CHAPTER IV

REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

This Page Intentionally Left Blank

MAJOR PLANT SECTIONS

A1. Reactor Vessel (Boiling Water Reactor)

A2. Reactor Vessel (Pressurized Water Reactor)

B1. Reactor Vessel Internals (Boiling Water Reactor)

B2. Reactor Vessel Internals (PWR) - Westinghouse

B3. Reactor Vessel Internals (PWR) - Combustion Engineering

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

C1. Reactor Coolant Pressure Boundary (Boiling Water Reactor)

C2. Reactor Coolant System and Connected Lines (Pressurized Water Reactor)

D1. Steam Generator (Recirculating)

D2. Steam Generator (Once-Through)

(refined outline to be added when issued for public comment)

Explanation of September 30, 2004 changes in preliminary interim draft chapter outline and aging management review (AMR) tables: Within the AMR tables, this update process increases license renewal review efficiency by:

- Consolidating components (combining similar or equivalent components with matching materials, environment and AMP into a single line-item),
- Increasing consistency between **M**aterial/**E**nvironment/**A**ging effects/aging management **P**rogram (MEAP) combinations between systems (some existing MEAPs had multiple definitions that, based on the aging effect, could be broadened to envelope these into a singe MEAP),
- Correcting any inconsistencies in the 2001 edition of the GALL Report,
- Updating references to the appropriate aging management programs, and
- Incorporating line-item changes based on approved staff SER positions or interim staff guidance.

The principal effect of this change is that the tables present the MEAP combinations at a higher level, and the prior detail within a structure or component line item is no longer explicitly presented. Consequently, the identifiers for subcomponents within a line item are no longer presented in the tables. As a result, the introductory listings of these subcomponents (originally in text preceding each table) have been deleted.

The following AMR tables contain a revised "Item" column and a new column titled "Link", which was not contained in the July 2001 revision. The "Item" number is a unique identifier that is used for traceability and, as mentioned above, no longer presents the detailed subcomponent identification. The link identifies the original item in the current version of the GALL Report when applicable (items added to this list refer to bases statements not yet available).

By January 30, 2005, the NRC staff plans to issue a revised GALL Report (NUREG-1801) and SRP-LR (NUREG-1800) for public comment. NRC anticipates re-numbering the line-items to provide an improved unique identifier as part of the public comment document. Also as part of the public comment process, the NRC will issue a NUREG documenting the basis for the proposed changes to the GALL Report and the SRP-LR. This NUREG bases document will be an aid for those reviewing the revised documents to understand what was changed and the basis for the proposed changes.

A1. REACTOR VESSEL (BOILING WATER REACTOR)

Systems, Structures, and Components

This section comprises the boiling water reactor (BWR) pressure vessel and consists of the vessel shell and flanges; attachment welds; the top and bottom heads; nozzles (including safe ends) for the reactor coolant recirculating system and connected systems such as high and low pressure core spray, high and low pressure coolant injection, main steam, and feedwater systems; penetrations for CRD stub tubes, instrumentation, standby liquid control, flux monitor, and drain lines; and control rod drive mechanism housings. The support skirt and attachment welds for vessel supports are also included in the table. Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all structures and components that comprise the reactor vessel are governed by Group A Quality Standards.

System Interfaces

The systems that interface with the reactor vessel include the reactor vessel internals (IV.B1), the reactor coolant pressure boundary (IV.C1), the emergency core cooling system (V.D2), and standby liquid control system (VII.E2).

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
A1 Reactor Vessel (BWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-68	IV.A1.4-a	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	Cracking/ stress corrosion cracking and 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
R-66	IV.A1.3-c	Nozzles Control rod drive return line	Steel (without lining/coating or with degraded lining/coating)	Reactor coolant	Cracking/ cyclic loading	Chapter XI.M6, “BWR Control Rod Drive Return Line Nozzle”	No
R-65	IV.A1.3-b	Nozzles Feedwater	Steel (without lining/coating or with degraded lining/coating)	Reactor coolant	Cracking/ cyclic loading	Chapter XI.M5, “BWR Feedwater Nozzle”	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
A1 Reactor Vessel (BWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-67	IV.A1.3-e	Nozzles Low pressure coolant injection or RHR injection mode	Steel	Reactor coolant and 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 1017 n/cm2 (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 (BWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-69	IV.A1.5-a	Penetrations Control rod drive stub tubes Instrumentation Jet pump instrument Standby liquid control Flux monitor Drain line	Stainless steel, nickel alloy	Reactor coolant	Cracking/ 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
R-04	IV.A1.2-b IV.A1.2-a IV.A1.3-a IV.A1.4-b IV.A1.1-b IV.A1.5-b IV.A1.6-a	Piping, piping components, and piping elements	Steel, stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	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						
A1	Reactor Vessel (BWR)						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
R-70	IV.A1.7-a	Support skirt and attachment welds	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
R-60	IV.A1.1-c	Top head enclosure Closure studs and nuts	High strength low alloy steel Maximum tensile strength < 1172 MPa (<170 Ksi)	Air with reactor coolant leakage	Cracking/ stress corrosion cracking and intergranular stress corrosion cracking	Chapter XI.M3, “Reactor Head Closure Studs”	No
R-61	IV.A1.1-d	Top head enclosure Vessel flange leak detection line	Stainless steel, nickel alloy	Air with reactor coolant leakage	Cracking/ stress corrosion cracking and 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
R-59	IV.A1.1-a	Top head enclosure (without cladding) Top head Nozzles (vent, top head spray or RCIC, and spare)	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

