.3.6	Control rod guide tube (CRGT) assembly CRGT pipe and flange CRGT spacer casting CRGT spacer screws Flange-to-upper grid screws CRGT rod guide tubes CRGT rod guide sectors	Pipe and flange: type 304 stainless steel; spacer casting: CF-3M; guide tubes and sectors: type 304L; screws: Gr. B-8 stainless steel	Chemically treated borated water up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	For components for which a fatigue analysis has been performed for the 40-year period, fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes, TLAA
B4.4-a B4.4.1 B4.4.3 B4.4.4	Core support shield assembly Core support shield cylinder (top and bottom flange) Outlet and vent valve (VV) nozzles VV body and retaining ring	Shield cylinder: Type 304; nozzles: stainless steel forging, CF-8; VV body: CF-8; VV ring: type 15-5PH forging	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M16, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No

IV Reactor Vessel, Internals, and Reactor Coolant System
B4. Reactor Vessel Internals (PWR) – Babcock and Wilcox

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B4.4-b B4.4.2 B4.4.5	Core support shield assembly Core support shield-to-core barrel bolts VV assembly locking device	Bolts: Gr. 660 (A-286), Gr. 688 (X-750); VV locking device: Gr. B-8 or B-8M	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M16, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
B4.4-c B4.4.1 B4.4.2 B4.4.4 B4.4.5	Core support shield assembly Core support shield cylinder (top and bottom flange) Core support shield-to-core barrel bolts VV retaining ring VV assembly locking device	Shield cylinder: type 304; bolts: A-286, X-750; VV ring: type 15-5PH forging; locking device: Gr. B-8 or B-8M	Chemically treated borated water up to 340°C (644°F)	Changes in dimensions/ Void swelling	A plant-specific aging management program is to be evaluated. The applicant is to provide a plant-specific AMP or participate in industry programs to investigate aging effects and determine appropriate AMP. Otherwise, the applicant is to provide the basis for concluding that void swelling is not an issue for the component.	Yes, plant specific

IV Reactor Vessel, Internals, and Reactor Coolant System
B4. Reactor Vessel Internals (PWR) – Babcock and Wilcox

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B4.4-d B4.4.1 B4.4.2 B4.4.3 B4.4.5	Core support shield assembly Core support shield cylinder (top and bottom flange) Core support shield-to-core barrel bolts Outlet and vent valve (VV) nozzles VV assembly locking device	Shield cylinder: type 304; bolts: A-286, X-750; nozzles: stainless steel forging; VV ring: type 15-5PH forging; locking device: Gr. B-8 or B-8M	Chemically treated borated water up to 340°C (644°F) neutron fluence $>10^{17}$ n/cm^2 (E >1 MeV)	Loss of fracture toughness/ Neutron irradiation embrittlement, void swelling	Chapter XI.M16, "PWR Vessel Internals"	No

IV Reactor Vessel, Internals, and Reactor Coolant System
B4. Reactor Vessel Internals (PWR) – Babcock and Wilcox

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B4.4-e B4.4.1 B4.4.2 B4.4.3 B4.4.4	Core support shield assembly Core support shield cylinder (top and bottom flange) Core support shield-to-core barrel bolts Outlet and vent valve (VV) nozzles VV body and retaining ring	Shield cylinder: type 304; bolts: A-286, X-750; nozzles: stainless steel forging, CF-8; VV body: CF-8; VV ring: type 15-5PH forging; locking device: Gr. B-8 or B-8M	Chemically treated borated water up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	For components for which a fatigue analysis has been performed for the 40-year period, fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes, TLAA
B4.4-f B4.4.1 B4.4.5	Core support shield assembly Core support shield cylinder (top flange) VV assembly locking device	Top flange: type 304, VV locking device: Gr. B-8 or B-8M	Chemically treated borated water up to 340°C (644°F)	Loss of material/ Wear	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components	No
B4.4-g B4.4.3 B4.4.4	Core support shield assembly Outlet and vent valve nozzles VV body and retaining ring	Cast austenitic stainless steel CF-8	Chemically treated borated water up to 340°C fluence $>10^{17}$ n/cm^2 (E >1 MeV)	Loss of fracture toughness/ Thermal aging and neutron irradiation embrittlement, void swelling	Chapter XI.M13, "Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)"	No

IV Reactor Vessel, Internals, and Reactor Coolant System
B4. Reactor Vessel Internals (PWR) – Babcock and Wilcox

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B4.4-h B4.4.2	Core support shield assembly Core support shield-to-core barrel bolts	Gr. 660 (A-286), Gr. 688 (X-750)	Chemically treated borated water up to 340°C (644°F)	Loss of preload/ Stress relaxation	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M14, "Loose Part Monitoring"	No
B4.5-a B4.5.1 B4.5.4	Core barrel assembly Core barrel cylinder (top and bottom flange) Baffle plates and formers	CB cylinder: type 304 forging, baffle plates and formers: type 304 stainless steel	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M16, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
B4.5-b B4.5.2 B4.5.3	Core barrel assembly Lower internals assembly-to-core barrel bolts Core barrel-to-thermal shield bolts	A-286, X-750	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M16, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
B4.5-c B4.5.1 B4.5.2 B4.5.3 B4.5.4	Core barrel assembly Core barrel cylinder (top and bottom flange) Lower internals assembly-to-core barrel bolts Core barrel-to-thermal shield bolts Baffle plates and formers	CB cylinder: type 304 forging; CB bolts: A-286, X-750; baffle plates and formers: type 304 stainless steel	Chemically treated borated water up to 340°C (644°F)	Changes in dimensions/ Void swelling	A plant-specific aging management program is to be evaluated. The applicant is to provide a plant-specific AMP or participate in industry programs to investigate aging effects and determine appropriate AMP. Otherwise, the applicant is to provide the basis for concluding that void swelling is not an issue for the component.	Yes, plant specific

IV Reactor Vessel, Internals, and Reactor Coolant System
B4. Reactor Vessel Internals (PWR) – Babcock and Wilcox

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B4.5-d B4.5.1 B4.5.2 B4.5.3 B4.5.4	Core barrel assembly Core barrel cylinder (top and bottom flange) Lower internals assembly-to-core barrel bolts Core barrel-to-thermal shield bolts Baffle plates and formers	CB cylinder: type 304 forging; CB bolts: A-286, X-750; baffle plates and formers: type 304 stainless steel	Chemically treated borated water up to 340°C (644°F) neutron fluence of greater than 10^{17} n/cm^2 (E>1 MeV)	Loss of fracture toughness/ Neutron irradiation embrittlement, void swelling	Chapter XI.M16, "PWR Vessel Internals"	No
B4.5-e B4.5.2 B4.5.3	Core barrel assembly Lower internals assembly-to-core barrel bolts Core barrel-to-thermal shield bolts	A-286, X-750	Chemically treated borated water up to 340°C (644°F)	Loss of preload/ Stress relaxation	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M14, "Loose Part Monitoring"	No

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B4. Reactor Vessel Internals (PWR) – Babcock and Wilcox

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B4.5-f B4.5.1 B4.5.2 B4.5.3 B4.5.4 B4.5.5	Core barrel assembly Core barrel cylinder (top and bottom flange) Lower internals assembly-to-core barrel bolts Core barrel-to-thermal shield bolts Baffle plates and formers Baffle/former bolts and screws	CB cylinder: type 304 forging; CB bolts: A-286, X-750; baffle plates and formers: type 304 stainless steel; baffle/ former bolts and screws: Gr. B-8 stainless steel	Chemically treated borated water up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	For components for which a fatigue analysis has been performed for the 40-year period, fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes, TLAA

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B4.5-g B4.5.5	Core barrel assembly Baffle/former bolts and screws	Gr. B-8 stainless steel	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking, irradiation-assisted stress corrosion cracking	A plant-specific aging management program is to be evaluated. Historically the VT-3 visual examinations have not identified baffle/former bolt cracking because cracking occurs at the juncture of the bolt head and shank, which is not accessible for visual inspection. However, recent UT examinations of the baffle/former bolts have identified cracking in several plants. The industry is currently addressing the issue of baffle bolt cracking in the PWR Materials Reliability Project, Issues Task Group (ITG) activities to determine, develop, and implement the necessary steps and plans to manage the applicable aging effects on a plant-specific basis.	Yes, plant specific
B4.5-h B4.5.5	Core barrel assembly Baffle/former bolts and screws	Gr. B-8 stainless steel	Chemically treated borated water up to 340°C (644°F)	Changes in dimensions/ Void swelling	A plant-specific aging management program is to be evaluated. The applicant is to provide a plant-specific AMP or participate in industry programs to investigate aging effects and determine appropriate AMP. Otherwise, the applicant is to provide the basis for concluding that void swelling is not an issue for the component.	Yes, plant specific
B4.5-i B4.5.5	Core barrel assembly Baffle/former bolts and screws	Gr. B-8 stainless steel	Chemically treated borated water up to 340°C fluence $>10^{17}$ n/cm^2 (E >1 MeV)	Loss of fracture toughness/ Neutron irradiation embrittlement, void swelling	A plant-specific aging management program is to be evaluated.	Yes, plant specific

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B4. Reactor Vessel Internals (PWR) – Babcock and Wilcox

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B4.5-j B4.5.5	Core barrel assembly Baffle/former bolts and screws	Gr. B-8 stainless steel	Chemically treated borated water up to 340°C (644°F)	Loss of preload/ Stress relaxation	A plant-specific aging management program is to be evaluated. Visual inspection (VT-3) is to be augmented to detect relevant conditions of stress relaxation because only the heads of the baffle/former bolts are visible, and a plant-specific aging management program is thus required.	Yes, plant specific
B4.6-a B4.6.1 B4.6.2 B4.6.4 B4.6.5 B4.6.6 B4.6.8 B4.6.9 B4.6.10 B4.6.11	Lower grid assembly Lower grid rib section Fuel assembly support pads Lower grid flow dist. plate Orifice plugs Lower grid and shell forgings Guide blocks Shock pads Support post pipes Incore guide tube spider castings	Type 304 stainless steel, cast austenitic stainless steel (CASS)	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M16, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
B4.6-b B4.6.3 B4.6.7 B4.6.8 B4.6.9	Lower grid assembly Lower grid rib-to-shell forging screws Lower internals assembly-to-thermal shield bolts Guide blocks and bolts Shock pads and bolts	Lower internal assembly-to-thermal shield bolts: A-286, X-750; Other bolts and screws: Gr. B-8 stainless steel	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M16, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No

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B4. Reactor Vessel Internals (PWR) – Babcock and Wilcox

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B4.6-c B4.6.1 B4.6.2 B4.6.3 B4.6.4 B4.6.5 B4.6.6 B4.6.7 B4.6.8 B4.6.9 B4.6.10 B4.6.11	Lower grid assembly Lower grid rib section Fuel assembly support pads Lower grid rib-to-shell forging screws Lower grid flow dist. plate Orifice plugs Lower grid and shell forgings Lower internals assembly-to-thermal shield bolts Guide blocks and bolts Shock pads and bolts Support post pipes Incore guide tube spider castings	Lower internals assembly-to-thermal shield bolts: A-286, X-750; other bolts and screws: Gr. B-8 stainless steel; spider castings: cast austenitic stainless steel	Chemically treated borated water up to 340°C (644°F)	Changes in dimensions/ Void swelling	A plant-specific aging management program is to be evaluated. The applicant is to provide a plant-specific AMP or participate in industry programs to investigate aging effects and determine appropriate AMP. Otherwise, the applicant is to provide the basis for concluding that void swelling is not an issue for the component.	Yes, plant specific

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B4.6-d B4.6.1 B4.6.2 B4.6.3 B4.6.4 B4.6.5 B4.6.6 B4.6.7 B4.6.8 B4.6.9 B4.6.10	Lower grid assembly Lower grid rib section Fuel assembly support pads Lower grid rib-to-shell forging screws Lower grid flow dist. plate Orifice plugs Lower grid and shell forgings Lower internals assembly-to-thermal shield bolts Guide blocks and bolts Shock pads and bolts Support post pipes	Type 304 stainless steel , lower internals assembly-to-thermal shield bolts: A-286, X-750; other bolts and screws: Gr. B-8 stainless steel	Chemically treated borated water up to 340°C (644°F) neutron fluence $>10^{17}$ n/cm^2 (E >1 MeV)	Loss of fracture toughness/ Neutron irradiation embrittlement, void swelling	Chapter XI.M16, "PWR Vessel Internals"	No
B4.6-e B4.6.11	Lower grid assembly Incore guide tube spider castings	Cast austenitic stainless steel (CASS)	Chemically treated borated water up to 340°C (644°F)	Loss of fracture toughness/ Thermal aging and neutron irradiation embrittlement, void swelling	Chapter XI.M13, "Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)"	No

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B4. Reactor Vessel Internals (PWR) – Babcock and Wilcox

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B4.6-f B4.6.1 B4.6.2 B4.6.3 B4.6.4 B4.6.5 B4.6.6 B4.6.7 B4.6.8 B4.6.9 B4.6.10 B4.6.11	Lower grid assembly Lower grid rib section Fuel assembly support pads Lower grid rib-to-shell forging screws Lower grid flow dist. plate Orifice plugs Lower grid and shell forgings Lower internals assembly-to-thermal shield bolts Guide blocks and bolts Shock pads and bolts Support post pipes Incore guide tube spider castings	Type 304 stainless steel; lower internals assembly-to-thermal shield bolts: A-286, X-750; other bolts and screws: Gr. B-8 stainless steel; spider castings: CASS	Chemically treated borated water up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	For components for which a fatigue analysis has been performed for the 40-year period, fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes, TLAA
B4.6-g B4.6.3 B4.6.7	Lower grid assembly Lower grid rib-to-shell forging screws Lower internals assembly-to-thermal shield bolts	Shell forging screws: Gr. B-8 stainless steel; thermal shield bolts: A-286, X-750	Chemically treated borated water up to 340°C (644°F)	Loss of preload/ Stress relaxation	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M14, "Loose Part Monitoring"	No
B4.6-h B4.6.2 B4.6.8	Lower grid assembly Fuel assembly support pads Guide blocks	Type 304 stainless steel	Chemically treated borated water up to 340°C (644°F)	Loss of material/ Wear	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components	No

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B4. Reactor Vessel Internals (PWR) – Babcock and Wilcox

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B4.7-a B4.7.1 B4.7.3 B4.7.4	Flow distributor assembly Flow distributor head and flange Incore guide support plate Clamping ring	Type 304 stainless steel	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M16, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
B4.7-b B4.7.2	Flow distributor assembly Shell forging-to-flow distributor bolts	A-286, X-750	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M16, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
B4.7-c B4.7.1 B4.7.2 B4.7.3 B4.7.4	Flow distributor assembly Flow distributor head and flange Shell forging-to-flow distributor bolts Incore guide support plate Clamping ring	Type 304 stainless steel; bolts: A-286, X-750	Chemically treated borated water up to 340°C (644°F)	Changes in dimensions/ Void swelling	A plant-specific aging management program is to be evaluated. The applicant is to provide a plant-specific AMP or participate in industry programs to investigate aging effects and determine appropriate AMP. Otherwise, the applicant is to provide the basis for concluding that void swelling is not an issue for the component.	Yes, plant specific
B4.7-d B4.7.1 B4.7.2 B4.7.3 B4.7.4	Flow distributor assembly Flow distributor head and flange Shell forging-to-flow distributor bolts Incore guide support plate Clamping ring	Type 304 stainless steel ; bolts: A-286, X-750	Chemically treated borated water up to 340°C (644°F) neutron fluence $>10^{17}$ n/cm^2 (E >1 MeV)	Loss of fracture toughness/ Neutron irradiation embrittlement, void swelling	Chapter XI.M16, "PWR Vessel Internals"	No

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B4. Reactor Vessel Internals (PWR) – Babcock and Wilcox

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B4.7-e B4.7.2	Flow distributor assembly Shell forging to flow distributor bolts	A-286, X-750	Chemically treated borated water up to 340°C (644°F)	Loss of preload/ Stress relaxation	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M14, "Loose Part Monitoring"	No
B4.8-a	Thermal shield	Stainless steel	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M16, "PWR Vessel Internals," and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
B4.8-b	Thermal shield	Stainless steel	Chemically treated borated water up to 340°C (644°F)	Changes in dimensions/ Void swelling	A plant-specific aging management program is to be evaluated. The applicant is to provide a plant-specific AMP or participate in industry programs to investigate aging effects and determine appropriate AMP. Otherwise, the applicant is to provide the basis for concluding that void swelling is not an issue for the component.	Yes, plant specific
B4.8-c	Thermal shield	Stainless steel	Chemically treated borated water up to 340°C fluence $>10^{17}$ n/cm^2 (E >1 MeV)	Loss of fracture toughness/ Neutron irradiation embrittlement, void swelling	Chapter XI.M16, "PWR Vessel Internals"	No

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C1. REACTOR COOLANT PRESSURE BOUNDARY (BOILING WATER REACTOR)

C1.1 Piping and Fittings

C1.1.1 Main Steam
C1.1.2 Feedwater
C1.1.3 High Pressure Coolant Injection (HPCI) System
C1.1.4 Reactor Core Isolation Cooling (RCIC) System
C1.1.5 Recirculation
C1.1.6 Residual Heat Removal (RHR) System
C1.1.7 Low Pressure Coolant Injection (LPCI) System
C1.1.8 Low Pressure Core Spray (LPCS) System
C1.1.9 High Pressure Core Spray (HPCS) System
C1.1.10 Lines to Isolation Condenser
C1.1.11 Lines to Reactor Water Cleanup (RWC) and Standby Liquid Control (SLC) Systems
C1.1.12 Steam Line to HPCI and RCIC Pump Turbine
C1.1.13 Small Bore Piping Less than NPS 4

C1.2 Recirculation Pump

C1.2.1 Casing
C1.2.2 Cover
C1.2.3 Seal Flange
C1.2.4 Closure Bolting

C1.3 Valves

C1.3.1 Body
C1.3.2 Bonnet
C1.3.3 Seal Flange
C1.3.4 Closure Bolting

C1.4 Isolation Condenser

C1.4.1 Tubing
C1.4.2 Tubesheet
C1.4.3 Channel Head
C1.4.4 Shell

C1. REACTOR COOLANT PRESSURE BOUNDARY (BOILING WATER REACTOR)

Systems, Structures, and Components

This section comprises the boiling water reactor (BWR) primary coolant pressure boundary and consists of the reactor coolant recirculation system and portions of other systems connected to the pressure vessel extending to the second containment isolation valve or to the first anchor point outside containment. The connected systems include the residual heat removal (RHR), low–pressure core spray (LPCS), high–pressure core spray (HPCS), low–pressure coolant injection (LPCI), high–pressure coolant injection (HPCI), reactor core isolation cooling (RCIC), isolation condenser (IC), reactor water cleanup (RWC), standby liquid control system (SLC), feedwater (FW), and main steam (MS) systems, and the steam line to the HPCI and RCIC pump turbines. Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all systems, structures, and components that comprise the reactor coolant pressure boundary are governed by Group A Quality Standards.

Pump and valve internals perform their intended functions with moving parts or with a change in configuration, or are subject to replacement based on qualified life or specified time period. Therefore, they are not subject to an aging management review, pursuant to 10 CFR 54.21(a)(1).

System Interfaces

The systems that interface with the reactor coolant pressure boundary include the reactor pressure vessel (IV.A1), the emergency core cooling system (V.D2), the standby liquid control system (VII.E2), the reactor water cleanup system (VII.E3), the shutdown cooling system (older plants) (VII.E4), the main steam system (VIII.B2), and the feedwater system (VIII.D2).

IV Reactor Vessel, Internals, and Reactor Coolant System
C1. Reactor Coolant Pressure Boundary (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
C1.1-a C1.1.1 C1.1.12	Piping and fittings Main steam Steam line to HPCI and RCIC pump turbine	Carbon steel SA106-Gr B, SA333-Gr 6, SA155-Gr KCF70	288°C (550°F) steam	Wall thinning/ Flow-accelerated corrosion	Chapter XI.M17, "Flow-Accelerated Corrosion"	No
C1.1-b C1.1.1	Piping and fittings Main steam	Carbon steel SA106-Gr B, SA333-Gr 6, SA155-Gr KCF70	288°C (550°F) steam	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii), and for addressing environmental effects on fatigue. See Chapter X.M1 of this report for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes, TLAA
C1.1-c C1.1.2	Piping and fittings Feedwater	Carbon steel SA106-Gr B, SA333-Gr 6, SA155-Gr KCF70	Up to 225°C, (437°F) reactor coolant water	Wall thinning/ Flow-accelerated corrosion	Chapter XI.M17, "Flow-Accelerated Corrosion"	No

IV **Reactor Vessel, Internals, and Reactor Coolant System**

C1. Reactor Coolant Pressure Boundary (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
C1.1-d C1.1.2	Piping and fittings Feedwater	Carbon steel SA106-Gr B, SA333-Gr 6, SA155-Gr KCF70	Up to 225°C (437°F) reactor coolant water	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii), and for addressing environmental effects on fatigue. See Chapter X.M1 of this report for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes, TLAA
C1.1-e C1.1.3 C1.1.4	Piping and fittings High pressure coolant injection Reactor core isolation cooling	Carbon steel SA106-Gr B, SA333-Gr 6, SA155-Gr KCF70	288°C (550°F) reactor coolant water or steam	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii), and for addressing environmental effects on fatigue. See Chapter X.M1 of this report for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes, TLAA

IV **Reactor Vessel, Internals, and Reactor Coolant System**
C1. Reactor Coolant Pressure Boundary (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
C1.1-f C1.1.5 C1.1.6 C1.1.7 C1.1.8 C1.1.9 C1.1.10 C1.1.11	Piping and fittings Recirculation Residual heat removal Low pressure coolant injection Low pressure core spray High pressure core spray Lines to isolation condenser Lines to reactor water cleanup and standby liquid control systems	Stainless steel (e.g., type 304, 316, or 316NG); cast austenitic stainless steel; nickel alloys (e.g., alloys 600, 182, or 82)	288°C (550°F) reactor coolant water or steam	Crack initiation and growth/ Stress corrosion cracking, inter-granular stress corrosion cracking	Chapter XI.M7, "BWR Stress Corrosion Cracking" and Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515)	No
C1.1-g C1.1.6 C1.1.7 C1.1.8 C1.1.9 C1.1.10 C1.1.11	Piping and fittings Residual heat removal Low pressure coolant injection Low pressure core spray High pressure core spray Lines to isolation condenser Lines to reactor water cleanup and standby liquid control systems	Cast austenitic stainless steel	288°C (550°F) reactor coolant water or steam	Loss of fracture toughness/ Thermal aging embrittlement	Chapter XI.M12, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)"	No

IV Reactor Vessel, Internals, and Reactor Coolant System
C1. Reactor Coolant Pressure Boundary (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
C1.1-h C1.1.5 C1.1.6 C1.1.7 C1.1.8 C1.1.9 C1.1.10 C1.1.11	Piping and fittings Recirculation Residual heat removal Low pressure coolant injection Low pressure core spray High pressure core spray Lines to isolation condenser Lines to reactor water cleanup and standby liquid control systems	Carbon steel, cast austenitic stainless steel, stainless steel	288°C (550°F) reactor coolant water or steam	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii), and for addressing environmental effects on fatigue. See Chapter X.M1 of this report for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes, TLAA

