IV Reactor Vessel, Internals, and Reactor Coolant System
A1. Reactor Vessel (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
A1.1-a A1.1.1 A1.1.2	Top head enclosure (without cladding) Top head Nozzles (vent, top head spray or RCIC, and spare)	SA302-Gr B, SA533-Gr B, SA336	288°C (550°F) steam	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)	No
	Top head enclosure (without cladding) Top head Nozzles (vent, top head spray or RCIC, and spare)	Carbon steel	Reactor coolant	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)	No
A1.1-b A1.1.3	Top head enclosure Head flange	SA302-Gr B, SA533-Gr B, SA336, with or without stainless-steel cladding	288°C (550°F) steam	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
A1. Reactor Vessel (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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
A1.1-c A1.1.4	Top head enclosure Closure studs and nuts	SA193-Gr. B7, SA540-Gr. B23/24, SA320-Gr. L43 (AISI 4340), SA194-Gr. 7; maximum tensile strength <1172 MPa (<170 Ksi)	Air, leaking reactor coolant water and/or steam at 288°C (550°F)	Crack initiation and growth/ Stress corrosion cracking, intergranular stress corrosion cracking	Chapter XI.M3, “Reactor Head Closure Studs”	No

IV Reactor Vessel, Internals, and Reactor Coolant System
A1. Reactor Vessel (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
	Top head enclosure Closure studs and nuts	High strength low alloy steel	Air with reactor coolant leakage	Crack initiation and growth/ Stress corrosion cracking, intergranular stress corrosion cracking	Chapter XI.M3, “Reactor Head Closure Studs”	No
A1.1-d A1.1.5	Top head enclosure Vessel flange leak detection line	Stainless steel, Ni alloys	Leaking reactor coolant water and/or steam up to 288°C (550°F)	Crack initiation and growth/ Stress corrosion cracking, intergranular stress corrosion cracking	A plant-specific aging management program is to be evaluated because existing programs may not be able to mitigate or detect crack initiation and growth due to SCC of vessel flange leak detection line.	Yes, plant specific
	Top head enclosure Vessel flange leak detection line	Stainless steel, nickel alloy	Air with reactor coolant leakage	Crack initiation and growth/ Stress corrosion cracking, intergranular stress corrosion cracking	A plant-specific aging management program is to be evaluated because existing programs may not be able to mitigate or detect crack initiation and growth due to SCC of vessel flange leak detection line.	Yes, plant specific
A1.2-a A1.2.1 A1.2.2	Vessel shell Vessel flange Upper shell	SA302-Gr B, SA533-Gr B, SA336 with stainless steel cladding	288°C (550°F) steam	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
A1. Reactor Vessel (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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
A1.2-b A1.2.3 A1.2.4 A1.2.5 A1.2.6	Vessel shell Intermediate nozzle shell Intermediate beltline shell Lower shell Beltline welds	SA302-Gr B, SA533-Gr B with 308, 309, 308L, 309L cladding	288°C (550°F) reactor coolant water max $5x10^9$ $n/cm^2 \cdot s$	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
A1. Reactor Vessel (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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
A1. Reactor Vessel (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
A1.2-c A1.2.4 A1.2.6	Vessel shell Intermediate beltline shell Beltline welds	SA302-Gr B, SA533-Gr B with 308, 309, 308L, 309L cladding; and low–alloy steel weldments	288°C (550°F) reactor coolant water $5x10^8$ - $5x10^9$ $n/cm^2 \cdot s$	Loss of fracture toughness/ Neutron irradiation embrittlement	Neutron irradiation embrittlement is a time dependent aging mechanism to be evaluated for the period of extended operation for all ferritic materials that have a neutron fluence exceeding 10^{17} n/cm^2 (E >1 MeV) at the end of the license renewal term. Aspects of this evaluation may involve a TLAA. In accordance with approved BWRVIP-74, the TLAA is to evaluate the impact of neutron embrittlement on: (a) the adjusted reference temperature, the plant's pressure-temperature limits, (b) the need for inservice inspection of circumferential welds, and (c) the Charpy upper shelf energy or the equivalent margins analyses performed in accordance with 10 CFR 50, Appendix G. Additionally, the applicant is to monitor axial beltline weld embrittlement. One acceptable method is to determine that the mean RT_{NDT} of the axial beltline welds at the end of the extended period of operation is less than the value specified by the staff in its May 7, 2000 letter. 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
A1. Reactor Vessel (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
	Vessel shell Intermediate beltline shell Beltline welds	Carbon steel with or without stainless steel cladding	Neutron flux	Loss of fracture toughness/ Neutron irradiation embrittlement	Neutron irradiation embrittlement is a time dependent aging mechanism to be evaluated for the period of extended operation for all ferritic materials that have a neutron fluence exceeding 10^{17} n/cm^2 (E >1 MeV) at the end of the license renewal term. Aspects of this evaluation may involve a TLAA. In accordance with approved BWRVIP-74, the TLAA is to evaluate the impact of neutron embrittlement on: (a) the adjusted reference temperature, the plant's pressure-temperature limits, (b) the need for inservice inspection of circumferential welds, and (c) the Charpy upper shelf energy or the equivalent margins analyses performed in accordance with 10 CFR 50, Appendix G. Additionally, the applicant is to monitor axial beltline weld embrittlement. One acceptable method is to determine that the mean RT_{NDT} of the axial beltline welds at the end of the extended period of operation is less than the value specified by the staff in its May 7, 2000 letter. 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
A1. Reactor Vessel (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
A1.2-d A1.2.4 A1.2.6	Vessel shell Intermediate beltline shell Beltline welds	SA302-Gr B, SA533-Gr B with 308, 309, 308L, 309L cladding; and low–alloy steel weldments	288°C (550°F) reactor coolant water $5x10^8$ - $5x10^9$ $n/cm^2 \cdot s$	Loss of fracture toughness/ Neutron irradiation embrittlement	Chapter XI.M31, “Reactor Vessel Surveillance”	Yes, plant specific
	Vessel shell Intermediate beltline shell Beltline welds	Carbon steel with or without stainless steel cladding	Neutron flux	Loss of fracture toughness/ Neutron irradiation embrittlement	Chapter XI.M31, “Reactor Vessel Surveillance”	Yes, plant specific
A1.2-e A1.2.7	Vessel shell Attachment welds	Stainless steel, Inconel 182	288°C (550°F) reactor coolant water	Crack initiation and growth/ Stress corrosion cracking, intergranular stress corrosion cracking	Chapter XI.M4, “BWR Vessel ID Attachment Welds,” and Chapter XI.M2, “Water Chemistry,” for BWR water in BWRVIP-29 (EPRI TR-103515)	No
	Vessel shell Attachment welds	Stainless steel, nickel alloy	Reactor coolant	Crack initiation and growth/ Stress corrosion cracking, intergranular stress corrosion cracking	Chapter XI.M4, “BWR Vessel ID Attachment Welds,” and Chapter XI.M2, “Water Chemistry,” for BWR water in BWRVIP-29 (EPRI TR-103515)	No

IV Reactor Vessel, Internals, and Reactor Coolant System
A1. Reactor Vessel (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
A1.3-a A1.3.1	Nozzles Main steam	SA508-Cl2 with or without stainless-steel cladding	288°C (550°F) steam	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
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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
A1.3-b A1.3.2	Nozzles Feedwater	SA508-Cl2 with or without stainless steel cladding	Up to 288°C (550°F), reactor coolant water	Crack initiation and growth/ Cyclic loading	Chapter XI.M5, “BWR Feedwater Nozzle”	No

IV Reactor Vessel, Internals, and Reactor Coolant System
A1. Reactor Vessel (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
	Nozzles Feedwater	Carbon steel with or without stainless steel cladding	Reactor coolant	Crack initiation and growth/ Cyclic loading	Chapter XI.M5, "BWR Feedwater Nozzle"	No
A1.3-c A1.3.3	Nozzles Control rod drive return line	SA508-Cl2 with or without stainless steel cladding	Up to 288°C (550°F), reactor coolant water	Crack initiation and growth/ Cyclic loading	Chapter XI.M6, "BWR Control Rod Drive Return Line Nozzle"	No
	Nozzles Control rod drive return line	Carbon steel with or without stainless steel cladding	Reactor coolant	Crack initiation and growth/ Cyclic loading	Chapter XI.M6, "BWR Control Rod Drive Return Line Nozzle"	No
A1.3-d A1.3.2 A1.3.3	Nozzles Feedwater Control rod drive return line	SA508-Cl2 with or without stainless steel cladding	Up to 288°C (550°F), reactor coolant water	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
A1. Reactor Vessel (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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
A1. Reactor Vessel (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
A1.3-e A1.3.4	Nozzles Low pressure coolant injection or RHR injection mode	SA508-Cl2	Up to 288°C reactor coolant water $5x10^8$ - $5x10^9$ $n/cm^2 \cdot s$	Loss of fracture toughness/ Neutron irradiation embrittlement	Neutron irradiation embrittlement is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation 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. In accordance with approved BWRVIP-74, the TLAA is to evaluate the impact of neutron embrittlement on: (a) the adjusted reference temperature, the plant's pressure-temperature limits, (b) the Charpy upper shelf energy, and (c) the equivalent margins analyses performed in accordance with 10 CFR 50, Appendix G. The applicant may choose to demonstrate that the materials of the nozzles are not controlling for the TLAA evaluations. 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
A1. Reactor Vessel (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
	Nozzles Low pressure coolant injection or RHR injection mode	Carbon steel	Neutron flux	Loss of fracture toughness/ Neutron irradiation embrittlement	Neutron irradiation embrittlement is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation 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. In accordance with approved BWRVIP-74, the TLAA is to evaluate the impact of neutron embrittlement on: (a) the adjusted reference temperature, the plant's pressure-temperature limits, (b) the Charpy upper shelf energy, and (c) the equivalent margins analyses performed in accordance with 10 CFR 50, Appendix G. The applicant may choose to demonstrate that the materials of the nozzles are not controlling for the TLAA evaluations. 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
A1.4-a A1.4.1 A1.4.2 A1.4.3 A1.4.4 A1.4.5	Nozzle safe ends High pressure core spray Low pressure core spray Control rod drive return line Recirculating water Low pressure coolant injection or RHR injection mode	Stainless steel, SB-166 (Inconel 182 butter, and Inconel 82 or 182 weld)	Up to 288°C (550°F), reactor coolant 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

IV Reactor Vessel, Internals, and Reactor Coolant System
A1. Reactor Vessel (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
	Nozzle safe ends High pressure core spray Low pressure core spray Control rod drive return line Recirculating water Low pressure coolant injection or RHR injection mode	Stainless steel, nickel alloy	Reactor coolant	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
A1.4-b A1.4.3	Nozzle safe ends Control rod drive return line	Stainless steel, SB-166 (Inconel 182 butter, and Inconel 82 or 182 weld)	Up to 288°C (550°F), reactor coolant water	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
A1. Reactor Vessel (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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
A1.5-a A1.5.1 A1.5.2 A1.5.3 A1.5.4 A1.5.5 A1.5.6	Penetrations Control rod drive stub tubes Instrumentation Jet pump instrument Standby liquid control Flux monitor Drain line	Stainless steel, SB-167	Up to 288°C (550°F), reactor coolant water	Crack initiation and growth/ Stress corrosion cracking, intergranular stress corrosion cracking, cyclic loading	Chapter XI.M8, "BWR Penetrations," and Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515)	No
	Penetrations Control rod drive stub tubes Instrumentation Jet pump instrument Standby liquid control Flux monitor Drain line	Stainless steel, nickel alloy	Reactor coolant	Crack initiation and growth/ Stress corrosion cracking, intergranular stress corrosion cracking, cyclic loading	Chapter XI.M8, "BWR Penetrations," and Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515)	No

IV Reactor Vessel, Internals, and Reactor Coolant System
A1. Reactor Vessel (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
A1.5-b A1.5.1 A1.5.2 A1.5.3 A1.5.4 A1.5.5 A1.5.6	Penetrations Control rod drive stub tubes Instrumentation Jet pump instrument Standby liquid control Flux monitor Drain line	Stainless steel, SB-167	Up to 288°C (550°F), reactor coolant water	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
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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
A1. Reactor Vessel (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
A1.6-a	Bottom head	SA302-Gr B, SA533-Gr B with 308, 309, 308L, 309L cladding	Up to 288°C (550°F) reactor coolant water	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
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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
A1. Reactor Vessel (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
A1.7-a	Support skirt and attachment welds	SA533-Gr B (Welds low-alloy steel)	Ambient temperature 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
	Support skirt and attachment welds	Carbon steel	Air – indoor uncontrolled	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

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
R-17	Piping and components external surfaces and bolting	Carbon steel	Air with boric acid leakage	Loss of material/ Boric acid corrosion	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
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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
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
	Closure head Stud assembly	High strength low alloy steel	Air with reactor coolant leakage	Crack initiation and growth/ Stress corrosion cracking	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-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
	Closure head Stud assembly	High strength low alloy steel	Air with reactor coolant leakage	Loss of material/ Wear	Chapter XI.M3, “Reactor Head Closure Studs”	No
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
	Closure head Stud assembly	High strength low alloy steel	Air with reactor coolant leakage	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

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-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
	Closure head Vessel flange leak detection line	Stainless steel	Air with reactor coolant leakage	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
	Control rod drive head penetration Nozzle	Nickel alloy	Reactor coolant	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

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
A2.2-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
	Control rod drive head penetration Pressure housing	Stainless steel; cast austenitic stainless steel, nickel alloy	Reactor coolant	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
A2. Reactor Vessel (Pressurized Water Reactor)

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
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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
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

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
	Control rod drive head penetration Pressure housing	Cast austenitic stainless steel	Reactor coolant	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
	Control rod drive head penetration Flange bolting	Stainless steel	Air with reactor coolant leakage	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
	Control rod drive head penetration Flange bolting	Stainless steel	Air with reactor coolant leakage	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
	Control rod drive head penetration Flange bolting	Stainless steel	Air with reactor coolant leakage	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