IV	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM						
A1	Reactor Vessel (BWR)						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
R-64	IV.A1.2-e	Vessel shell Attachment welds	Stainless steel, nickel alloy	Reactor coolant	Cracking/ stress corrosion cracking and 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 (BWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-62	IV.A1.2-c	Vessel shell Intermediate beltline shell Beltline welds	Steel (without lining/coating or with degraded lining/coating)	Reactor coolant and 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 1017 n/cm2 (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 RTNDT 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 (BWR)						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
R-63	IV.A1.2-d	Vessel shell Intermediate beltline shell Beltline welds	Steel (without lining/coating or with degraded lining/coating)	Reactor coolant and neutron flux	Loss of fracture toughness/ neutron irradiation embrittlement	Chapter XI.M31, “Reactor Vessel Surveillance”	Yes, plant specific

A2. REACTOR VESSEL (PRESSURIZED WATER REACTOR)

Systems, Structures, and Components

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

System Interfaces

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

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
A2 Reactor Vessel (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-71	IV.A2.1-c	Closure head Stud assembly	High strength low alloy steel Maximum tensile strength < 1172 MPa (<170 Ksi)	Air with reactor coolant leakage	Cracking/ stress corrosion cracking	Chapter XI.M3, “Reactor Head Closure Studs”	No
R-73	IV.A2.1-e	Closure head Stud assembly	High strength low alloy steel Maximum tensile strength < 1172 MPa (<170 Ksi)	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
R-72	IV.A2.1-d	Closure head Stud assembly	High strength low alloy steel Maximum tensile strength < 1172 MPa (<170 Ksi)	Air with reactor coolant leakage	Loss of material/ wear	Chapter XI.M3, “Reactor Head Closure Studs”	No
R-74	IV.A2.1-f	Closure head Vessel flange leak detection line	Stainless steel	Air with reactor coolant leakage	Cracking/ 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	Yes, plant specific

IV A2	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel (PWR)						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
						detection line.	
R-78	IV.A2.2-e	Control rod drive head penetration Flange bolting	Stainless steel	Air with reactor coolant leakage	Cracking/ stress corrosion cracking	Chapter XI.M18, “Bolting Integrity”	No
R-79	IV.A2.2-f	Control rod drive head penetration Flange bolting	Stainless steel	Air with reactor coolant leakage	Loss of material/ wear	Chapter XI.M18, “Bolting Integrity”	No
R-80	IV.A2.2-g	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
R-75	IV.A2.2-a	Control rod drive head penetration Nozzle	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 and Chapter XI.M2, “Water Chemistry,” for PWR primary water in EPRI TR-105714 and plant specific AMP consistent with applicant commitments to NRC Order EA-03-009 or any subsequent regulatory requirements.	Yes, plant specific

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
A2 Reactor Vessel (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-77	IV.A2.2-d	Control rod drive head penetration Pressure housing	Cast austenitic stainless steel	Reactor coolant >250°C (>482°F)	Loss of fracture toughness/ thermal aging embrittlement	Chapter XI.M12 “Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)”	No
R-76	IV.A2.2-b	Control rod drive head penetration Pressure housing	Stainless steel, cast austenitic stainless steel, nickel alloy	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-88	IV.A2.6-a	Core support pads/core guide lugs	Nickel alloy	Reactor coolant	Cracking/ 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
R-17	IV.A2.8-b IV.A2.1-a IV.A2.5-e	External surfaces	Steel	Air with borated water leakage	Loss of material/ boric acid corrosion	Chapter XI.M10, “Boric Acid Corrosion”	No
R-83	IV.A2.4-b	Nozzle safe ends Inlet Outlet Safety injection	Stainless steel, cast austenitic stainless steel, nickel alloy and associated welds and buttering	Reactor coolant	Cracking/ 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 (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-81	IV.A2.3-a	Nozzles Inlet Outlet Safety injection	Steel with stainless steel cladding	Reactor coolant and 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 1017 n/cm2 (E >1 MeV) at the end of the license renewal term. The TLAA is to evaluate the impact of neutron embrittlement on: (a) the RTPTS 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
R-82	IV.A2.3-b	Nozzles Inlet Outlet Safety injection	Steel with stainless steel cladding	Reactor coolant and 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
A2 Reactor Vessel (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-90	IV.A2.7-b	Penetrations Head vent pipe (top head) Instrument tubes (top head)	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 and Chapter XI.M2, “Water Chemistry,” for PWR primary water in EPRI TR-105714 and plant specific AMP consistent with applicant commitments to NRC Order EA-03-009 or any subsequent regulatory requirements.	Yes, plant specific
R-89	IV.A2.7-a	Penetrations Instrument tubes (bottom head)	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 and Chapter XI.M2, “Water Chemistry,” for PWR primary water in EPRI TR-105714 and plant specific AMP consistent with applicant commitments to NRC Bulletin BL-03-02 or any subsequent regulatory requirements.	Yes, plant specific