IV Reactor Vessel, Internals, and Reactor Coolant System
C1. Reactor Coolant Pressure Boundary (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
C1.1-i C1.1.13	Piping and fittings Small bore piping less than NPS 4	Stainless steel, carbon steel	288°C (550°F) reactor coolant water	Crack initiation and growth/ Stress corrosion cracking, inter-granular stress corrosion cracking, thermal and mechanical loading	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515) Inspection in accordance with ASME Section XI does not require volumetric examination of pipes less than NPS 4. A plant-specific destructive examination or a nondestructive examination (NDE) that permits inspection of the inside surfaces of the piping is to be conducted to ensure that cracking has not occurred and the component intended function will be maintained during the extended period of operation. The AMPs are to be augmented by verifying that service-induced weld cracking is not occurring in the small-bore piping less than NPS 4, including pipe, fittings, and branch connections. See Chapter XI.M32, "One-Time Inspection" for an acceptable verification method.	Yes, parameters monitored/ inspected and detection of aging effects are to be evaluated

IV Reactor Vessel, Internals, and Reactor Coolant System
C1. Reactor Coolant Pressure Boundary (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
C1.2-a C1.2.1 C1.2.2 C1.2.3	Recirculation pump Casing Cover Seal flange	Cast austenitic stainless steel, stainless steel	288°C (550°F) reactor coolant water	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii), and for addressing environmental effects on fatigue. See Chapter X.M1 of this report for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes, TLAA
C1.2-b C1.2.1	Recirculation pump Casing	Cast austenitic stainless steel	288°C (550°F) reactor coolant water	Crack initiation and growth/ stress corrosion cracking, inter-granular stress corrosion cracking	Chapter XI.M7, "BWR Stress Corrosion Cracking" and Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515)	No
C1.2-c C1.2.1	Recirculation pump Casing	Cast austenitic stainless steel	288°C (550°F) reactor coolant water	Loss of fracture toughness/ Thermal aging embrittlement	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components For pump casings, screening for susceptibility to thermal aging is not required. The ASME Section XI inspection requirements are sufficient for managing the effects of loss of fracture toughness due to thermal aging embrittlement of CASS valve bodies.	No

IV **Reactor Vessel, Internals, and Reactor Coolant System**
C1. Reactor Coolant Pressure Boundary (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
C1.2-d C1.2.3 C1.2.4	Recirculation pump Seal flange Closure bolting	Flange: stainless steel; bolting: high-strength low-alloy steel SA193 Gr. B7	Air with metal temperature up to 288°C (550°F)	Loss of material/ Wear	Chapter XI.M18, "Bolting Integrity"	No
C1.2-e C1.2.4	Recirculation pump Closure bolting	High-strength low-alloy steel SA193 Gr. B7	Air with metal temperature up to 288°C (550°F)	Loss of preload/ Stress relaxation	Chapter XI.M18, "Bolting Integrity"	No
C1.2-f C1.2.4	Recirculation pump Closure bolting	High-strength low-alloy steel SA193 Gr. B7	Air with metal temperature up to 288°C (550°F)	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation; check Code limits for allowable cycles (less than 7000 cycles) of thermal stress range. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes, TLAA
C1.3-a C1.3.1	Valves (check, control, hand, motor-operated, relief, and containment isolation) Body	Carbon steel	288°C (550°F) reactor coolant water	Wall thinning/ Flow-accelerated corrosion	Chapter XI.M17, "Flow-Accelerated Corrosion"	No

IV Reactor Vessel, Internals, and Reactor Coolant System
C1. Reactor Coolant Pressure Boundary (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
C1.3-b C1.3.1 C1.3.2	Valves (check, control, hand, motor-operated, relief, and containment isolation) Body Bonnet	Cast austenitic stainless steel	288°C (550°F) reactor coolant water	Loss of fracture toughness/ Thermal aging embrittlement	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components For valve bodies, screening for susceptibility to thermal aging is not required. The ASME Section XI inspection requirements are sufficient for managing the effects of loss of fracture toughness due to thermal aging embrittlement of CASS valve bodies.	No
C1.3-c C1.3.1 C1.3.2	Valves (check, control, hand, motor-operated, relief, and containment isolation) Body Bonnet	Cast austenitic stainless steel, stainless steel	288°C (550°F) reactor coolant water	Crack initiation and growth/ Stress corrosion cracking, inter-granular stress corrosion cracking	Chapter XI.M7, "BWR Stress Corrosion Cracking" and Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515)	No
C1.3-d C1.3.1 C1.3.2 C1.3.3	Valves (check, control, hand, motor-operated, relief, and containment isolation) Body Bonnet Seal flange	Carbon steel, cast austenitic stainless steel, stainless steel	288°C (550°F) reactor coolant water	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii), and for addressing environmental effects on fatigue. See Chapter X.M1 of this report for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes, TLAA

IV **Reactor Vessel, Internals, and Reactor Coolant System**
C1. Reactor Coolant Pressure Boundary (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
C1.3-e C1.3.4	Valves Closure bolting	Flange: carbon steel, stainless steel; bolting: high-strength low-alloy steel	Air with metal temperature up to 288°C (550°F)	Loss of material/ Wear	Chapter XI.M18, "Bolting Integrity"	No
C1.3-f C1.3.4	Valves Closure bolting	High-strength low-alloy steel SA193 GrB7	Air with metal temperature up to 288°C (550°F)	Loss of preload/ Stress relaxation	Chapter XI.M18, "Bolting Integrity"	No
C1.3-g C1.3.4	Valves Closure bolting	High-strength low-alloy steel SA193 GrB7	Air with metal temperature up to 288°C (550°F)	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation; check Code limits for allowable cycles (less than 7000 cycles) of thermal stress range. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes, TLAA

IV **Reactor Vessel, Internals, and Reactor Coolant System**
C1. Reactor Coolant Pressure Boundary (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
C1.4-a C1.4.1 C1.4.2 C1.4.3 C1.4.4	Isolation condenser Tubing Tubesheet Channel head Shell	Tubes: stainless steel; tubesheet: carbon steel, stainless steel; channel head: carbon steel, stainless steel; shell: carbon steel	Tube side: steam; shell side: demineralized water	Crack initiation and growth/ Stress corrosion cracking, cyclic loading	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515) The AMP in Chapter XI.M1 is to be augmented to detect cracking due to stress corrosion cracking and cyclic loading or loss of material due to pitting and crevice corrosion, and verification of the effectiveness of the program is required to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period of operation. An acceptable verification program is to include temperature and radioactivity monitoring of the shell side water, and eddy current testing of tubes.	Yes, plant specific

IV Reactor Vessel, Internals, and Reactor Coolant System
C1. Reactor Coolant Pressure Boundary (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
C1.4-b C1.4.1 C1.4.2 C1.4.3 C1.4.4	Isolation condenser Tubing Tubesheet Channel head Shell	Tubes: stainless steel; tubesheet: carbon steel, stainless steel; channel head: carbon steel, stainless steel; shell: carbon steel	Tube side: steam; shell side: demineralized water	Loss of material/ General, pitting, and crevice corrosion	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515) The AMP in Chapter XI.M1 is to be augmented to detect cracking due to stress corrosion cracking and cyclic loading or loss of material due to pitting and crevice corrosion, and verification of the effectiveness of the program is required to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period of operation. An acceptable verification program is to include temperature and radioactivity monitoring of the shell side water, and eddy current testing of tubes.	Yes, plant specific

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C2. REACTOR COOLANT SYSTEM AND CONNECTED LINES (PRESSURIZED WATER REACTOR)

C2.1 Reactor Coolant System Piping and Fittings

C2.1.1 Cold Leg
C2.1.2 Hot Leg
C2.1.3 Surge Line
C2.1.4 Spray Line
C2.1.5 Small-Bore RCS Piping, Fittings, and Branch Connections Less than NPS 4

C2.2 Connected Systems Piping and Fittings

C2.2.1 Residual Heat Removal (RHR) or Low Pressure Injection System (Decay Heat Removal [DHR]/ Shutdown System)
C2.2.2 Core Flood System (CFS)
C2.2.3 High Pressure Injection System (Makeup & Letdown Functions)
C2.2.4 Chemical and Volume Control System
C2.2.5 Sampling System
C2.2.6 Drains and Instrument Lines
C2.2.7 Nozzles and Safe Ends
C2.2.8 Small-Bore Piping, Fittings, and Branch Connections Less than NPS 4 in Connected Systems

C2.3 Reactor Coolant Pump

C2.3.1 Casing
C2.3.2 Cover
C2.3.3 Closure Bolting

C2.4 Valves (Check, Control, Hand, Motor-Operated, Relief, and Containment Isolation)

C2.4.1 Body
C2.4.2 Bonnet
C2.4.3 Closure Bolting

C2.5 Pressurizer

C2.5.1 Shell/Heads
C2.5.2 Spray Line Nozzle
C2.5.3 Surge Line Nozzle
C2.5.4 Spray Head
C2.5.5 Thermal Sleeves
C2.5.6 Instrument Penetrations
C2.5.7 Safe Ends
C2.5.8 Manway and Flanges
C2.5.9 Manway and Flange Bolting
C2.5.10 Heater Sheaths and Sleeves
C2.5.11 Support Keys, Skirt, and Shear Lugs
C2.5.12 Integral Support

C2.6 Pressurizer Relief Tank

C2.6.1 Tank Shell and Heads
C2.6.2 Flanges and Nozzles

C2. REACTOR COOLANT SYSTEM AND CONNECTED LINES (PRESSURIZED WATER REACTOR)

Systems, Structures, and Components

This section comprises the pressurized water reactor (PWR) primary coolant pressure boundary and consists of the reactor coolant system and portions of other connected systems generally extending up to and including the second containment isolation valve or to the first anchor point and including the containment isolation valves, the reactor coolant pump, valves, pressurizer, and the pressurizer relief tank. The connected systems include the residual heat removal (RHR) or low pressure injection system, high pressure injection system, sampling system, and the small-bore piping. With respect to other systems such as the core flood spray (CFS) or the safety injection tank (SIT) and the chemical and volume control system (CVCS), the isolation valves associated with the boundary between ASME Code class 1 and 2 are located inside the containment. Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," and with the exception of the pressurizer relief tank, which is governed by Group B Quality Standards, all systems, structures, and components that comprise the reactor coolant system are governed by Group A Quality Standards. The recirculating pump seal water heat exchanger is discussed in V.D1.

Pump and valve internals perform their intended functions with moving parts or with a change in configuration, or are subject to replacement based on qualified life or specified time period. Therefore, they are not subject to an aging management review, pursuant to 10 CFR 54.21(a)(1).

System Interfaces

The systems that interface with the reactor coolant pressure boundary include the reactor pressure vessel (IV.A2), the steam generators (IV.D1 and IV.D2), the emergency core cooling system (V.D1), and the chemical and volume control system (VII.E1).

IV Reactor Vessel, Internals, and Reactor Coolant System
C2. Reactor Coolant System and Connected Lines (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
C2.1-a C2.1.1 C2.1.2	Reactor coolant system piping and fittings Cold leg Hot leg	Stainless steel, cast austenitic stainless steel, carbon steel with stainless steel cladding	Chemically treated borated water up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii), and for addressing environmental effects on fatigue. See Chapter X.M1 of this report for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes, TLAA
C2.1-b C2.1.3 C2.1.4	Reactor coolant system piping and fittings Surge line Spray line	Surge line: stainless steel, cast austenitic stainless steel; spray line: stainless steel	Chemically treated borated water up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii), and for addressing environmental effects on fatigue. See Chapter X.M1 of this report for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes, TLAA

IV Reactor Vessel, Internals, and Reactor Coolant System
C2. Reactor Coolant System and Connected Lines (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
C2.1-c C2.1.1 C2.1.2 C2.1.3 C2.1.4	Reactor coolant system piping and fittings Cold leg Hot leg Surge line Spray line	Stainless steel, stainless steel cladding on carbon steel	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking (stainless steel piping), cyclic loading	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
C2.1-d C2.1.1 C2.1.2	Reactor coolant system piping and fittings Cold leg Hot leg (external surfaces)	Carbon steel	Air, leaking chemically treated borated water	Loss of material/ Boric acid corrosion of external surfaces	Chapter XI.M10, "Boric Acid Corrosion"	No
C2.1-e C2.1.1 C2.1.2 C2.1.3	Reactor coolant system piping and fittings Cold leg Hot leg Surge line	Cast austenitic stainless steel	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking	Monitoring and control of primary water chemistry in accordance with the guidelines in EPRI TR-105714 (Rev. 3 or later revisions or update) minimize the potential of SCC, and material selection according to the NUREG-0313, Rev. 2 guidelines of ≤0.035% C and ≥7.5% ferrite has reduced susceptibility to SCC. For CASS components that do not meet either one of the above guidelines, a plant-specific aging management program is to be evaluated. The program is to include (a) adequate inspection methods to ensure detection of cracks, and (b) flaw evaluation methodology for CASS components that are susceptible to thermal aging embrittlement.	Yes, plant specific
C2.1-f C2.1.1 C2.1.2 C2.1.3	Reactor coolant system piping and fittings Cold-leg Hot-leg Surge line	Cast austenitic stainless steel	Chemically treated borated water up to 340°C (644°F)	Loss of fracture toughness/ Thermal aging embrittlement	Chapter XI.M12, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)"	No

IV **Reactor Vessel, Internals, and Reactor Coolant System**
C2. Reactor Coolant System and Connected Lines (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
C2.1-g C2.1.5	Reactor coolant system piping and fittings RCS piping, fittings, and branch connections less than NPS 4	Stainless steel	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking, thermal and mechanical loading	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714 Inspection in accordance with ASME Section XI does not require volumetric examination of pipes less than NPS 4. A plant-specific destructive examination or a nondestructive examination (NDE) that permits inspection of the inside surfaces of the piping is to be conducted to ensure that cracking has not occurred and the component intended function will be maintained during the extended period of operation. The AMPs are to be augmented by verifying that service-induced weld cracking is not occurring in the small-bore piping less than NPS 4, including pipe, fittings, and branch connections. See Chapter XI.M32, "One-Time Inspection" for an acceptable verification method.	Yes, parameters monitored/ inspected and detection of aging effects are to be evaluated

IV Reactor Vessel, Internals, and Reactor Coolant System
C2. Reactor Coolant System and Connected Lines (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
C2.2-a C2.2.1 C2.2.2 C2.2.3 C2.2.4	Connected systems piping and fittings Residual heat removal Core flood system High pressure injection system Chemical and volume control system	Stainless steel	Chemically treated borated water up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii), and for addressing environmental effects on fatigue. See Chapter X.M1 of this report for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes, TLAA
C2.2-b C2.2.5 C2.2.6	Connected systems piping and fittings Sampling system Drains and instrument lines	Carbon steel with stainless steel cladding, stainless steel	Chemically treated borated water up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii), and for addressing environmental effects on fatigue. See Chapter X.M1 of this report for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes, TLAA

IV Reactor Vessel, Internals, and Reactor Coolant System
C2. Reactor Coolant System and Connected Lines (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
C2.2-c C2.2.7	Connected systems piping and fittings Nozzles and safe ends	Stainless steel, cast austenitic stainless steel	Chemically treated borated water up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii), and for addressing environmental effects on fatigue. See Chapter X.M1 of this report for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes, TLAA
C2.2-d C2.2.5 C2.2.6	Connected systems piping and fittings Sampling system Drains and instrument lines (external surfaces)	Carbon steel	Air, leaking chemically treated borated water	Loss of material/ Boric acid corrosion of external surfaces	Chapter XI.M10, "Boric Acid Corrosion"	No
C2.2-e C2.2.7	Connected systems piping and fittings Nozzles and safe ends	Cast austenitic stainless steel	Chemically treated borated water up to 340°C (644°F)	Loss of fracture toughness/ Thermal aging embrittlement	Chapter XI.M12, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)"	No

IV Reactor Vessel, Internals, and Reactor Coolant System
C2. Reactor Coolant System and Connected Lines (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
C2.2-f C2.2.1 C2.2.2 C2.2.3 C2.2.4 C2.2.5 C2.2.6 C2.2.7	Connected systems piping and fittings Residual heat removal Core flood system High pressure injection system Chemical and volume control system Sampling system Drains and instrument lines Nozzles and safe ends	Stainless steel	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
C2.2-g C2.2.7	Connected systems piping and fittings Nozzles and safe ends	Cast austenitic stainless steel	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking	Monitoring and control of primary water chemistry in accordance with the guidelines in EPRI TR-105714 (Rev. 3 or later revisions or update) minimize the potential of SCC, and material selection according to the NUREG-0313, Rev. 2 guidelines of ≤0.035% C and ≥7.5% ferrite has reduced susceptibility to SCC. For CASS components that do not meet either one of the above guidelines, a plant-specific aging management program is to be evaluated. The program is to include (a) adequate inspection methods to ensure detection of cracks, and (b) flaw evaluation methodology for CASS components that are susceptible to thermal aging embrittlement.	Yes, plant specific

IV Reactor Vessel, Internals, and Reactor Coolant System
C2. Reactor Coolant System and Connected Lines (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
C2.2-h C2.2.8	Connected systems piping and fittings Small-bore piping, fittings, and branch connections less than NPS 4 in connected systems	Stainless steel, carbon steel	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking, thermal and mechanical loading	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714 Inspection in accordance with ASME Section XI does not require volumetric examination of pipes less than NPS 4. A plant-specific destructive examination or a nondestructive examination (NDE) that permits inspection of the inside surfaces of the piping is to be conducted to ensure that cracking has not occurred and the component intended function will be maintained during the period of extended operation. The AMPs are to be augmented by verifying that service-induced weld cracking is not occurring in the small-bore piping less than NPS 4, including pipe, fittings, and branch connections. See Chapter XI.M32, "One-Time Inspection" for an acceptable verification method.	Yes, parameters monitored/ inspected and detection of aging effects are to be evaluated

IV Reactor Vessel, Internals, and Reactor Coolant System
C2. Reactor Coolant System and Connected Lines (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
C2.3-a C2.3.1 C2.3.2	Reactor coolant pump Casing Cover	Bowl: cast austenitic stainless steel CF-8 or CF-8M, carbon steel with stainless steel cladding; cover: stainless steel	Chemically treated borated water up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii), and for addressing environmental effects on fatigue. See Chapter X.M1 of this report for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes, TLAA
C2.3-b C2.3.1	Reactor coolant pump Casing	Cast austenitic stainless steel CF-8 or CF-8M, carbon steel with stainless steel cladding	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking	Monitoring and control of primary water chemistry in accordance with the guidelines in EPRI TR-105714 (Rev. 3 or later revisions or update) minimize the potential of SCC, and material selection according to the NUREG-0313, Rev. 2 guidelines of ≤0.035% C and ≥7.5% ferrite has reduced susceptibility to SCC. For CASS components that do not meet either one of the above guidelines, see Chapter XI.M1, "ASME Section XI, Subsections IWB, IWC, and IWD."	No
C2.3-c C2.3.1	Reactor coolant pump Casing	Cast austenitic stainless steel CF-8 or CF-8M	Chemically treated borated water up to 340°C (644°F)	Loss of fracture toughness/ Thermal aging embrittlement	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components For pump casings, screening for susceptibility to thermal aging is not required.	No

IV **Reactor Vessel, Internals, and Reactor Coolant System**
C2. Reactor Coolant System and Connected Lines (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
C2.3-d C2.3.3	Reactor coolant pump Closure bolting	High-strength low-alloy steel SA540 GrB23, SA193 GrB7	Air with metal temperature up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii). See Chapter X.M1 of this report for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes, TLAA
C2.3-e C2.3.3	Reactor coolant pump Closure bolting	High-strength low-alloy steel SA540 GrB23, SA193 GrB7	Air, leaking chemically treated borated water or steam up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking	Chapter XI.M18, "Bolting Integrity"	No
C2.3-f C2.3.3	Reactor coolant pump Closure bolting	High-strength low-alloy steel SA540 GrB23, SA193 GrB7	Air, leaking chemically treated borated water or steam up to 340°C (644°F)	Loss of material/ Boric acid corrosion of external surfaces	Chapter XI.M10, "Boric Acid Corrosion"	No
C2.3-g C2.3.3	Reactor coolant pump Closure bolting	High-strength low-alloy steel SA540 GrB23, SA193 GrB7	Air with metal temperature up to 340°C (644°F)	Loss of preload/ Stress relaxation	Chapter XI.M18, "Bolting Integrity"	No

IV **Reactor Vessel, Internals, and Reactor Coolant System**
C2. Reactor Coolant System and Connected Lines (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
C2.4-a C2.4.1 C2.4.2	Valves (check, control, hand, motor operated, relief, and containment isolation) Body Bonnet	Cast austenitic stainless steel CF-8M, SA182 F316, SA582 Type 416	Chemically treated borated water up to 340ºC (644ºF)	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii). See Chapter X.M1 of this report, for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes, TLAA
C2.4-b C2.4.1	Valves (check, control, hand, motor operated, relief, and containment isolation) Body	Cast austenitic stainless steel CF-8M	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking	Monitoring and control of primary water chemistry in accordance with the guidelines in EPRI TR-105714 (Rev. 3 or later revisions or update) minimize the potential of SCC, and material selection according to the NUREG-0313, Rev. 2 guidelines of ≤0.035% C and ≥7.5% ferrite has reduced susceptibility to SCC. For CASS components that do not meet either one of the above guidelines, see Chapter XI.M1, "ASME Section XI, Subsections IWB, IWC, and IWD."	No
C2.4-c C2.4.1	Valves (check, control, hand, motor operated, relief, and containment isolation) Body	Cast austenitic stainless steel CF-8M	Chemically treated borated water up to 340°C (644°F)	Loss of fracture toughness/ Thermal aging embrittlement	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components For valve body, screening for susceptibility to thermal aging is not required.	No

IV Reactor Vessel, Internals, and Reactor Coolant System
C2. Reactor Coolant System and Connected Lines (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
C2.4-d C2.4.3	Valves Closure bolting	High-strength low-alloy steel, stainless steel	Air with metal temperature up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii). See Chapter X.M1 of this report for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes, TLAA
C2.4-e C2.4.3	Valves Closure bolting	High-strength low-alloy steel, stainless steel	Air, leaking chemically treated borated water or steam	Crack initiation and growth/ Stress corrosion cracking	Chapter XI.M18, "Bolting Integrity"	No
C2.4-f C2.4.3	Valves Closure bolting	High-strength low-alloy steel	Air, leaking chemically treated borated water or steam	Loss of material/ Boric acid corrosion of external surfaces	Chapter XI.M10, "Boric Acid Corrosion"	No
C2.4-g C2.4.3	Valves Closure bolting	High-strength low-alloy steel, stainless steel	Air with metal temperature up to 340°C (644°F)	Loss of preload/ Stress relaxation	Chapter XI.M18, "Bolting Integrity"	No

IV **Reactor Vessel, Internals, and Reactor Coolant System**
C2. Reactor Coolant System and Connected Lines (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
C2.5-a C2.5.1	Pressurizer Shell/heads	Low-alloy steel with stainless steel or alloy 600 cladding	Chemically treated borated water or saturated steam 290-343°C (554-650°F)	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii), and for addressing environmental effects on fatigue. See Chapter X.M1 of this report for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes, TLAA
C2.5-b C2.5.1	Pressurizer Shell/heads (outer surfaces)	Low-alloy steel	Air, leaking chemically treated borated water or steam up to 340°C (644°F)	Loss of material/ Boric acid corrosion of external surfaces	Chapter XI.M10, "Boric Acid Corrosion"	No