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
	Nozzles Inlet Outlet Safety injection	Carbon steel with stainless steel cladding	Neutron flux	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
	Nozzles Inlet Outlet Safety injection	Carbon steel with stainless steel cladding	Neutron flux	Loss of fracture toughness/ Neutron irradiation embrittlement	Chapter XI.M31, "Reactor Vessel Surveillance"	Yes, plant specific
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
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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
A2. Reactor Vessel (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
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
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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
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
	Nozzle safe ends Inlet Outlet Safety injection	Stainless steel, cast austenitic stainless steel, nickel alloy	Reactor coolant	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

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-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
	Vessel shell Upper shell Intermediate and lower shell (including beltline welds)	Carbon steel with stainless steel cladding	Neutron flux	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, 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-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
	Vessel shell Upper shell Intermediate and lower shell (including beltline welds)	SA508-Cl 2 forgings clad with stainless steel using a high-heat-input welding process	Reactor coolant	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

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-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
	Vessel shell Upper shell Intermediate and lower shell (including beltline welds)	Carbon steel with stainless steel cladding	Neutron flux	Loss of fracture toughness/ Neutron irradiation embrittlement	Chapter XI.M31, "Reactor Vessel Surveillance"	Yes, plant specific
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

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
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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
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
R-17	Piping and components external surfaces and bolting	Carbon steel	Air with boric acid leakage	Loss of material/ Boric acid corrosion	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
	Vessel shell Vessel flange	Carbon steel	Reactor coolant	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
A2. Reactor Vessel (Pressurized Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
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
	Core support pads/core guide lugs	Nickel alloy	Reactor coolant	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
	Penetrations Instrument tubes (bottom head)	Nickel alloy	Reactor coolant	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-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
	Penetrations Head vent pipe(top head) Instrument tubes (top head)	Nickel alloy	Reactor coolant	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

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
A2.8-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
	Pressure vessel support Skirt support	Carbon steel	Air – indoor uncontrolled	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
R-17	Piping and components external surfaces and bolting	Carbon steel	Air with boric acid leakage	Loss of material/ Boric acid corrosion	Chapter XI.M10**,** **"**Boric Acid Corrosion**"**	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-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
	Core shroud and core plate Core shroud (upper, central, lower)	Stainless steel	Reactor coolant	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

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
	Core shroud and core plate Core plate Core plate bolts (used in early BWRs)	Stainless steel	Reactor coolant	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
R-53	Reactor vessel internals components	Stainless steel, cast austenitic stainless steel, nickel alloy	Reactor coolant	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
	Core shroud and core plate Access hole cover (welded covers)	Nickel alloy	Reactor coolant	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

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-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
	Core shroud and core plate Access hole cover (mechanical covers)	Nickel alloy	Reactor coolant	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
	Core shroud and core plate Shroud support structure (shroud support cylinder, shroud support plate, shroud support legs)	Nickel alloy	Reactor coolant	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
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
	Core shroud and core plate LPCI coupling	Stainless steel	Reactor coolant	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

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
B1.2-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
	Top guide	Stainless steel	Reactor coolant	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
R-53	Reactor vessel internals components	Stainless steel, cast austenitic stainless steel, nickel alloy	Reactor coolant	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
B1.3-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
	Core spray lines and spargers Core spray lines (headers) Spray rings Spray nozzles Thermal sleeves	Stainless steel	Reactor coolant	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

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-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
R-53	Reactor vessel internals components	Stainless steel, cast austenitic stainless steel, nickel alloy	Reactor coolant	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
	Jet pump assemblies Thermal sleeve Inlet header Riser brace arm Holddown beams Inlet elbow Mixing assembly Diffuser Castings	Nickel alloy, cast austenitic stainless steel, stainless steel	Reactor coolant	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
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
R-53	Reactor vessel internals components	Stainless steel, cast austenitic stainless steel, nickel alloy	Reactor coolant	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

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
	Jet pump assemblies Castings	Cast austenitic stainless steel	Reactor coolant	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
	Jet pump assemblies Jet pump sensing line	Stainless steel	Reactor coolant	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
	Fuel supports and control rod drive assemblies Orificed fuel support	Cast austenitic stainless steel	Reactor coolant	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
R-53	Reactor vessel internals components	Stainless steel, cast austenitic stainless steel, nickel alloy	Reactor coolant	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
B1.5-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
	Fuel supports and control rod drive assemblies Control rod drive housing	Stainless steel	Reactor coolant	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

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
	Instrumentation Intermediate range monitor (IRM) dry tubes Source range monitor (SRM) dry tubes Incore neutron flux monitor guide tubes	Stainless steel	Reactor coolant	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
R-53	Reactor vessel internals components	Stainless steel, cast austenitic stainless steel, nickel alloy	Reactor coolant	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-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
	Upper internals assembly Upper support plate Upper core plate Hold-down spring	Stainless steel	Reactor coolant	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
	Upper internals assembly Upper support plate Upper core plate Hold-down spring	Stainless steel	Reactor coolant	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
B2.1-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
R-53	Reactor vessel internals components	Stainless steel, cast austenitic stainless steel, nickel alloy	Reactor coolant	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
	Upper internals assembly Hold-down spring	Stainless steel	Reactor coolant	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
	Upper internals assembly Upper support column	Stainless steel, cast austenitic stainless steel	Reactor coolant	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
	Upper internals assembly Upper support column	Stainless steel, cast austenitic stainless steel	Reactor coolant	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
B2.1-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
	Upper internals assembly Upper support column (only cast austenitic stainless steel portions)	Cast austenitic stainless steel	Reactor coolant and neutron flux	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
R-53	Reactor vessel internals components	Stainless steel, cast austenitic stainless steel, nickel alloy	Reactor coolant	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
	Upper internals assembly Upper support column bolts Upper core plate alignment pins Fuel alignment pins	Stainless steel, nickel alloy	Reactor coolant	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

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
	Upper internals assembly Upper support column bolts Upper core plate alignment pins Fuel alignment pins	Stainless steel, nickel alloy	Reactor coolant	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
	Upper internals assembly Upper support column bolts	Stainless steel, nickel alloy	Reactor coolant	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
	Upper internals assembly Upper core plate alignment pins	Stainless steel, nickel alloy	Reactor coolant	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
R-53	Reactor vessel internals components	Stainless steel, cast austenitic stainless steel, nickel alloy	Reactor coolant	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
	RCCA guide tube assemblies RCCA guide tubes	Stainless steel	Reactor coolant	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
B2.2-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
	RCCA guide tube assemblies RCCA guide tubes	Stainless steel	Reactor coolant	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
R-53	Reactor vessel internals components	Stainless steel, cast austenitic stainless steel, nickel alloy	Reactor coolant	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
B2.2-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
	RCCA guide tube assemblies RCCA guide tube bolts RCCA guide tube support pins	Stainless steel, nickel alloy	Reactor coolant	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

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
	RCCA guide tube assemblies RCCA guide tube bolts, RCCA guide tube support pins	Stainless steel, nickel alloy	Reactor coolant	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
R-53	Reactor vessel internals components	Stainless steel, cast austenitic stainless steel, nickel alloy	Reactor coolant	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

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B2. Reactor Vessel Internals (PWR) – Westinghouse

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
	Core barrel Core barrel (CB) CB flange (upper) CB outlet nozzles Thermal shield	Stainless steel	Reactor coolant	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
B2.3-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
	Core barrel Core barrel (CB) CB flange (upper) CB outlet nozzles Thermal shield	Stainless steel	Reactor coolant	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

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B2. Reactor Vessel Internals (PWR) – Westinghouse

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
	Core barrel Core barrel (CB) CB flange (upper) CB outlet nozzles Thermal shield	Stainless steel	Reactor coolant and neutron flux	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
R-53	Reactor vessel internals components	Stainless steel, cast austenitic stainless steel, nickel alloy	Reactor coolant	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

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B2. Reactor Vessel Internals (PWR) – Westinghouse

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
	Baffle/former assembly Baffle and former plates	Stainless steel	Reactor coolant	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
B2.4-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
	Baffle/former assembly Baffle and former plates	Stainless steel	Reactor coolant	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

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B2. Reactor Vessel Internals (PWR) – Westinghouse

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
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

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B2. Reactor Vessel Internals (PWR) – Westinghouse

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
	Baffle/former assembly Baffle/former bolts	Stainless steel	Reactor coolant 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
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

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B2. Reactor Vessel Internals (PWR) – Westinghouse

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
	Baffle/former assembly Baffle/former bolts	Stainless steel	Reactor coolant	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
	Baffle/former assembly Baffle and former plates	Stainless steel	Reactor coolant and neutron flux	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
	Baffle/former assembly Baffle/former bolts	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/ Neutron irradiation embrittlement	A plant-specific aging management program is to be evaluated.	Yes, plant specific

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B2. Reactor Vessel Internals (PWR) – Westinghouse

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
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
R-53	Reactor vessel internals components	Stainless steel, cast austenitic stainless steel, nickel alloy	Reactor coolant	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-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
	Baffle/former assembly Baffle/former bolts	Stainless steel, nickel alloy	Reactor coolant	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

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B2. Reactor Vessel Internals (PWR) – Westinghouse

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
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
	Lower internal assembly Lower core plate Radial keys and clevis inserts	Stainless steel	Reactor coolant	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
	Lower internal assembly Lower core plate Radial keys and clevis inserts	Stainless steel	Reactor coolant	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

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B2. Reactor Vessel Internals (PWR) – Westinghouse

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
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
	Lower internal assembly Lower core plate	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/ Neutron irradiation embrittlement, void swelling	Chapter XI.M16, “PWR Vessel Internals”	No
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
R-53	Reactor vessel internals components	Stainless steel, cast austenitic stainless steel, nickel alloy	Reactor coolant	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

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B2. Reactor Vessel Internals (PWR) – Westinghouse

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
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
	Lower internal assembly Fuel alignment pins Lower support plate column bolts Clevis insert bolts	Stainless steel, nickel alloy	Reactor coolant	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

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B2. Reactor Vessel Internals (PWR) – Westinghouse

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
	Lower internal assembly Fuel alignment pins Lower support plate column bolts Clevis insert bolts	Stainless steel, nickel alloy	Reactor coolant	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
	Lower internal assembly Fuel alignment pins Lower support plate column bolts Clevis insert bolts	Stainless steel, nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness/ Neutron irradiation embrittlement, void swelling	Chapter XI.M16, "PWR Vessel Internals"	No
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
	Lower internal assembly Lower support plate column bolts	Stainless steel, nickel alloy	Reactor coolant	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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B2. Reactor Vessel Internals (PWR) – Westinghouse

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
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
	Lower internal assembly Clevis insert bolts	Stainless steel, nickel alloy	Reactor coolant	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
R-53	Reactor vessel internals components	Stainless steel, cast austenitic stainless steel, nickel alloy	Reactor coolant	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
B2.5-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
	Lower internal assembly Lower support forging or casting Lower support plate columns	Stainless steel, cast austenitic stainless steel	Reactor coolant	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
	Lower internal assembly Lower support forging or casting Lower support plate columns	Stainless steel, cast austenitic stainless steel	Reactor coolant	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
	Lower internal assembly Lower support casting Lower support plate columns	Cast austenitic stainless steel	Reactor coolant and neutron flux	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
	Lower internal assembly Lower support forging Lower support plate columns	Stainless steel	Reactor coolant and neutron flux	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

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
	Lower internal assembly Radial keys and clevis Inserts	Stainless steel	Reactor coolant	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
R-53	Reactor vessel internals components	Stainless steel, cast austenitic stainless steel, nickel alloy	Reactor coolant	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
B2.6-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
	Instrumentation support structures Flux thimble guide tubes	Stainless steel	Reactor coolant	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
B2.6-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
	Instrumentation support structures Flux thimble guide tubes	Stainless steel	Reactor coolant	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
B2.6-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

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
	Instrumentation support structures Flux thimble	Stainless steel	Reactor coolant	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

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
	Upper Internals Assembly Upper guide structure support plate Fuel alignment plate Fuel alignment plate guide lugs and guide lug inserts	Stainless steel	Reactor coolant	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
	Upper Internals Assembly Upper guide structure support plate Fuel alignment plate Fuel alignment plate guide lugs and guide lug inserts	Stainless steel	Reactor coolant	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
B3.1-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
	Upper Internals Assembly Fuel alignment plate Fuel alignment plate guide lugs and their lugs Hold-down ring	Stainless steel	Reactor coolant	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
	CEA Shroud Assemblies CEA shroud	Stainless steel, cast austenitic stainless steel	Reactor coolant	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
B3.2-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

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
	CEA Shroud Assemblies CEA shrouds bolts	Stainless steel, nickel alloy	Reactor coolant	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
	CEA shroud assemblies CEA shroud CEA shrouds bolts	Stainless steel, cast austenitic stainless steel, nickel alloy	Reactor coolant	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
	CEA shroud assemblies CEA shroud extension shaft guides	Stainless steel	Reactor coolant	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
B3.2-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
	CEA shroud assemblies CEA shroud	Cast austenitic stainless steel	Reactor coolant and neutron flux	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
B3.2-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

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
R-54	Reactor vessel internals components	Stainless steel, cast austenitic stainless steel, nickel alloy	Reactor coolant	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
	CEA shroud assemblies CEA shrouds bolts	Stainless steel, nickel alloy	Reactor coolant	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

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
	Core support barrel Core support barrel Core support barrel upper flange	Stainless steel	Reactor coolant	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
	Core support barrel Core support barrel Core support barrel upper flange	Stainless steel	Reactor coolant	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
B3.3-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

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B3. Reactor Vessel Internals (PWR) – Combustion Engineering