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
A2 Reactor Vessel (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-04	IV.A2.2-c IV.A2.3-c IV.A2.5-d IV.A2.4-a IV.A2.1-b	Piping, piping components, and piping elements	Steel, stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	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-91	IV.A2.8-a	Pressure vessel support Skirt support	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 (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-85	IV.A2.5-b	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 (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-84	IV.A2.5-a	Vessel shell Upper shell Intermediate and lower shell (including beltline welds)	Steel with stainless steel cladding	Reactor coolant and 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 1017 n/cm2 (E >1 MeV) at the end of the license renewal term. The TLAA is to evaluate the impact of neutron embrittlement on: (a) the RTPTS 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 A2	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel (PWR)						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
R-86	IV.A2.5-c	Vessel shell Upper shell Intermediate and lower shell (including beltline welds)	Steel with stainless steel cladding	Reactor coolant and neutron flux	Loss of fracture toughness/ neutron irradiation embrittlement	Chapter XI.M31, "Reactor Vessel Surveillance"	Yes, plant specific
R-87	IV.A2.5-f	Vessel shell Vessel flange	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

B1. REACTOR VESSEL INTERNALS (BOILING WATER REACTOR)

Systems, Structures, and Components

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

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

System Interfaces

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

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B1 Reactor Vessel Internals (BWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-95	IV.B1.1-e	Core shroud and core plate Access hole cover (mechanical covers)	Nickel alloy	Reactor coolant	Cracking/ 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
R-94	IV.B1.1-d	Core shroud and core plate Access hole cover (welded covers)	Nickel alloy	Reactor coolant	Cracking/ 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
R-93	IV.B1.1-b	Core shroud and core plate Core plate Core plate bolts (used in early BWRs)	Stainless steel	Reactor coolant	Cracking/ 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 (BWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-92	IV.B1.1-a	Core shroud and core plate Core shroud (upper, central, lower)	Stainless steel	Reactor coolant	Cracking/ 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
R-96	IV.B1.1-f	Core shroud and core plate Shroud support structure (shroud support cylinder, shroud support plate, shroud support legs)	Nickel alloy	Reactor coolant	Cracking/ 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 B1	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (BWR)						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
R-97	IV.B1.1-g	Core shroud and core plate Shroud support structure (shroud support cylinder, shroud support plate, shroud support legs)	Stainless steel	Reactor coolant	Cracking/ 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
R-99	IV.B1.3-a	Core spray lines and spargers Core spray lines (headers) Spray rings Spray nozzles Thermal sleeves	Stainless steel	Reactor coolant	Cracking/ 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
R-104	IV.B1.5-c	Fuel supports and control rod drive assemblies Control rod drive housing	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking and 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 (BWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-103	IV.B1.5-a	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
R-105	IV.B1.6-a	Instrumentation Intermediate range monitor (IRM) dry tubes Source range monitor (SRM) dry tubes Incore neutron flux monitor guide tubes	Stainless steel	Reactor coolant	Cracking/ 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
R-101	IV.B1.4-c	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
R-102	IV.B1.4-d	Jet pump assemblies Jet pump sensing line	Stainless steel	Reactor coolant	Cracking/ cyclic loading	A plant-specific aging management program is to be evaluated.	Yes, plant specific

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B1 Reactor Vessel Internals (BWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-100	IV.B1.4-a	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	Cracking/ 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
R-53	IV.B1.3-b IV.B1.2-b IV.B1.5-b IV.B1.4-b IV.B1.1-c IV.B1.6-b	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 (BWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-98	IV.B1.2-a	Top guide	Stainless steel	Reactor coolant	Cracking/ 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) For top guides with neutron fluence exceeding the IASCC threshold (5x1020, E>IMeV) inspect ten (10) percent of the top guide locations using enhanced visual inspection technique, EVT-1 within 12 years, one-half (5 percent) to be completed within 6 years. Locations selected for examination will be areas that have exceeded the neutron fluence threshold. The extent and frequency of examination of the top guide is similar to the examination of the control rod drive housing guide tube in BWRVIP-47.	No

This Page Intentionally Left Blank

B2. REACTOR VESSEL INTERNALS (PWR) - WESTINGHOUSE

Systems, Structures, and Components

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

System Interfaces

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

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-124	IV.B2.4-b	Baffle/former assembly Baffle and former plates	Stainless steel	Reactor coolant	Changes in dimensions/Void swelling	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-123	IV.B2.4-a	Baffle/former assembly Baffle and former plates	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” for PWR primary water in EPRI TR-105714 and the applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-127	IV.B2.4-e	Baffle/former assembly Baffle and former plates	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/ neutron irradiation embrittlement, void swelling	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-126	IV.B2.4-d	Baffle/former assembly Baffle/former bolts	Stainless steel	Reactor coolant	Changes in dimensions/Void swelling	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV B2	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) - Westinghouse						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
R-125	IV.B2.4-c,	Baffle/former assembly Baffle/former bolts	Stainless steel	Reactor coolant and high fluence (>1 x 10E21 n/cm2 E >0.1 MeV)	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-128	IV.B2.4-f	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
R-129	IV.B2.4-h	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