IV **Reactor Vessel, Internals, and Reactor Coolant System**
C2. Reactor Coolant System and Connected Lines (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
C2.5-c C2.5.1	Pressurizer Shell/heads	Low-alloy steel with type 308, 308L, or 309 stainless steel or alloy 82 or 182 cladding	Chemically treated borated water or saturated steam 290-343°C (554-650°F)	Crack initiation and growth/ Stress corrosion cracking, cyclic loading	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714 Cracks in the pressurizer cladding could propagate from cyclic loading into the ferrite base metal and weld metal. However, because the weld metal between the surge nozzle and the vessel lower head is subjected to the maximum stress cycles and the area is periodically inspected as part of the ISI program, the existing AMP is adequate for managing the effect of pressurizer clad cracking.	No
C2.5-d C2.5.2 C2.5.4	Pressurizer Spray line nozzle Spray head	Nozzle: carbon steel or low-alloy steel with stainless steel cladding; spray head: alloy 600, stainless steel, cast austenitic stainless steel	Chemically treated borated water or saturated steam 290-343°C (554-650°F)	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii), and for addressing environmental effects on fatigue. See Chapter X.M1 of this report for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes, TLAA

IV **Reactor Vessel, Internals, and Reactor Coolant System**
C2. Reactor Coolant System and Connected Lines (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
C2.5-e C2.5.3	Pressurizer Surge line nozzle	Carbon steel or low-alloy steel with stainless steel cladding, cast austenitic stainless steel	Chemically treated borated water up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii), and for addressing environmental effects on fatigue. See Chapter X.M1 of this report for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes, TLAA
C2.5-f C2.5.5 C2.5.6 C2.5.7	Pressurizer Thermal sleeves Instrument penetrations Safe ends	Thermal sleeves: alloy 600; penetrations: Alloy 600, stainless steel; safe ends: stainless steel	Chemically treated borated water or saturated steam 290-343°C (554-650°F)	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii), and for addressing environmental effects on fatigue. See Chapter X.M1 of this report for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes, TLAA

IV Reactor Vessel, Internals, and Reactor Coolant System
C2. Reactor Coolant System and Connected Lines (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
C2.5-g C2.5.2 C2.5.3 C2.5.6	Pressurizer Spray line nozzle Surge line nozzle Instrument penetrations	Carbon steel or low-alloy steel with stainless steel cladding; or stainless steel	Chemically treated borated water or saturated steam 290-343°C (554-650°F)	Crack initiation and growth/ Stress corrosion cracking	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714 Cracks in the pressurizer cladding could propagate from cyclic loading into the ferrite base metal and weld metal. However, because the weld metal between the surge nozzle and the vessel lower head is subjected to the maximum stress cycles and the area is periodically inspected as part of the ISI program, the existing AMP is adequate for managing the effect of pressurizer clad cracking.	No
C2.5-h C2.5.7	Pressurizer Safe ends	Stainless steel	Chemically treated borated water or saturated steam 290-343°C (554-650°F)	Crack initiation and growth/ Stress corrosion cracking	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No

IV Reactor Vessel, Internals, and Reactor Coolant System
C2. Reactor Coolant System and Connected Lines (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
C2.5-i C2.5.3	Pressurizer Surge line nozzle	Cast austenitic stainless steel	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking	Monitoring and control of primary water chemistry in accordance with the guidelines in EPRI TR-105714 (Rev. 3 or later revisions or update) minimize the potential of SCC, and material selection according to the NUREG-0313, Rev. 2 guidelines of ≤0.035% C and ≥7.5% ferrite has reduced susceptibility to SCC. For CASS components that do not meet either one of the above guidelines, a plant-specific aging management program is to be evaluated. The program is to include (a) adequate inspection methods to ensure detection of cracks, and (b) flaw evaluation methodology for CASS components that are susceptible to thermal aging embrittlement.	Yes, plant specific
C2.5-j C2.5.4	Pressurizer Spray head	Alloy 600, stainless steel, cast austenitic stainless steel	Chemically treated borated water or saturated steam 290-343°C (554-650°F)	Crack initiation and growth/ Primary water stress corrosion cracking, stress corrosion cracking	A plant-specific aging management program is to be evaluated.	Yes, plant specific
C2.5-k C2.5.6	Pressurizer Instrument penetrations	Alloy 600	Chemically treated borated water or saturated steam 290-343°C (554-650°F)	Crack initiation and growth/ Primary water stress corrosion cracking (PWSCC)	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components, Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714 and the applicant is to provide a plant-specific AMP or participate in industry programs to determine appropriate AMP for PWSCC of Inconel 182 weld.	Yes, an AMP for PWSCC of Inconel 182 weld is to be evaluated

IV Reactor Vessel, Internals, and Reactor Coolant System
C2. Reactor Coolant System and Connected Lines (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
C2.5-l C2.5.3 C2.5.4	Pressurizer Surge line nozzle Spray head	Cast austenitic stainless steel	Chemically treated borated water or saturated steam 290-343°C (554-650°F)	Loss of fracture toughness/ Thermal aging embrittlement	Chapter XI.M12, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)"	No
C2.5-m C2.5.8	Pressurizer Manway and flanges	Low-alloy steel with type 308, 308L, or 309 stainless steel cladding; or alloy 82 or 182 cladding	Chemically treated borated water or saturated steam 290-343°C (554-650°F)	Crack initiation and growth/ Stress corrosion cracking, primary water stress corrosion cracking	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
C2.5-n C2.5.9	Pressurizer Manway and flange bolting	High-strength low-alloy steel	Air, leaking chemically treated borated water or steam up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking	Chapter XI.M18, "Bolting Integrity"	No
C2.5-o C2.5.8 C2.5.9	Pressurizer Manway and flanges Manway and flange bolting	Low-alloy steel, High-strength low-alloy steel	Air, leaking chemically treated borated water or steam up to 340°C (644°F)	Loss of material/ Boric acid corrosion of external surfaces	Chapter XI.M10, "Boric Acid Corrosion"	No
C2.5-p C2.5.9	Pressurizer Manway and flange bolting	High-strength low-alloy steel	Air, leaking chemically treated borated water or steam up to 340°C (644°F)	Loss of preload/ Stress relaxation	Chapter XI.M18, "Bolting Integrity"	No

IV Reactor Vessel, Internals, and Reactor Coolant System
C2. Reactor Coolant System and Connected Lines (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
C2.5-q C2.5.10	Pressurizer Heater sheaths and sleeves	Alloy 600 or austenitic stainless steel	Chemically treated borated water up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii), and for addressing environmental effects on fatigue. See Chapter X.M1 of this report for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes, TLAA
C2.5-r C2.5.10	Pressurizer Heater sheaths and sleeves	Austenitic stainless steel	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
C2.5-s C2.5.10	Pressurizer Heater sheaths and sleeves	Alloy 600	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Primary water stress corrosion cracking (PWSCC)	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components, Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714 and the applicant is to provide a plant-specific AMP or participate in industry programs to determine appropriate AMP for PWSCC of Inconel 182 weld.	Yes, AMP for PWSCC of Inconel 182 weld is to be evaluated

IV **Reactor Vessel, Internals, and Reactor Coolant System**
C2. Reactor Coolant System and Connected Lines (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
C2.5-t C2.5.11	Pressurizer Support keys, skirt, and shear lugs	Carbon steel, low-alloy steel	Air, with metal temperatures up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
C2.5-u C2.5.12	Pressurizer Integral support	Carbon steel	Air, leaking chemically treated borated water	Loss of material/ Boric acid corrosion of external surfaces	Chapter XI.M10, "Boric Acid Corrosion"	No
C2.5-v C2.5.12	Pressurizer Integral support	Carbon steel, stainless steel	Air	Crack initiation and growth/ Cyclic loading	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components	No
C2.5-w C2.5.12	Pressurizer Integral support	Carbon steel, stainless steel	Air	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
C2.6-a C2.6.1 C2.6.2	Pressurizer relief tank Tank shell and heads Flanges and nozzles	Carbon steel with type 304 stainless steel cladding	Chemically treated borated water at 93°C (200°F)	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii). See Chapter X.M1 of this report, for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes, TLAA

IV Reactor Vessel, Internals, and Reactor Coolant System
C2. Reactor Coolant System and Connected Lines (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
C2.6-b C2.6.1	Pressurizer relief tank Tank shell and heads (external surfaces)	Carbon steel	Air, leaking chemically treated borated water at 93°C (200°F)	Loss of material/ Boric acid corrosion of external surfaces	Chapter XI.M10, "Boric Acid Corrosion"	No
C2.6-c C2.6.1 C2.6.2	Pressurizer relief tank Tank shell and heads Flanges and nozzles	Carbon steel with type 304 stainless steel cladding	Chemically treated borated water at 93°C (200°F)	Crack initiation and growth/ Stress corrosion cracking	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 2 components and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No

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D1. STEAM GENERATOR (RECIRCULATING)

D1.1 Pressure Boundary and Structural

- D1.1.1 Top Head
- D1.1.2 Steam Nozzle and Safe End
- D1.1.3 Upper and Lower Shell
- D1.1.4 Transition Cone
- D1.1.5 Feedwater Nozzle and Safe End
- D1.1.6 Feedwater Impingement Plate and Support
- D1.1.7 Secondary Manway and Handhole Bolting
- D1.1.8 Lower Head
- D1.1.9 Primary Nozzles and Safe Ends
- D1.1.10 Instrument Nozzles
- D1.1.11 Primary Manway (Cover and Bolting)

D1.2 Tube Bundle

- D1.2.1 Tubes and Sleeves
- D1.2.2 Tube Support Lattice Bars (Combustion Engineering)
- D1.2.3 Tube Plugs
- D1.2.4 Tube Support Plates

D1.3 Upper Assembly and Separators

- D1.3.1 Feedwater Inlet Ring and Support

D1. STEAM GENERATOR (RECIRCULATING)

Systems, Structures, and Components

This section consists of the recirculating-type steam generators, as found in Westinghouse and Combustion Engineering pressurized water reactors (PWRs), including all internal components and water/steam nozzles and safe ends. Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," the primary water side (tube side) of the steam generator is governed by Group A Quality Standards, and the secondary water side is governed by Group B Quality Standards.

System Interfaces

The systems that interface with the steam generators include the reactor coolant system and connected lines (IV.C2), the containment isolation components (V.C), the main steam system (VIII.B1), the feedwater system (VIII.D1), the steam generator blowdown system (VIII.F), and the auxiliary feedwater system (VIII.G).

IV Reactor Vessel, Internals, and Reactor Coolant System
D1. Steam Generator (Recirculating)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
D1.1-a D1.1.1 D1.1.2	Pressure boundary and structural Top head Steam nozzle and safe end	Low-alloy steel	Up to 300°C (572°F) steam	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes, TLAA
D1.1-b D1.1.3 D1.1.4 D1.1.5 D1.1.6	Pressure boundary and structural Upper and lower shell Transition cone FW nozzle and safe end FW impingement plate and support	Carbon steel, low-alloy steel	Up to 300°C (572°F) secondary-side water chemistry at 5.3-7.2 MPa	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes, TLAA
D1.1-c D1.1.3 D1.1.4	Pressure boundary and structural Upper and lower shell Transition cone	Carbon steel, low-alloy steel	Up to 300°C (572°F) secondary-side water chemistry at 5.3-7.2 MPa	Loss of material/ General, pitting, and crevice corrosion	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 2 components and Chapter XI.M2, "Water Chemistry," for PWR secondary water in EPRI TR-102134 As noted in NRC Information Notice IN 90-04, general and pitting corrosion of the shell exists, the program recommendations may not be sufficient to detect general and pitting corrosion, and additional inspection procedures are to be developed, if required.	Yes, detection of aging effects is to be evaluated
D1.1-d D1.1.2 D1.1.5	Pressure boundary and structural Steam nozzle and safe end FW nozzle and safe end	Carbon steel	Up to 300°C (572°F) steam or secondary-side water chemistry at 5.3-7.2 MPa	Wall thinning/ Flow-accelerated corrosion	Chapter XI.M17, "Flow-Accelerated Corrosion"	No

IV **Reactor Vessel, Internals, and Reactor Coolant System**
D1. Steam Generator (Recirculating)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
D1.1-e D1.1.6	Pressure boundary and structural Feedwater impingement plate and support	Carbon steel	Up to 300°C (572°F) secondary-side water chemistry	Loss of section thickness/ Erosion	A plant-specific aging management program is to be evaluated.	Yes, plant specific
D1.1-f D1.1.7	Pressure boundary and structural Secondary manway and handhole bolting	Low-alloy steel	Air, with metal temperature up to 340°C (644°F)	Loss of preload/ Stress relaxation	Chapter XI.M18, "Bolting Integrity"	No
D1.1-g D1.1.8	Pressure boundary and structural Lower head (external surfaces)	Low-alloy steel	Air, leaking chemically treated borated water or steam up to 340°C (644°F)	Loss of material/ Boric acid corrosion of external surfaces	Chapter XI.M10, "Boric Acid Corrosion"	No
D1.1-h D1.1.8 D1.1.9	Pressure boundary and structural Lower head Primary nozzles and safe ends	Carbon steel with stainless steel cladding, safe ends: stainless steel	Chemically treated borated water up to 340°C (644°F) and 15.2 MPa	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii), and for addressing environmental effects on fatigue. See Chapter X.M1 of this report for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes, TLAA

IV Reactor Vessel, Internals, and Reactor Coolant System
D1. Steam Generator (Recirculating)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
D1.1-i D1.1.9	Pressure boundary and structural Primary nozzles and safe ends	Carbon steel with stainless steel cladding, safe ends: stainless steel (NiCrFe buttering, and stainless steel or NiCrFe weld)	Chemically treated borated water at temperatures up to 340°C (644°F) and 15.2 MPa	Crack initiation and growth/ Stress corrosion cracking, primary water stress corrosion cracking	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
D1.1-j D1.1.10	Pressure boundary and structural Instrument nozzles	Alloy 600	Chemically treated borated water up to 340°C (644°F) and 15.5 MPa	Crack initiation and growth/ Primary water stress corrosion cracking	A plant-specific aging management program is to be evaluated.	Yes, plant specific
D1.1-k D1.1.11	Pressure boundary and structural Primary manway (cover and bolting)	Carbon steel, low-alloy steel	Air, leaking chemically treated borated water and/or steam up to 340°C (644°F)	Loss of material/ Boric acid corrosion of external surfaces	Chapter XI.M10, "Boric Acid Corrosion"	No

IV Reactor Vessel, Internals, and Reactor Coolant System
D1. Steam Generator (Recirculating)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
D1.1-l D1.1.11	Pressure boundary and structural Primary manway (bolting only)	Carbon steel, low-alloy steel	Air, leaking chemically treated borated water and/or steam up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking	Chapter XI.M18, "Bolting Integrity"	No
D1.2-a D1.2.1	Tube bundle Tubes and sleeves	Alloy 600	Chemically treated borated water up to 340°C (644°F) and 15.5 MPa	Crack initiation and growth/ Primary water stress corrosion cracking	Chapter XI.M19, "Steam Generator Tubing Integrity" and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714 All PWR licensees have committed voluntarily to a SG degradation management program described in NEI 97-06; these guidelines are currently under NRC staff review. An AMP based on the recommendations of staff-approved NEI 97-06 guidelines, or other alternate regulatory basis for SG degradation management, is to be developed and incorporated in the plant technical specifications.	Yes, effective-ness of the AMP is to be evaluated

IV Reactor Vessel, Internals, and Reactor Coolant System
D1. Steam Generator (Recirculating)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
D1.2-b D1.2.1	Tube bundle Tubes and sleeves	Alloy 600	Up to 300°C (572°F) secondary-side water chemistry at 5.3-7.2 MPa	Crack initiation and growth/ Outer diameter stress corrosion cracking	Chapter XI.M19, "Steam Generator Tubing Integrity" and Chapter XI.M2, "Water Chemistry," for PWR secondary water in EPRI TR-102134 All PWR licensees have committed voluntarily to a SG degradation management program described in NEI 97-06; these guidelines are currently under NRC staff review. An AMP based on the recommendations of staff-approved NEI 97-06 guidelines, or other alternate regulatory basis for SG degradation management, is to be developed and incorporated in the plant technical specifications.	Yes, effective-ness of the AMP is to be evaluated
D1.2-c D1.2.1	Tube bundle Tubes and sleeves	Alloy 600	Up to 300°C (572°F) secondary-side water chemistry at 5.3-7.2 MPa	Crack initiation and growth/ Intergranular attack	Chapter XI.M19, "Steam Generator Tubing Integrity" and Chapter XI.M2, "Water Chemistry," for PWR secondary water in EPRI TR-102134 All PWR licensees have committed voluntarily to a SG degradation management program described in NEI 97-06; these guidelines are currently under NRC staff review. An AMP based on the recommendations of staff-approved NEI 97-06 guidelines, or other alternate regulatory basis for SG degradation management, is to be developed and incorporated in the plant technical specifications.	Yes, effective-ness of the AMP is to be evaluated

IV Reactor Vessel, Internals, and Reactor Coolant System
D1. Steam Generator (Recirculating)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
D1.2-d D1.2.1	Tube bundle Tubes and sleeves	Alloy 600	ID chemically treated borated water up to 340°C (644°F); OD up to 300°C (572°F) secondary-side water chemistry	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii), and for addressing environmental effects on fatigue. See Chapter X.M1 of this report for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes, TLAA
D1.2-e D1.2.1	Tube bundle Tubes and sleeves	Alloy 600	Up to 300°C (572°F) secondary-side water chemistry at 5.3-7.2 MPa	Loss of section thickness/ Fretting and wear	Chapter XI.M19, "Steam Generator Tubing Integrity" and Chapter XI.M2, "Water Chemistry," for PWR secondary water in EPRI TR-102134 All PWR licensees have committed voluntarily to a SG degradation management program described in NEI 97-06; these guidelines are currently under NRC staff review. An AMP based on the recommendations of staff-approved NEI 97-06 guidelines, or other alternate regulatory basis for SG degradation management, is to be developed and incorporated in the plant technical specifications.	Yes, effectiveness of the AMP is to be evaluated

IV **Reactor Vessel, Internals, and Reactor Coolant System**
D1. Steam Generator (Recirculating)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
D1.2-f D1.2.1	Tube bundle Tubes and sleeves (exposed to phosphate chemistry)	Alloy 600	Up to 300°C (572°F) secondary-side water chemistry at 5.3-7.2 MPa	Loss of material/ Wastage and pitting corrosion	Chapter XI.M19, "Steam Generator Tubing Integrity" and Chapter XI.M2, "Water Chemistry," for PWR secondary water in EPRI TR-102134 All PWR licensees have committed voluntarily to a SG degradation management program described in NEI 97-06; these guidelines are currently under NRC staff review. An AMP based on the recommendations of staff-approved NEI 97-06 guidelines, or other alternate regulatory basis for SG degradation management, is to be developed and incorporated in the plant technical specifications.	Yes, effective-ness of the AMP is to be evaluated

IV Reactor Vessel, Internals, and Reactor Coolant System
D1. Steam Generator (Recirculating)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
D1.2-g D1.2.1	Tube bundle Tubes	Alloy 600	Up to 300°C (572°F) secondary-side water chemistry at 5.3-7.2 MPa	Denting/ Corrosion of carbon steel tube support plate	Chapter XI.M19, "Steam Generator Tubing Integrity" and Chapter XI.M2, "Water Chemistry," for PWR secondary water in EPRI TR-102134. For plants where analyses were completed in response to NRC Bulletin 88-02 "Rapidly Propagating Cracks in SG Tubes," the results of those analyses have to be reconfirmed for the period of license renewal. All PWR licensees have committed voluntarily to a SG degradation management program described in NEI 97-06; these guidelines are currently under NRC staff review. An AMP based on the recommendations of staff-approved NEI 97-06 guidelines, or other alternate regulatory basis for SG degradation management, is to be developed and incorporated in the plant technical specifications.	Yes, effective-ness of the AMP is to be evaluated
D1.2-h D1.2.2	Tube bundle Tube support lattice bars	Carbon steel	Up to 300°C (572°F) secondary-side water chemistry at 5.3-7.2 MPa	Loss of section thickness/ Flow-accelerated corrosion	A plant-specific aging management program is to be evaluated.	Yes, plant specific

IV Reactor Vessel, Internals, and Reactor Coolant System
D1. Steam Generator (Recirculating)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
D1.2-i D1.2.3	Tube bundle Tube plugs (mechanical) (Westinghouse)	Alloy 600, alloy 690	Chemically treated borated water at temperatures up to 340°C (644°F) and 15.5 MPa	Crack initiation and growth/ Primary water stress corrosion cracking	Chapter XI.M19, "Steam Generator Tubing Integrity" and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714 All PWR licensees have committed voluntarily to a SG degradation management program described in NEI 97-06; these guidelines are currently under NRC staff review. An AMP based on the recommendations of staff-approved NEI 97-06 guidelines, or other alternate regulatory basis for SG degradation management, is to be developed and incorporated in the plant technical specifications.	Yes, effective-ness of the AMP is to be evaluated
D1.2-j D1.2.3	Tube bundle Tube plugs (mechanical) (Babcock and Wilcox)	Alloy 600, alloy 690	Chemically treated borated water at temperatures up to 340°C (644°F) and 15.5 MPa	Crack initiation and growth/ Primary water stress corrosion cracking	Chapter XI.M19, "Steam Generator Tubing Integrity" and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714 All PWR licensees have committed voluntarily to a SG degradation management program described in NEI 97-06; these guidelines are currently under NRC staff review. An AMP based on the recommendations of staff-approved NEI 97-06 guidelines or other alternate regulatory basis for SG degradation management, is to be developed and incorporated in the plant technical specifications.	Yes, effective-ness of the AMP is to be evaluated

IV **Reactor Vessel, Internals, and Reactor Coolant System**
D1. Steam Generator (Recirculating)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
D1.2-k D1.2.4	Tube bundle Tube support plates	Carbon steel	Up to 300°C (572°F) secondary-side water chemistry at 5.3-7.2 MPa	Ligament cracking/ Corrosion	Chapter XI.M19, "Steam Generator Tubing Integrity" and Chapter XI.M2, "Water Chemistry," for PWR secondary water in EPRI TR-102134 All PWR licensees have committed voluntarily to a SG degradation management program described in NEI 97-06; these guidelines are currently under NRC staff review. An AMP based on the recommendations of staff-approved NEI 97-06 guidelines, or other alternate regulatory basis for SG degradation management, is to be developed and incorporated in the plant technical specifications.	Yes, effective-ness of the AMP is to be evaluated
D1.3-a D1.3.1	Upper assembly and separators Feedwater inlet ring and support	Carbon steel	Up to 300°C (572°F) secondary-side water chemistry at 5.3-7.2 MPa	Loss of material/ Flow-accelerated corrosion	A plant-specific aging management program is to be evaluated. As noted in Combustion Engineering (CE) Information Notice (IN) 90-04 and NRC IN 91-19 and LER 50-362/90-05-01, this form of degradation has been detected only in certain CE System 80 steam generators.	Yes, plant specific

D2. STEAM GENERATOR (ONCE–THROUGH)

D2.1 Pressure Boundary and Structural

- D2.1.1 Upper and Lower Heads
- D2.1.2 Tube Sheets
- D2.1.3 Primary Nozzles
- D2.1.4 Shell Assembly
- D2.1.5 Feed Water and Auxiliary Feed Water Nozzles and Safe Ends
- D2.1.6 Steam Nozzles and Safe Ends
- D2.1.7 Primary Side Drain Nozzles
- D2.1.8 Secondary Side Nozzles (Vent, Drain, and Instrumentation)
- D2.1.9 Primary Manways (Cover and Bolting)
- D2.1.10 Secondary Manways and Handholes (Cover and Bolting)

D2.2 Tube Bundle

- D2.2.1 Tubes and Sleeves
- D2.2.2 Tube Plugs

D2. STEAM GENERATOR (ONCE-THROUGH)

Systems, Structures, and Components

This section consists of the once-through type steam generators, as found in Babcock & Wilcox pressurized water reactors (PWRs), including all internal components and water/steam nozzles and safe ends. Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," the primary water side (tube side) of the steam generator is governed by Group A Quality Standards, and the secondary water side is governed by Group B Quality Standards.