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
	Core support barrel Core support barrel Core support barrel upper flange	Stainless steel	Reactor coolant and neutron flux	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
	Core support barrel Core support barrel upper flange Core support barrel alignment keys	Stainless steel	Reactor coolant	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
	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	Reactor coolant	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
B3.4-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
	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, nickel alloy	Reactor coolant	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-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
	Core shroud assembly Core shroud assembly Core shroud tie rods (core support plate attached by welds in later plants)	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/ Neutron irradiation embrittlement, void swelling	Chapter XI.M16, "PWR Vessel Internals"	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-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
R-54	Reactor vessel internals components	Stainless steel, cast austenitic stainless steel, nickel alloy	Reactor coolant	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
	Core shroud assembly Core shroud assembly bolts (later plants are welded)	Stainless steel, nickel alloy	Reactor coolant	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
B3.4-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
	Core shroud assembly Core shroud assembly bolts (later plants are welded)	Stainless steel, nickel alloy	Reactor coolant	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

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
	Core shroud assembly Core shroud assembly bolts (later plants are welded)	Stainless steel, nickel alloy	Reactor coolant and neutron flux	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
	Core shroud assembly Core shroud assembly bolts Core shroud tie rods	Stainless steel, nickel alloy	Reactor coolant	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
	Lower internal assembly Core support plate Lower support structure beam assemblies Core support column Core support barrel snubber assemblies	Stainless steel	Reactor coolant	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
	Lower internal Assembly Fuel alignment pins Core support column bolts	Stainless steel, nickel alloy	Reactor coolant	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

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
	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, cast austenitic stainless steel, nickel alloy	Reactor coolant	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
	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, nickel alloy	Reactor coolant and neutron flux	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
	Lower internal assembly Fuel alignment pins Core support barrel snubber assemblies	Stainless steel, nickel alloy	Reactor coolant	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
	Lower internal assembly Core support column	Cast austenitic stainless steel	Reactor coolant	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
R-54	Reactor vessel internals components	Stainless steel, cast austenitic stainless steel, nickel alloy	Reactor coolant	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
B4.1-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
	Plenum cover and plenum cylinder Plenum cover assembly Plenum cylinder Reinforcing plates	Stainless steel	Reactor coolant	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
	Plenum cover and plenum cylinder Top flange-to-cover bolts Bottom flange-to-upper grid screws	Stainless steel	Reactor coolant	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
B4.1-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
	Plenum cover and plenum cylinder Plenum cover assembly Plenum cylinder Reinforcing plates Top flange-to-cover bolts Bottom flange-to-upper grid screws	Stainless steel	Reactor coolant	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
R-54	Reactor vessel internals components	Stainless steel, cast austenitic stainless steel, nickel alloy	Reactor coolant	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
B4.2-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
	Upper grid assembly Upper grid rib section Upper grid ring forging Fuel assembly support pads Plenum rib pads	Stainless steel	Reactor coolant	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

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
	Upper grid assembly Rib- to-ring screws	Stainless steel	Reactor coolant	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
	Upper grid assembly Upper grid rib section Upper grid ring forging Fuel assembly support pads Plenum rib pads Rib-to-ring screws	Stainless steel	Reactor coolant	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
R-54	Reactor vessel internals components	Stainless steel, cast austenitic stainless steel, nickel alloy	Reactor coolant	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
B4.2-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
	Upper grid assembly Upper grid rib section Upper grid ring forging Fuel assembly support pads Plenum rib pads Rib-to-ring screws	Stainless steel	Reactor coolant	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
	Upper grid assembly Fuel assembly support pads Plenum rib pads	Stainless steel	Reactor coolant	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
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-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
	Control rod guide tube (CRGT) assembly CRGT pipe and flange CRGT spacer casting CRGT rod guide tubes CRGT rod guide sectors	Stainless steel, cast austenitic stainless steel	Reactor coolant	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
	Control rod guide tube (CRGT) assembly CRGT spacer screws Flange-to-upper grid screws	Stainless steel	Reactor coolant	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
	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	Stainless steel, cast austenitic stainless steel	Reactor coolant	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

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
	Control rod guide tube (CRGT) assembly CRGT spacer casting	Cast austenitic stainless steel	Reactor coolant and neutron flux	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
	Control rod guide tube (CRGT) assembly Flange-to-upper grid screws	Stainless steel	Reactor coolant	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
B4.3-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

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
R-54	Reactor vessel internals components	Stainless steel, cast austenitic stainless steel, nickel alloy	Reactor coolant	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
	Core support shield assembly Core support shield cylinder (top and bottom flange) Outlet and vent valve (VV) nozzles VV body and retaining ring	Stainless steel, type 15-5PH forging	Reactor coolant	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
	Core support shield assembly Core support shield-to-core barrel bolts VV assembly locking device	Stainless steel, nickel alloy	Reactor coolant	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
	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	Stainless steel, nickel alloy, type 15-5PH forging	Reactor coolant	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
B4.4-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
	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	Stainless steel, nickel alloy, type 15-5PH forging	Reactor coolant and neutron flux	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
	Reactor vessel internals components	Stainless steel, cast austenitic stainless steel, nickel alloy, type 15-5PH forging	Reactor coolant	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
B4.4-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
	Core support shield assembly Core support shield cylinder (top flange) VV assembly locking device	Stainless steel	Reactor coolant	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
	Core support shield assembly Outlet and vent valve nozzles VV body and retaining ring	Cast austenitic stainless steel	Reactor coolant and neutron flux	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
B4.4-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

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
	Core support shield assembly Core support shield-to-core barrel bolts	Stainless steel, nickel alloy	Reactor coolant	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
	Core barrel assembly Core barrel cylinder (top and bottom flange) Baffle plates and formers	Stainless steel	Reactor coolant	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

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
	Core barrel assembly Lower internals assembly-to-core barrel bolts Core barrel-to-thermal shield bolts	Stainless steel, nickel alloy	Reactor coolant	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
	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	Stainless steel, nickel alloy	Reactor coolant	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
	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	Stainless steel, nickel alloy	Reactor coolant and neutron flux	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
	Core barrel assembly Lower internals assembly-to-core barrel bolts Core barrel-to-thermal shield bolts	Stainless steel, nickel alloy	Reactor coolant	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
B4.5-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
R-54	Reactor vessel internals components	Stainless steel, cast austenitic stainless steel, nickel alloy	Reactor coolant	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
B4.5-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
	Core barrel assembly Baffle/former bolts and screws	Stainless steel	Reactor coolant	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

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-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
	Core barrel assembly Baffle/former bolts and screws	Stainless steel	Reactor coolant	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
	Core barrel assembly Baffle/former bolts and screws	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/ Neutron irradiation embrittlement, void swelling	A plant-specific aging management program is to be evaluated.	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-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
	Core barrel assembly Baffle/former bolts and screws	Stainless steel	Reactor coolant	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

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
	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	Stainless steel; cast austenitic stainless steel	Reactor coolant	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
	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	Stainless steel, nickel alloy	Reactor coolant	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
B4.6-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
	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	Stainless steel; cast austenitic stainless steel, nickel alloy	Reactor coolant	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
B4.6-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
	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	Stainless steel, nickel alloy	Reactor coolant and neutron flux	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
B4.6-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
	Lower grid assembly Incore guide tube spider castings	Cast austenitic stainless steel	Reactor coolant	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
B4.6-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

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
R-54	Reactor vessel internals components	Stainless steel, cast austenitic stainless steel, nickel alloy	Reactor coolant	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
	Lower grid assembly Lower grid rib-to-shell forging screws Lower internals assembly-to-thermal shield bolts	Stainless steel, nickel alloy	Reactor coolant	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

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
	Lower grid assembly Fuel assembly support pads Guide blocks	Stainless steel	Reactor coolant	Loss of material/ Wear	Chapter XI.M1, “ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD,” for Class 1 components	No
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
	Flow distributor assembly Flow distributor head and flange Incore guide support plate Clamping ring	Stainless steel	Reactor coolant	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
	Flow distributor assembly Shell forging-to-flow distributor bolts	Stainless steel, nickel alloy	Reactor coolant	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
B4.7-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
	Flow distributor assembly Flow distributor head and flange Shell forging-to-flow distributor bolts Incore guide support plate Clamping ring	Stainless steel, nickel alloy	Reactor coolant	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
	Flow distributor assembly Flow distributor head and flange Shell forging-to-flow distributor bolts Incore guide support plate Clamping ring	Stainless steel, nickel alloy	Reactor coolant and neutron flux	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
B4.7-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
	Flow distributor assembly Shell forging to flow distributor bolts	Stainless steel, nickel alloy	Reactor coolant	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
	Thermal shield	Stainless steel	Reactor coolant	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

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
	Thermal shield	Stainless steel	Reactor coolant	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
	Thermal shield	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/ Neutron irradiation embrittlement, void swelling	Chapter XI.M16, “PWR Vessel Internals”	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-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
R-23	Piping, fittings and components susceptible to flow-accelerated corrosion	Carbon steel	Reactor coolant	Loss of material/ 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

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
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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
R-23	Piping, fittings and components susceptible to flow-accelerated corrosion	Carbon steel	Reactor coolant	Loss of material/ 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
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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-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
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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
R-22	Piping, fittings and components greater than or equal to 4 inch nominal diameter	Stainless steel	Reactor coolant	Cracking	Chapter XI.M7, "BWR Stress Corrosion Cracking" and Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515)	No
R-20	Piping, fittings and components greater than or equal to 4 inch nominal diameter	Cast austenitic stainless steel	Reactor coolant	Cracking	Chapter XI.M7, "BWR Stress Corrosion Cracking" and Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515)	No
R-21	Piping, fittings and components greater than or equal to 4 inch nominal diameter	Nickel-alloy	Reactor coolant	Cracking	Chapter XI.M7, "BWR Stress Corrosion Cracking" and Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515)	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-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
R-52	Class 1 piping, fittings and components	Cast austenitic stainless steel	Reactor coolant > 482°F	Loss of fracture toughness/ Thermal aging embrittlement	Chapter XI.M12, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)"	No
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
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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
R-03	Class 1 piping, fittings and branch connections less than NPS 4	Stainless steel	Reactor coolant	Crack initiation and growth/ Stress corrosion cracking, inter-granular 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) 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.	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
R-55	Class 1 piping, fittings and branch connections less than NPS 4	Stainless steel, carbon steel	Reactor coolant	Crack initiation and growth/ Thermal and mechanical loading	Chapter XI.M1, “ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD,” for Class 1 components 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
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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

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
R-20	Piping, fittings and components greater than or equal to 4 inch nominal diameter	Cast austenitic stainless steel	Reactor coolant	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
R-08	Class 1 pump casings, and valve bodies and bonnets	Cast austenitic stainless steel	Reactor coolant > 482°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 and 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 pump casings and 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
R-29	Pump and valve seal flanges	Stainless steel, carbon steel	System temperature up to 288°C (550°F)	Loss of material/ Wear	Chapter XI.M18, “Bolting Integrity”	No
R-26	Pump and valve closure bolting	Carbon steel	System 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
R-27	Pump and valve closure bolting	Carbon steel	System temperature up to 288°C (550°F)	Loss of preload/ Stress relaxation	Chapter XI.M18, “Bolting Integrity”	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-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
R-28	Pump and valve closure bolting	Carbon steel	System temperature up to 288°C (550°F)	Cumulative fatigue damage	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
R-23	Piping, fittings and components susceptible to flow-accelerated corrosion	Carbon steel	Reactor coolant	Loss of material/ 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
R-08	Class 1 pump casings, and valve bodies and bonnets	Cast austenitic stainless steel	Reactor coolant > 482°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 and 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 pump casings and 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

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
R-22	Piping, fittings and components greater than or equal to 4 inch nominal diameter	Stainless steel	Reactor coolant	Cracking	Chapter XI.M7, "BWR Stress Corrosion Cracking" and Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515)	No
R-20	Piping, fittings and components greater than or equal to 4 inch nominal diameter	Cast austenitic stainless steel	Reactor coolant	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
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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
C1.3-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
R-29	Pump and valve seal flanges	Stainless steel, carbon steel	System temperature up to 288°C (550°F)	Loss of material/ Wear	Chapter XI.M18, “Bolting Integrity”	No
R-26	Pump and valve closure bolting	Carbon steel	System temperature up to 288°C (550°F)	Loss of material/ Wear	Chapter XI.M18, “Bolting Integrity”	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-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
R-27	Pump and valve closure bolting	Carbon steel	System 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
R-28	Pump and valve closure bolting	Carbon steel	System temperature up to 288°C (550°F)	Cumulative fatigue damage	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
R-15	Isolation condenser tube side components	Stainless steel, carbon steel	Reactor coolant	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) 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

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
R-16	Isolation condenser tube side components	Stainless steel, carbon steel	Reactor coolant	Loss of material	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
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
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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-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
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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-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

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
R-30	Reactor coolant system piping and fittings Cold leg Hot leg Surge line Spray line	Stainless steel, carbon steel with stainless steel cladding	Reactor coolant	Cracking/ 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
R-56	Reactor coolant system piping and fittings Cold leg Hot leg Surge line Spray line	Stainless steel, carbon steel with stainless steel cladding	Reactor coolant	Cracking/ Cyclic loading	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components	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
R-17	Piping and components external surfaces and bolting	Carbon steel	Air with boric acid leakage	Loss of material/ Boric acid corrosion	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
C2.1-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

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
R-05	Class 1 piping, fittings and components	Cast austenitic stainless steel	Reactor coolant	Cracking/ 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
R-52	Class 1 piping, fittings and components	Cast austenitic stainless steel	Reactor coolant > 482°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
R-02	Class 1 piping, fittings and branch connections less than NPS 4	Stainless steel	Reactor coolant	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 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.	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
R-57	Class 1 piping, fittings and branch connections less than NPS 4	Stainless steel, carbon steel with stainless steel cladding	Reactor coolant	Crack initiation and growth/ Thermal and mechanical loading	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components 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
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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-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
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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