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-121	IV.B2.3-b	Core barrel Core barrel (CB) CB flange (upper) CB outlet nozzles Thermal shield	Stainless steel	Reactor coolant	Changes in dimensions/Void swelling	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-120	IV.B2.3-a	Core barrel Core barrel (CB) CB flange (upper) CB outlet nozzles Thermal shield	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” for PWR primary water in EPRI TR-105714 and the applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV B2	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) - Westinghouse						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
R-122	IV.B2.3-c	Core barrel Core barrel (CB) CB flange (upper) CB outlet nozzles Thermal shield	Stainless steel	Reactor coolant >250°C (>482°F) and neutron flux	Loss of fracture toughness/ neutron irradiation embrittlement, void swelling	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV B2	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) - Westinghouse						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
R-145	IV.B2.6-c	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
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-144	IV.B2.6-b	Instrumentation support structures Flux thimble guide tubes	Stainless steel	Reactor coolant	Changes in dimensions/Void swelling	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-143	IV.B2.6-a	Instrumentation support structures Flux thimble guide tubes	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” for PWR primary water in EPRI TR-105714 and the applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-137	IV.B2.5-i	Lower internal assembly Clevis insert bolts	Stainless steel, nickel alloy	Reactor coolant	Loss of preload/ stress relaxation	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-134	IV.B2.5-f	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	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM						
B2	Reactor Vessel Internals (PWR) - Westinghouse						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
R-133	IV.B2.5-e	Lower internal assembly Fuel alignment pins Lower support plate column bolts Clevis insert bolts	Stainless steel, nickel alloy	Reactor coolant	Cracking/ stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” for PWR primary water in EPRI TR-105714 and the applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-135	IV.B2.5-g	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	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-132	IV.B2.5-c	Lower internal assembly Lower core plate	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/ neutron irradiation embrittlement, void swelling	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-131	IV.B2.5-b	Lower internal assembly Lower core plate Radial keys and clevis inserts	Stainless steel	Reactor coolant	Changes in dimensions/Void swelling	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
R-130	IV.B2.5-a	Lower internal assembly Lower core plate Radial keys and clevis inserts	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” for PWR primary water in EPRI TR-105714 and the applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-140	IV.B2.5-m	Lower internal assembly Lower support casting Lower support plate columns	Cast austenitic stainless steel	Reactor coolant >250°C (>482°F) 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
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-141	IV.B2.5-n	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	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-139	IV.B2.5-l	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	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-138	IV.B2.5-k	Lower internal assembly Lower support forging or casting Lower support plate columns	Stainless steel, cast austenitic stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714 and the applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-136	IV.B2.5-h	Lower internal assembly Lower support plate column bolts	Stainless steel, nickel alloy	Reactor coolant	Loss of preload/ stress relaxation	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV B2	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) - Westinghouse						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
R-142	IV.B2.5-o	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
R-118	IV.B2.2-d	RCCA guide tube assemblies RCCA guide tube bolts RCCA guide tube support pins	Stainless steel, nickel alloy	Reactor coolant	Cracking/ stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714 and the applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV B2	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) - Westinghouse						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
R-119	IV.B2.2-e	RCCA guide tube assemblies RCCA guide tube bolts, RCCA guide tube support pins	Stainless steel, nickel alloy	Reactor coolant	Changes in dimensions/Void swelling	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-117	IV.B2.2-b	RCCA guide tube assemblies RCCA guide tubes	Stainless steel	Reactor coolant	Changes in dimensions/Void swelling	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM						
B2	Reactor Vessel Internals (PWR) - Westinghouse						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
R-116	IV.B2.2-a	RCCA guide tube assemblies RCCA guide tubes	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714 and the applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-53	IV.B2.1-c IV.B2.1-m IV.B2.5-p IV.B2.3-d IV.B2.2-c IV.B2.5-d IV.B2.2-f IV.B2.1-h IV.B2.5-j IV.B2.4-g	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 B2	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) - Westinghouse						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
R-108	IV.B2.1-d	Upper internals assembly Hold-down spring	Stainless steel	Reactor coolant	Loss of preload/ stress relaxation	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-115	IV.B2.1-l	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	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-110	IV.B2.1-f	Upper internals assembly Upper support column	Stainless steel, cast austenitic stainless steel	Reactor coolant	Changes in dimensions/Void swelling	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-109	IV.B2.1-e	Upper internals assembly Upper support column	Stainless steel, cast austenitic stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” for PWR primary water in EPRI TR-105714 and the applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM						
B2	Reactor Vessel Internals (PWR) - Westinghouse						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
R-111	IV.B2.1-g	Upper internals assembly Upper support column (only cast austenitic stainless steel portions)	Cast austenitic stainless steel	Reactor coolant >250°C (>482°F) 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
R-114	IV.B2.1-k	Upper internals assembly Upper support column bolts	Stainless steel, nickel alloy	Reactor coolant	Loss of preload/ stress relaxation	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM						
B2	Reactor Vessel Internals (PWR) - Westinghouse						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
R-113	IV.B2.1-j	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	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-112	IV.B2.1-i	Upper internals assembly Upper support column bolts Upper core plate alignment pins Fuel alignment pins	Stainless steel, nickel alloy	Reactor coolant	Cracking/ stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” for PWR primary water in EPRI TR-105714 and the applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-107	IV.B2.1-b	Upper internals assembly Upper support plate Upper core plate Hold-down spring	Stainless steel	Reactor coolant	Changes in dimensions/Void swelling	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-106	IV.B2.1-a	Upper internals assembly Upper support plate Upper core plate Hold-down spring	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” for PWR primary water in EPRI TR-105714 and the applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

B3. REACTOR VESSEL INTERNALS (PWR) - COMBUSTION ENGINEERING

Systems, Structures, and Components

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

System Interfaces

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

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B3 Reactor Vessel Internals (PWR) - Combustion Engineering