System Interfaces

The systems that interface with the steam generators include the reactor coolant system and connected lines (IV.C2), the main steam system (VIII.B1), the feedwater system (VIII.D1), the steam generator blowdown system (VIII.F), and the auxiliary feedwater system (VIII.G).

IV Reactor Vessel, Internals, and Reactor Coolant System
D2. Steam Generator (Once-Through)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
D2.1-a D2.1.1 D2.1.2	Pressure boundary and structural Upper and lower heads Tube sheets	Low-alloy steel with stainless steel (head) and alloy 82/182 (tubesheet) cladding	Chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
D2.1-b D2.1.1 D2.1.3	Pressure boundary and structural Upper and lower heads (external surfaces) Primary nozzles	Low-alloy steel	Air, leaking chemically treated borated water and/or steam up to 340°C (644°F)	Loss of material/ Boric acid corrosion of external surfaces	Chapter XI.M10, "Boric Acid Corrosion"	No
D2.1-c D2.1.3	Pressure boundary and structural Primary nozzles	Low-alloy steel with stainless steel cladding	Chemically treated borated water up to 340°C (644°F) and 15.2 MPa	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii), and for addressing environmental effects on fatigue. See Chapter X.M1 of this report for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes, TLAA

IV **Reactor Vessel, Internals, and Reactor Coolant System**
D2. Steam Generator (Once-Through)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
D2.1-d D2.1.4 D2.1.5	Pressure boundary and structural Shell assembly Feedwater (FW) and auxiliary FW (AFW) nozzles and safe ends	Carbon steel	Up to 300°C secondary-side water chemistry at 5.3-7.2 MPa	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes TLAA
D2.1-e D2.1.4	Pressure boundary and structural Shell assembly	Carbon steel	Up to 300°C (572°F) secondary-side water chemistry at 5.3-7.2 MPa	Loss of material/ General, pitting, and crevice corrosion	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 2 components and Chapter XI.M2, "Water Chemistry," for PWR secondary water in EPRI TR-102134 As noted in NRC Information Notice 90-04, general and pitting corrosion of the shell exists, the AMP guidelines in Chapter XI.M1 may not be sufficient to detect general and pitting corrosion, and additional inspection procedures may be required.	Yes, detection of aging effects is to be evaluated
D2.1-f D2.1.5 D2.1.6	Pressure boundary and structural FW and AFW nozzles and safe ends Steam nozzles and safe ends	Carbon steel	Up to 300°C (572°F) steam or secondary-side water chemistry at 5.3-7.2 MPa	Wall thinning/ Flow-accelerated corrosion	Chapter XI.M17, "Flow-Accelerated Corrosion"	No
D2.1-g D2.1.6	Pressure boundary and structural Steam nozzles and safe ends	Carbon steel	Up to 300°C (572°F) steam	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes, TLAA

IV **Reactor Vessel, Internals, and Reactor Coolant System**
D2. Steam Generator (Once-Through)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
D2.1-h D2.1.7	Pressure boundary and structural Primary side drain nozzles	Alloy 600	Chemically treated borated water up to 340°C (644°F) and 15.2 MPa	Crack initiation and growth/ Primary water stress corrosion cracking	A plant-specific aging management program is to be evaluated.	Yes, plant specific
D2.1-i D2.1.8	Pressure boundary and structural Secondary side nozzles (vent, drain, and instrumentation)	Alloy 600	Up to 300°C (572°F) secondary-side water chemistry at 5.3-7.2 MPa	Crack initiation and growth/ Stress corrosion cracking	A plant-specific aging management program is to be evaluated.	Yes, plant specific
D2.1-j D2.1.4 D2.1.5 D2.1.6 D2.1.9 D2.1.10	Pressure boundary and structural External surfaces of shell assembly FW and AFW nozzles and safe ends Steam nozzles and safe ends Primary manways (cover and bolting) Secondary manways and handholes (cover and bolting)	Carbon steel, low-alloy steel	Air, leaking chemically treated borated water and/or steam at temperatures up to 340°C (644°F)	Loss of material/ Boric acid corrosion of external surfaces	Chapter XI.M10, "Boric Acid Corrosion"	No
D2.1-k D2.1.9 D2.1.10	Pressure boundary and structural Primary manways (bolting only) Secondary manways and handholes (bolting only)	Low-alloy steel	Air, with metal temperatures up to 340°C (644°F)	Loss of preload/ Stress relaxation	Chapter XI.M18, "Bolting Integrity"	No

IV **Reactor Vessel, Internals, and Reactor Coolant System**
D2. Steam Generator (Once-Through)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
D2.1-l D2.1.10	Pressure boundary and structural Secondary manways and handholes (cover only)	Carbon steel	Air, leaking secondary-side water and/or steam at temperatures up to 300°C (572°F)	Wall thinning/ Erosion	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 2 components	No
D2.2-a D2.2.1	Tube bundle (Babcock and Wilcox) Tubes and sleeves	Alloy 600	Chemically treated borated water up to 340°C (644°F) and 15.2 MPa	Crack initiation and growth/ Primary water stress corrosion cracking	Chapter XI.M19, "Steam Generator Tubing Integrity" and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714 All PWR licensees have committed voluntarily to a SG degradation management program described in NEI 97-06; these guidelines are currently under NRC staff review. An AMP based on the recommendations of staff-approved NEI 97-06 guidelines, or other alternate regulatory basis for SG degradation management, is to be developed and incorporated in the plant technical specifications.	Yes, effectiveness of the AMP is to be evaluated

IV Reactor Vessel, Internals, and Reactor Coolant System
D2. Steam Generator (Once-Through)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
D2.2-b D2.2.1	Tube bundle (Babcock and Wilcox) Tubes and sleeves	Alloy 600	Up to 300°C (572°F) secondary-side water chemistry at 5.3-7.2 MPa	Crack initiation and growth/ Outer diameter stress corrosion cracking	Chapter XI.M19, "Steam Generator Tubing Integrity" and Chapter XI.M2, "Water Chemistry," for PWR secondary water in EPRI TR-102134 All PWR licensees have committed voluntarily to a SG degradation management program described in NEI 97-06; these guidelines are currently under NRC staff review. An AMP based on the recommendations of staff-approved NEI 97-06 guidelines, or other alternate regulatory basis for SG degradation management, is to be developed and incorporated in the plant technical specifications.	Yes, effective-ness of the AMP is to be evaluated
D2.2-c D2.2.1	Tube bundle (Babcock and Wilcox) Tubes and sleeves	Alloy 600	Up to 300°C (572°F) secondary-side water chemistry at 5.3-7.2 MPa	Crack initiation and growth/ Intergranular attack	Chapter XI.M19, "Steam Generator Tubing Integrity" and Chapter XI.M2, "Water Chemistry," for PWR secondary water in EPRI TR-102134 All PWR licensees have committed voluntarily to a SG degradation management program described in NEI 97-06; these guidelines are currently under NRC staff review. An AMP based on the recommendations of staff-approved NEI 97-06 guidelines, or other alternate regulatory basis for SG degradation management, is to be developed and incorporated in the plant technical specifications.	Yes, effective-ness of the AMP is to be evaluated

IV **Reactor Vessel, Internals, and Reactor Coolant System**
D2. Steam Generator (Once-Through)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
D2.2-d D2.2.1	Tube bundle (Babcock and Wilcox) Tubes and sleeves	Alloy 600	Up to 300°C (572°F) secondary-side water chemistry at 5.3-7.2 MPa	Loss of section thickness/ Fretting and wear	Chapter XI.M19, "Steam Generator Tubing Integrity" All PWR licensees have committed voluntarily to a SG degradation management program described in NEI 97-06; these guidelines are currently under NRC staff review. An AMP based on the recommendations of staff-approved NEI 97-06 guidelines, or other alternate regulatory basis for SG degradation management, is to be developed and incorporated in the plant technical specifications.	Yes, effectiveness of the AMP is to be evaluated
D2.2-e D2.2.1	Tube bundle (Babcock and Wilcox) Tubes and sleeves	Alloy 600	Up to 300°C (572°F) secondary-side water chemistry at 5.3-7.2 MPa	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes, TLAA

IV Reactor Vessel, Internals, and Reactor Coolant System
D2. Steam Generator (Once-Through)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
D2.2-f D2.2.2	Tube bundle Tube plugs (mechanical) (Westinghouse)	Alloy 600, alloy 690	Chemically treated borated water at temperatures up to 340°C (644°F) and 15.5 MPa	Crack initiation and growth/ Primary water stress corrosion cracking	Chapter XI.M19, "Steam Generator Tubing Integrity" and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714 All PWR licensees have committed voluntarily to a SG degradation management program described in NEI 97-06; these guidelines are currently under NRC staff review. An AMP based on the recommendations of staff-approved NEI 97-06 guidelines, or other alternate regulatory basis for SG degradation management, is to be developed and incorporated in the plant technical specifications.	Yes, effectiveness of the AMP is to be evaluated
D2.2-g D2.2.2	Tube bundle Tube plugs (mechanical) (Babcock and Wilcox)	Alloy 600, alloy 690	Chemically treated borated water at temperatures up to 340°C (644°F) and 15.5 MPa	Crack initiation and growth/ Primary water stress corrosion cracking	Chapter XI.M19, "Steam Generator Tubing Integrity" and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714 All PWR licensees have committed voluntarily to a SG degradation management program described in NEI 97-06; these guidelines are currently under NRC staff review. An AMP based on the recommendations of staff-approved NEI 97-06 guidelines, or other alternate regulatory basis for SG degradation management, is to be developed and incorporated in the plant technical specifications.	Yes, effectiveness of the AMP is to be evaluated

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CHAPTER V

ENGINEERED SAFETY FEATURES

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A. Containment Spray System (Pressurized Water Reactor)

B. Standby Gas Treatment System (Boiling Water Reactor)

C. Containment Isolation Components

D1. Emergency Core Cooling System (Pressurized Water Reactor)

D2. Emergency Core Cooling System (Boiling Water Reactor)

E. Carbon Steel Components

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A. CONTAINMENT SPRAY SYSTEM (PRESSURIZED WATER REACTOR)

A.1 Piping, Fittings and Miscellaneous Items

A.1.1 Piping and Fittings up to Isolation Valve
A.1.2 Flow Orifice/Elements
A.1.3 Temperature Elements/Indicators
A.1.4 Bolting
A.1.5 Eductors

A.2 Headers and Spray Nozzles

A.2.1 Piping and Fittings
A.2.2 Flow Orifice
A.2.3 Headers
A.2.4 Spray Nozzles

A.3 Pumps

A.3.1 Bowl/Casing
A.3.2 Bolting

A.4 Valves (Hand, Control, Check, Motor-Operated, and Containment Isolation) in Containment Spray System

A.4.1 Body and Bonnet
A.4.2 Bolting

A.5 Valves (Hand, Control, and Containment Isolation) in Headers and Spray Nozzles

A.5.1 Body and Bonnet
A.5.2 Bolting

A.6 Containment Spray Heat Exchanger

A.6.1 Bonnet/Cover
A.6.2 Tubing
A.6.3 Shell
A.6.4 Case/Cover
A.6.5 Bolting

A. CONTAINMENT SPRAY SYSTEM (PRESSURIZED WATER REACTORS)

Systems, Structures, and Components

This section comprises the containment spray system for pressurized water reactors (PWRs) designed to lower the pressure, temperature, and gaseous radioactivity (iodine) content of the containment atmosphere following a design basis event. Spray systems using chemically treated borated water are reviewed. The system consists of piping and valves, including containment isolation valves, flow elements and orifices, pumps, spray nozzles, eductors, and the containment spray system heat exchanger (some plants).

Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all components that comprise the containment spray system outside or inside the containment are governed by Group B Quality Standards.

Pumps and valve internals perform their intended functions with moving parts or with a change in configuration, or are subject to replacement based on qualified life or specified time period. Accordingly, they are not subject to an aging management review, pursuant to 10 CFR 54.21(a)(1).

Aging management programs for the degradation of external surfaces of carbon steel components are included in V.E.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The systems that interface with the containment spray system are the PWR emergency core cooling (V.D1), and open- or closed-cycle cooling water systems (VII.C1 or VII.C2).

V Engineered Safety Features

A. Containment Spray System (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
A.1-a A.1.1 A.1.2 A.1.3	Piping, fittings and miscellaneous items Piping and fittings up to isolation valve Flow orifice/elements Temperature elements/ indicators	Stainless steel	Chemically treated borated water at temperature < 93°C (200°F)	Crack initiation and growth/ Stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
A.1-b A.1.4	Containment spray system Bolting	Carbon steel, low-alloy steel	Air, leaking chemically treated borated water	Loss of material/ Boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No
A.1-c A.1.5	Containment spray system Eductors	Stainless steel	Chemically treated borated water at temperature < 93°C (200°F)	Crack initiation and growth/ Stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
A.2-a A.2.1 A.2.2 A.2.3 A.2.4	Headers and spray nozzles Piping and fittings Flow orifice Headers Spray nozzles	Carbon steel	Air	Loss of material/ General corrosion	A plant-specific aging management program is to be evaluated.	Yes, plant specific
A.3-a A.3.1	Pump Bowl/casing	Stainless steel	Chemically treated borated water at temperature < 93°C (200°F)	Crack initiation and growth/ Stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
A.3-b A.3.2	Pump Bolting	Carbon steel, low-alloy steel	Air, leaking chemically treated borated water	Loss of material/ Boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No

V Engineered Safety Features
A. Containment Spray System (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
A.4-a A.4.1	Valves (hand, control, check, motor-operated, and containment isolation) in containment spray system Body and bonnet	Stainless steel	Chemically treated borated water at temperature < 93°C (200°F)	Crack initiation and growth/ Stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
A.4-b A.4.2	Valves (hand, control, check, motor-operated, and containment isolation) in containment spray system Bolting	Carbon steel, low-alloy steel	Air, leaking chemically treated borated water	Loss of material/ Boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No
A.5-a A.5.1	Valves (hand, control and containment isolation) in headers and spray nozzles Body and bonnet	Carbon Steel	Air	Loss of material/ General corrosion	A plant-specific aging management program is to be evaluated.	Yes, plant specific
A.5-b A.5.2	Valves (hand, control and containment isolation) in headers and spray nozzles Bolting	Carbon steel, low-alloy steel	Air, leaking chemically treated borated water	Loss of material/ Boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No
A.6-a A.6.1 A.6.2 A.6.3 A.6.4	Containment spray heat exchanger (serviced by open-cycle cooling water) Bonnet/cover Tubing Shell Case/cover	Carbon steel, stainless steel	Chemically treated borated water on one side and open-cycle cooling water (raw water) on the other side	Loss of material/ General and microbiologically influenced corrosion and biofouling	Chapter XI.M20, "Open-Cycle Cooling Water System"	No

V Engineered Safety Features
A. Containment Spray System (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
A.6-b A.6.2	Containment spray heat exchanger (serviced by open-cycle cooling water) Tubing	Carbon steel, stainless steel	Chemically treated borated water on one side and open-cycle cooling water (raw water) on the other side	Buildup of deposit/ Biofouling	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
A.6-c A.6.1 A.6.2 A.6.3 A.6.4	Containment spray heat exchanger (serviced by closed-cycle cooling water) Bonnet/cover Tubing Shell Case/cover	Carbon steel, stainless steel	Chemically treated borated water on tube side and closed-cycle cooling water on shell side	Loss of material/ General, pitting and crevice corrosion	Chapter XI.M21, "Closed-Cycle Cooling Water System"	No
A.6-d A.6.3 A.6.4 A.6.5	Containment spray heat exchanger Shell Case/cover (external surfaces) Bolting	Carbon steel, low-alloy steel	Air, leaking chemically treated borated water	Loss of material/ Boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No

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B. STANDBY GAS TREATMENT SYSTEM (BOILING WATER REACTOR)

B.1 Ductwork

- B.1.1 Duct, Fittings, Access Doors, and Closure Bolts
- B.1.2 Equipment Frames and Housing
- B.1.3 Seals between Ducts and Fan
- B.1.4 Seals in Dampers and Doors

B.2 Filters

- B.2.1 Housing and Supports
- B.2.2 Elastomer Seals

B. STANDBY GAS TREATMENT SYSTEM (BOILING WATER REACTOR)

Systems, Structures, and Components

This section comprises the standby gas treatment system found in boiling water reactors (BWRs) and consist of ductwork, filters, and fans. Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all components that comprise the standby gas treatment system are governed by Group B Quality Standards.

With respect to charcoal absorber filters, these items are to be addressed consistent with the NRC position on consumables, provided in the NRC letter from Christopher I. Grimes to Douglas J. Walters of NEI, dated March 10, 2000. Specifically, components that function as system filters are typically replaced based on performance or condition monitoring that identifies whether these components are at the end of their qualified lives and may be excluded, on a plant-specific basis, from an aging management review under 10 CFR 54.21(a)(1)(ii). The application is to identify the standards that are relied on for replacement as part of the methodology description, for example, NFPA standards for fire protection equipment.

Aging management programs for the degradation of external surfaces of carbon steel components are included in V.E.

System Interfaces

There are no system interfaces with the standby gas treatment system addressed in this section.

V Engineered Safety Features
B. Standby Gas Treatment Systems (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B.1-a B.1.1 B.1.2	Ductwork Fittings, access doors, and closure bolts Equipment frames and housing	Carbon steel	Internal: occasional exposure to moist air; external: ambient plant air environment	Loss of material/ General corrosion	A plant-specific aging management program is to be evaluated.	Yes, plant specific
B.1-b B.1.3 B.1.4	Ductwork Seals between ducts and fan Seals in dampers and doors	Elastomer (Neoprene)	Internal: occasional exposure to moist air; external: ambient plant air environment	Hardening and loss of strength/ Elastomer degradation	A plant-specific aging management program is to be evaluated.	Yes, plant specific
B.2-a B.2.1	Filters Housing and supports	Carbon steel	Internal: occasional exposure to moist air; external: ambient plant air environment	Loss of material/ General corrosion	A plant-specific aging management program is to be evaluated.	Yes, plant specific
B.2-b B.2.2	Filters Elastomer seals	Elastomers (Neoprene and similar materials)	Occasional exposure to moist air	Hardening and loss of strength/ Elastomer degradation	A plant-specific aging management program is to be evaluated.	Yes, plant specific

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C. CONTAINMENT ISOLATION COMPONENTS

C.1 Isolation Barriers

C.1.1 Valve Body and Bonnet
C.1.2 Pipe Penetrations

C. CONTAINMENT ISOLATION COMPONENTS

Systems, Structures, and Components

This section comprises the containment isolation components found in all designs of boiling water reactors (BWR) and pressurized water reactors (PWR) in the United States. The system consists of isolation barriers in lines for BWR and PWR nonsafety systems such as the plant heating, waste gas, plant drain, liquid waste, and cooling water systems. Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all components that comprise the containment isolation components are governed by Group A or B Quality Standards.

The aging management programs for hatchways, hatch doors, penetration sleeves, penetration bellows, seals, gaskets, and anchors are addressed in II.A and II.B. The containment isolation valves for in-scope systems are addressed in the appropriate sections in IV, VII, and VIII.

Aging management programs for the degradation of external surfaces of carbon steel components are included in V.E.

System Interfaces

There are no system interfaces with the containment isolation components addressed in this section.

V Engineered Safety Features
C. Containment Isolation Components

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
C.1-a C.1.1 C.1.2	BWR and PWR isolation barriers Valve body and bonnet Pipe penetrations (piping between two isolation valves)	Carbon steel and low-alloy steel	Inside surface: treated or raw water, liquid waste; outside surface: ambient air	Loss of material/ General, pitting, crevice and microbiologically influenced corrosion and biofouling	A plant-specific aging management program is to be evaluated. See IN 85-30 for evidence of microbiologically influenced corrosion.	Yes, plant specific
C.1-b C.1.1 C.1.2	BWR and PWR isolation barriers Valve body and bonnet Pipe penetrations (piping between two isolation valves)	Stainless steel	Inside surface: treated or raw water, liquid waste; outside surface: ambient air	Loss of material/ Pitting, crevice and microbiologically influenced corrosion and biofouling	A plant-specific aging management program is to be evaluated. See IN 85-30 for evidence of microbiologically influenced corrosion.	Yes, plant specific

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D1. EMERGENCY CORE COOLING SYSTEM (PRESSURIZED WATER REACTOR)

D1.1 Piping and Fittings

- D1.1.1 Core Flood System (CFS)
- D1.1.2 Residual Heat Removal (RHR) or Shutdown Cooling (SDC)
- D1.1.3 High-Pressure Safety Injection (HPSI)
- D1.1.4 Low-Pressure Safety Injection (LPSI)
- D1.1.5 Connecting lines to Chemical and Volume Control System (CVCS) and Spent Fuel Pool (SFP) Cooling
- D1.1.6 Lines to Emergency Sump
- D1.1.7 Bolting for Flange Connections

D1.2 HPSI and LPSI Pumps

- D1.2.1 Bowl/Casing
- D1.2.2 Bolting
- D1.2.3 Orifice

D1.3 RWT Circulation Pump

- D1.3.1 Bolting

D1.4 Valves

- D1.4.1 Body and Bonnet
- D1.4.2 Bolting

D1.5 Heat Exchangers (RCP, HPSI, and LPSI Pump Seals; and RHR or SDC)

- D1.5.1 Bonnet/Cover
- D1.5.2 Tubing
- D1.5.3 Shell
- D1.5.4 Case/Cover
- D1.5.5 Bolting

D1.6 Heat Exchangers (RWT Heating)

- D1.6.1 Bonnet/Cover
- D1.6.2 Tubing
- D1.6.3 Shell
- D1.6.4 Bolting

D1.7 Safety Injection Tank (Accumulator)

- D1.7.1 Shell
- D1.7.2 Manway
- D1.7.3 Penetrations/Nozzles

D1.8 Refueling Water Tank (RWT)

- D1.8.1 Shell
- D1.8.2 Manhole
- D1.8.3 Penetrations/Nozzles
- D1.8.4 Bolting
- D1.8.5 Buried Portion of Tank

D1. EMERGENCY CORE COOLING SYSTEM (PRESSURIZED WATER REACTORS)

Systems, Structures, and Components

This section comprises the emergency core cooling systems for pressurized water reactors (PWRs) designed to cool the reactor core and provide safe shutdown following a design basis accident. They consist of the core flood (CFS), residual heat removal (RHR) (or shutdown cooling (SDC)), high-pressure safety injection (HPSI), low-pressure safety injection (LPSI), and spent fuel pool (SFP) cooling systems; the lines to the chemical and volume control system (CVCS); the emergency sump, the HPSI and LPSI pumps; the pump seal coolers; the RHR heat exchanger; and the refueling water tank (RWT). Stainless steel components are not subject to significant general, pitting, and crevice corrosion in borated water and, therefore, for these stainless steel components, loss of material due to corrosion in borated water is not included in this section.