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
R-17	Piping and components external surfaces and bolting	Carbon steel	Air with boric acid leakage	Loss of material/ Boric acid corrosion	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
R-52	Class 1 piping, fittings and components	Cast austenitic stainless steel	Reactor coolant > 482°F	Loss of fracture toughness/ Thermal aging embrittlement	Chapter XI.M12, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)"	No
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
R-07	Class 1 piping, fittings and components	Stainless steel, carbon steel with stainless steel or nickel-alloy cladding, nickel-alloy	Reactor coolant	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
C2.2-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
R-05	Class 1 piping, fittings and components	Cast austenitic stainless steel	Reactor coolant	Cracking/ 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
R-02	Class 1 piping, fittings and branch connections less than NPS 4	Stainless steel, carbon steel with stainless steel cladding	Reactor coolant	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 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.	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
R-57	Class 1 piping, fittings and branch connections less than NPS 4	Stainless steel, carbon steel with stainless steel cladding	Reactor coolant	Crack initiation and growth/ Thermal and mechanical loading	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components 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
C2.3-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
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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
C2.3-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
R-09	Class 1 pump casings and valve bodies	Cast austenitic stainless steel, carbon steel with stainless steel cladding	Reactor coolant	Cracking/ 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
R-08	Class 1 pump casings and valve bodies	Cast austenitic stainless steel	Reactor coolant > 482°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 and 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 pump casings and valve bodies.	No
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
R-18	Piping and components external surfaces and bolting	Stainless steel, carbon steel	System temperature up to 340°C (644°F)	Cumulative fatigue damage	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
C2.3-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
R-11	Closure bolting	High-strength low-alloy steel, stainless steel	Air with reactor coolant leakage	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
R-17	Piping and components external surfaces and bolting	Carbon steel	Air with boric acid leakage	Loss of material/ Boric acid corrosion	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
R-12	Closure bolting	High-strength low-alloy steel, stainless steel	Air with reactor coolant leakage	Loss of preload	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
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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
C2.4-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
R-09	Class 1 pump casings and valve bodies	Cast austenitic stainless steel, carbon steel with stainless steel cladding	Reactor coolant	Cracking/ 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
R-08	Class 1 pump casings and valve bodies	Cast austenitic stainless steel	Reactor coolant > 482°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 and 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 pump casings and valve bodies.	No
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
R-18	Piping and components external surfaces and bolting	Stainless steel, carbon steel	System temperature up to 340°C (644°F)	Cumulative fatigue damage	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
C2.4-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
R-11	Closure bolting	High-strength low-alloy steel, stainless steel	Air with reactor coolant leakage	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
R-17	Piping and components external surfaces and bolting	Carbon steel	Air with boric acid leakage	Loss of material/ Boric acid corrosion	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
R-12	Closure bolting	High-strength low-alloy steel, stainless steel	Air with reactor coolant leakage	Loss of preload	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
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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
R-17	Piping and components external surfaces and bolting	Carbon steel	Air with boric acid leakage	Loss of material/ Boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No
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
R-25	Pressurizer components	Carbon steel with stainless steel or nickel-alloy cladding; or stainless steel	Reactor coolant	Cracking/ 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
R-58	Pressurizer components	Carbon steel with stainless steel or nickel-alloy cladding; or stainless steel	Reactor coolant	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
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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-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

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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
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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-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

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
R-25	Pressurizer components	Carbon steel with stainless steel or nickel-alloy cladding; or stainless steel	Reactor coolant	Cracking/ 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
R-58	Pressurizer components	Carbon steel with stainless steel or nickel-alloy cladding; or stainless steel	Reactor coolant	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-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
R-07	Class 1 piping, fittings and components	Stainless steel, carbon steel with stainless steel or nickel-alloy cladding, nickel-alloy	Reactor coolant	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-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

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
R-05	Class 1 piping, fittings and components	Cast austenitic stainless steel	Reactor coolant	Cracking/ 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
R-24	Pressurizer Spray head	Nickel-alloy, stainless steel, cast austenitic stainless steel	Reactor coolant	Cracking/ Primary water stress corrosion cracking, stress corrosion cracking	A plant-specific aging management program is to be evaluated.	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
C2.5-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
R-06	Class 1 piping, fittings and components	Nickel-alloy	Reactor coolant	Cracking/ Primary water stress corrosion cracking	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
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
R-52	Class 1 piping, fittings and components	Cast austenitic stainless steel	Reactor coolant > 482°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
C2.5-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
R-07	Class 1 piping, fittings and components	Stainless steel, carbon steel with stainless steel or nickel-alloy cladding, nickel-alloy	Reactor coolant	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
R-11	Closure bolting	High-strength low-alloy steel, stainless steel	Air with reactor coolant leakage	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
R-17	Piping and components external surfaces and bolting	Carbon steel	Air with boric acid leakage	Loss of material/ Boric acid corrosion	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-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
R-12	Closure bolting	High-strength low-alloy steel, stainless steel	Air with reactor coolant leakage	Loss of preload	Chapter XI.M18, “Bolting Integrity”	No
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

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
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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
R-07	Class 1 piping, fittings and components	Stainless steel, carbon steel with stainless steel or nickel-alloy cladding, nickel-alloy	Reactor coolant	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-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
R-06	Class 1 piping, fittings and components	Nickel-alloy	Reactor coolant	Cracking/ Primary water stress corrosion cracking	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
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

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
R-18	Piping and components external surfaces and bolting	Stainless steel, carbon steel	System temperature up to 340°C (644°F)	Cumulative fatigue damage	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
C2.5-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
R-17	Piping and components external surfaces and bolting	Carbon steel	Air with boric acid leakage	Loss of material/ Boric acid corrosion	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
R-19	Pressurizer Integral support	Stainless steel, carbon steel	System temperature up to 340°C (644°F)	Cracking/ 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

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
R-18	Piping and components external surfaces and bolting	Stainless steel, carbon steel	System temperature up to 340°C (644°F)	Cumulative fatigue damage	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
C2.6-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
R-13	General piping and components	Carbon steel with stainless steel cladding	Treated borated water	Cumulative fatigue damage	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
R-17	Piping and components external surfaces and bolting	Carbon steel	Air with boric acid leakage	Loss of material/ Boric acid corrosion	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
R-14	General piping, fittings and components	Stainless steel, carbon steel with stainless steel cladding	Treated borated water >140°F	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

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
R-33	Steam generator components	Carbon steel	Secondary feedwater/ steam	Cumulative fatigue damage	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-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
R-33	Steam generator components	Carbon steel	Secondary feedwater/ steam	Cumulative fatigue damage	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

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-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 recommenda-tions 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
R-34	Steam generator shell assembly	Carbon steel	Secondary feedwater/ steam	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 AMP guidelines in Chapter XI.M1 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
R-37	Pressure boundary and structural Steam nozzle and safe end FW nozzle and safe end	Carbon steel	Secondary feedwater/ steam	Loss of material/ Flow-accelerated corrosion	Chapter XI.M17, "Flow-Accelerated Corrosion"	No
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
R-39	Steam generator feedwater impingement plate and support	Carbon steel	Secondary feedwater	Loss of material/ 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
R-32	Steam generator closure bolting	Carbon steel	System 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
R-17	Piping and components external surfaces and bolting	Carbon steel	Air with boric acid leakage	Loss of material/ Boric acid corrosion	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-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
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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
R-07	Class 1 piping, fittings and components	Stainless steel, carbon steel with stainless steel or nickel-alloy cladding, nickel-alloy	Reactor coolant	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
R-01	Class 1 fittings and components	Nickel-alloy	Reactor coolant	Cracking/ Primary water stress corrosion cracking	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
D1.1-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
R-17	Piping and components external surfaces and bolting	Carbon steel	Air with boric acid leakage	Loss of material/ Boric acid corrosion	Chapter XI.M10**, “**Boric Acid Corrosion**”**	No
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
R-10	Closure bolting	Carbon steel	Air with reactor coolant leakage	Cracking	Chapter XI.M18, “Bolting Integrity”	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
D1.2-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
R-44	Tubes and sleeves	Nickel-alloy	Reactor coolant	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 for alloy 600 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
R-47	Tubes and sleeves	Nickel-alloy	Secondary feedwater/ steam	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 for alloy 600 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-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
R-48	Tubes and sleeves	Nickel-alloy	Secondary feedwater/ steam	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 for alloy 600 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
R-45	Tubes and sleeves	Nickel-alloy	Reactor coolant and Secondary feedwater/ 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
D1. Steam Generator (Recirculating)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
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, effective-ness of the AMP is to be evaluated
R-49	Tubes and sleeves	Nickel-alloy	Secondary feedwater/ steam	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, effective-ness of the AMP for alloy 600 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
R-50	Tubes and sleeves (exposed to phosphate chemistry)	Nickel-alloy	Secondary feedwater/ steam	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 for alloy 600 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

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
R-43	Tubes	Nickel-alloy	Secondary feedwater/ steam	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 for alloy 600 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
R-41	Tube support lattice bars	Carbon steel	Secondary feedwater/ steam	Loss of material/ 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
R-40	Tube plugs	Nickel-alloy	Reactor coolant	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 for alloy 600 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-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
R-40	Tube plugs	Nickel-alloy	Reactor coolant	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 for alloy 600 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
R-42	Tube support plates	Carbon steel	Secondary feedwater/ steam	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

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
D1.3-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
R-51	Upper assembly and separators Feedwater inlet ring and support	Carbon steel	Secondary feedwater/ steam	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

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
R-35	Steam generator components	Carbon steel with stainless steel or nickel-alloy cladding	Reactor coolant	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
R-17	Piping and components external surfaces and bolting	Carbon steel	Air with boric acid leakage	Loss of material/ Boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	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-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
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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
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

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
R-33	Steam generator components	Carbon steel	Secondary feedwater/ steam	Cumulative fatigue damage	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
R-34	Steam generator shell assembly	Carbon steel	Secondary feedwater/ steam	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 AMP guidelines in Chapter XI.M1 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

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-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
R-38	Pressure boundary and structural FW and AFW nozzles and safe ends Steam nozzles and safe ends	Carbon steel	Secondary feedwater/ steam	Loss of material/ 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
R-33	Steam generator components	Carbon steel	Secondary feedwater/ steam	Cumulative fatigue damage	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-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
R-01	Class 1 fittings and components	Nickel-alloy	Reactor coolant	Cracking/ Primary water stress corrosion cracking	A plant-specific aging management program is to be evaluated.	Yes, plant specific

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-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
R-36	Steam generator components	Nickel-alloy	Secondary feedwater/ steam	Cracking/ 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
R-17	Piping and components external surfaces and bolting	Carbon steel	Air with boric acid leakage	Loss of material/ Boric acid corrosion	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
R-32	Steam generator closure bolting	Carbon steel	System 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
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
R-31	Secondary manways and handholes (cover only)	Carbon steel	Air, with leaking secondary-side water and/or steam	Loss of material/ 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, 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
R-44	Tubes and sleeves	Nickel-alloy	Reactor coolant	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 for alloy 600 is to be evaluated
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

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
R-47	Tubes and sleeves	Nickel-alloy	Secondary feedwater/ steam	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 for alloy 600 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
R-48	Tubes and sleeves	Nickel-alloy	Secondary feedwater/ steam	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 for alloy 600 is to be evaluated
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, 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
R-49	Tubes and sleeves	Nickel-alloy	Secondary feedwater/ steam	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, effective-ness of the AMP for alloy 600 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
R-46	Tubes and sleeves	Nickel-alloy	Reactor coolant and Secondary feedwater/ steam	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, effective-ness of the AMP is to be evaluated
R-40	Tube plugs	Nickel-alloy	Reactor coolant	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 for alloy 600 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-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, effective-ness of the AMP is to be evaluated
R-40	Tube plugs	Nickel-alloy	Reactor coolant	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 for alloy 600 is to be evaluated

General Material Types

Material	Description
Aluminum	Pure aluminum
Aluminum alloys	Alloys of aluminum
Carbon steel	For a given environment, carbon steel, alloy steel, and cast iron exhibit the same aging effects, even though the rates of aging may vary. Consequently, these metal types may be considered the same for aging management reviews. Gray cast iron is also susceptible to selective leaching and high strength low alloy steel is also susceptible to stress corrosion cracking. Therefore, when these aging effects are being considered, these materials are specifically mentioned; otherwise they are considered part of the general category of carbon steel. (References 5, 6)
Cast austenitic stainless steel	Cast stainless steels containing ferrite in an austenitic matrix
Copper alloy < 15 % Zn	Copper, copper nickel, brass, bronze <15% Zn, Aluminum bronze < 8% Al – These materials are resistant to stress corrosion cracking, selective leaching and pitting and crevice corrosion. (References 5, 6) May be identified simply as copper alloy when these aging mechanisms are not at issue.
Copper alloy >15% Zn	Copper, brass and other alloys >15% Zn, Aluminum bronze > 8% Al – These materials are susceptible to stress corrosion cracking, selective leaching (except for inhibited brass) and pitting and crevice corrosion. (References 5, 6) May be identified simply as copper alloy when these aging mechanisms are not at issue.
Elastomers	Elastomers include rubber, EPT, EPDM, PTFE, ETFE, viton, vitril, neoprene, silicone elastomer, etc.
Galvanized steel	Zinc coated carbon steel
Glass	All glass materials
Soils	Earthen structures
Nickel-alloy	Nickel based iron alloys such as Alloy 600, Alloy 690, Inconel
Reinforced concrete	Concrete with embedded steel reinforcement

Stainless steel	Wrought or forged austenitic stainless steel

Environment Categories

Environment[1]	Description
Air – indoor controlled (Int/Ext)	Indoor air in a humidity controlled (e.g., air conditioned) environment.
Air – indoor uncontrolled (Int/Ext)	Indoor air on systems with temperatures higher than the dew point – Condensation can occur but only rarely – equipment surfaces are normally dry.
Air – indoor uncontrolled > 95°F (Int/Ext)	Indoor air above thermal stress threshold for elastomers
Air with boric acid leakage	Air and untreated borated water leakage on indoor or outdoor systems with temperatures above or below the dew point
Air with reactor coolant leakage	Air and reactor coolant or steam leakage on high temperature systems
Air with steam or water leakage	Air and untreated steam or water leakage on indoor or outdoor systems with temperatures above or below the dew point
Air – outdoor (Int/Ext)	Exposed to air and local weather conditions including salt spray where applicable
Air and steam	Exposed normally to air and periodically to steam
Condensation (Int/Ext)	Air and condensation on surfaces of indoor systems with temperatures below the dew point – for exterior surfaces and interior surfaces in communication ambient indoor air, condensation is considered untreated water due to potential for surface contamination.
Condensation with boric acid leakage	Air and condensation with the potential for boric acid leakage on surfaces of indoor systems with temperatures below the dew point – condensation is considered untreated water due to potential for surface contamination

[1] For environments listed with (Int/Ext), the component information description should identify whether the surface is internal or external. This information is important because it indicates the applicability of direct visual observation of the surface for aging management. For the remaining environments, this distinction need not be made since the environment must be internal to some barrier that precludes direct observation of the surface.