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-153	IV.B3.2-e	CEA Shroud Assemblies	Cast austenitic stainless steel	Reactor coolant >250°C (>482°F) 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
R-149	IV.B3.2-a	CEA Shroud Assemblies	Stainless steel, cast austenitic stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714 and the applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-152	IV.B3.2-d	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	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-151	IV.B3.2-c	CEA Shroud Assemblies CEA shrouds bolts	Stainless steel, cast austenitic stainless steel, nickel alloy	Reactor coolant	Changes in dimensions/Void swelling	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-150	IV.B3.2-b	CEA Shroud Assemblies CEA shrouds bolts	Stainless steel, nickel alloy	Reactor coolant	Cracking/ stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” for PWR primary water in EPRI TR-105714 and the applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B3 Reactor Vessel Internals (PWR) - Combustion Engineering

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-154	IV.B3.2-g	CEA Shroud Assemblies CEA shrouds bolts	Stainless steel, nickel alloy	Reactor coolant	Loss of preload/ stress relaxation	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-161	IV.B3.4-c	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	Reactor coolant and neutron flux	Loss of fracture toughness/ neutron irradiation embrittlement, void swelling	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B3 Reactor Vessel Internals (PWR) - Combustion Engineering

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-163	IV.B3.4-f	Core shroud assembly Core shroud assembly bolts (later plants are welded)	Stainless steel, nickel alloy	Reactor coolant	Changes in dimensions/Void swelling	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-162	IV.B3.4-e	Core shroud assembly Core shroud assembly bolts (later plants are welded)	Stainless steel, nickel alloy	Reactor coolant	Cracking/ stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” for PWR primary water in EPRI TR-105714 and the applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM						
B3	Reactor Vessel Internals (PWR) - Combustion Engineering						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
R-164	IV.B3.4-g	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	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-165	IV.B3.4-h	Core shroud assembly Core shroud assembly bolts Core shroud tie rods	Stainless steel, nickel alloy	Reactor coolant	Loss of preload/ stress relaxation	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV B3	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) - Combustion Engineering						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
R-159	IV.B3.4-a	Core shroud assembly Core shroud tie rods (core support plate attached by welds in later plants)	Stainless steel, cast austenitic stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” for PWR primary water in EPRI TR-105714 and the applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-160	IV.B3.4-b	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	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B3 Reactor Vessel Internals (PWR) - Combustion Engineering

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-158	IV.B3.3-b	Core support barrel Core support barrel upper flange	Stainless steel	Reactor coolant	Changes in dimensions/Void swelling	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-155	IV.B3.3-a	Core support barrel Core support barrel upper flange	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” for PWR primary water in EPRI TR-105714 and the applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B3 Reactor Vessel Internals (PWR) - Combustion Engineering

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-157	IV.B3.3-a	Core support barrel Core support barrel upper flange	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/ neutron irradiation embrittlement, void swelling	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-156	IV.B3.3-b	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
R-171	IV.B3.5-f	Lower internal assembly Core support column	Cast austenitic stainless steel	Reactor coolant >250°C (>482°F) 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	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-168	IV.B3.5-c	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	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV B3	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) - Combustion Engineering						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
R-169	IV.B3.5-d	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	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B3 Reactor Vessel Internals (PWR) - Combustion Engineering

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-166	IV.B3.5-a	Lower internal assembly Core support plate Lower support structure beam assemblies Core support column Core support barrel snubber assemblies	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714 and the applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-170	IV.B3.5-e	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	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-167	IV.B3.5-b	Lower internal Assembly Fuel alignment pins Core support column bolts	Stainless steel, nickel alloy	Reactor coolant	Cracking/ stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” for PWR primary water in EPRI TR-105714 and the applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-54	IV.B3.2-f IV.B3.5-g IV.B3.4-d	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. 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	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-148	IV.B3.1-c	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
R-147	IV.B3.1-b	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	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B3 Reactor Vessel Internals (PWR) - Combustion Engineering

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-146	IV.B3.1-a	Upper Internals Assembly Upper guide structure support plate Fuel alignment plate Fuel alignment plate guide lugs and guide lug inserts	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” for PWR primary water in EPRI TR-105714 and the applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

This Page Intentionally Left Blank

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

Systems, Structures, and Components

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

System Interfaces

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

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B4 Reactor Vessel Internals (PWR) – Babcock & Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-125	IV.B4.5-g	Baffle/former assembly Baffle/former bolts	Stainless steel	Reactor coolant and high fluence (>1 x 10E21 n/cm2 E >0.1 MeV)	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-180	IV.B4.3-a	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	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” for PWR primary water in EPRI TR-105714 and the applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV B4	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) – Babcock & Wilcox						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
R-182	IV.B4.3-c	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	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-183	IV.B4.3-d	Control rod guide tube (CRGT) assembly CRGT spacer casting	Cast austenitic stainless steel	Reactor coolant >250°C (>482°F) 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
B4 Reactor Vessel Internals (PWR) – Babcock & Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-181	IV.B4.3-b	Control rod guide tube (CRGT) assembly CRGT spacer screws Flange-to-upper grid screws	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714 and the applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-184	IV.B4.3-e	Control rod guide tube (CRGT) assembly Flange-to-upper grid screws	Stainless steel	Reactor coolant	Loss of preload/ stress relaxation	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM						
B4	Reactor Vessel Internals (PWR) – Babcock & Wilcox						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
R-199	IV.B4.5-h	Core barrel assembly Baffle/former bolts and screws	Stainless steel	Reactor coolant	Changes in dimensions/Void swelling	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B4 Reactor Vessel Internals (PWR) – Babcock & Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-198	IV.B4.5-g	Core barrel assembly Baffle/former bolts and screws	Stainless steel	Reactor coolant	Cracking/ 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
R-201	IV.B4.5-j	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