Based on Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” all components that comprise the emergency core cooling system are governed by Group B Quality Standards. Portions of the RHR, HPSI, and LPSI systems and the CVCS extending from the reactor coolant system up to and including the second containment isolation valve are governed by Group A Quality Standards and covered in IV.C2.

Pumps and valve internals perform their intended functions with moving parts or with a change in configuration, or are subject to replacement based on qualified life or specified time period. Accordingly, they are not subject to an aging management review, pursuant to 10 CFR 54.21(a)(1).

Aging management programs for the degradation of external surfaces of carbon steel components are included in V.E.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The systems that interface with the emergency core cooling system include the reactor coolant system and connected lines (IV.C2), the containment spray system (V.A), the spent fuel pool cooling and cleanup system (VII.A3), the closed-cycle cooling water system (VII.C2), the ultimate heat sink (VII.C3), the chemical and volume control system (VII.E1), and the open-cycle cooling water system (service water system) (VII.C1).

V Engineered Safety Features
D1. Emergency Core Cooling System (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
D1.1-a D1.1.1 D1.1.2 D1.1.3 D1.1.4 D1.1.5 D1.1.6	Piping and fittings Core flood system Residual heat removal or shutdown cooling High-pressure safety injection Low-pressure safety injection Connecting lines to chemical and volume control system Spent fuel pool cooling lines to emergency sump	Stainless steel	Chemically treated borated water at temperature < 93°C (200°F)	Crack initiation and growth/ Stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
D1.1-b D1.1.1 D1.1.2 D1.1.3 D1.1.4 D1.1.5 D1.1.6	Piping and fittings Core flood system Residual heat removal or shutdown cooling High-pressure safety injection Low-pressure safety injection Connecting lines to chemical and volume control system Spent fuel pool cooling lines to emergency sump	Cast austenitic stainless steel	Chemically treated borated water at temperature 25–340°C (77-644°F)	Loss of fracture toughness/ Thermal aging embrittlement	Chapter XI.M12, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)"	No
D1.1-c D1.1.1 D1.1.2 D1.1.3 D1.1.4	Piping and fittings Core flood system Residual heat removal or shutdown cooling High-pressure safety injection Low-pressure safety injection	Stainless steel	Chemically treated borated water at temperature < 93°C (200°F)	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3, "Metal Fatigue" for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes, TLAA
D1.1-d D1.1.7	Piping and fittings Bolting for flange connections in items D1.1.1 through D1.1.6	Nuts: carbon steel; bolts/studs: alloy steel	Air, leaking chemically treated borated water	Loss of material/ Boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No

V Engineered Safety Features

D1. Emergency Core Cooling System (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
D1.2-a D1.2.1	HPSI and LPSI pumps Bowl/casing	Stainless steel, carbon steel with stainless steel cladding	Chemically treated borated water at temperature < 93°C (200°F)	Crack initiation and growth/ Stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
D1.2-b D1.2.1 D1.2.2	HPSI and LPSI pumps Bowl/casing (external surfaces) Bolting	Casing: carbon steel with stainless steel cladding; nuts: carbon steel; bolts/studs: alloy steel	Air, leaking chemically treated borated water	Loss of material/ Boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No
D1.2-c D1.2.3	HPSI and LPSI pumps Orifice (miniflow recirculation)	Stainless steel	Chemically treated borated water at temperature < 93°C (200°F)	Loss of material/ Erosion	A plant-specific aging management program is to be evaluated for erosion of the orifice due to extended use of the centrifugal HPSI pump for normal charging. See LER 50-275/94-023 for evidence of erosion.	Yes, plant specific
D1.3-a D1.3.1	RWT circulation pump Bolting	Nuts: carbon steel; bolts/studs: alloy steel	Air, leaking chemically treated borated water	Loss of material/ Boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No
D1.4-a D1.4.1	Valves (check, control, hand, motor operated, and relief valves) Body and bonnet	Stainless steel, carbon steel with stainless steel cladding	Chemically treated borated water at temperature < 93°C (200°F)	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3, "Metal Fatigue" for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes, TLAA

V Engineered Safety Features
D1. Emergency Core Cooling System (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
D1.4-b D1.4.1	Valves (check, control, hand, motor operated, and relief valves) Body and bonnet	Stainless steel, carbon steel with stainless steel cladding	Chemically treated borated water at temperature < 93°C (200°F)	Crack initiation and growth/ Stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
D1.4-c D1.4.1 D1.4.2	Valves (check, control, hand, motor operated, and relief valves) Body and bonnet (external surfaces) Bolting	Body and bonnet: carbon steel; nuts: carbon steel; bolts/studs: alloy steel	Air, leaking chemically treated borated water	Loss of material/ Boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No
D1.5-a D1.5.1 D1.5.2 D1.5.3 D1.5.4	Heat exchangers (reactor coolant pump seal, HPSI pump seal, LPSI pump seal, RHR or SDC) Bonnet/cover Tubing Shell Case/cover	Bonnet/ cover and tubing: stainless steel; shell: carbon steel; case/cover: cast iron	Chemically treated borated water; and treated component cooling water	Loss of material/ Pitting and crevice corrosion	Chapter XI.M21, "Closed-Cycle Cooling Water System"	No
D1.5-b D1.5.3 D1.5.4 D1.5.5	Heat exchangers (RCP seal, HPSI pump seal, LPSI pump seal, RHR or SDC) Shell Case/cover (external surfaces) Bolting	Shell: carbon steel; case/cover: cast iron; nuts: carbon steel; bolts/studs: alloy steel	Air, leaking chemically treated borated water	Loss of material/ Boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No

V Engineered Safety Features
D1. Emergency Core Cooling System (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
D1.6-a D1.6.1 D1.6.2 D1.6.3	Heat exchanger (RWT heating) serviced by closed-cycle cooling water Bonnet/cover Tubing Shell	Bonnet/ cover and tubing: stainless steel; shell: carbon steel	Chemically treated borated water and treated component cooling water	Loss of material/ Pitting and crevice corrosion	Chapter XI.M21, "Closed-Cycle Cooling Water System"	No
D1.6-b D1.6.1 D1.6.2 D1.6.3	Heat exchanger (RWT Heating) serviced by open-cycle cooling water Bonnet/cover Tubing Shell	Carbon steel, stainless steel	Chemically treated borated water on one side and open-cycle cooling water (raw water) on the other side	Loss of material/ General (carbon steel only), pitting, crevice, and microbiologically influenced corrosion and biofouling	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
D1.6-c D1.6.2	Heat exchanger (RWT heating) serviced by open-cycle cooling water Tubing	Carbon steel, stainless steel	Chemically treated borated water on one side and open-cycle cooling water (raw water) on the other side	Buildup of deposit/ Biofouling	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
D1.6-d D1.6.3 D1.6.4	Heat exchanger (RWT heating) Shell (external surface) Bolting	Shell: carbon steel; nuts: carbon steel; bolts/studs: alloy steel	Air, leaking chemically treated borated water	Loss of Material/ Boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No

V Engineered Safety Features
D1. Emergency Core Cooling System (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
D1.7-a D1.7.1 D1.7.2 D1.7.3	Safety injection tank (accumulator) Shell Manway Penetrations/ nozzles (all external surface)	Carbon steel with stainless steel cladding	Air, leaking chemically treated borated water	Loss of material/ Boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No
D1.7-b D1.7.3	Safety injection tank (accumulator) Penetrations/nozzles	Carbon steel with stainless steel cladding	Chemically treated borated water at temperature < 93°C (200°F)	Crack initiation and growth/ Stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
D1.8-a D1.8.1 D1.8.2 D1.8.3	Refueling water tank (RWT) Shell Manhole Penetrations/nozzles	Stainless steel	Chemically treated borated water at temperature < 93°C (200°F)	Crack initiation and growth/ Stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
D1.8-b D1.8.4	Refueling water tank (RWT) Bolting	Nuts: carbon steel; bolts/studs: alloy steel	Air, leaking chemically treated borated water	Loss of material/ Boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No
D1.8-c D1.8.5	Refueling water tank (RWT) Buried portion of tank (outer surface)	Stainless steel	Moisture, water	Loss of material/ Pitting and crevice corrosion	A plant-specific aging management program is to be evaluated for pitting and crevice corrosion of tank bottom because moisture and water can egress under the tank due to cracking of the perimeter seal from weathering.	Yes, plant specific

D2. EMERGENCY CORE COOLING SYSTEM (BWR)

D2.1 Piping and Fittings

- D2.1.1 High Pressure Coolant Injection (HPCI)
- D2.1.2 Reactor Core Isolation Cooling (RCIC)
- D2.1.3 High-Pressure Core Spray (HPCS)
- D2.1.4 Low-Pressure Core Spray (LPCS)
- D2.1.5 Low-Pressure Coolant Injection (LPCI) and Residual Heat Removal (RHR)
- D2.1.6 Lines to Suppression Chamber (SC)
- D2.1.7 Lines to Drywell and Suppression Chamber Spray System (DSCSS)
- D2.1.8 Automatic Depressurization System (ADS)
- D2.1.9 Lines to HPCI and RCIC Pump Turbine
- D2.1.10 Lines from HPCI and RCIC Pump Turbines to Condenser

D2.2 Pumps (HPCS or HPCI Main and Booster, LPCS, LPCI or RHR, and RCIC)

- D2.2.1 Bowl/Casing
- D2.2.2 Suction Head
- D2.2.3 Discharge Head

D2.3 Valves (Check, Control, Hand, Motor Operated, and Relief Valves)

- D2.3.1 Body and Bonnet

D2.4 Heat Exchangers (RHR and LPCI)

- D2.4.1 Tubes
- D2.4.2 Tubesheet
- D2.4.3 Channel Head
- D2.4.4 Shell

D2.5 Drywell and Suppression Chamber Spray System (DSCSS)

- D2.5.1 Piping and Fittings
- D2.5.2 Flow Orifice
- D2.5.3 Headers
- D2.5.4 Spray Nozzles

D2. EMERGENCY CORE COOLING SYSTEM (BOILING WATER REACTORS)

Systems, Structures, and Components

This section comprises the emergency core cooling systems for boiling water reactors (BWRs) designed to cool the reactor core and provide safe shutdown following a design basis accident. They consist of the high-pressure coolant injection (HPCI), reactor core isolation cooling (RCIC), high-pressure core spray (HPCS), automatic depressurization (ADS), low-pressure core spray (LPCS), low-pressure coolant injection (LPCI) and residual heat removal (RHR) systems, including various pumps and valves; the RHR heat exchangers; and the drywell and suppression chamber spray system (DSCSS).

Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all components that comprise the emergency core cooling system outside the containment are governed by Group B Quality Standards and the portion of the DSCSS inside the containment up to the isolation valve is governed by Group A Quality Standard. Portions of the HPCI, RCIC, HPCS, LPCS, and LPCI (or RHR) systems extending from the reactor vessel up to and including the second containment isolation valve are governed by Group A Quality Standards and covered in IV.C1.

Pumps and valve internals perform their intended functions with moving parts or with a change in configuration, or are subject to replacement based on qualified life or specified time period. Accordingly, they are not subject to an aging management review, pursuant to 10 CFR 54.21(a)(1).

Aging management programs for the degradation of external surfaces of carbon steel components are included in V.E.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The systems that interface with the emergency core cooling system include the reactor vessel (IV.A1), the reactor coolant pressure boundary (IV.C1), the feedwater system (VIII.D2), the condensate system (VIII.E), the closed-cycle cooling water system (VII.C2), the open-cycle cooling water system (VII.C1), and the ultimate heat sink (VII.C3).

V Engineered Safety Features

D2. Emergency Core Cooling system (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
D2.1-a D2.1.1 D2.1.2 D2.1.3 D2.1.4 D2.1.5 D2.1.6 D2.1.7	Piping and fittings High-pressure coolant injection Reactor core isolation cooling High-pressure core spray Low-pressure core spray Low-pressure coolant injection and residual heat removal Lines to suppression chamber Lines to drywell and suppression chamber spray system	Carbon steel	25–288°C (77-550°F) demineralized water	Loss of material/ General, pitting, and crevice corrosion	Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515) The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated
D2.1-b D2.1.1	Piping and fittings HPCI	Carbon steel, stainless steel	25–288°C (77-550°F) demineralized water	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3, "Metal Fatigue" for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes, TLAA
D2.1-c D2.1.1 D2.1.2 D2.1.3 D2.1.4 D2.1.5 D2.1.6 D2.1.7	Piping and fittings HPCI RCIC HPCS LPCS LPCI and RHR Lines to SC Lines to DSCSS	Stainless steel	25–288°C (77-550°F) demineralized water	Crack initiation and growth/ Stress corrosion cracking, intergranular stress corrosion cracking	Chapter XI.M7, "BWR Stress Corrosion Cracking," and Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515)	No

V Engineered Safety Features
D2. Emergency Core Cooling system (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
D2.1-d D2.1.1 D2.1.2 D2.1.3 D2.1.4 D2.1.5 D2.1.6 D2.1.7	Piping and fittings HPCI RCIC HPCS LPCS LPCI and RHR Lines to SC Lines to DSCSS	Cast austenitic stainless steel	25–288°C (77-550°F) demineralized water	Loss of fracture toughness/ Thermal aging embrittlement	Chapter XI.M12, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)"	No
D2.1-e D2.1.8	Piping and fittings Automatic depressurization system	Carbon steel, stainless steel	Moist containment atmosphere (air/nitrogen), steam, or demineralized water	Loss of material/ General (carbon steel only), pitting, and crevice corrosion	A plant-specific aging management program is to be evaluated.	Yes, plant specific
D2.1-f D2.1.9 D2.1.10	Piping and fittings Lines to HPCI and RCIC pump turbine Lines from HPCI and RCIC pump turbine to torus or wetwell	Carbon steel	Air and steam up to 320°C (608°F)	Wall thinning/ Flow-accelerated corrosion	Chapter XI.M17, "Flow-Accelerated Corrosion"	No
D2.2-a D2.2.1 D2.2.2 D2.2.3	Pumps HPCS or HPCI main and booster, LPCS, LPCI or RHR, and RCIC Bowl/casing Suction head Discharge head	Carbon steel casting, carbon steel	25–288°C (77-550°F) demineralized water	Loss of material/ General, pitting, and crevice corrosion	Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515) The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated
D2.3-a D2.3.1	Valves (check, control, hand, motor operated, and relief valves) Body and bonnet	Carbon steel forging, carbon steel casting	25–288°C (77-550°F) demineralized water	Wall thinning/ Flow-accelerated corrosion	Chapter XI.M17, "Flow-Accelerated Corrosion"	No

V Engineered Safety Features
D2. Emergency Core Cooling system (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
D2.3-b D2.3.1	Valves (check, control, hand, motor operated, and relief valves) Body and bonnet	Carbon steel forging, carbon steel casting	25–288°C (77-550°F) demineralized water	Loss of material/ General, pitting, and crevice corrosion	Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515) The AMP is to be augmented by verification of its effectiveness of the water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated
D2.3-c D2.3.1	Valves (check, control, hand, motor operated, and relief valves) Body and bonnet	Stainless steel forging, stainless steel casting	25–288°C (77-550°F) demineralized water	Crack initiation and growth/ Stress corrosion cracking	Chapter XI.M7, "BWR Stress Corrosion Cracking," and Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515)	No
D2.4-a D2.4.1 D2.4.2 D2.4.3 D2.4.4	Heat exchangers (RHR and LPCI) (serviced by open-cycle cooling water) Tubes Tubesheet Channel head Shell	Carbon steel, stainless steel	Demineralized water on one side; open-cycle cooling water (raw water) on the other side	Loss of material/ General (carbon steel only), pitting, crevice, and microbiologically influenced corrosion, and biofouling	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
D2.4-b D2.4.1	Heat exchangers (RHR and LPCI) (serviced by open-cycle cooling water) Tubes	Carbon steel, stainless steel	Demineralized water on one side; open cycle cooling water (raw water) on the other side	Buildup of deposit/ Biofouling	Chapter XI.M20, "Open-Cycle Cooling Water System"	No

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
D2.4-c D2.4.1 D2.4.2 D2.4.3 D2.4.4	Heat exchangers (RHR and LPCI) (serviced by closed-cycle cooling water) Tubes Tubesheet Channel head Shell	Carbon steel, stainless steel	Demineralized water on one side; closed-cycle cooling water (treated water) on the other side	Loss of material/ General (carbon steel only), pitting, and crevice corrosion	Chapter XI.M21, "Closed-Cycle Cooling Water System"	No
D2.5-a D2.5.1 D2.5.2 D2.5.3 D2.5.4	Drywell and suppression chamber spray system Piping and fittings Flow orifice Headers Spray nozzles	Carbon steel	Air	Loss of material/ General corrosion	A plant-specific aging management program is to be evaluated.	Yes, plant specific
D2.5-b D2.5.1 D2.5.2 D2.5.3 D2.5.4	Drywell and suppression chamber spray system Piping and fittings Flow orifice Headers Spray nozzles	Carbon steel	Air	Plugging of flow orifice and spray nozzles/ General corrosion	A plant-specific aging management program is to be evaluated.	Yes, plant specific

E. CARBON STEEL COMPONENTS

E.1 Carbon Steel Components

E.1.1 External Surfaces

E.2 Closure Bolting

E.2.1 In High-Pressure or High-Temperature Systems

E. CARBON STEEL COMPONENTS

Systems, Structures, and Components

This section includes the aging management programs for the degradation of external surface of all carbon steel structures and components including closure boltings in the engineered safety features in pressurized water reactors (PWRs) and boiling water reactors (BWRs). For the carbon steel components in PWRs, this section addresses only boric acid corrosion of external surfaces as a result of the dripping borated water that is leaking from an adjacent PWR component. Boric acid corrosion can also occur for carbon steel components containing borated water due to leakage; such components and the related aging management program are covered in the appropriate major plant sections in V.

System Interfaces

The structures and components covered in this section belong to the engineered safety features in PWRs and BWRs. (For example, see System Interfaces in V.A to V.D2 for details.)

V Engineered Safety Features
E. Carbon Steel Components

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
E.1-a E.1.1	Carbon steel components (PWRs) External surfaces	Carbon steel, low-alloy steel	Air, leaking and dripping chemically treated borated water up to 340°C (644°F)	Loss of material/ Boric acid corrosion of external surfaces	Chapter XI.M10, "Boric Acid Corrosion"	No
E.1-b E.1.1	Carbon steel components (PWRs and BWRs) External surfaces	Carbon steel, low-alloy steel	Air, moisture, and humidity < 100°C (212°F)	Loss of material/ General corrosion	A plant-specific aging management program is to be evaluated.	Yes, plant specific
E.2-a E.2.1	Closure bolting In high-pressure or high-temperature systems	Carbon steel, low-alloy steel	Air, moisture, humidity, and leaking fluid	Loss of material/ General corrosion	Chapter XI.M18, "Bolting Integrity"	No
E.2-b E.2.1	Closure bolting In high-pressure or high-temperature systems	Carbon steel, low-alloy steel	Air, moisture, humidity, and leaking fluid	Crack initiation and growth/ Cyclic loading, stress corrosion cracking	Chapter XI.M18, "Bolting Integrity"	No

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CHAPTER VI

ELECTRICAL COMPONENTS

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A. Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

B. Equipment Subject to 10 CFR 50.49 Environmental Qualification Requirements

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A. ELECTRICAL CABLES AND CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS

A.1 Conductor Insulation

A.1.1 Electrical cables and connections exposed to an adverse localized environment caused by heat, radiation, or moisture

A.1.2 Electrical cables used in instrumentation circuits that are sensitive to reduction in conductor insulation resistance (IR) exposed to an adverse localized environment caused by heat, radiation, or moisture

A.1.3 Inaccessible medium-voltage (2kV to 15kV) cables (e.g., installed in conduit or direct buried) exposed to an adverse localized environment caused by exposure to moisture and voltage

A.2 Connector Contacts

A.2.1 Electrical connectors exposed to borated water leakage

A. ELECTRICAL CABLES AND CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS

Systems, Structures and Components

This section addresses electrical cables and connections that are not subject to the environmental qualification requirements of 10 CFR 50.49, and that are installed in power and instrumentation and control (I&C) applications. The power cables and connections addressed are low-voltage (<1000V) and medium-voltage (2kV to 15kV). High voltage (>15kV) power cables and connections have unique, specialized constructions and must be evaluated on an application specific basis.

Electrical cables and their required terminations (i.e., connections) are typically reviewed as a single commodity. The types of connections included in this review are splices, mechanical connectors, and terminal blocks. This common review is translated into program actions, which treat cables and connections in the same manner.

Electrical cables and connections that are in the plant's environmental qualification (EQ) program are addressed in VI.B.

System Interfaces

Electrical cables and connections functionally interface with all plant systems that rely on electric power or instrumentation and control. Electrical cables and connections also interface with and are supported by structural commodities (e.g., cable trays, conduit, cable trenches, cable troughs, duct banks, cable vaults and manholes) that are reviewed, as appropriate, in the Structures and Components Supports section.

VI Electrical Components

A. Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
A.1-a A.1.1	Conductor insulation Conductor insulation for electrical cables and connections	Various organic polymers (e.g., EPR, SR, EPDM, XLPE)	Adverse localized environment caused by heat, radiation, or moisture in the presence of oxygen	Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced insulation resistance, electrical failure / (Thermal/ thermoxidative) degradation of organics, radiolysis and photolysis (UV sensitive materials only) of organics; radiation-induced oxidation, moisture intrusion	Chapter XI.E1, "Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements"	No

VI **Electrical Components**

A. Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
A.1-b A.1.2	Conductor insulation Conductor insulation for electrical cables used in instrumentation circuits that are sensitive to reduction in conductor insulation resistance (IR)	Various organic polymers (e.g., EPR, SR, EPDM, XLPE)	Adverse localized environment caused by heat, radiation, or moisture in the presence of oxygen	Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced insulation resistance, electrical failure / (Thermal/ thermoxidative) degradation of organics, radiation-induced oxidation, moisture intrusion	Chapter XI.E2, "Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits"	No
A.1-c A.1.3	Conductor insulation Conductor insulation for inaccessible medium-voltage (2kV to 15kV) cables (e.g., installed in conduit or direct buried)	Various organic polymers (e.g., EPR, SR, EPDM, XLPE)	Adverse localized environment caused by exposure to moisture and voltage	Formation of water trees, localized damage, leading to electrical failure (breakdown of insulation)/ Moisture intrusion, water trees	Chapter XI.E3, "Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements"	No
A.2-a A.2.1	Connector contacts Connector contacts for electrical connectors exposed to borated water leakage	Various metals used for electrical contacts	Exposure to borated water leakage	Corrosion of connector contact surfaces / Intrusion of borated water	Chapter XI.M10, "Boric Acid Corrosion"	No

B. EQUIPMENT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS

B.1 Equipment Subject to 10 CFR 50.49 Environmental Qualification Requirements

B.1.1 Electrical Equipment Subject to 10 CFR 50.49 Environmental Qualification Requirements

B. EQUIPMENT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS

Systems, Structures and Components

The Nuclear Regulatory Commission (NRC) has established nuclear station environmental qualification (EQ) requirements in 10 CFR Part 50 Appendix A, Criterion 4, and in 10 CFR 50.49. 10 CFR 50.49 specifically requires that an EQ program be established to demonstrate that certain electrical components located in harsh plant environments (i.e., those areas of the plant that could be subject to the harsh environmental effects of a loss of coolant accident [LOCA], high energy line breaks [HELBs] or post-LOCA radiation) are qualified to perform their safety function in those harsh environments after the effects of inservice aging. 10 CFR 50.49 requires that the effects of significant aging mechanisms be addressed as part of environmental qualification. Components in the EQ program have a qualified life, and the components are replaced at the end of that qualified life, if it is shorter than the current operating term. The qualified life may be extended by methods such as refurbishment or reanalysis, but the licensee is required by the EQ regulation (10 CFR 50.49) to replace the component when its qualified life has expired.