Closed cycle cooling water	Treated water subject to the closed cycle cooling water chemistry program
Concrete	Components embedded in concrete
Dried Air	Air that has been treated to reduce the dew point well below the system operating temperature
Exhaust gases	Gas present in a diesel engine exhaust
Gas	Inert gases such as carbon dioxide, freon, halon, nitrogen
Fuel oil	Fuel oil used for combustion engines
Lubricating oil	Lubricating oil for plant equipment with possible water contamination
Neutron flux	Reactor core environment for ferritic materials that will result in a neutron fluence exceeding 10^{17} n/cm^2 (E >1 MeV) at the end of the license renewal term.
Raw water	Raw untreated fresh or salt water
Reactor coolant	Water in the reactor coolant system and connected systems at or near full operating temperature – includes steam for BWRs
Reactor coolant > 482°F	Water in the reactor coolant system and connected systems above thermal embrittlement threshold for CASS
Sand and concrete	Sand/concrete base for tanks
Soil	External environment for components buried in the soil, including groundwater in the soil
Secondary feedwater/steam	PWR feedwater or steam at or near full operating temperature subject to the secondary water chemistry program
Steam	Steam, subject to BWR water chemistry program or PWR secondary plant water chemistry program
Treated borated water	Treated water with boric acid

Treated borated water >140°F	Treated water with boric acid above SCC threshold for stainless steel
Treated borated water >482°F	Treated water with boric acid above thermal embrittlement threshold for CASS
Treated water	Treated or demineralized water – This environment is used where the context of the MEAP combination makes the type of treated water apparent; e.g., if the program is for PWR secondary water chemistry, the treated water is from the PWR secondary system.
Treated water >140°F	Treated water above SCC threshold for stainless steel
Treated water >482°F	Treated water above thermal embrittlement threshold for CASS
Untreated water	Water that may contain contaminants including oil and boric acid depending on the location – includes originally treated water that is not monitored by a chemistry program

Temperature Thresholds

Temperature	Threshold	Basis
95°F	Thermal stresses for elastomers	In general, if the ambient temperature is less than about 95°F, then thermal aging may be considered not significant for rubber, butyl rubber, neoprene, nitrile rubber, silicone elastomer, fluoroelastomer, EPR, and EPDM (Reference 8).
140°F	SCC for stainless steel	In general, SCC very rarely occurs in austenitic stainless steels below 140°F (Reference 1, 2). Although SCC has been observed in stagnant, oxygenated borated water systems at lower temperatures than this 140°F threshold, all of these instances have identified a significant presence of contaminants (halogens, specifically chlorides) in the failed components. With a harsh enough environment (significant contamination), SCC can occur in austenitic stainless steel at ambient temperature. However, these conditions are considered event driven, resulting from a breakdown of chemistry controls. Further discussion of this threshold is provided in Reference 7.
482°F	Thermal embrittlement for CASS	CASS materials subjected to sustained temperatures below 250°C (482°F) will not result in a reduction of room temperature Charpy impact energy below 50 ft-lb for exposure times of approximately 300,000 hours (for CASS with ferrite content of 40%) and approximately 2,500,000 hours for CASS with ferrite content of 14%) [Figure 1; Reference 4]. For a maximum exposure time of approximately 420,000 hours (48 EFPY), a screening temperature of 482°F is conservatively chosen because (1) the majority of nuclear grade materials are expected to contain a ferrite content well below 40%, and (2) the 50 ft-lb limit is very conservative when applied to cast austenitic materials. It is typically applied to ferritic materials (e.g., 10 CFR 50 Appendix G). For CASS components in the reactor coolant pressure boundary, this threshold is supported by NUREG-1801 XI.M12, with the exception of niobium-containing steels which require evaluation on a case-by-case basis.

New Aging Effect Terms

Change in material properties	This effect covers all degradation of a material's properties considered important for its intended function
Reduction of heat transfer	Reduction of heat transfer from fouling by the buildup (from whatever source) on the heat transfer surface.
Macrofouling	Biofouling listed in NUREG-1801 as aging mechanism is assumed to be the plugging of components due to biological growth or material. Although plugging of a component affects only flow, an active intended function outside the purview of license renewal, the term macrofouling is used to address fouling that causes plugging as opposed to fouling that causes loss of heat transfer, and includes plugging from any source, including biological.

References

1. D. Peckner and I. M. Bernstein, Eds., Handbook of Stainless Steels, McGraw-Hill, New York, 1977.
2. Metals Handbook, Ninth Edition, Volume 13, Corrosion, American Society of Metals, Copyright 1987.
3. Not Used
4. R. Nickell, M. A. Rinckel, “Evaluation of Thermal Aging Embrittlement for Cast Austenitic Stainless Steel Components,” TR-106092, Research Project 2643-33, Final Report, March 1996.
5. Metals Handbook, Desk Edition, American Society for Metals, Materials Park, OH, 1985.
6. M. G. Fontana, Corrosion Engineering, Third Edition, Copyright 1986, McGraw Hill.
7. License Renewal Application for St. Lucie Units 1 and 2, November 30, 2001, Appendix C.
8. Aging Management Guideline for Commercial Nuclear Power Plants – Electrical and Mechanical Penetrations, EPRI, Palo Alto, CA: 2002. 1003456

IV Reactor Vessel, Internals, and Reactor Coolant System
A1. Reactor Vessel (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
	Top head enclosure (without cladding) Top head Nozzles (vent, top head spray or RCIC, and spare)	Carbon steel	Reactor coolant	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)	No
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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
	Top head enclosure Closure studs and nuts	High strength low alloy steel	Air with reactor coolant leakage	Crack initiation and growth/ Stress corrosion cracking, intergranular stress corrosion cracking	Chapter XI.M3, “Reactor Head Closure Studs”	No
	Top head enclosure Vessel flange leak detection line	Stainless steel, nickel alloy	Air with reactor coolant leakage	Crack initiation and growth/ Stress corrosion cracking, intergranular stress corrosion cracking	A plant-specific aging management program is to be evaluated because existing programs may not be able to mitigate or detect crack initiation and growth due to SCC of vessel flange leak detection line.	Yes, plant specific

IV Reactor Vessel, Internals, and Reactor Coolant System
A1. Reactor Vessel (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
	Vessel shell Intermediate beltline shell Beltline welds	Carbon steel with or without stainless steel cladding	Neutron flux	Loss of fracture toughness/ Neutron irradiation embrittlement	Neutron irradiation embrittlement is a time dependent aging mechanism to be evaluated for the period of extended operation for all ferritic materials that have a neutron fluence exceeding 10^{17} n/cm^2 (E >1 MeV) at the end of the license renewal term. Aspects of this evaluation may involve a TLAA. In accordance with approved BWRVIP-74, the TLAA is to evaluate the impact of neutron embrittlement on: (a) the adjusted reference temperature, the plant's pressure-temperature limits, (b) the need for inservice inspection of circumferential welds, and (c) the Charpy upper shelf energy or the equivalent margins analyses performed in accordance with 10 CFR 50, Appendix G. Additionally, the applicant is to monitor axial beltline weld embrittlement. One acceptable method is to determine that the mean RT_{NDT} of the axial beltline welds at the end of the extended period of operation is less than the value specified by the staff in its May 7, 2000 letter. 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
	Vessel shell Intermediate beltline shell Beltline welds	Carbon steel with or without stainless steel cladding	Neutron flux	Loss of fracture toughness/ Neutron irradiation embrittlement	Chapter XI.M31, "Reactor Vessel Surveillance"	Yes, plant specific

IV Reactor Vessel, Internals, and Reactor Coolant System
A1. Reactor Vessel (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
	Vessel shell Attachment welds	Stainless steel, nickel alloy	Reactor coolant	Crack initiation and growth/ Stress corrosion cracking, intergranular stress corrosion cracking	Chapter XI.M4, “BWR Vessel ID Attachment Welds,” and Chapter XI.M2, “Water Chemistry,” for BWR water in BWRVIP-29 (EPRI TR-103515)	No
	Nozzles Feedwater	Carbon steel with or without stainless steel cladding	Reactor coolant	Crack initiation and growth/ Cyclic loading	Chapter XI.M5, “BWR Feedwater Nozzle”	No
	Nozzles Control rod drive return line	Carbon steel with or without stainless steel cladding	Reactor coolant	Crack initiation and growth/ Cyclic loading	Chapter XI.M6, “BWR Control Rod Drive Return Line Nozzle”	No

IV Reactor Vessel, Internals, and Reactor Coolant System
A1. Reactor Vessel (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
	Nozzles Low pressure coolant injection or RHR injection mode	Carbon steel	Neutron flux	Loss of fracture toughness/ Neutron irradiation embrittlement	Neutron irradiation embrittlement is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation 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. In accordance with approved BWRVIP-74, the TLAA is to evaluate the impact of neutron embrittlement on: (a) the adjusted reference temperature, the plant's pressure-temperature limits, (b) the Charpy upper shelf energy, and (c) the equivalent margins analyses performed in accordance with 10 CFR 50, Appendix G. The applicant may choose to demonstrate that the materials of the nozzles are not controlling for the TLAA evaluations. 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
	Nozzle safe ends High pressure core spray Low pressure core spray Control rod drive return line Recirculating water Low pressure coolant injection or RHR injection mode	Stainless steel, nickel alloy	Reactor coolant	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

IV Reactor Vessel, Internals, and Reactor Coolant System
A1. Reactor Vessel (Boiling Water Reactor)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
	Penetrations Control rod drive stub tubes Instrumentation Jet pump instrument Standby liquid control Flux monitor Drain line	Stainless steel, nickel alloy	Reactor coolant	Crack initiation and growth/ Stress corrosion cracking, intergranular stress corrosion cracking, cyclic loading	Chapter XI.M8, "BWR Penetrations," and Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515)	No
	Support skirt and attachment welds	Carbon steel	Air – indoor uncontrolled	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

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
R-17	Piping and components external surfaces and bolting	Carbon steel	Air with boric acid leakage	Loss of material/ Boric acid corrosion	Chapter XI.M10**, "**Boric Acid Corrosion**"**	No
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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
	Closure head Stud assembly	High strength low alloy steel	Air with reactor coolant leakage	Crack initiation and growth/ Stress corrosion cracking	Chapter XI.M3, "Reactor Head Closure Studs"	No
	Closure head Stud assembly	High strength low alloy steel	Air with reactor coolant leakage	Loss of material/ Wear	Chapter XI.M3, "Reactor Head Closure Studs"	No
	Closure head Stud assembly	High strength low alloy steel	Air with reactor coolant leakage	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

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
	Closure head Vessel flange leak detection line	Stainless steel	Air with reactor coolant leakage	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
	Control rod drive head penetration Nozzle	Nickel alloy	Reactor coolant	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
	Control rod drive head penetration Pressure housing	Stainless steel; cast austenitic stainless steel, nickel alloy	Reactor coolant	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
	Control rod drive head penetration Pressure housing	Cast austenitic stainless steel	Reactor coolant	Loss of fracture toughness/ Thermal aging embrittlement	Chapter XI.M12 “Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)”	No
	Control rod drive head penetration Flange bolting	Stainless steel	Air with reactor coolant leakage	Crack initiation and growth/ Stress corrosion cracking	Chapter XI.M18, “Bolting Integrity”	No
	Control rod drive head penetration Flange bolting	Stainless steel	Air with reactor coolant leakage	Loss of material/ Wear	Chapter XI.M18, “Bolting Integrity”	No
	Control rod drive head penetration Flange bolting	Stainless steel	Air with reactor coolant leakage	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
	Nozzles Inlet Outlet Safety injection	Carbon steel with stainless steel cladding	Neutron flux	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
	Nozzles Inlet Outlet Safety injection	Carbon steel with stainless steel cladding	Neutron flux	Loss of fracture toughness/ Neutron irradiation embrittlement	Chapter XI.M31, "Reactor Vessel Surveillance"	Yes, plant specific
	Nozzle safe ends Inlet Outlet Safety injection	Stainless steel, cast austenitic stainless steel, nickel alloy	Reactor coolant	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

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
	Vessel shell Upper shell Intermediate and lower shell (including beltline welds)	Carbon steel with stainless steel cladding	Neutron flux	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, 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
	Vessel shell Upper shell Intermediate and lower shell (including beltline welds)	SA508-Cl 2 forgings clad with stainless steel using a high-heat-input welding process	Reactor coolant	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
	Vessel shell Upper shell Intermediate and lower shell (including beltline welds)	Carbon steel with stainless steel cladding	Neutron flux	Loss of fracture toughness/ Neutron irradiation embrittlement	Chapter XI.M31, "Reactor Vessel Surveillance"	Yes, plant specific
	Vessel shell Vessel flange	Carbon steel	Reactor coolant	Loss of material/ Wear	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components	No
	Core support pads/core guide lugs	Nickel alloy	Reactor coolant	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
	Penetrations Instrument tubes (bottom head)	Nickel alloy	Reactor coolant	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
	Penetrations Head vent pipe(top head) Instrument tubes (top head)	Nickel alloy	Reactor coolant	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
	Pressure vessel support Skirt support	Carbon steel	Air – indoor uncontrolled	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