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B4 Reactor Vessel Internals (PWR) – Babcock & Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-200	IV.B4.5-i	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
R-193	IV.B4.5-a	Core barrel assembly Core barrel cylinder (top and bottom flange) Baffle plates and formers	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” for PWR primary water in EPRI TR-105714 and the applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM						
B4	Reactor Vessel Internals (PWR) – Babcock & Wilcox						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
R-195	IV.B4.5-c	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	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-196	IV.B4.5-d	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	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV B4	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) – Babcock & Wilcox						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
R-194	IV.B4.5-b	Core barrel assembly Lower internals assembly-to-core barrel bolts Core barrel-to-thermal shield bolts	Stainless steel, nickel alloy	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” for PWR primary water in EPRI TR-105714 and the applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-197	IV.B4.5-e	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	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM						
B4	Reactor Vessel Internals (PWR) – Babcock & Wilcox						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
R-190	IV.B4.4-f	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
R-188	IV.B4.4-d	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, PH Stainless Steel forging	Reactor coolant and neutron flux	Loss of fracture toughness/ neutron irradiation embrittlement, void swelling	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B4 Reactor Vessel Internals (PWR) – Babcock & Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-187	IV.B4.4-c	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, PH Stainless Steel forging	Reactor coolant	Changes in dimensions/Void swelling	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-185	IV.B4.4-a	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, PH stainless steel forging, CASS	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714 and the applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B4 Reactor Vessel Internals (PWR) – Babcock & Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-192	IV.B4.4-h	Core support shield assembly Core support shield-to-core barrel bolts	Stainless steel, nickel alloy	Reactor coolant	Loss of preload/ stress relaxation	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-186	IV.B4.4-b	Core support shield assembly Core support shield-to-core barrel bolts VV assembly locking device	Stainless steel, nickel alloy	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” for PWR primary water in EPRI TR-105714 and the applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B4 Reactor Vessel Internals (PWR) – Babcock & Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-191	IV.B4.4-g	Core support shield assembly Outlet and vent valve nozzles VV body and retaining ring	Cast austenitic stainless steel	Reactor coolant >250°C (>482°F) 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
R-209	IV.B4.7-a	Flow distributor assembly Flow distributor head and flange Incore guide support plate Clamping ring	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714 and the applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV B4	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) – Babcock & Wilcox						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
R-212	IV.B4.7-d	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	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-211	IV.B4.7-c	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	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM						
B4	Reactor Vessel Internals (PWR) – Babcock & Wilcox						

Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
R-210	IV.B4.7-b	Flow distributor assembly Shell forging-to-flow distributor bolts	Stainless steel, nickel alloy	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714 and the applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-213	IV.B4.7-e	Flow distributor assembly Shell forging-to-flow distributor bolts	Stainless steel, nickel alloy	Reactor coolant	Loss of preload/ stress relaxation	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM						
B4	Reactor Vessel Internals (PWR) – Babcock & Wilcox						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
R-208	IV.B4.6-h	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
R-206	IV.B4.6-e	Lower grid assembly Incore guide tube spider castings	Cast austenitic stainless steel	Reactor coolant >250°C (>482°F) 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
B4 Reactor Vessel Internals (PWR) – Babcock & Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-202	IV.B4.6-a	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	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” for PWR primary water in EPRI TR-105714 and the applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B4 Reactor Vessel Internals (PWR) – Babcock & Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-204	IV.B4.6-c	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	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B4 Reactor Vessel Internals (PWR) – Babcock & Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-205	IV.B4.6-d	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	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B4 Reactor Vessel Internals (PWR) – Babcock & Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-203	IV.B4.6-b	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	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” for PWR primary water in EPRI TR-105714 and the applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-207	IV.B4.6-g	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	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B4 Reactor Vessel Internals (PWR) – Babcock & Wilcox

Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
R-172	IV.B4.1-a	Plenum cover and plenum cylinder Plenum cover assembly Plenum cylinder Reinforcing plates	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714 and the applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-174	IV.B4.1-c	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	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B4 Reactor Vessel Internals (PWR) – Babcock & Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-173	IV.B4.1-b	Plenum cover and plenum cylinder Top flange-to-cover bolts Bottom flange-to-upper grid screws	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” for PWR primary water in EPRI TR-105714 and the applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-54	IV.B4.6-f IV.B4.5-f IV.B4.1-d IV.B4.2-d IV.B4.3-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. 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 B4	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel Internals (PWR) – Babcock & Wilcox						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
R-189	IV.B4.4-e	Reactor vessel internals components	Stainless steel, cast austenitic stainless steel, nickel alloy, PH Stainless Steel 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
R-215	IV.B4.8-b	Thermal shield	Stainless steel	Reactor coolant	Changes in dimensions/Void swelling	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B4 Reactor Vessel Internals (PWR) – Babcock & Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-216	IV.B4.8-c	Thermal shield	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/ neutron irradiation embrittlement, void swelling	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-214	IV.B4.8-a	Thermal shield	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” for PWR primary water in EPRI TR-105714 and the applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM						
B4	Reactor Vessel Internals (PWR) – Babcock & Wilcox						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
R-179	IV.B4.2-f	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
R-176	IV.B4.2-b	Upper grid assembly Rib- to-ring screws	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” for PWR primary water in EPRI TR-105714 and the applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B4 Reactor Vessel Internals (PWR) – Babcock & Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-175	IV.B4.2-a	Upper grid assembly Upper grid rib section Upper grid ring forging Fuel assembly support pads Plenum rib pads	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714 and the applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-177	IV.B4.2-c	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	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B4 Reactor Vessel Internals (PWR) – Babcock & Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-178	IV.B4.2-e	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	Applicant must provide a commitment which includes the following elements: (1) to participate in industry programs for investigating and managing aging effects applicable to Reactor Internals, (2) to evaluate and implement the results of the industry programs as applicable to the Reactor Internals design and, (3) to submit, for NRC review and approval an inspection plan for Reactor Internals, as based on industry recommendation, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.