System Interfaces

Equipment subject to 10 CFR 50.49 environmental qualification requirements functionally interface with all plant systems that rely on electric power or instrumentation and control.

VI Electrical Components
B. Equipment Subject to 10 CFR 50.49 Environmental Qualification Requirements

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B.1-a B.1.1	Equipment subject to 10 CFR 50.49 environmental qualification requirements Electrical equipment subject to 10 CFR 50.49 EQ requirements	Various polymeric and metallic materials	Adverse localized environment caused by heat, radiation, oxygen, moisture, or voltage	Various degradation/ Various mechanisms	EQ is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.4, "Environmental Qualification (EQ) of Electrical Equipment," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii). See Chapter X.E1, "Environmental Qualification (EQ) of Electric Components," of this report for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes, TLAA

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CHAPTER VII

AUXILIARY SYSTEMS

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MAJOR PLANT SECTIONS

A1. New Fuel Storage

A2. Spent Fuel Storage

A3. Spent Fuel Pool Cooling and Cleanup (PWR)

A4. Spent Fuel Pool Cooling and Cleanup (BWR)

A5. Suppression Pool Cleanup System (BWR)

B. Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems

C1. Open-Cycle Cooling Water System (Service Water System)

C2. Closed-Cycle Cooling Water System

C3. Ultimate Heat Sink

D. Compressed Air System

E1. Chemical and Volume Control System (PWR)

E2. Standby Liquid Control System (BWR)

E3. Reactor Water Cleanup System (BWR)

E4. Shutdown Cooling System (Older BWR)

F1. Control Room Area Ventilation System

F2. Auxiliary and Radwaste Area Ventilation System

F3. Primary Containment Heating and Ventilation System

F4. Diesel Generator Building Ventilation System

G. Fire Protection

H1. Diesel Fuel Oil System

H2. Emergency Diesel Generator System

I. Carbon Steel Components

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A1. NEW FUEL STORAGE

A1.1 New Fuel Rack

A1.1.1 New Fuel Rack Assembly

A1. NEW FUEL STORAGE

Systems, Structures, and Components

This section comprises those structures and components used for new fuel storage, and includes carbon steel new fuel storage racks located in the auxiliary building or the fuel handling building. The racks are exposed to the temperature and humidity in the auxiliary building. The racks are generally painted with a protective coating. Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all components used for new fuel storage are governed by Group C Quality Standards.

Aging management programs for the degradation of external surfaces of carbon steel components are included in VII.I.

System Interfaces

No other systems discussed in this report interface with those used for new fuel storage.

VII Auxiliary Systems
A1. New Fuel Storage

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
A1.1-a A1.1.1	New fuel rack New fuel rack assembly	Carbon steel	Indoors: exposed to variable temperature and humidity inside the auxiliary building or fuel handling building	Loss of material/ General, pitting, and crevice corrosion	Chapter XI.S6, "Structures Monitoring Program"	No

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A2. SPENT FUEL STORAGE

A2.1 Spent Fuel Storage Rack

A2.1.1 Neutron-Absorbing Sheets
A2.1.2 Storage Rack

A2. SPENT FUEL STORAGE

Systems, Structures, and Components

This section comprises those structures and components used for spent fuel storage and include stainless steel spent fuel storage racks and neutron absorbing materials (e.g., Boraflex, Boral, or boron-steel sheets, if used) submerged in chemically treated oxygenated (BWR) or borated (PWR) water. The intended function of a spent fuel rack is to separate spent fuel assemblies. Boraflex sheets fastened to the storage cells provide for neutron absorption and help maintain subcriticality of spent fuel assemblies in the spent fuel pool.

Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all components used for spent fuel storage are governed by Group C Quality Standards. In some plants, the Boraflex has been replaced by Boral or boron steel.

Aging management programs for the degradation of external surfaces of carbon steel components are included in VII.I.

System Interfaces

No other systems discussed in this report interface with those used for spent fuel storage.

VII Auxiliary Systems
A2. Spent Fuel Storage

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
A2.1-a A2.1.1	Spent fuel storage racks Neutron-absorbing sheets	Boraflex	Chemically treated oxygenated (BWR) or borated (PWR) water	Reduction of neutron-absorbing capacity/ Boraflex degradation	Chapter XI.M22, "Boraflex Monitoring"	No
A2.1-b A2.1.1	Spent fuel storage racks Neutron-absorbing sheets	Boral, boron steel	Chemically treated oxygenated (BWR) or borated (PWR) water	Reduction of neutron-absorbing capacity and loss of material/ General corrosion	A plant-specific aging management program is to be evaluated.	Yes, plant specific
A2.1-c A2.1.2	Spent fuel storage racks Storage racks	Stainless steel	Chemically treated oxygenated (BWR) or borated (PWR) water	Crack initiation and growth/ Stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515), or PWR primary water in EPRI TR-105714	No

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A3. SPENT FUEL POOL COOLING AND CLEANUP (PRESSURIZED WATER REACTOR)

A3.1 Piping

A3.1.1 Closure Bolting

A3.2 Filter

A3.2.1 Housing
A3.2.2 Closure Bolting
A3.2.3 Elastomer Lining

A3.3 Valves (Check and Hand Valves)

A3.3.1 Body and Bonnet
A3.3.2 Closure Bolting
A3.3.3 Elastomer Lining (Hand Valves Only)

A3.4 Heat Exchanger

A3.4.1 Shell and Access Cover
A3.4.2 Channel Head and Access Cover
A3.4.3 Closure Bolting

A3.5 Ion Exchanger

A3.5.1 Shell
A3.5.2 Nozzles
A3.5.3 Closure Bolting
A3.5.4 Elastomer Lining

A3.6 Pump

A3.6.1 Closure Bolting

A3. SPENT FUEL POOL COOLING AND CLEANUP (PRESSURIZED WATER REACTOR)

Systems, Structures, and Components

This section comprises the PWR spent fuel pool cooling and cleanup system and consists of piping, valves, heat exchangers, filters, linings, demineralizers, and pumps. The system contains borated water. Stainless steel components are not subject to significant aging degradation in borated water and, therefore, are not included in this section. The system removes heat from the spent fuel pool, and transfers heat to the closed-cycle cooling water system, which in turn transfers heat to the open-cycle cooling water system. Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all components that comprise the PWR spent fuel pool cooling and cleanup system are governed by Group C Quality Standards.

With respect to filters, these items are to be addressed consistent with the NRC position on consumables, provided in the NRC letter from Christopher I. Grimes to Douglas J. Walters of NEI, dated March 10, 2000. Specifically, components that function as system filters are typically replaced based on performance or condition monitoring that identifies whether these components are at the end of their qualified lives and may be excluded, on a plant-specific basis, from an aging management review under 10 CFR 54.21(a)(1)(ii). The application is to identify the standards that are relied on for replacement as part of the methodology description, for example, NFPA standards for fire protection equipment.

Pump and valve internals perform their intended functions with moving parts or with a change in configuration, or are subject to replacement based on qualified life or specified time period. Accordingly, they are not subject to an aging management review, pursuant to 10 CFR 54.21(a)(1).

Aging management programs for the degradation of external surfaces of carbon steel components are included in VII.I.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The systems that interface with the PWR spent fuel cooling and cleanup system are the PWR emergency core cooling system (V.D1), the closed-cycle cooling water system (VII.C2), and the PWR chemical and volume control system (VII.E1).

VII Auxiliary Systems
A3. Spent Fuel Pool Cooling and Cleanup (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
A3.1-a A3.1.1	Piping Closure bolting	Carbon steel, low-alloy steel	Air, leaking chemically treated borated water	Loss of material/ Boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No
A3.2-a A3.2.1	Filter Housing	Carbon steel with elastomer lining	Chemically treated borated water	Loss of material/ Pitting and crevice corrosion (only for carbon steel after lining degradation)	Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714 The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated
A3.2-b A3.2.1	Filter Housing (external surface)	Carbon steel	Air, leaking chemically treated borated water	Loss of material/ Boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No
A3.2-c A3.2.2	Filter Closure bolting	Carbon steel, low-alloy steel	Air, leaking chemically treated borated water	Loss of material/ Boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No
A3.2-d A3.2.3	Filter Elastomer lining	Elastomers	Chemically treated borated water	Hardening, cracking/ Elastomer degradation	A plant-specific aging management program that determines and assesses the qualified life of the linings in the environment is to be evaluated.	Yes, plant specific
A3.3-a A3.3.1	Valves (check and hand valves) Body and bonnet	Carbon steel with elastomer lining	Chemically treated borated water	Loss of material/ Pitting and crevice corrosion (only for carbon steel after lining degradation)	Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714 The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated

VII Auxiliary Systems

A3. Spent Fuel Pool Cooling and Cleanup (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
A3.3-b A3.3.1	Valves (check and hand valves) Body and bonnet	Carbon steel with stainless steel cladding	Chemically treated borated water	Crack initiation and growth/ Stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714	No
A3.3-c A3.3.1 A3.3.2	Valves (check and hand valves) Body and bonnet (external surface) Closure bolting	Body: carbon steel; bolting: carbon steel or low-alloy steel	Air, leaking chemically treated borated water	Loss of material/ Boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No
A3.3-d A3.3.3	Valves (hand valve only) Elastomer lining	Elastomers	Chemically treated borated water	Hardening, cracking/ Elastomer degradation	A plant-specific aging management program that determines and assesses the qualified life of the linings in the environment is to be evaluated.	Yes, plant specific
A3.4-a A3.4.1 A3.4.2	Heat exchanger (serviced by closed-cycle cooling water system) Shell and access cover Channel head and access cover	Carbon steel	Shell side: closed-cycle cooling water (treated water)	Loss of material/ General, pitting and crevice corrosion	Chapter XI.M21, "Closed-Cycle Cooling Water System"	No
A3.4-b A3.4.1 A3.4.2 A3.4.3	Heat exchanger (serviced by closed-cycle cooling water system) Shell and access cover Channel head and access cover (external surface) Closure bolting	Carbon steel, low-alloy steel	Air, leaking chemically treated borated water	Loss of material/ Boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No

VII **Auxiliary Systems**
A3. Spent Fuel Pool Cooling and Cleanup (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
A3.5-a A3.5.1 A3.5.2	Ion exchanger (demineralizer) Shell Nozzles	Carbon steel with elastomer lining	Chemically treated borated water	Loss of material/ Pitting and crevice corrosion (only for carbon steel after lining degradation)	Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714 The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated
A3.5-b A3.5.1 A3.5.2 A3.5.3	Ion exchanger (demineralizer) Shell (external surface) Nozzles (external surface) Closure bolting	Carbon steel, low-alloy steel	Air, leaking chemically treated borated water	Loss of material/ Boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No
A3.5-c A3.5.4	Ion exchanger (demineralizer) Elastomer lining	Elastomers	Chemically treated borated water	Hardening, cracking/ Elastomer degradation	A plant-specific aging management program that determines and assesses the qualified life of the linings in the environment is to be evaluated.	Yes, plant specific
A3.6-a A3.6.1	Pump Closure bolting	Carbon steel, low-alloy steel	Air, leaking chemically treated borated water	Loss of material/ Boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No

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A4. SPENT FUEL POOL COOLING AND CLEANUP (BOILING WATER REACTOR)

A4.1 Piping

- A4.1.1 Piping, Fittings, and Flanges

A4.2 Filter

- A4.2.1 Housing
- A4.2.2 Elastomer Lining

A4.3 Valves (Check and Hand Valves)

- A4.3.1 Body and Bonnet
- A4.3.2 Elastomer Lining (Hand Valves Only)

A4.4 Heat Exchanger

- A4.4.1 Shell and Access Cover
- A4.4.2 Channel Head and Access Cover
- A4.4.3 Tubes
- A4.4.4 Tubesheet

A4.5 Ion Exchanger

- A4.5.1 Shell
- A4.5.2 Nozzles
- A4.5.3 Elastomer Lining

A4.6 Pump

- A4.6.1 Casing

A4. SPENT FUEL POOL COOLING AND CLEANUP (BOILING WATER REACTOR)

Systems, Structures, and Components

This section comprises the BWR spent fuel pool cooling and cleanup system and consists of piping, valves, heat exchangers, filters, linings, demineralizers, and pumps. The system contains chemically treated oxygenated water. The system removes heat from the spent fuel pool, and transfers the heat to the closed-cycle cooling water system, which in turn transfers the heat to the open-cycle cooling water system. Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all components that comprise the BWR spent fuel pool cooling and cleanup system are governed by Group C Quality Standards.

With respect to filters, these items are to be addressed consistent with the NRC position on consumables, provided in the NRC letter from Christopher I. Grimes to Douglas J. Walters of NEI, dated March 10, 2000. Specifically, components that function as system filters are typically replaced based on performance or condition monitoring that identifies whether these components are at the end of their qualified lives and may be excluded, on a plant-specific basis, from an aging management review under 10 CFR 54.21(a)(1)(ii). The application is to identify the standards that are relied on for replacement as part of the methodology description, for example, NFPA standards for fire protection equipment.

Pump and valve internals perform their intended functions with moving parts or with a change in configuration, or are subject to replacement based on qualified life or specified time period. Accordingly, they are not subject to an aging management review, pursuant to 10 CFR 54.21(a)(1).

Aging management programs for the degradation of external surfaces of carbon steel components are included in VII.I.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The systems that interface with the BWR spent fuel cooling and cleanup system are the closed-cycle cooling water system (VII.C2) and the condensate system (VIII.E).

VII Auxiliary Systems
A4. Spent Fuel Pool Cooling and Cleanup (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
A4.1-a A4.1.1	Piping Piping, fittings, and flanges	Stainless steel	Chemically treated oxygenated water up to 50°C (125°F)	Loss of material/ Pitting and crevice corrosion	Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515) The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated
A4.2-a A4.2.1	Filter Housing	Stainless steel; carbon steel with elastomer lining, or stainless steel cladding	Chemically treated oxygenated water up to 50°C (125°F)	Loss of material/ Pitting and crevice corrosion (only for carbon steel after lining/cladding degradation)	Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515) The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated
A4.2-b A4.2.2	Filter Elastomer lining	Elastomers	Chemically treated oxygenated water up to 50°C (125°F)	Hardening, cracking/ Elastomer degradation	A plant-specific aging management program that determines and assesses the qualified life of the linings in the environment is to be evaluated.	Yes, plant specific
A4.3-a A4.3.1	Valves (check and hand valves) Body and bonnet	Stainless steel; carbon steel with elastomer lining, or stainless steel cladding	Chemically treated oxygenated water up to 50°C (125°F)	Loss of material/ Pitting and crevice corrosion (only for carbon steel after lining/cladding degradation)	Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515) The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated

VII **Auxiliary Systems**
A4. Spent Fuel Pool Cooling and Cleanup (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
A4.3-b A4.3.2	Valves (hand valve only) Elastomer lining	Elastomers	Chemically treated oxygenated water up to 50°C (125°F)	Hardening, cracking/ Elastomer degradation	A plant-specific aging management program that determines and assesses the qualified life of the linings in the environment is to be evaluated.	Yes, plant specific
A4.4-a A4.4.1 A4.4.2 A4.4.3	Heat exchanger (serviced by closed-cycle cooling water system) Shell and access cover Channel head and access cover Tubes	Carbon steel	Shell side: closed-cycle cooling water	Loss of material/ General, pitting and crevice corrosion	Chapter XI.M21, "Closed-Cycle Cooling Water System"	No
A4.4-b A4.4.2 A4.4.3 A4.4.4	Heat exchanger (serviced by closed-cycle cooling water system) Channel head and access cover Tubes Tubesheet	Channel head and access cover: stainless steel, carbon steel with stainless steel cladding; tubes and tubesheet: stainless steel	Demineralized oxygenated water	Loss of material/ General, pitting and crevice corrosion	Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515) The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated
A4.5-a A4.5.1 A4.5.2	Ion exchanger (demineralizer) Shell Nozzles	Stainless steel, carbon steel with elastomer lining	Demineralized oxygenated water	Loss of Material/ Pitting and crevice corrosion (only for carbon steel after lining degradation)	Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515) The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated

VII Auxiliary Systems

A4. Spent Fuel Pool Cooling and Cleanup (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
A4.5-b A4.5.3	Ion exchanger (demineralizer) Elastomer lining	Elastomers	Chemically treated oxygenated water up to 50°C (125°F)	Hardening, cracking/ Elastomer degradation	A plant-specific aging management program that determines and assesses the qualified life of the linings in the environment is to be evaluated.	Yes, plant specific
A4.6-a A4.6.1	Pump Casing	Stainless steel, carbon steel (with stainless steel cladding)	Demineralized oxygenated water	Loss of material/ Pitting and crevice corrosion	Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515) The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated

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A5. SUPPRESSION POOL CLEANUP SYSTEM (BOILING WATER REACTOR)

Systems, Structures, and Components

This section comprises the suppression pool cleanup system, which maintains water quality in the suppression pool in boiling water reactors (BWRs). The components of this system include piping, filters, valves, and pumps. These components are fabricated of carbon, low-alloy, or austenitic stainless steel. Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," the components that comprise the suppression pool cleanup system are governed by the same Group C Quality Standards Group as the corresponding components in the spent fuel pool cooling and cleanup system (VII.A4).

Pump and valve internals perform their intended functions with moving parts or with a change in configuration, or are subject to replacement based on qualified life or specified time period. Accordingly, they are not subject to an aging management review, pursuant to 10 CFR 54.21(a)(1).

Aging management programs for the degradation of external surfaces of carbon steel components are included in VII.I.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The system that interfaces with the suppression pool cleanup system is the BWR containment (II.B), or BWR emergency core cooling system (V.D2).

Evaluation Summary

The suppression pool cleanup system in BWRs is similar to the spent fuel pool cooling and cleanup system (VII.A4), and the components in the two systems are identical or very similar. The reader is therefore referred to the section for the spent fuel storage pool system for a listing of aging effects, aging mechanisms, and aging management programs that are to be applied to the suppression pool cleanup system components. (The only component in VII.A4 that may not be applicable to the suppression pool cleanup system is the heat exchanger [VII.A4.4].)

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B. OVERHEAD HEAVY LOAD AND LIGHT LOAD (RELATED TO REFUELING) HANDLING SYSTEMS

B.1 Cranes Including Bridge and Trolley (that fall within the scope of 10 CFR 54)

B.1.1 Structural Girders

B.2 Rail System

B.2.1 Rail

B. OVERHEAD HEAVY LOAD AND LIGHT LOAD (RELATED TO REFUELING) HANDLING SYSTEMS

Systems, Structures, and Components

Most commercial nuclear facilities have between fifty and one hundred cranes. Many of these cranes are industrial grade cranes that must meet the requirements of 29 CRF Volume XVII, Part 1910, and Section 1910.179. They do not fall within the scope of 10 CFR Part 54.4 and therefore are not required to be part of the integrated plant assessment (IPA). Normally fewer than ten cranes fall within the scope of 10 CFR Part 54.4. These cranes must all comply with the requirements provided in 10 CFR Part 50.65 and Reg. Guide 1.160 for monitoring the effectiveness of maintenance at nuclear power plants.

The Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems (the Program) must demonstrate that the testing and the monitoring of the maintenance programs have been completed to ensure that the structures, systems, and components of these cranes are capable of sustaining their rated loads during the period of extended operation. The inspection is also to evaluate whether the usage of the cranes or hoists has been sufficient to warrant additional fatigue analysis. It should be noted that many of the systems and components of these cranes can be classified as moving parts or as components which change configuration, or, they may be subject to replacement based on a qualified life. In any of these cases, they will not fall within the scope of this Aging Management Review (AMR). The primary components that this program is concerned with are the structural girders and beams that make up the bridge and the trolley.

Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all components that comprise the overhead heavy load and light load handling systems are governed by Group C Quality Standards.

Aging management programs for the degradation of external surface of carbon steel components are included in VII.I.

System Interfaces

No other systems discussed in this report interface with the overhead heavy load and light load (related to refueling) handling systems. Physical interfaces exist with the supporting structure. The direct interface is at the connection to the structure.

VII Auxiliary Systems

B. Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
B.1-a B.1.1	Cranes including bridge and trolley (for cranes that fall within the scope of 10 CFR 54) Structural girders	Structural steel A-36, A-7, or A-285	Air at 100% relative humidity and 49°C (120°F)	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation for structural girders of cranes that fall within the scope of 10 CFR 54. See the Standard Review Plan, Section 4.7, "Other Plant-Specific Time-Limited Aging Analyses," for generic guidance for meeting the requirements of 10 CFR 54.21 (c).	Yes, TLAA
B.1-b B.1.1	Cranes including bridge and trolley (for cranes that fall within the scope of 10 CFR 54) Structural girders	Structural steel A-36, A-7, or A-285	Air at 100% relative humidity and 49°C (120°F)	Loss of material/ General corrosion	Chapter XI.M23, "Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems"	No
B.2-a B.2.1	Rail system Rail	Structural steel A-759	Air at 100% relative humidity and 49°C (120°F)	Loss of material/ Wear	Chapter XI.M23, "Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems"	No

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C1. OPEN-CYCLE COOLING WATER SYSTEM (SERVICE WATER SYSTEM)

C1.1 Piping

C1.1.1 Piping and Fittings
C1.1.2 Underground Piping and Fittings

C1.2 Valves

C1.2.1 Body and Bonnet

C1.3 Heat Exchanger

C1.3.1 Shell
C1.3.2 Channel
C1.3.3 Channel Head and Access Cover
C1.3.4 Tubesheet
C1.3.5 Tubes

C1.4 Flow Orifice

C1.4.1 Body

C1.5 Pump

C1.5.1 Casing

C1.6 Basket Strainer

C1.6.1 Body

C1. OPEN-CYCLE COOLING WATER SYSTEM (SERVICE WATER SYSTEM)

Systems, Structures, and Components

This section comprises the open-cycle cooling water (OCCW) (or service water) system, which consists of piping, heat exchangers, pumps, flow orifices, basket strainers, and valves, including containment isolation valves. Because the characteristics of an OCCW system may be unique to each facility, the OCCW system is defined as a system or systems that transfer heat from safety-related systems, structures, and components (SSCs) to the ultimate heat sink (UHS) such as a lake, ocean, river, spray pond, or cooling tower. The AMPs described in this section apply to any such system, provided the service conditions and materials of construction are identical to those identified in the section. The system removes heat from the closed-cycle cooling water system and, in some plants, other auxiliary systems and components such as steam turbine bearing oil coolers, or miscellaneous coolers in the condensate system. The only heat exchangers addressed in this section are those removing heat from the closed-cycle cooling system. Heat exchangers for removing heat from other auxiliary systems and components are addressed in their respective systems, such as those for the steam turbine bearing oil coolers (VIII.A) and for the condensate system coolers (VIII.E).

Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all components that comprise the open-cycle cooling water system are governed by Group C Quality Standards, with the exception of those forming part of the containment penetration boundary which are governed by Group B Quality Standards.

Pump and valve internals perform their intended functions with moving parts or with a change in configuration, or are subject to replacement based on qualified life or specified time period. Accordingly, they are not subject to an aging management review, pursuant to 10 CFR 54.21(a)(1).

Aging management programs for the degradation of external surfaces of carbon steel components are included in VII.I.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The systems that may interface with the open-cycle cooling water system include the closed-cycle cooling water system (VII.C2), the ultimate heat sink (VII.C3), the emergency diesel generator system (VII.H2), the containment spray system (V.A), the PWR steam generator blowdown system (VIII.F), the condensate system (VIII.E), the auxiliary feedwater system (PWR) (VIII.G), the emergency core cooling system (PWR) (V.D1), and the emergency core cooling system (BWR) (V.D2).

VII Auxiliary Systems
C1. Open-Cycle Cooling Water System (Service Water System)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
C1.1-a C1.1.1	Piping Piping and fittings (with or without internal lining or coating)	Carbon steel (for fresh water only) aluminum-bronze, brass, copper-nickel, stainless steel	Raw, untreated salt water or fresh water	Loss of material/ General (only for carbon steel without lining/coating or with degraded lining/coating), selective leaching (only for aluminum-bronze, brass, and copper-nickel), pitting, crevice, galvanic, microbiologically influenced corrosion and biofouling	Chapter XI.M20, “Open-Cycle Cooling Water System” and Chapter XI.M33, “Selective Leaching of Materials”	No
C1.1-b C1.1.2	Piping Underground piping and fittings (external surface, with or without organic coating or wrapping)	Carbon steel	Soil	Loss of material/ General, pitting, crevice, and microbiologically influenced corrosion	Chapter XI.M28, “Buried Piping and Tanks Surveillance,” or Chapter XI.M34, “Buried Piping and Tanks Inspection”	No Yes, detection of aging effects and operating experience are to be further evaluated
C1.1-c C1.1.2	Piping Underground piping and fittings (external surface, with or without organic coating or wrapping)	Cast iron	Soil	Loss of material/ selective leaching and general corrosion	Chapter XI.M33, “Selective Leaching of Materials”	No

VII Auxiliary Systems
C1. Open-Cycle Cooling Water System (Service Water System)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
C1.2-a C1.2.1	Valves (check, hand, control, and containment isolation valves) Body and bonnet (with or without internal lining or coating)	Bronze, aluminum-bronze, stainless steel, carbon steel (fresh water only)	Raw, untreated salt water or fresh water	Loss of material/ General (only carbon steel without lining/coating or with degraded lining/coating), selective leaching (only for bronze, aluminum-bronze), pitting, crevice, microbiologically influenced corrosion and biofouling	Chapter XI.M20, "Open-Cycle Cooling Water System," and Chapter XI.M33, "Selective Leaching of Materials"	No
C1.3-a C1.3.1 C1.3.2 C1.3.3 C1.3.4 C1.3.5	Heat exchanger (between open-cycle and closed-cycle cooling water systems) Shell Channel Channel head and access cover Tube sheet Tubes	Shell, channel, channel head and access cover: carbon steel; tube sheet: aluminum-bronze; tubes: copper-nickel, aluminum brass	Shell side: treated water; tube side: raw untreated salt or fresh water	Loss of material/ General (only for carbon steel), selective leaching (only for aluminum-bronze, copper-nickel, and aluminum brass), galvanic, pitting, crevice, microbiologically influenced corrosion and biofouling	Chapter XI.M20, "Open-Cycle Cooling Water System" and Chapter XI.M33, "Selective Leaching of Materials"	No
C1.3-b C1.3.5	Heat exchanger (between open-cycle and closed-cycle cooling water systems) Tubes	Copper-nickel, aluminum brass	Shell side: treated water/ tube side: raw untreated salt or fresh water	Buildup of deposit/ Biofouling	Chapter XI.M20, "Open-Cycle Cooling Water System"	No

VII **Auxiliary Systems**
C1. Open-Cycle Cooling Water System (Service Water System)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
C1.4-a C1.4.1	Flow orifice Body	Stainless steel	Raw, untreated salt water or fresh water	Loss of material/ Pitting, crevice, microbiologically influenced corrosion and biofouling	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
C1.5-a C1.5.1	Pump Casing	Cast steel, Carbon steel	Raw, untreated salt water or fresh water	Loss of material/ General, selective leaching (for cast steel), pitting, crevice, microbiologically influenced corrosion and biofouling	Chapter XI.M20, "Open-Cycle Cooling Water System," and Chapter XI.M33, "Selective Leaching of Materials"	No
C1.6-a C1.6.1	Basket strainer Body	Carbon steel, stainless steel	Raw, untreated salt water or fresh water	Loss of material/ General (for carbon steel only), pitting, crevice, microbiologically influenced corrosion and biofouling	Chapter XI.M20, "Open-Cycle Cooling Water System"	No

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C2. CLOSED-CYCLE COOLING WATER SYSTEM

C2.1 Piping

C2.1.1 Pipe, Fittings, and Flanges

C2.2 Valves (Check, Hand, Control, Relief, Solenoid, and Containment Isolation)

C2.2.1 Body and Bonnet

C2.3 Pump

C2.3.1 Casing

C2.4 Tank

C2.4.1 Shell

C2.5 Flow Orifice

C2.5.1 Body

C2. CLOSED-CYCLE COOLING WATER SYSTEM

Systems, Structures, and Components

This section comprises the closed-cycle cooling water (CCCW) system, which consists of piping, radiation elements, temperature elements, heat exchangers, pumps, tanks, flow orifices, and valves, including containment isolation valves. The system contains chemically treated demineralized water. The closed-cycle cooling water system is designed to remove heat from various auxiliary systems and components such as the chemical and volume control system and the spent fuel cooling system to the open-cycle cooling water system (VII.C1). A CCCW system is defined as part of the service water system that does not reject heat directly to a heat sink and that has water chemistry control and is not subject to significant sources of contamination.

Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all components in the closed-cycle cooling water system are classified as Group C quality Standards, with the exception of those forming part of the containment penetration boundary which are Group B.

The aging management programs (AMPs) for the heat exchanger between the closed-cycle and the open-cycle cooling water systems are addressed in the open-cycle cooling water system (VII.C1). The AMPs for the heat exchangers between the closed-cycle cooling water system and the interfacing auxiliary systems are included in the evaluations of their respective systems, such as those for the PWR and BWR spent fuel pool cooling and cleanup systems (VII.A3 and VII.A4, respectively) and the chemical and volume control system (VII.E1).

Pump and valve internals perform their intended functions with moving parts or with a change in configuration, or are subject to replacement based on qualified life or specified time period. Accordingly, they are not subject to an aging management review, pursuant to 10 CFR 54.21(a)(1).

Aging management programs for the degradation of external surfaces of carbon steel components are included in VII.I.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The systems that interface with the closed-cycle cooling water system include the open-cycle cooling water system (VII.C1), the PWR spent fuel pool cooling and cleanup system (VII.A3), the BWR spent fuel pool cooling and cleanup system (VII.A4), the chemical and volume control system (VII.E1), the BWR reactor water cleanup system (VII.E3), the shutdown cooling system (older BWR, VII.E5), the primary containment heating and ventilation system (VII.F3), the fire protection (VII.G), the emergency diesel generator system (VII.H2), the PWR containment spray system (V.A), the PWR and BWR emergency core cooling systems (V.D1 and V.D2), the PWR steam generator blowdown system (VIII.F), the condensate system (VIII.E), and the PWR auxiliary feedwater system (VIII.G).

VII **Auxiliary Systems**

C2. Closed-Cycle Cooling Water System

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
C2.1-a C2.1.1	Piping Pipe, fittings, and flanges	Carbon steel	35°C (95°F) treated water	Loss of material/ General, pitting, and crevice corrosion	Chapter XI.M21, "Closed-Cycle Cooling Water System"	No
C2.2-a C2.2.1	Valves (check, hand, control, relief, solenoid, and containment isolation valves) Body and bonnet	Carbon steel, stainless steel	35°C (95°F) treated water	Loss of material/ General (only for carbon steel), pitting and crevice corrosion	Chapter XI.M21, "Closed-Cycle Cooling Water System"	No
C2.3-a C2.3.1	Pump Casing	Carbon steel, cast iron	35°C (95°F) treated water	Loss of material/ General (only for carbon steel), selective leaching (for cast iron only), pitting and crevice corrosion	Chapter XI.M21, "Closed-Cycle Cooling Water System," and Chapter XI.M33, "Selective Leaching of Materials"	No
C2.4-a C2.4.1	Tank Shell	Carbon steel	35°C (95°F) treated water	Loss of material/ General, pitting and crevice corrosion	Chapter XI.M21, "Closed-Cycle Cooling Water System"	No
C2.5-a C2.5.1	Flow orifice Body	Carbon steel	35°C (95°F) treated water	Loss of material/ General, pitting and crevice corrosion	Chapter XI.M21, "Closed-Cycle Cooling Water System"	No

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C3. ULTIMATE HEAT SINK

C3.1 Piping

C3.1.1 Piping and Fittings

C3.2 Valves (Check, Hand, and Control)

C3.2.1 Body and Bonnet

C3.3 Pump

C3.3.1 Casing

C3. ULTIMATE HEAT SINK

Systems, Structures, and Components

The ultimate heat sink (UHS) consists of a lake, ocean, river, spray pond, or cooling tower and provides sufficient cooling water for safe reactor shutdown and reactor cooldown via the residual heat removal system or other similar system. Due to the varying configurations of connections to lakes, oceans, and rivers, a plant specific aging management program (AMP) is required. Appropriate AMPs shall be provided to trend and project (1) deterioration of earthen dams and impoundments; (2) rate of silt deposition; (3) meteorological, climatological, and oceanic data since obtaining the Final Safety Analysis Report (FSAR) data; (4) water level extremes for plants located on rivers; and (5) aging degradation of all upstream and downstream dams affecting the UHS.

The systems, structures and components included in this section consist of piping, valves, and pumps. The cooling tower is addressed in this report on water-control structures (III.A6). The ultimate heat sink absorbs heat from the residual heat removal system or other similar system. Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," the piping and valves used for the ultimate heat sink are governed by Group C Quality Standards.

Pump and valve internals perform their intended functions with moving parts or with a change in configuration, or are subject to replacement based on qualified life or specified time period. Accordingly, they are not subject to an aging management review, pursuant to 10 CFR 54.21(a)(1).

Aging management programs for the degradation of external surfaces of carbon steel components are included in VII.I.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The systems that interface with the ultimate heat sink include the open-cycle cooling water system (VII.C1) and the emergency core cooling systems (V.D1 and V.D2).

VII Auxiliary Systems
C3. Ultimate Heat Sink

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
C3.1-a C3.1.1	Piping Piping and fittings (with or without internal lining or coating)	Carbon steel, brass, copper-nickel	Raw, untreated fresh water	Loss of material/ General (only for carbon steel without internal lining or coating), selective leaching (only for brass, copper-nickel), pitting, crevice and microbiologically influenced corrosion	Chapter XI.M20, "Open-Cycle Cooling Water System," and Chapter XI.M33, "Selective Leaching of Materials"	No
C3.2-a C3.2.1	Valves (check, hand, and control) Body and bonnet (with or without internal lining or coating)	Bronze, stainless steel, carbon steel	Raw, untreated fresh water	Loss of material/ General (only for carbon steel), selective leaching (for bronze), pitting, crevice and microbiologically influenced corrosion	Chapter XI.M20, "Open-Cycle Cooling Water System," and Chapter XI.M33, "Selective Leaching of Materials"	No
C3.3-a C3.3.1	Pump Casing (with or without internal lining or coating)	Carbon steel	Raw, untreated fresh water	Loss of material/ General, pitting, crevice and microbiologically influenced corrosion	Chapter XI.M20, "Open-Cycle Cooling Water System"	No

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D. COMPRESSED AIR SYSTEM

D.1 Piping

D.1.1 Piping and Fittings
D.1.2 Closure Bolting

D.2 Valves (including check valves and containment isolation)

D.2.1 Body and Bonnet
D.2.2 Closure Bolting

D.3 Air Receiver

D.3.1 Shell and Access Cover
D.3.2 Closure Bolting

D.4 Pressure Regulators

D.4.1 Body and Bonnet

D.5 Filter

D.5.1 Shell and Access Cover
D.5.2 Closure Bolting

D.6 Dryer

D.6.1 Shell and Access Cover
D.6.2 Closure Bolting

D. COMPRESSED AIR SYSTEM

Systems, Structures, and Components

This section comprises the compressed air system, which consists of piping, valves (including containment isolation valves), air receiver, pressure regulators, filters, and dryers. The system components and piping are located in various buildings at most nuclear power plants. Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste- Containing Components of Nuclear Power Plants," all components of the compressed air system are classified as Group D Quality Standards, with the exception of those forming part of the containment penetration boundary which are Group B. However, the cleanliness of these components and high air quality is to be maintained because the air provides the motive power for instruments and active components (some of them safety-related) that may not function properly if nonsafety Group D equipment is contaminated.

With respect to filters, these items are to be addressed consistent with the NRC position on consumables, provided in the NRC letter from Christopher I. Grimes to Douglas J. Walters of NEI, dated March 10, 2000. Specifically, components that function as system filters are typically replaced based on performance or condition monitoring that identifies whether these components are at the end of their qualified lives and may be excluded, on a plant-specific basis, from an aging management review under 10 CFR 54.21(a)(1)(ii). The application is to identify the standards that are relied on for replacement as part of the methodology description, for example, NFPA standards for fire protection equipment.

Pump and valve internals perform their intended functions with moving parts or with a change in configuration, or are subject to replacement based on qualified life or specified time period. Accordingly, they are not subject to an aging management review, pursuant to 10 CFR 54.21(a)(1).

Aging management programs for the degradation of external surfaces of carbon steel components are included in VII.I.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

Various other systems discussed in this report may interface with the compressed air system.

VII Auxiliary Systems
D. Compressed Air System

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
D.1-a D.1.1 D.1.2	Piping Piping and fittings Closure bolting	Carbon steel and low-alloy steel	Saturated air	Loss of material/ General and pitting corrosion	Chapter XI.M24, "Compressed Air Monitoring"	No
D.2-a D.2.1 D.2.2	Valves (including check valves and containment isolation valves) Body and bonnet Closure bolting	Carbon steel	Saturated air	Loss of material/ General and pitting corrosion	Chapter XI.M24, "Compressed Air Monitoring"	No
D.3-a D.3.1 D.3.2	Air receiver Shell and access cover Closure bolting	Carbon steel	Saturated air	Loss of material/ General and pitting corrosion	Chapter XI.M24, "Compressed Air Monitoring"	No
D.4-a D.4.1	Pressure regulators Body and bonnet	Carbon steel	Saturated air	Loss of material/ General and pitting corrosion	Chapter XI.M24, "Compressed Air Monitoring"	No
D.5-a D.5.1 D.5.2	Filter Shell and access cover Closure bolting	Carbon steel	Saturated air	Loss of material/ General and pitting corrosion	Chapter XI.M24, "Compressed Air Monitoring"	No
D.6-a D.6.1 D.6.2	Dryer Shell and access cover Closure bolting	Carbon steel	Moist air	Loss of material/ General and pitting corrosion	Chapter XI.M24, "Compressed Air Monitoring"	No

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E1. CHEMICAL AND VOLUME CONTROL SYSTEM (PRESSURIZED WATER REACTOR)

E1.1 High-Pressure Piping (1500-psig rating)

- E1.1.1 Pipe, Fittings and Flanges
- E1.1.2 Closure Bolting

E1.2 Low-Pressure Piping (150-psig rating)

- E1.2.1 Closure Bolting

E1.3 High-Pressure Valves (check, control, hand, motor operated, pressure control and relief)

- E1.3.1 Body and Bonnet
- E1.3.2 Closure Bolting

E1.4 Low-Pressure Valves (check, control, hand, motor operated, pressure control and relief)

- E1.4.1 Closure Bolting

E1.5 High-Pressure Pump

- E1.5.1 Casing
- E1.5.2 Closure Bolting

E1.6 Low-Pressure Pump

- E1.6.1 Closure Bolting

E1.7 Regenerative Heat Exchanger

- E1.7.1 Channel Head and Access Cover
- E1.7.2 Tubesheet
- E1.7.3 Tubes
- E1.7.4 Shell and Access Cover
- E1.7.5 Closure Bolting

E1.8 Letdown Heat Exchanger

- E1.8.1 Channel Head and Access Cover
- E1.8.2 Tubesheet
- E1.8.3 Tubes
- E1.8.4 Shell and Access Cover
- E1.8.5 Closure Bolting

E1.9 Basket Strainers

- E1.9.1 Closure Bolting

E1.10 Volume Control Tank

E1.10.1 Closure Bolting

E1. CHEMICAL AND VOLUME CONTROL SYSTEM (PRESSURIZED WATER REACTOR)

Systems, Structures, and Components

This section comprises a portion of the pressurized water reactor (PWR) chemical and volume control system (CVCS). The portion of the PWR CVCS covered in this section extends from the isolation valves associated with the reactor coolant pressure boundary (and Code change as discussed below) to the volume control tank. This portion of the PWR CVCS consists of high- and low-pressure piping and valves (including the containment isolation valves), regenerative and letdown heat exchangers, pumps, basket strainers, and the volume control tank. The system contains chemically treated borated water; the shell side of the letdown heat exchanger contains closed-cycle cooling water (treated water). The effects of pitting and crevice corrosion on stainless steel components are not significant in chemically treated borated water and, therefore, are not included in this section.

Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all components that comprise the CVCS are governed by Group C Quality Standards. Portions of the CVCS extending from the reactor coolant system up to and including the isolation valves associated with reactor coolant pressure boundary are governed by Group A Quality Standards and covered in IV.C2.

Pump and valve internals perform their intended functions with moving parts or with a change in configuration, or are subject to replacement based on qualified life or specified time period. Accordingly, they are not subject to an aging management review, pursuant to 10 CFR 54.21(a)(1).

Aging management programs for the degradation of external surfaces of carbon steel components are included in VII.I.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The systems that interface with the chemical and volume control system include the reactor coolant system (IV.C2), the emergency core cooling system (V.D1), the spent fuel pool cooling system (VII.A3), and the closed-cycle cooling water system (VII.C2).

VII Auxiliary Systems

E1. Chemical and Volume Control System (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
E1.1-a E1.1.1 E1.1.2	High-pressure piping (1500-psig rating) Pipe, fittings, and flanges Closure bolting	Pipe, fittings, and flanges: stainless steel; closure bolting: low-alloy steel, carbon steel	Chemically treated borated water up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3, "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes, TLAA
E1.1-b E1.1.2	High-pressure piping (1500-psig rating) Closure bolting	Low-alloy steel, carbon steel	Air, leaking chemically treated borated water	Loss of Material/ Boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No
E1.2-a E1.2.1	Low-pressure piping (150-psig rating) Closure bolting	Low-alloy steel, carbon steel	Air, leaking chemically treated borated water	Loss of material/ Boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No
E1.3-a E1.3.1 E1.3.2	High-pressure valves (check, control, hand, motor operated, pressure control, and relief valves) Body and bonnet Closure bolting	Body and bonnet: stainless steel; closure bolting: carbon steel, low-alloy steel	Chemically treated borated water up to 340°C (644°F).	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3, "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes, TLAA
E1.3-b E1.3.2	High-pressure valves (check, control, hand, motor operated, pressure control, and relief valves) Closure bolting	Low-alloy steel, carbon steel	Air, leaking chemically treated borated water	Loss of material/ Boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No
E1.4-a E1.4.1	Low-pressure valves (check, control, hand, motor operated, pressure control, and relief valves) Closure bolting	Low-alloy steel, carbon steel	Air, leaking chemically treated borated water	Loss of material/ Boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No

VII **Auxiliary Systems**
E1. Chemical and Volume Control System (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
E1.5-a E1.5.1 E1.5.2	High-pressure pump Casing Closure bolting	Casing: stainless steel; closure bolting: carbon steel, low-alloy steel	Chemically treated borated water	Crack initiation and growth/ cracking	A plant-specific aging management program is to be evaluated.	Yes, plant specific
E1.5-b E1.5.2	High-pressure pump Closure bolting	Low-alloy steel, carbon steel	Air, leaking chemically treated borated water	Loss of material/ Boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No
E1.6-a E1.6.1	Low-pressure pump Closure bolting	Low-alloy steel, carbon steel	Air, leaking chemically treated borated water	Loss of material/ Boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No
E1.7-a E1.7.1 E1.7.2 E1.7.3 E1.7.4 E1.7.5	Regenerative heat exchanger Channel head (including channel weld) and access cover Tubesheet Tubes Shell and access cover Closure bolting	Stainless steel; closure bolting (low-alloy steel, carbon steel)	Tube and shell side: chemically treated borated water up to 340°C (644°F)	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3, "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes, TLAA
E1.7-b E1.7.5	Regenerative heat exchanger Closure bolting	Low-alloy steel, carbon steel	Both sides: chemically treated borated water up to 340°C (644°F)	Loss of material/ Boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No

VII Auxiliary Systems
E1. Chemical and Volume Control System (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
E1.7-c E1.7.1 E1.7.2 E1.7.3 E1.7.4	Regenerative heat exchanger Channel head (including channel weld) and access cover Tubesheet Tubes Shell and access cover	Stainless steel	Both sides: chemically treated borated water up to 340°C (644°F)	Crack initiation and growth/ Stress corrosion cracking, cyclic loading	Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714 The AMP is to be augmented by verifying the absence of cracking due to stress corrosion cracking and cyclic loading, or loss of material due to pitting and crevice corrosion. An acceptable verification program is to include temperature and radioactivity monitoring of the shell side water, and eddy current testing of tubes.	Yes, plant specific
E1.8-a E1.8.1 E1.8.2 E1.8.3 E1.8.4 E1.8.5	Letdown heat exchanger (serviced by closed-cycle cooling water) Channel head (including channel weld) and access cover Tubesheet Tubes Shell and access cover Closure bolting	Stainless steel, carbon steel	Tube side: chemically treated borated water up to 340°C (644°F); shell side: closed-cycle cooling water (treated water)	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3, "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes, TLAA

VII Auxiliary Systems

E1. Chemical and Volume Control System (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
E1.8-b E1.8.1 E1.8.2 E1.8.3	Letdown heat exchanger (serviced by closed-cycle cooling water) Channel head (including channel weld) and access cover Tubesheet Tubes	Stainless steel	Tube side: chemically treated borated water up to 340°C (644°F); shell side: closed-cycle cooling water	Crack initiation and growth/ Stress corrosion cracking, cyclic loading	Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714 The AMP is to be augmented by verifying the absence of cracking due to stress corrosion cracking and cyclic loading, or loss of material due to pitting and crevice corrosion. An acceptable verification program is to include temperature and radioactivity monitoring of the shell side water, and eddy current testing of tubes. (See Oconee operating experience, License Renewal Application, Revision 2, June 1998, p. 3.4-26)	Yes, plant specific
E1.8-c E1.8.4	Letdown heat exchanger (serviced by closed-cycle cooling water) Shell and access cover	Carbon steel	Closed-cycle cooling water	Loss of material/ Pitting and crevice corrosion	Chapter XI.M21, "Closed-Cycle Cooling Water System"	No
E1.8-d E1.8.5	Letdown heat exchanger (serviced by closed-cycle cooling water) Closure bolting	Low-alloy steel, carbon steel	Air, leaking chemically treated borated water	Loss of material/ Boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No
E1.9-a E1.9.1	Basket strainer Closure bolting	Low-alloy steel, carbon steel	Air, leaking chemically treated borated water	Loss of material/ Boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No
E1.10-a E1.10.1	Volume control tank Closure bolting	Low-alloy steel, carbon steel	Air, leaking chemically treated borated water	Loss of material/ Boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No

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E2. STANDBY LIQUID CONTROL SYSTEM (BOILING WATER REACTOR)

E2.1 Piping

E2.1.1 Piping and Fittings

E2.2 Solution Storage

E2.2.1 Tank
E2.2.2 Tank Heaters

E2.3 Valves (Pump Suction, Relief, Injection, Containment Isolation, and Explosive Actuated Discharge)

E2.3.1 Body and Bonnet

E2.4 Injection Pumps

E2.4.1 Casing

E2. STANDBY LIQUID CONTROL SYSTEM (BOILING WATER REACTOR)

Systems, Structures, and Components

This section comprises the portion of the standby liquid control (SLC) system extending from the containment isolation valve to the solution storage tank. The system serves as a backup reactivity control system in all boiling water reactors (BWRs). The major components of this system are the piping, the solution storage tank, the solution storage tank heaters, valves, and pumps. All of the components from the storage tank to the explosive actuated discharge valve operate in contact with a sodium pentaborate ($Na_2B_{10}O_{16} \cdot 10H_2O$) solution.

Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all components that comprise the standby liquid control system are governed by Group B Quality Standards. The portions of the standby liquid control system extending from the reactor coolant pressure boundary up to and including the containment isolation valves are governed by Group A Quality Standards and covered in IV.C1.

Pump and valve internals perform their intended functions with moving parts or with a change in configuration, or are subject to replacement based on qualified life or specified time period. Accordingly, they are not subject to an aging management review, pursuant to 10 CFR 54.21(a)(1).

Aging management programs for the degradation of external surfaces of carbon steel components are included in VII.I.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The system that interfaces with the standby liquid control system is the BWR reactor pressure vessel (IV.A1). If used, the standby liquid control system would inject sodium pentaborate solution into the pressure vessel near the bottom of the reactor core.

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
E2.1-a E2.1.1	Piping Piping and fittings in contact with sodium pentaborate solution	Stainless steel	Sodium pentaborate solution at 21 - 32 °C (70 - 90°F) (≈24,500 ppm B)	Crack initiation and growth/ Stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515)	No
E2.2-a E2.2.1 E2.2.2	Solution storage Tank Tank heaters	Stainless steel	Sodium pentaborate solution at 21 - 32 °C (70 - 90°F) (≈24,500 ppm B)	Crack initiation and growth/ Stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515)	No
E2.3-a E2.3.1	Valves (pump suction, relief, injection, containment isolation, and explosive actuated discharge valves) Body and bonnet	Stainless steel	Sodium pentaborate solution at 21 - 32 °C (70 - 90°F) (≈24,500 ppm B)	Crack initiation and growth/ Stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515)	No
E2.4-a E2.4.1	Injection pumps Casing	Stainless steel	Sodium pentaborate solution at 21 - 32 °C (70 - 90°F) (≈24,500 ppm B)	Crack initiation and growth/ Stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515)	No

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E3. REACTOR WATER CLEANUP SYSTEM (BWR)

E3.1 Piping

E3.1.1 Piping and Fittings (Beyond Second Isolation Valves)

E3.2 Reactor Water Cleanup (RWCU) Pump

E3.2.1 Casing

E3.3 Regenerative Heat Exchanger

E3.3.1 Channel Head and Access Cover
E3.3.2 Tubesheet
E3.3.3 Tubes
E3.3.4 Shell and Access Cover

E3.4 Non-Regenerative Heat Exchanger

E3.4.1 Channel Head and Access Cover
E3.4.2 Tubesheet
E3.4.3 Tubes
E3.4.4 Shell and Access Cover

E3. REACTOR WATER CLEANUP SYSTEM (BOILING WATER REACTOR)

Systems, Structures, and Components

This section comprises the reactor water cleanup (RWCU) system, which provides for cleanup and particulate removal from the recirculating reactor coolant in all boiling water reactors (BWRs). Some plants may not include the RWCU system in the scope of license renewal, while other plants may include the RWCU system because it is associated with safety-related functions.

Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," the portion of the RWCU system extending from the reactor coolant recirculation system up to and including the containment isolation valves forms the primary pressure boundary, and is governed by Group A Quality Standards and covered in IV.C1. The remainder of the system outboard of the isolation valves is governed by Group C Quality Standards. In this table, only aging management programs for RWCU-related piping and components outboard of the isolation valves are evaluated. The aging management program for containment isolation valves in the RWCU system is evaluated in IV.C1, which concern the reactor coolant pressure boundary in BWRs.

Pump and valve internals perform their intended functions with moving parts or with a change in configuration, or are subject to replacement based on qualified life or specified time period. Accordingly, they are not subject to an aging management review, pursuant to 10 CFR 54.21(a)(1).

Aging management programs for the degradation of external surfaces of carbon steel components are included in VII.I.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The systems that interface with the BWR reactor water cleanup system include the reactor coolant pressure boundary (IV.C1), the closed-cycle cooling water system (VII.C2), and the condensate system (VIII.E).

VII Auxiliary Systems

E3. Reactor Water Cleanup System (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
E3.1-a E3.1.1	Piping Piping and fittings (beyond second isolation valve)	Stainless steel: types 304, 316, or 316NG	Oxygenated water 93°C -288°C (200°F-550°F)	Crack initiation and growth/ Stress corrosion cracking, intergranular stress corrosion cracking	Chapter XI.M25, "BWR Reactor Water Cleanup System"	No
E3.1-b E3.1.1	Piping Piping and fittings (beyond second isolation valve)	Stainless steel: types 304, 316, or 316NG	Oxygenated water 93°C -288°C (200°F-550°F)	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3, "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes, TLAA
E3.2-a E3.2.1	Reactor water cleanup (RWCU) pump Casing	Cast austenitic stainless steel	Oxygenated water 93°C -288°C (200°F-550°F)	Crack initiation and growth/ Stress corrosion cracking, intergranular stress corrosion cracking	Chapter XI.M25, "BWR Reactor Water Cleanup System"	No
E3.2-b E3.2.1	RWCU pump Casing	Cast austenitic stainless steel, stainless steel	Oxygenated water 93°C -288°C (200°F-550°F)	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3, "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes, TLAA
E3.2-c E3.2.2	RWCU pump Closure bolting	High strength low-alloy steel	Air, Leaking oxygenated water	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3, "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes, TLAA

VII Auxiliary Systems
E3. Reactor Water Cleanup System (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
E3.3-d E3.3.1 E3.3.2 E3.3.3 E3.3.4	Regenerative heat exchanger Channel head and access cover Tubesheet Tubes Shell and access cover	Channel head and access cover, tubesheet, tubes: stainless steel; Shell and access cover: high strength low-alloy steel with stainless steel cladding	Oxygenated water at 288°C (550°F) and 10 MPa max.	Crack initiation and growth/ Stress corrosion cracking	A plant-specific aging management program is to be evaluated.	Yes, plant specific
E3.4-a E3.4.1 E3.4.2 E3.4.3 E3.4.4	Non-regenerative heat exchanger (serviced by closed-cycle cooling water) Channel head and access cover Tubesheet Tubes Shell and access cover	Channel head and access cover, tubesheet, tubes: stainless steel; Shell and access cover: high strength low-alloy steel with stainless steel cladding	Reactor coolant water at 288°C (550°F) and 10 MPa max.	Crack initiation and growth/ Stress corrosion cracking	A plant-specific aging management program is to be evaluated.	Yes, plant specific
E3.4-b E3.4.4	Non-regenerative heat exchanger (serviced by closed-cycle cooling water) Shell and access cover	High strength low-alloy steel with stainless steel cladding	Reactor coolant water at 10 MPa max.	Loss of material/ Microbiologically influenced corrosion (for portions of the RWCU system <93°C [200°F])	Chapter XI.M21, "Closed-Cycle Cooling Water System"	No

E4. SHUTDOWN COOLING SYSTEM (OLDER BWR)

E4.1 Piping

E4.1.1 Piping and Fittings

E4.2 Pump

E4.2.1 Casing

E4.3 Valves (Check, Control, Hand, Motor Operated, and Relief)

E4.3.1 Body and Bonnet

E4.4 Heat Exchanger

E4.4.1 Channel Head and Access Cover
E4.4.2 Tubesheet
E4.4.3 Tubes
E4.4.4 Shell and Access Cover

E4. SHUTDOWN COOLING SYSTEM (OLDER BWR)

Systems, Structures, and Components

This section comprises the shutdown cooling (SDC) system for older vintage boiling water reactors (BWRs) and consists of piping and fittings, the SDC system pump, the heat exchanger, and valves.

Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all components that comprise the SDC system are governed by Group B Quality Standards. Portions of the SDC system extending from the reactor coolant pressure boundary up to and including the containment isolation valves are governed by Group A Quality Standards and covered in IV.C1.

Pump and valve internals perform their intended functions with moving parts or with a change in configuration, or are subject to replacement based on qualified life or specified time period. Accordingly, they are not subject to an aging management review, pursuant to 10 CFR 54.21(a)(1).

Aging management programs for the degradation of external surfaces of carbon steel components are included in VII.I.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The systems that interface with the SDC system include the reactor coolant pressure boundary (IV.C1) and the closed-cycle cooling water system (VII.C2).

VII Auxiliary Systems
E4. Shutdown Cooling System (Older Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
E4.1-a E4.1.1	Piping Piping and fittings	Carbon steel, stainless steel	Oxygenated water, up to 288°C (550°F)	Loss of material/ Pitting and crevice corrosion	Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515) The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated
E4.1-b E4.1.1	Piping Piping and fittings	Carbon steel, stainless steel	Oxygenated water, up to 288°C (550°F)	Cumulative fatigue damage/ Fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3, "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes TLAA
E4.1-c E4.1.1	Piping Piping and fittings	Stainless steel	Oxygenated water, up to 288°C (550°F)	Crack initiation and growth/ Stress corrosion cracking	Chapter XI.M7, "BWR Stress Corrosion Cracking" and Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515)	No
E4.2-a E4.2.1	Pump Casing	Carbon steel	Oxygenated water, up to 288°C (550°F)	Loss of material/ Pitting and crevice corrosion	Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515) The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated
E4.3-a E4.3.1	Valves (check, control, hand, motor operated, and relief valves) Body and bonnet	Stainless steel forging, stainless steel casting	Oxygenated water, up to 288°C (550°F)	Crack initiation and growth/ Stress corrosion cracking	Chapter XI.M7, "BWR Stress Corrosion Cracking" and Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515)	No

VII Auxiliary Systems
E4. Shutdown Cooling System (Older Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
E4.4-a E4.4.1 E4.4.2 E4.4.3 E4.4.4	Heat exchanger (serviced by closed-cycle cooling water system) Channel head and access cover Tubesheet Tubes Shell and access cover	Channel head and access cover: carbon steel; tubesheet: carbon steel (stainless steel cladding on channel side; Tubes: stainless steel; Shell: carbon steel	Reactor coolant water, and closed-cycle cooling water	Loss of material/ Pitting, crevice and microbiologically influenced corrosion	Chapter XI.M21, "Closed-Cycle Cooling Water System"	No

F1. CONTROL ROOM AREA VENTILATION SYSTEM

F1.1 Duct

- F1.1.1 Duct Fittings, Access Doors, and Closure Bolts
- F1.1.2 Equipment Frames and Housing
- F1.1.3 Flexible Collars between Ducts and Fans
- F1.1.4 Seals in Dampers and Doors

F1.2 Air Handler Heating/Cooling

- F1.2.1 Heating/Cooling Coils

F1.3 Piping

- F1.3.1 Piping and Fittings

F1.4 Filters

- F1.4.1 Housing and Supports
- F1.4.2 Elastomer Seals

F1. CONTROL ROOM AREA VENTILATION SYSTEM

Systems, Structures, and Components

This section comprises the control room area ventilation system (with warm moist air as the normal environment), which contains ducts, piping and fittings, equipment frames and housings, flexible collars and seals, filters, and heating and cooling air handlers. Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all components that comprise the control room area ventilation system are governed by Group B Quality Standards.

With respect to filters and seals, these items are to be addressed consistent with the NRC position on consumables, provided in the NRC letter from Christopher I. Grimes to Douglas J. Walters of NEI, dated March 10, 2000. Specifically, components that function as system filters and seals are typically replaced based on performance or condition monitoring that identifies whether these components are at the end of their qualified lives and may be excluded, on a plant-specific basis, from an aging management review under 10 CFR 54.21(a)(1)(ii). The application is to identify the standards that are relied on for replacement as part of the methodology description, for example, NFPA standards for fire protection equipment.

Aging management programs for the degradation of external surfaces of carbon steel components are included in VII.I.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The system that interfaces with the control room area ventilation system is the auxiliary and radwaste area ventilation system (VII.F2). The cooling coils receive their cooling water from other systems, such as the hot water heating system or the chilled water cooling system.

VII Auxiliary Systems
F1. Control Room Area Ventilation System

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
F1.1-a F1.1.1 F1.1.2	Duct Duct fittings, access doors, and closure bolts Equipment frames and housing	Carbon steel (galvanized or painted) bolts: plated carbon steel	Warm, moist air	Loss of material/ General, pitting, crevice corrosion, and microbiologically influenced corrosion (for duct [drip-pan] and piping for moisture drainage)	A plant-specific aging management program is to be evaluated.	Yes, plant specific
F1.1-b F1.1.3 F1.1.4	Duct Flexible collars between ducts and fans Seals in dampers and doors	Elastomer (Neoprene)	Warm, moist air	Hardening and loss of strength/ Elastomer degradation	A plant-specific aging management program is to be evaluated.	Yes, plant specific
F1.1-c F1.1.3 F1.1.4	Duct Flexible collars between ducts and fans Seals in dampers and doors	Elastomer (Neoprene)	Warm, moist air	Loss of material/ Wear	A plant-specific aging management program is to be evaluated.	Yes, plant specific
F1.2-a F1.2.1	Air handler heating/ cooling Heating/ cooling coils	Copper/ nickel	Warm, moist air	Loss of material/ Pitting and crevice corrosion	A plant-specific aging management program is to be evaluated.	Yes, plant specific
F1.3-a F1.3.1	Piping Piping and fittings	Carbon steel	Hot or cold treated water	Loss of material/ General, pitting, crevice corrosion	Chapter XI.M21, "Closed-Cycle Cooling Water System"	No
F1.4-a F1.4.1	Filters Housing and supports	Carbon steel, stainless steel	Warm, moist air	Loss of material/ General (only for carbon steel), pitting, and crevice corrosion	A plant-specific aging management program is to be evaluated.	Yes, plant specific
F1.4-b F1.4.2	Filters Elastomer seals	Elastomers (Neoprene and similar materials)	Warm, moist air	Hardening and loss of strength/ Elastomer degradation	A plant-specific aging management program is to be evaluated.	Yes, plant specific

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F2. AUXILIARY AND RADWASTE AREA VENTILATION SYSTEM

F2.1 Duct

- F2.1.1 Duct Fittings, Access Doors, and Closure Bolts
- F2.1.2 Equipment Frames and Housing
- F2.1.3 Flexible Collars between Ducts and Fans
- F2.1.4 Seals in Dampers and Doors

F2.2 Air Handler Heating/Cooling

- F2.2.1 Heating/Cooling Coils

F2.3 Piping

- F2.3.1 Piping and Fittings

F2.4 Filters

- F2.4.1 Housing and Supports
- F2.4.2 Elastomer Seals

F2. Auxiliary and Radwaste Area Ventilation System

Systems, Structures, and Components

This section comprises the auxiliary and radwaste areas ventilation systems (with warm moist air as the normal environment) and contains ducts, piping and fittings, equipment frames and housings, flexible collars and seals, filters, and heating and cooling air handlers. Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all components that comprise the auxiliary and radwaste area ventilation system are governed by Group B Quality Standards.

With respect to filters and seals, these items are to be addressed consistent with the NRC position on consumables, provided in the NRC letter from Christopher I. Grimes to Douglas J. Walters of NEI, dated March 10, 2000. Specifically, components that function as system filters and seals are typically replaced based on performance or condition monitoring that identifies whether these components are at the end of their qualified lives and may be excluded, on a plant-specific basis, from an aging management review under 10 CFR 54.21(a)(1)(ii). The application is to identify the standards that are relied on for replacement as part of the methodology description, for example, NFPA standards for fire protection equipment.

Aging management programs for the degradation of external surfaces of carbon steel components are included in VII.I.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The systems that interface with the auxiliary and radwaste area ventilation system are the control room area ventilation system (VII.F1) and the diesel generator building ventilation system (VII.F4). The cooling coils receive their cooling water from other systems, such as the hot water heating system or the chilled water cooling system.

VII Auxiliary Systems
F2. Auxiliary and Radwaste Area Ventilation System

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
F2.1-a F2.1.1 F2.1.2	Duct Duct fittings, access doors, and closure bolts Equipment frames and housing	Carbon steel (galvanized or painted) bolts: plated carbon steel	Warm, moist air	Loss of material/ General, pitting, crevice corrosion, and microbiologically influenced corrosion (for duct [drip-pan] and piping for moisture drainage)	A plant-specific aging management program is to be evaluated.	Yes, plant specific
F2.1-b F2.1.3 F2.1.4	Duct Flexible collars between ducts and fans Seals in dampers and doors	Elastomer (Neoprene)	Warm, moist air	Hardening and loss of strength/ Elastomer degradation	A plant-specific aging management program is to be evaluated.	Yes, plant specific
F2.1-c F2.1.3 F2.1.4	Duct Flexible collars between ducts and fans Seals in dampers and doors	Elastomer (Neoprene)	Warm, moist air	Loss of material/ Wear	A plant-specific aging management program is to be evaluated.	Yes, plant specific
F2.2-a F2.2.1	Air handler heating/ cooling Heating/ cooling coils	Copper/ nickel	Warm, moist air	Loss of material/ Pitting and crevice corrosion	A plant-specific aging management program is to be evaluated.	Yes, plant specific
F2.3-a F2.3.1	Piping Piping and fittings	Carbon steel	Hot or cold treated water	Loss of material/ General, pitting, crevice corrosion	Chapter XI.M21, "Closed-Cycle Cooling Water System"	No
F2.4-a F2.4.1	Filters Housing and supports	Carbon steel, stainless steel	Warm, moist air	Loss of material/ General (only for carbon steel), pitting and crevice corrosion	A plant-specific aging management program is to be evaluated.	Yes, plant specific
F2.4-b F2.4.2	Filters Elastomer seals	Elastomers (Neoprene and similar materials)	Warm, moist air	Hardening and loss of strength/ Elastomer degradation	A plant-specific aging management program is to be evaluated.	Yes, plant specific

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F3. PRIMARY CONTAINMENT HEATING AND VENTILATION SYSTEM

F3.1 Duct

F3.1.1 Duct Fittings, Access Doors, and Closure Bolts
F3.1.2 Equipment Frames and Housing
F3.1.3 Flexible Collars between Ducts and Fans
F3.1.4 Seals in Dampers and Doors

F3.2 Air Handler Heating/Cooling

F3.2.1 Heating/Cooling Coils

F3.3 Piping

F3.3.1 Piping and Fittings

F3.4 Filters

F3.4.1 Housing and Supports
F3.4.2 Elastomer Seals

F3. PRIMARY CONTAINMENT HEATING AND VENTILATION SYSTEM

Systems, Structures, and Components

This section comprises the primary containment heating and ventilation system (with warm moist air as the normal environment), which contains ducts, piping and fittings, equipment frames and housings, flexible collars and seals, filters, and heating and cooling air handlers. Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all components that comprise the primary containment heating and ventilation system are governed by Group C Quality Standards.

With respect to filters and seals, these items are to be addressed consistent with the NRC position on consumables, provided in the NRC letter from Christopher I. Grimes to Douglas J. Walters of NEI, dated March 10, 2000. Specifically, components that function as system filters and seals are typically replaced based on performance or condition monitoring that identifies whether these components are at the end of their qualified lives and may be excluded, on a plant-specific basis, from an aging management review under 10 CFR 54.21(a)(1)(ii). The application is to identify the standards that are relied on for replacement as part of the methodology description, for example, NFPA standards for fire protection equipment.

Aging management programs for the degradation of external surfaces of carbon steel components are included in VII.I.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The systems that interface with the primary containment heating and ventilation system are the closed-cycle cooling water system (VII.C2) and the PWR and BWR containments (II.A and II.B, respectively). The cooling coils receive their cooling water from other systems, such as the hot water heating system or the chilled water cooling system.

VII Auxiliary Systems
F3. Primary Containment Heating and Ventilation System

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
F3.1-a F3.1.1 F3.1.2	Duct Duct fittings, access doors and closure bolts Equipment frames and housing	Carbon steel (galvanized or painted) bolts: plated carbon steel	Warm, moist air	Loss of material/ General, pitting, crevice corrosion, and microbiologically influenced corrosion (for duct [drip-pan] and piping for moisture drainage)	A plant-specific aging management program is to be evaluated.	Yes, plant specific
F3.1-b F3.1.3 F3.1.4	Duct Flexible collars between ducts and fans Seals in dampers and doors	Elastomer (Neoprene)	Warm, moist air	Hardening and loss of strength/ Elastomer degradation	A plant-specific aging management program is to be evaluated.	Yes, plant specific
F3.1-c F3.1.3 F3.1.4	Duct Flexible collars between ducts and fans Seals in dampers and doors	Elastomer (Neoprene)	Warm, moist air	Loss of material/ Wear	A plant-specific aging management program is to be evaluated.	Yes, plant specific
F3.2-a F3.2.1	Air handler heating/ cooling Heating/ cooling coils	Copper/nickel	Warm, moist air	Loss of material/ Pitting and crevice corrosion	A plant-specific aging management program is to be evaluated.	Yes, plant specific
F3.3-a F3.3.1	Piping Piping and fittings	Carbon steel	Hot or cold treated water	Loss of material/ General, pitting, crevice corrosion	Chapter XI.M21, "Closed-Cycle Cooling Water System"	No
F3.4-a F3.4.1	Filters Housing and supports	Carbon steel, stainless steel	Warm, moist air	Loss of material/ General (only for carbon steel), pitting and crevice corrosion	A plant-specific aging management program is to be evaluated.	Yes, plant specific
F3.4-b F3.4.2	Filters Elastomer seals	Elastomers (Neoprene and similar materials)	Warm, moist air	Hardening and loss of strength/ Elastomer degradation	A plant-specific aging management program is to be evaluated.	Yes, plant specific

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