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
	Core shroud and core plate Core shroud (upper, central, lower)	Stainless steel	Reactor coolant	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
	Core shroud and core plate Core plate Core plate bolts (used in early BWRs)	Stainless steel	Reactor coolant	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
R-53	Reactor vessel internals components	Stainless steel, cast austenitic stainless steel, nickel alloy	Reactor coolant	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
	Core shroud and core plate Access hole cover (welded covers)	Nickel alloy	Reactor coolant	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
	Core shroud and core plate Access hole cover (mechanical covers)	Nickel alloy	Reactor coolant	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
	Core shroud and core plate Shroud support structure (shroud support cylinder, shroud support plate, shroud support legs)	Nickel alloy	Reactor coolant	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
	Core shroud and core plate LPCI coupling	Stainless steel	Reactor coolant	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
	Top guide	Stainless steel	Reactor coolant	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
	Core spray lines and spargers Core spray lines (headers) Spray rings Spray nozzles Thermal sleeves	Stainless steel	Reactor coolant	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

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
	Jet pump assemblies Thermal sleeve Inlet header Riser brace arm Holddown beams Inlet elbow Mixing assembly Diffuser Castings	Nickel alloy, cast austenitic stainless steel, stainless steel	Reactor coolant	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
	Jet pump assemblies Castings	Cast austenitic stainless steel	Reactor coolant	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
	Jet pump assemblies Jet pump sensing line	Stainless steel	Reactor coolant	Crack initiation and growth/ cyclic loading	A plant-specific aging management program is to be evaluated.	Yes, plant specific
	Fuel supports and control rod drive assemblies Orificed fuel support	Cast austenitic stainless steel	Reactor coolant	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
	Fuel supports and control rod drive assemblies Control rod drive housing	Stainless steel	Reactor coolant	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

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
	Instrumentation Intermediate range monitor (IRM) dry tubes Source range monitor (SRM) dry tubes Incore neutron flux monitor guide tubes	Stainless steel	Reactor coolant	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

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
	Upper internals assembly Upper support plate Upper core plate Hold-down spring	Stainless steel	Reactor coolant	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
	Upper internals assembly Upper support plate Upper core plate Hold-down spring	Stainless steel	Reactor coolant	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
R-53	Reactor vessel internals components	Stainless steel, cast austenitic stainless steel, nickel alloy	Reactor coolant	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
	Upper internals assembly Hold-down spring	Stainless steel	Reactor coolant	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
	Upper internals assembly Upper support column	Stainless steel, cast austenitic stainless steel	Reactor coolant	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
	Upper internals assembly Upper support column	Stainless steel, cast austenitic stainless steel	Reactor coolant	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
	Upper internals assembly Upper support column (only cast austenitic stainless steel portions)	Cast austenitic stainless steel	Reactor coolant and neutron flux	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
	Upper internals assembly Upper support column bolts Upper core plate alignment pins Fuel alignment pins	Stainless steel, nickel alloy	Reactor coolant	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
B2. Reactor Vessel Internals (PWR) – Westinghouse

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
	Upper internals assembly Upper support column bolts Upper core plate alignment pins Fuel alignment pins	Stainless steel, nickel alloy	Reactor coolant	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
	Upper internals assembly Upper support column bolts	Stainless steel, nickel alloy	Reactor coolant	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
	Upper internals assembly Upper core plate alignment pins	Stainless steel, nickel alloy	Reactor coolant	Loss of material/ Wear	Chapter XI.M1, “ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD,” for Class 1 components	No
	RCCA guide tube assemblies RCCA guide tubes	Stainless steel	Reactor coolant	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
	RCCA guide tube assemblies RCCA guide tubes	Stainless steel	Reactor coolant	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
	RCCA guide tube assemblies RCCA guide tube bolts RCCA guide tube support pins	Stainless steel, nickel alloy	Reactor coolant	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
	RCCA guide tube assemblies RCCA guide tube bolts, RCCA guide tube support pins	Stainless steel, nickel alloy	Reactor coolant	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
	Core barrel Core barrel (CB) CB flange (upper) CB outlet nozzles Thermal shield	Stainless steel	Reactor coolant	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
	Core barrel Core barrel (CB) CB flange (upper) CB outlet nozzles Thermal shield	Stainless steel	Reactor coolant	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
	Core barrel Core barrel (CB) CB flange (upper) CB outlet nozzles Thermal shield	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/ Neutron irradiation embrittlement, void selling	Chapter XI.M16, “PWR Vessel Internals”	No
	Baffle/former assembly Baffle and former plates	Stainless steel	Reactor coolant	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
	Baffle/former assembly Baffle and former plates	Stainless steel	Reactor coolant	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

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
	Baffle/former assembly Baffle/former bolts	Stainless steel	Reactor coolant 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
	Baffle/former assembly Baffle/former bolts	Stainless steel	Reactor coolant	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
	Baffle/former assembly Baffle and former plates	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/ Neutron irradiation embrittlement, void swelling	Chapter XI.M16, “PWR Vessel Internals”	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
	Baffle/former assembly Baffle/former bolts	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/ Neutron irradiation embrittlement	A plant-specific aging management program is to be evaluated.	Yes, plant specific
	Baffle/former assembly Baffle/former bolts	Stainless steel, nickel alloy	Reactor coolant	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
	Lower internal assembly Lower core plate Radial keys and clevis inserts	Stainless steel	Reactor coolant	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
	Lower internal assembly Lower core plate Radial keys and clevis inserts	Stainless steel	Reactor coolant	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
	Lower internal assembly Lower core plate	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/ Neutron irradiation embrittlement, void swelling	Chapter XI.M16, “PWR Vessel Internals”	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
	Lower internal assembly Fuel alignment pins Lower support plate column bolts Clevis insert bolts	Stainless steel, nickel alloy	Reactor coolant	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
	Lower internal assembly Fuel alignment pins Lower support plate column bolts Clevis insert bolts	Stainless steel, nickel alloy	Reactor coolant	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
	Lower internal assembly Fuel alignment pins Lower support plate column bolts Clevis insert bolts	Stainless steel, nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness/ Neutron irradiation embrittlement, void swelling	Chapter XI.M16, “PWR Vessel Internals”	No
	Lower internal assembly Lower support plate column bolts	Stainless steel, nickel alloy	Reactor coolant	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
	Lower internal assembly Clevis insert bolts	Stainless steel, nickel alloy	Reactor coolant	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
	Lower internal assembly Lower support forging or casting Lower support plate columns	Stainless steel, cast austenitic stainless steel	Reactor coolant	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
	Lower internal assembly Lower support forging or casting Lower support plate columns	Stainless steel, cast austenitic stainless steel	Reactor coolant	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
	Lower internal assembly Lower support casting Lower support plate columns	Cast austenitic stainless steel	Reactor coolant and neutron flux	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
	Lower internal assembly Lower support forging Lower support plate columns	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/ Neutron irradiation embrittlement, void swelling	Chapter XI.M16, “PWR Vessel Internals”	No
	Lower internal assembly Radial keys and clevis Inserts	Stainless steel	Reactor coolant	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
	Instrumentation support structures Flux thimble guide tubes	Stainless steel	Reactor coolant	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
	Instrumentation support structures Flux thimble guide tubes	Stainless steel	Reactor coolant	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
	Instrumentation support structures Flux thimble	Stainless steel	Reactor coolant	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

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
	Upper Internals Assembly Upper guide structure support plate Fuel alignment plate Fuel alignment plate guide lugs and guide lug inserts	Stainless steel	Reactor coolant	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
	Upper Internals Assembly Upper guide structure support plate Fuel alignment plate Fuel alignment plate guide lugs and guide lug inserts	Stainless steel	Reactor coolant	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
	Upper Internals Assembly Fuel alignment plate Fuel alignment plate guide lugs and their lugs Hold-down ring	Stainless steel	Reactor coolant	Loss of material/ Wear	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components	No
	CEA Shroud Assemblies CEA shroud	Stainless steel, cast austenitic stainless steel	Reactor coolant	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
	CEA Shroud Assemblies CEA shrouds bolts	Stainless steel, nickel alloy	Reactor coolant	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
	CEA shroud assemblies CEA shroud CEA shrouds bolts	Stainless steel, cast austenitic stainless steel, nickel alloy	Reactor coolant	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
	CEA shroud assemblies CEA shroud extension shaft guides	Stainless steel	Reactor coolant	Loss of material/ Wear	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components	No
	CEA shroud assemblies CEA shroud	Cast austenitic stainless steel	Reactor coolant and neutron flux	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
R-54	Reactor vessel internals components	Stainless steel, cast austenitic stainless steel, nickel alloy	Reactor coolant	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
	CEA shroud assemblies CEA shrouds bolts	Stainless steel, nickel alloy	Reactor coolant	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
	Core support barrel Core support barrel Core support barrel upper flange	Stainless steel	Reactor coolant	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
	Core support barrel Core support barrel Core support barrel upper flange	Stainless steel	Reactor coolant	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
	Core support barrel Core support barrel Core support barrel upper flange	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/ Neutron irradiation embrittlement, void swelling	Chapter XI.M16, “PWR Vessel Internals”	No
	Core support barrel Core support barrel upper flange Core support barrel alignment keys	Stainless steel	Reactor coolant	Loss of material/ Wear	Chapter XI.M1, “ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD,” for Class 1 components	No
	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	Reactor coolant	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
	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, nickel alloy	Reactor coolant	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
	Core shroud assembly Core shroud assembly Core shroud tie rods (core support plate attached by welds in later plants)	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/ Neutron irradiation embrittlement, void swelling	Chapter XI.M16, “PWR Vessel Internals”	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
	Core shroud assembly Core shroud assembly bolts (later plants are welded)	Stainless steel, nickel alloy	Reactor coolant	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
	Core shroud assembly Core shroud assembly bolts (later plants are welded)	Stainless steel, nickel alloy	Reactor coolant	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
	Core shroud assembly Core shroud assembly bolts (later plants are welded)	Stainless steel, nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness/ Neutron irradiation embrittlement, void swelling	Chapter XI.M16, “PWR Vessel Internals”	No
	Core shroud assembly Core shroud assembly bolts Core shroud tie rods	Stainless steel, nickel alloy	Reactor coolant	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
B3. Reactor Vessel Internals (PWR) – Combustion Engineering

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
	Lower internal assembly Core support plate Lower support structure beam assemblies Core support column Core support barrel snubber assemblies	Stainless steel	Reactor coolant	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
	Lower internal Assembly Fuel alignment pins Core support column bolts	Stainless steel, nickel alloy	Reactor coolant	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
	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, cast austenitic stainless steel, nickel alloy	Reactor coolant	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
	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, nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness/ Neutron irradiation embrittlement, void swelling	Chapter XI.M16, “PWR Vessel Internals”	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
	Lower internal assembly Fuel alignment pins Core support barrel snubber assemblies	Stainless steel, nickel alloy	Reactor coolant	Loss of material/ Wear	Chapter XI.M1, “ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD,” for Class 1 components	No
	Lower internal assembly Core support column	Cast austenitic stainless steel	Reactor coolant	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
	Plenum cover and plenum cylinder Plenum cover assembly Plenum cylinder Reinforcing plates	Stainless steel	Reactor coolant	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
	Plenum cover and plenum cylinder Top flange-to-cover bolts Bottom flange-to-upper grid screws	Stainless steel	Reactor coolant	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
	Plenum cover and plenum cylinder Plenum cover assembly Plenum cylinder Reinforcing plates Top flange-to-cover bolts Bottom flange-to-upper grid screws	Stainless steel	Reactor coolant	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
R-54	Reactor vessel internals components	Stainless steel, cast austenitic stainless steel, nickel alloy	Reactor coolant	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
	Upper grid assembly Upper grid rib section Upper grid ring forging Fuel assembly support pads Plenum rib pads	Stainless steel	Reactor coolant	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
	Upper grid assembly Rib- to-ring screws	Stainless steel	Reactor coolant	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
	Upper grid assembly Upper grid rib section Upper grid ring forging Fuel assembly support pads Plenum rib pads Rib-to-ring screws	Stainless steel	Reactor coolant	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
	Upper grid assembly Upper grid rib section Upper grid ring forging Fuel assembly support pads Plenum rib pads Rib-to-ring screws	Stainless steel	Reactor coolant	Loss of fracture toughness/ Neutron irradiation embrittlement, void swelling	Chapter XI.M16, “PWR Vessel Internals”	No
	Upper grid assembly Fuel assembly support pads Plenum rib pads	Stainless steel	Reactor coolant	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
B4. Reactor Vessel Internals (PWR) – Babcock and Wilcox