This Page Intentionally Left Blank

C1. REACTOR COOLANT PRESSURE BOUNDARY (BOILING WATER REACTOR)

Systems, Structures, and Components

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

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

System Interfaces

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

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
C1 Reactor Coolant Pressure Boundary (BWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-03	IV.C1.1-i	Class 1 piping, fittings and branch connections < NPS 4	Stainless steel, Steel	Reactor coolant	Cracking/ stress corrosion cracking and intergranular 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 (BWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-55	IV.C1.1-i	Class 1 piping, fittings and branch connections < NPS 4	Stainless steel, Steel	Reactor coolant	Cracking/ 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-52	IV.C1.1-g	Class 1 piping, piping components, and piping elements	Cast austenitic stainless steel	Reactor coolant >250°C (>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
C1 Reactor Coolant Pressure Boundary (BWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-08	IV.C1.3-b IV.C1.2-c	Class 1 pump casings, and valve bodies and bonnets	Cast austenitic stainless steel	Reactor coolant >250°C (>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. Alternatively, the requirements of ASME Code Case N-481 for pump casings, are sufficient for managing the effects of loss of fracture toughness due to thermal aging embrittlement of CASS pump casings.	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
C1 Reactor Coolant Pressure Boundary (BWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-15	IV.C1.4-a	Isolation condenser tube side components	Stainless steel, Steel	Reactor coolant	Cracking/ stress corrosion cracking and intergranular 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) 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 (BWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-16	IV.C1.4-b	Isolation condenser tube side components	Stainless steel, 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) 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-23	IV.C1.3-a IV.C1.1-a IV.C1.1-c	Piping, piping components, and piping elements	Steel	Reactor coolant	Wall thinning/ flow-accelerated corrosion	Chapter XI.M17, "Flow-Accelerated Corrosion"	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
C1 Reactor Coolant Pressure Boundary (BWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-04	IV.C1.1-e IV.C1.1-h IV.C1.2-a IV.C1.1-b IV.C1.3-d IV.C1.1-d	Piping, piping components, and piping elements	Steel, stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	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-21	IV.C1.1-f	Piping, piping components, and piping elements greater than or equal to 4 NPS	Nickel alloy	Reactor coolant	Cracking/ stress corrosion cracking and 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
R-22	IV.C1.3-c IV.C1.1-f	Piping, piping components, and piping elements greater than or equal to 4 NPS	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking and 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
C1 Reactor Coolant Pressure Boundary (BWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-20	IV.C1.1-f IV.C1.3-c IV.C1.2-b	Piping, piping components, and piping elements greater than or equal to 4 NPS	Stainless steel, cast austenitic stainless steel	Reactor coolant	Cracking/ stress corrosion cracking and 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
R-27	IV.C1.3-f IV.C1.2-e	Pump and valve closure bolting	High- strength low-alloy steel SA 193 Gr. B7	System temperature up to 288°C (550°F)	Loss of preload/ stress relaxation	Chapter XI.M18, “Bolting Integrity”	No
R-28	IV.C1.2-f IV.C1.3-g	Pump and valve closure bolting	Steel	System 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-26	IV.C1.2-d IV.C1.3-e	Pump and valve closure bolting	Steel	System temperature up to 288°C (550°F)	Loss of material/ wear	Chapter XI.M18, “Bolting Integrity”	No
R-29	IV.C1.3-e IV.C1.2-d	Pump and valve seal flange closure bolting	Stainless steel, Steel	Air with metal temperature up to 288°C (550°F)	Loss of material/ wear	Chapter XI.M18, “Bolting Integrity”	No

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

Systems, Structures, and Components

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

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

System Interfaces

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

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
C2 Reactor Coolant System and Connected Lines (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-02	IV.C2.1-g IV.C2.2-h	Class 1 piping, fittings and branch connections < NPS 4	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 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 (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-57	IV.C2.2-h IV.C2.1-g	Class 1 piping, fittings and branch connections < NPS 4	Stainless steel/ steel with stainless steel cladding	Reactor coolant	Cracking/ 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-07	IV.C2.2-f IV.C2.5-h IV.C2.5-m IV.C2.5-r	Class 1 piping, fittings and primary nozzles, safe ends, manways, and flanges	Stainless steel, steel with stainless steel or nickel-alloy cladding, nickel-alloy	Reactor coolant	Cracking/ 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
C2 Reactor Coolant System and Connected Lines (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-05	IV.C2.5-i IV.C2.1-e IV.C2.2-g	Class 1 piping, piping components, and piping elements	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-52	IV.C2.2-e IV.C2.5-l IV.C2.1-f	Class 1 piping, piping components, and piping elements	Cast austenitic stainless steel	Reactor coolant >250°C (>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 (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-09	IV.C2.3-b IV.C2.4-b	Class 1 pump casings and valve bodies	CASS, 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