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
	Control rod guide tube (CRGT) assembly CRGT pipe and flange CRGT spacer casting CRGT rod guide tubes CRGT rod guide sectors	Stainless steel, cast austenitic stainless steel	Reactor coolant	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
	Control rod guide tube (CRGT) assembly CRGT spacer screws Flange-to-upper grid screws	Stainless steel	Reactor coolant	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
	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	Stainless steel, cast austenitic stainless steel	Reactor coolant	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
	Control rod guide tube (CRGT) assembly CRGT spacer casting	Cast austenitic stainless steel	Reactor coolant and neutron flux	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
	Control rod guide tube (CRGT) assembly Flange-to-upper grid screws	Stainless steel	Reactor coolant	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
	Core support shield assembly Core support shield cylinder (top and bottom flange) Outlet and vent valve (VV) nozzles VV body and retaining ring	Stainless steel, type 15-5PH forging	Reactor coolant	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
	Core support shield assembly Core support shield-to-core barrel bolts VV assembly locking device	Stainless steel, nickel alloy	Reactor coolant	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
	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	Stainless steel, nickel alloy, type 15-5PH forging	Reactor coolant	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
	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	Stainless steel, nickel alloy, type 15-5PH forging	Reactor coolant and neutron flux	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
	Reactor vessel internals components	Stainless steel, cast austenitic stainless steel, nickel alloy, type 15-5PH forging	Reactor coolant	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
	Core support shield assembly Core support shield cylinder (top flange) VV assembly locking device	Stainless steel	Reactor coolant	Loss of material/ Wear	Chapter XI.M1, “ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD,” for Class 1 components	No
	Core support shield assembly Outlet and vent valve nozzles VV body and retaining ring	Cast austenitic stainless steel	Reactor coolant and neutron flux	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
	Core support shield assembly Core support shield-to-core barrel bolts	Stainless steel, nickel alloy	Reactor coolant	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
	Core barrel assembly Core barrel cylinder (top and bottom flange) Baffle plates and formers	Stainless steel	Reactor coolant	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
	Core barrel assembly Lower internals assembly-to-core barrel bolts Core barrel-to-thermal shield bolts	Stainless steel, nickel alloy	Reactor coolant	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
	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	Stainless steel, nickel alloy	Reactor coolant	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
	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	Stainless steel, nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness/ Neutron irradiation embrittlement, void swelling	Chapter XI.M16, "PWR Vessel Internals"	No
	Core barrel assembly Lower internals assembly-to-core barrel bolts Core barrel-to-thermal shield bolts	Stainless steel, nickel alloy	Reactor coolant	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
	Core barrel assembly Baffle/former bolts and screws	Stainless steel	Reactor coolant	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
	Core barrel assembly Baffle/former bolts and screws	Stainless steel	Reactor coolant	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
	Core barrel assembly Baffle/former bolts and screws	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/ Neutron irradiation embrittlement, void swelling	A plant-specific aging management program is to be evaluated.	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
	Core barrel assembly Baffle/former bolts and screws	Stainless steel	Reactor coolant	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
	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	Stainless steel; cast austenitic stainless steel	Reactor coolant	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
	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	Stainless steel, nickel alloy	Reactor coolant	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
	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	Stainless steel; cast austenitic stainless steel, nickel alloy	Reactor coolant	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
	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	Stainless steel, nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness/ Neutron irradiation embrittlement, void swelling	Chapter XI.M16, “PWR Vessel Internals”	No
	Lower grid assembly Incore guide tube spider castings	Cast austenitic stainless steel	Reactor coolant	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
	Lower grid assembly Lower grid rib-to-shell forging screws Lower internals assembly-to-thermal shield bolts	Stainless steel, nickel alloy	Reactor coolant	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
	Lower grid assembly Fuel assembly support pads Guide blocks	Stainless steel	Reactor coolant	Loss of material/ Wear	Chapter XI.M1, “ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD,” for Class 1 components	No
	Flow distributor assembly Flow distributor head and flange Incore guide support plate Clamping ring	Stainless steel	Reactor coolant	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
	Flow distributor assembly Shell forging-to-flow distributor bolts	Stainless steel, nickel alloy	Reactor coolant	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
	Flow distributor assembly Flow distributor head and flange Shell forging-to-flow distributor bolts Incore guide support plate Clamping ring	Stainless steel, nickel alloy	Reactor coolant	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
	Flow distributor assembly Flow distributor head and flange Shell forging-to-flow distributor bolts Incore guide support plate Clamping ring	Stainless steel, nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness/ Neutron irradiation embrittlement, void swelling	Chapter XI.M16, “PWR Vessel Internals”	No
	Flow distributor assembly Shell forging to flow distributor bolts	Stainless steel, nickel alloy	Reactor coolant	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
	Thermal shield	Stainless steel	Reactor coolant	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
	Thermal shield	Stainless steel	Reactor coolant	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
	Thermal shield	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/ Neutron irradiation embrittlement, void swelling	Chapter XI.M16, “PWR Vessel Internals”	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
R-03	Class 1 piping, fittings and branch connections less than NPS 4	Stainless steel, carbon steel	Reactor coolant	Crack initiation and growth/ Stress corrosion cracking, inter-granular 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) 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.	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
R-55	Class 1 piping, fittings and branch connections less than NPS 4	Stainless steel, carbon steel	Reactor coolant	Crack initiation and growth/ Thermal and mechanical loading	Chapter XI.M1, “ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD,” for Class 1 components 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
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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
R-52	Class 1 piping, fittings and components	Cast austenitic stainless steel	Reactor coolant > 482°F	Loss of fracture toughness/ Thermal aging embrittlement	Chapter XI.M12, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)"	No
R-08	Class 1 pump casings, and valve bodies and bonnets	Cast austenitic stainless steel	Reactor coolant > 482°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 and 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 pump casings and 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
R-15	Isolation condenser tube side components	Stainless steel, carbon steel	Reactor coolant	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) 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
R-16	Isolation condenser tube side components	Stainless steel, carbon steel	Reactor coolant	Loss of material	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
R-20	Piping, fittings and components greater than or equal to 4 inch nominal diameter	Cast austenitic stainless steel	Reactor coolant	Cracking	Chapter XI.M7, “BWR Stress Corrosion Cracking” and Chapter XI.M2, “Water Chemistry,” for BWR water in BWRVIP-29 (EPRI TR-103515)	No
R-21	Piping, fittings and components greater than or equal to 4 inch nominal diameter	Nickel-alloy	Reactor coolant	Cracking	Chapter XI.M7, “BWR Stress Corrosion Cracking” and Chapter XI.M2, “Water Chemistry,” for BWR water in BWRVIP-29 (EPRI TR-103515)	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
R-22	Piping, fittings and components greater than or equal to 4 inch nominal diameter	Stainless steel	Reactor coolant	Cracking	Chapter XI.M7, “BWR Stress Corrosion Cracking” and Chapter XI.M2, “Water Chemistry,” for BWR water in BWRVIP-29 (EPRI TR-103515)	No
R-23	Piping, fittings and components susceptible to flow-accelerated corrosion	Carbon steel	Reactor coolant	Loss of material/ Flow-accelerated corrosion	Chapter XI.M17, “Flow-Accelerated Corrosion”	No
R-26	Pump and valve closure bolting	Carbon steel	System temperature up to 288°C (550°F)	Loss of material/ Wear	Chapter XI.M18, “Bolting Integrity”	No
R-27	Pump and valve closure bolting	Carbon steel	System temperature up to 288°C (550°F)	Loss of preload/ Stress relaxation	Chapter XI.M18, “Bolting Integrity”	No
R-28	Pump and valve closure bolting	Carbon steel	System temperature up to 288°C (550°F)	Cumulative fatigue damage	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
R-29	Pump and valve seal flanges	Stainless steel, carbon steel	System temperature up to 288°C (550°F)	Loss of material/ Wear	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
R-02	Class 1 piping, fittings and branch connections less than NPS 4	Stainless steel, carbon steel with stainless steel cladding	Reactor coolant	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 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.	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
R-57	Class 1 piping, fittings and branch connections less than NPS 4	Stainless steel, carbon steel with stainless steel cladding	Reactor coolant	Crack initiation and growth/ Thermal and mechanical loading	Chapter XI.M1, “ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD,” for Class 1 components 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
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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
R-05	Class 1 piping, fittings and components	Cast austenitic stainless steel	Reactor coolant	Cracking/ 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
R-52	Class 1 piping, fittings and components	Cast austenitic stainless steel	Reactor coolant > 482°F	Loss of fracture toughness/ Thermal aging embrittlement	Chapter XI.M12, “Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)”	No
R-06	Class 1 piping, fittings and components	Nickel-alloy	Reactor coolant	Cracking/ Primary water stress corrosion cracking	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
R-07	Class 1 piping, fittings and components	Stainless steel, carbon steel with stainless steel or nickel-alloy cladding, nickel-alloy	Reactor coolant	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
R-08	Class 1 pump casings and valve bodies	Cast austenitic stainless steel	Reactor coolant > 482°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 and 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 pump casings and valve bodies.	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
R-09	Class 1 pump casings and valve bodies	Cast austenitic stainless steel, carbon steel with stainless steel cladding	Reactor coolant	Cracking/ 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
R-11	Closure bolting	High-strength low-alloy steel, stainless steel	Air with reactor coolant leakage	Cracking	Chapter XI.M18, "Bolting Integrity"	No
R-12	Closure bolting	High-strength low-alloy steel, stainless steel	Air with reactor coolant leakage	Loss of preload	Chapter XI.M18, "Bolting Integrity"	No
R-13	General piping and components	Carbon steel with stainless steel cladding	Treated borated water	Cumulative fatigue damage	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
R-14	General piping, fittings and components	Stainless steel, carbon steel with stainless steel cladding	Treated borated water >140°F	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
R-17	Piping and components external surfaces and bolting	Carbon steel	Air with boric acid leakage	Loss of material/ Boric acid corrosion	Chapter XI.M10**, "**Boric Acid Corrosion**"**	No
R-18	Piping and components external surfaces and bolting	Stainless steel, carbon steel	System temperature up to 340°C (644°F)	Cumulative fatigue damage	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
R-24	Pressurizer Spray head	Nickel-alloy, stainless steel, cast austenitic stainless steel	Reactor coolant	Cracking/ Primary water stress corrosion cracking, stress corrosion cracking	A plant-specific aging management program is to be evaluated.	Yes, plant specific
R-19	Pressurizer Integral support	Stainless steel, carbon steel	System temperature up to 340°C (644°F)	Cracking/ Cyclic loading	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
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
R-25	Pressurizer components	Carbon steel with stainless steel or nickel-alloy cladding; or stainless steel	Reactor coolant	Cracking/ 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
R-58	Pressurizer components	Carbon steel with stainless steel or nickel-alloy cladding; or stainless steel	Reactor coolant	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
R-30	Reactor coolant system piping and fittings Cold leg Hot leg Surge line Spray line	Stainless steel, carbon steel with stainless steel cladding	Reactor coolant	Cracking/ 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
R-56	Reactor coolant system piping and fittings Cold leg Hot leg Surge line Spray line	Stainless steel, carbon steel with stainless steel cladding	Reactor coolant	Cracking/ Cyclic loading	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
D1. Steam Generator (Recirculating)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-01	Class 1 fittings and components	Nickel-alloy	Reactor coolant	Cracking/ Primary water stress corrosion cracking	A plant-specific aging management program is to be evaluated.	Yes, plant specific
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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
R-07	Class 1 piping, fittings and components	Stainless steel, carbon steel with stainless steel or nickel-alloy cladding, nickel-alloy	Reactor coolant	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
R-10	Closure bolting	Carbon steel	Air with reactor coolant leakage	Cracking	Chapter XI.M18, "Bolting Integrity"	No
R-17	Piping and components external surfaces and bolting	Carbon steel	Air with boric acid leakage	Loss of material/ Boric acid corrosion	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
R-32	Steam generator closure bolting	Carbon steel	System Temperature up to 340°C (644°F)	Loss of preload/ Stress relaxation	Chapter XI.M18, “Bolting Integrity”	No
R-33	Steam generator components	Carbon steel	Secondary feedwater/ steam	Cumulative fatigue damage	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
R-37	Pressure boundary and structural Steam nozzle and safe end FW nozzle and safe end	Carbon steel	Secondary feedwater/ steam	Loss of material/ Flow-accelerated corrosion	Chapter XI.M17, “Flow-Accelerated Corrosion”	No
R-39	Steam generator feedwater impingement plate and support	Carbon steel	Secondary feedwater	Loss of material/ Erosion	A plant-specific aging management program is to be evaluated.	Yes, plant specific
R-34	Steam generator shell assembly	Carbon steel	Secondary feedwater/ steam	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 AMP guidelines in Chapter XI.M1 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

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
R-40	Tube plugs	Nickel-alloy	Reactor coolant	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 for alloy 600 is to be evaluated
R-41	Tube support lattice bars	Carbon steel	Secondary feedwater/ steam	Loss of material/ 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
R-42	Tube support plates	Carbon steel	Secondary feedwater/ steam	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

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
R-43	Tubes	Nickel-alloy	Secondary feedwater/ steam	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 for alloy 600 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
R-44	Tubes and sleeves	Nickel-alloy	Reactor coolant	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 for alloy 600 is to be evaluated
R-45	Tubes and sleeves	Nickel-alloy	Reactor coolant and Secondary feedwater/ 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
D1. Steam Generator (Recirculating)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-47	Tubes and sleeves	Nickel-alloy	Secondary feedwater/ steam	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 for alloy 600 is to be evaluated
R-48	Tubes and sleeves	Nickel-alloy	Secondary feedwater/ steam	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 for alloy 600 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
R-49	Tubes and sleeves	Nickel-alloy	Secondary feedwater/ steam	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, effective-ness of the AMP for alloy 600 is to be evaluated
R-50	Tubes and sleeves (exposed to phosphate chemistry)	Nickel-alloy	Secondary feedwater/ steam	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 for alloy 600 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
R-51	Upper assembly and separators Feedwater inlet ring and support	Carbon steel	Secondary feedwater/ steam	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

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
R-01	Class 1 fittings and components	Nickel-alloy	Reactor coolant	Cracking/ Primary water stress corrosion cracking	A plant-specific aging management program is to be evaluated.	Yes, plant specific
R-04	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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
R-17	Piping and components external surfaces and bolting	Carbon steel	Air with boric acid leakage	Loss of material/ Boric acid corrosion	Chapter XI.M10**, “**Boric Acid Corrosion**”**	No
R-31	Secondary manways and handholes (cover only)	Carbon steel	Air, with leaking secondary-side water and/or steam	Loss of material/ Erosion	Chapter XI.M1, “ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD,” for Class 2 components	No
R-32	Steam generator closure bolting	Carbon steel	System 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
D2. Steam Generator (Once-Through)

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-33	Steam generator components	Carbon steel	Secondary feedwater/ steam	Cumulative fatigue damage	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
R-35	Steam generator components	Carbon steel with stainless steel or nickel-alloy cladding	Reactor coolant	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
R-36	Steam generator components	Nickel-alloy	Secondary feedwater/ steam	Cracking/ Stress corrosion cracking	A plant-specific aging management program is to be evaluated.	Yes, plant specific
R-38	Pressure boundary and structural FW and AFW nozzles and safe ends Steam nozzles and safe ends	Carbon steel	Secondary feedwater/ steam	Loss of material/ Flow-accelerated corrosion	Chapter XI.M17, "Flow-Accelerated Corrosion"	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
R-34	Steam generator shell assembly	Carbon steel	Secondary feedwater/ steam	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 AMP guidelines in Chapter XI.M1 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
R-40	Tube plugs	Nickel-alloy	Reactor coolant	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 for alloy 600 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
R-44	Tubes and sleeves	Nickel-alloy	Reactor coolant	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 for alloy 600 is to be evaluated
R-46	Tubes and sleeves	Nickel-alloy	Reactor coolant and Secondary feedwater/ steam	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
R-47	Tubes and sleeves	Nickel-alloy	Secondary feedwater/ steam	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 for alloy 600 is to be evaluated
R-48	Tubes and sleeves	Nickel-alloy	Secondary feedwater/ steam	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 for alloy 600 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
R-49	Tubes and sleeves	Nickel-alloy	Secondary feedwater/ steam	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, effective-ness of the AMP for alloy 600 is to be evaluated

IV Reactor Vessel, Internals, and Reactor Coolant System
Additional MEAP Combinations Not Currently Addressed by NUREG-1801

Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
	General piping and components	Carbon steel	Concrete	None	None	
	General piping and components	Cast austenitic stainless steel	Air – indoor uncontrolled (Ext)	None	None	
	General piping and components	Nickel-alloy	Air – indoor uncontrolled (Ext)	None	None	
	General piping and components	Stainless steel	Air – indoor uncontrolled (Ext)	None	None	
	General piping and components	Stainless steel	Air with boric acid leakage	None	None	
	General piping and components	Stainless steel	Concrete	None	None	
	General piping and components	Stainless steel	Gas	None	None	
	General piping and components	Stainless steel	Treated borated water	Loss of material	Chapter XI.M2, “Water Chemistry,” for PWR primary water in EPRI TR-105714	No

Line	Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-01	D1.1-j D2.1-h	Class 1 fittings and components	Nickel-alloy	Reactor coolant	Cracking/ Primary water stress corrosion cracking	A plant-specific aging management program is to be evaluated.	Yes, plant specific
R-02	C2.1-g C2.2-h	Class 1 piping, fittings and branch connections less than NPS 4	Stainless steel	Reactor coolant	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 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.	Yes, parameters monitored/ inspected and detection of aging effects are to be evaluated

Line	Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-03	C1.1-i	Class 1 piping, fittings and branch connections less than NPS 4	Stainless steel, carbon steel	Reactor coolant	Crack initiation and growth/ Stress corrosion cracking, inter-granular 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) 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.	Yes, parameter s monitored/ inspected and detection of aging effects are to be evaluated

Line	Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-04	A1.1-b A1.2-a A1.2-b A1.3-a A1.4-b A1.5-b A1.6-a A2.1-b A2.2-c A2.3-c A2.4-a A2.5-d C1.1-b C1.1-d C1.1-e C1.1-h C1.2-a C1.3-d C2.1-a C2.1-b C2.2-a C2.2-b C2.2-c C2.3-a C2.4-a C2.5-a C2.5-d C2.5-e C2.5-f C2.5-q D1.1-h D2.1-c	Class 1 piping, fittings and components	Carbon steel stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage	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

Line	Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-05	C2.1-e C2.2-g C2.5-i	Class 1 piping, fittings and components	Cast austenitic stainless steel	Reactor coolant	Cracking/ 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
R-06	C2.5-k C2.5-s	Class 1 piping, fittings and components	Nickel-alloy	Reactor coolant	Cracking/ Primary water stress corrosion cracking	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

Line	Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-07	C2.2-f C2.5-h C2.5-m C2.5-r D1.1-i	Class 1 piping, fittings and components	Stainless steel, carbon steel with stainless steel or nickel-alloy cladding, nickel-alloy	Reactor coolant	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
R-08	C1.2-c C1.3-b C2.3-c C2.4-c	Class 1 pump casings, and valve bodies and bonnets	Cast austenitic stainless steel	Reactor coolant > 482°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 and 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 pump casings and valve bodies.	No
R-09	C2.3-b C2.4-b	Class 1 pump casings and valve bodies	Cast austenitic stainless steel, carbon steel with stainless steel cladding	Reactor coolant	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

Line	Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-10	D1.1-l	Closure bolting	Carbon steel	Air with reactor coolant leakage	Cracking	Chapter XI.M18, “Bolting Integrity”	No
R-11	C2.3-e C2.4-e C2.5-n	Closure bolting	High-strength low-alloy steel, stainless steel	Air with reactor coolant leakage	Cracking	Chapter XI.M18, “Bolting Integrity”	No
R-12	C2.3-g C2.4-g C2.5-p	Closure bolting	High-strength low-alloy steel, stainless steel	Air with reactor coolant leakage	Loss of preload	Chapter XI.M18, “Bolting Integrity”	No
R-13	C2.6-a	General piping and components	Carbon steel with stainless steel cladding	Treated borated water	Cumulative fatigue damage	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
R-14	C2.6-c	General piping, fittings and components	Stainless steel, carbon steel with stainless steel cladding	Treated borated water >140°F	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

Line	Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-15	C1.4-a	Isolation condenser tube side components	Stainless steel, carbon steel	Reactor coolant	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) 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

Line	Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-16	C1.4-b	Isolation condenser tube side components	Stainless steel, carbon steel	Reactor coolant	Loss of material	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

Line	Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-17	A2.1-a A2.5-e A2.8-b C2.1-d C2.2-d C2.3-f C2.4-f C2.5-b C2.5-o C2.5-u C2.6-b D1.1-g D1.1-k D2.1-b D2.1-j	Piping and components external surfaces and bolting	Carbon steel	Air with boric acid leakage	Loss of material/ Boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No
R-18	C2.3-d C2.4-d C2.5-t C2.5-w	Piping and components external surfaces and bolting	Stainless steel, carbon steel	System temperature up to 340°C (644°F)	Cumulative fatigue damage	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
R-19	C2.5-v	Pressurizer Integral support	Stainless steel, carbon steel	System temperature up to 340°C (644°F)	Cracking/ Cyclic loading	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components	No
R-20	C1.1-f C1.2-b C1.3-c	Piping, fittings and components greater than or equal to 4 inch nominal diameter	Cast austenitic stainless steel	Reactor coolant	Cracking	Chapter XI.M7, "BWR Stress Corrosion Cracking" and Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515)	No

Line	Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-21	C1.1-f	Piping, fittings and components greater than or equal to 4 inch nominal diameter	Nickel-alloy	Reactor coolant	Cracking	Chapter XI.M7, “BWR Stress Corrosion Cracking” and Chapter XI.M2, “Water Chemistry,” for BWR water in BWRVIP-29 (EPRI TR-103515)	No
R-22	C1.1-f C1.3-c	Piping, fittings and components greater than or equal to 4 inch nominal diameter	Stainless steel	Reactor coolant	Cracking	Chapter XI.M7, “BWR Stress Corrosion Cracking” and Chapter XI.M2, “Water Chemistry,” for BWR water in BWRVIP-29 (EPRI TR-103515)	No
R-23	C1.1-a C1.1-c C1.3-a	Piping, fittings and components susceptible to flow-accelerated corrosion	Carbon steel	Reactor coolant	Loss of material/ Flow-accelerated corrosion	Chapter XI.M17, “Flow-Accelerated Corrosion”	No
R-24	C2.5-j	Pressurizer Spray head	Nickel-alloy, stainless steel, cast austenitic stainless steel	Reactor coolant	Cracking/ Primary water stress corrosion cracking, stress corrosion cracking	A plant-specific aging management program is to be evaluated.	Yes, plant specific
R-25	C2.5-c C2.5-g	Pressurizer components	Carbon steel with stainless steel or nickel-alloy cladding; or stainless steel	Reactor coolant	Cracking/ 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
R-26	C1.2-d C1.3-e	Pump and valve closure bolting	Carbon steel	System temperature up to 288°C (550°F)	Loss of material/ Wear	Chapter XI.M18, “Bolting Integrity”	No

Line	Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-27	C1.2-e C1.3-f	Pump and valve closure bolting	Carbon steel	System temperature up to 288°C (550°F)	Loss of preload/ Stress relaxation	Chapter XI.M18, “Bolting Integrity”	No
R-28	C1.2-f C1.3-g	Pump and valve closure bolting	Carbon steel	System temperature up to 288°C (550°F)	Cumulative fatigue damage	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
R-29	C1.2-d C1.3-e	Pump and valve seal flanges	Stainless steel, carbon steel	System temperature up to 288°C (550°F)	Loss of material/ Wear	Chapter XI.M18, “Bolting Integrity”	No
R-30	C2.1-c	Reactor coolant system piping and fittings Cold leg Hot leg Surge line Spray line	Stainless steel, carbon steel with stainless steel cladding	Reactor coolant	Cracking/ 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
R-31	D2.1-l	Secondary manways and handholes (cover only)	Carbon steel	Air, with leaking secondary-side water and/or steam	Loss of material/ Erosion	Chapter XI.M1, “ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD,” for Class 2 components	No
R-32	D1.1-f D2.1-k	Steam generator closure bolting	Carbon steel	System Temperature up to 340°C (644°F)	Loss of preload/ Stress relaxation	Chapter XI.M18, “Bolting Integrity”	No

Line	Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-33	D1.1-a D1.1-b D2.1-d D2.1-g	Steam generator components	Carbon steel	Secondary feedwater/ steam	Cumulative fatigue damage	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
R-34	D1.1-c D2.1-e	Steam generator shell assembly	Carbon steel	Secondary feedwater/ steam	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 AMP guidelines in Chapter XI.M1 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
R-35	D2.1-a	Steam generator components	Carbon steel with stainless steel or nickel-alloy cladding	Reactor coolant	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
R-36	D2.1-i	Steam generator components	Nickel-alloy	Secondary feedwater/ steam	Cracking/ Stress corrosion cracking	A plant-specific aging management program is to be evaluated.	Yes, plant specific

Line	Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-37	D1.1-d	Pressure boundary and structural Steam nozzle and safe end FW nozzle and safe end	Carbon steel	Secondary feedwater/ steam	Loss of material/ Flow-accelerated corrosion	Chapter XI.M17, "Flow-Accelerated Corrosion"	No
R-38	D2.1-f	Pressure boundary and structural FW and AFW nozzles and safe ends Steam nozzles and safe ends	Carbon steel	Secondary feedwater/ steam	Loss of material/ Flow-accelerated corrosion	Chapter XI.M17, "Flow-Accelerated Corrosion"	No
R-39	D1.1-e	Steam generator feedwater impingement plate and support	Carbon steel	Secondary feedwater	Loss of material/ Erosion	A plant-specific aging management program is to be evaluated.	Yes, plant specific
R-40	D1.2-i D1.2-j D2.2-f D2.2-g	Tube plugs	Nickel-alloy	Reactor coolant	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 for alloy 600 is to be evaluated
R-41	D1.2-h	Tube support lattice bars	Carbon steel	Secondary feedwater/ steam	Loss of material/ Flow-accelerated corrosion	A plant-specific aging management program is to be evaluated.	Yes, plant specific

Line	Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-42	D1.2-k	Tube support plates	Carbon steel	Secondary feedwater/ steam	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

Line	Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-43	D1.2-g	Tubes	Nickel-alloy	Secondary feedwater/ steam	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 for alloy 600 is to be evaluated

Line	Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-44	D1.2-a D2.2-a	Tubes and sleeves	Nickel-alloy	Reactor coolant	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 for alloy 600 is to be evaluated
R-45	D1.2-d	Tubes and sleeves	Nickel-alloy	Reactor coolant and Secondary feedwater/ 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

Line	Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-46	D2.2-e	Tubes and sleeves	Nickel-alloy	Reactor coolant and Secondary feedwater/ steam	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
R-47	D1.2-b D2.2-b	Tubes and sleeves	Nickel-alloy	Secondary feedwater/ steam	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 for alloy 600 is to be evaluated

Line	Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-48	D1.2-c D2.2-c	Tubes and sleeves	Nickel-alloy	Secondary feedwater/ steam	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 for alloy 600 is to be evaluated
R-49	D1.2-e D2.2-d	Tubes and sleeves	Nickel-alloy	Secondary feedwater/ steam	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, effective-ness of the AMP for alloy 600 is to be evaluated

Line	Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-50	D1.2-f	Tubes and sleeves (exposed to phosphate chemistry)	Nickel-alloy	Secondary feedwater/ steam	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 for alloy 600 is to be evaluated
R-51	D1.3-a	Upper assembly and separators Feedwater inlet ring and support	Carbon steel	Secondary feedwater/ steam	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
R-52	C1.1-g C2.1-f C2.2-e C2.5-l	Class 1 piping, fittings and components	Cast austenitic stainless steel	Reactor coolant > 482°F	Loss of fracture toughness/ Thermal aging embrittlement	Chapter XI.M12, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)"	No

Line	Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-53	B1.1-c B1.2-b B1.3-b B1.4-b B1.5-b B1.6-b B2.1-c B2.1-h B2.1-m B2.2-c B2.2-f B2.3-d B2.4-g B2.5-d B2.5-j B2.5-p	Reactor vessel internals components	Stainless steel, cast austenitic stainless steel, nickel alloy	Reactor coolant	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
R-54	B3.2-f B3.4-d B3.5-g B4.1-d B4.2-d B4.3-f B4.5-f B4.6-f	Reactor vessel internals components	Stainless steel, cast austenitic stainless steel, nickel alloy	Reactor coolant	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

Line	Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-55	C1.1-i	Class 1 piping, fittings and branch connections less than NPS 4	Stainless steel, carbon steel	Reactor coolant	Crack initiation and growth/ Thermal and mechanical loading	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components 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
R-56	C2.1-c	Reactor coolant system piping and fittings Cold leg Hot leg Surge line Spray line	Stainless steel, carbon steel with stainless steel cladding	Reactor coolant	Cracking/ Cyclic loading	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components	No

Line	Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-57	C2.1-g C2.2-h	Class 1 piping, fittings and branch connections less than NPS 4	Stainless steel, carbon steel with stainless steel cladding	Reactor coolant	Crack initiation and growth/ Thermal and mechanical loading	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components 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

Line	Item	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-58	C2.5-c C2.5-g	Pressurizer components	Carbon steel with stainless steel or nickel-alloy cladding; or stainless steel	Reactor coolant	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