IV	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM						
C2	Reactor Coolant System and Connected Lines (PWR)						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
R-08	IV.C2.4-c IV.C2.3-c	Class 1 pump casings, and valve bodies and bonnets	Cast austenitic stainless steel	Reactor coolant >250°C (>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. Alternatively, the requirements of ASME Code Case N-481 for pump casings, are sufficient for managing the effects of loss of fracture toughness due to thermal aging embrittlement of CASS pump casings.	No
R-11	IV.C2.3-e IV.C2.5-n IV.C2.4-e	Closure bolting	High-strength low-alloy steel, stainless steel	Air with reactor coolant leakage	Cracking/ stress corrosion cracking	Chapter XI.M18, “Bolting Integrity”	No
R-12	IV.C2.5-p IV.C2.3-g IV.C2.4-g	Closure bolting	High-strength low-alloy steel, 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
C2 Reactor Coolant System and Connected Lines (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-17	IV.C2.1-d IV.C2.5-b IV.C2.2-d IV.C2.6-b IV.C2.5-u IV.C2.5-o IV.C2.3-f IV.C2.4-f	External surfaces	Steel	Air with borated water leakage	Loss of material/ boric acid corrosion	Chapter XI.M10, “Boric Acid Corrosion”	No
R-18	IV.C2.3-d IV.C2.4-d IV.C2.5-w IV.C2.5-t	Piping and components external surfaces and bolting	Stainless steel, Steel	System 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

IV	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM						
C2	Reactor Coolant System and Connected Lines (PWR)						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
R-04	IV.C2.2-b IV.C2.5-d IV.C2.5-a IV.C2.4-a IV.C2.3-a IV.C2.2-c IV.C2.1-b IV.C2.1-a IV.C2.2-a IV.C2.5-f IV.C2.5-e IV.C2.5-q	Piping, piping components, and piping elements	Steel, stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	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-19	IV.C2.5-v	Pressurizer Integral support	Stainless steel, Steel	Air with metal temperature up to 288°C (550°F)	Cracking/ cyclic loading	Chapter XI.M1, “ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD,” for Class 1 components	No
R-24	IV.C2.5-j	Pressurizer Spray head	Nickel alloy, cast austenitic stainless steel, stainless steel	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
C2 Reactor Coolant System and Connected Lines (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-58	IV.C2.5-c IV.C2.5-g	Pressurizer components	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-25	IV.C2.5-g IV.C2.5-c	Pressurizer components	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 (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-06	IV.C2.5-k IV.C2.5-s	Pressurizer instrumentation penetrations and heater sheaths and sleeves	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 and Chapter XI.M2, “Water Chemistry,” for PWR primary water in EPRI TR-105714 and plant specific AMP consistent with applicant commitments to NRC Bulletin BL-04-01 or any subsequent regulatory requirements.	Yes, plant specific
R-14	IV.C2.6-c	Pressurizer relief tank Tank shell and heads Flanges and nozzles	Stainless steel/ steel with stainless steel cladding	Treated borated water >60°C (>140°F)	Cracking/ 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-13	IV.C2.6-a	Pressurizer relief tank Tank shell and heads Flanges and nozzles Same as above	Steel with stainless steel cladding	Treated borated water	Cumulative fatigue damage/ fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation. See the Standard Review Plan, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii). See Chapter X.M1 of this report, for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes, TLAA

IV C2	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Coolant System and Connected Lines (PWR)						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
R-56	IV.C2.1-c	Reactor coolant system piping and fittings Cold leg Hot leg Surge line Spray line	Stainless steel/ 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
R-30	IV.C2.1-c	Reactor coolant system piping and fittings Cold leg Hot leg Surge line Spray line	Stainless steel/ 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

This Page Intentionally Left Blank

D1. STEAM GENERATOR (RECIRCULATING)

Systems, Structures, and Components

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

System Interfaces

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

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D1 Steam Generator (Recirculating)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-07	IV.D1.1-i	Class 1 piping, fittings and primary nozzles, safe ends, manways, and flanges	Stainless steel, steel with stainless steel or nickel-alloy cladding, nickel-alloy	Reactor coolant	Cracking/ 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-10	IV.D1.1-l	Closure bolting	Steel	Air with reactor coolant leakage	Cracking/ stress corrosion cracking	Chapter XI.M18, "Bolting Integrity"	No
R-17	IV.D1.1-g IV.D1.1-k	External surfaces	Steel	Air with borated water leakage	Loss of material/ boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No
R-01	IV.D1.1-j	Instrument penetrations and primary side nozzles	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 and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714 and plant specific AMP consistent with applicant commitments to NRC Orders, Bulletins and Generic Letters associated with nickel alloys.	Yes, plant specific

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D1 Steam Generator (Recirculating)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-04	IV.D1.1-h	Piping, piping components, and piping elements	Steel, stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	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-37	IV.D1.1-d	Pressure boundary and structural Steam nozzle and safe end FW nozzle and safe end	Steel	Secondary feedwater/steam	Wall thinning/ flow-accelerated corrosion	Chapter XI.M17, "Flow-Accelerated Corrosion"	No
R-32	IV.D1.1-f	Steam generator closure bolting	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
D1 Steam Generator (Recirculating)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-33	IV.D1.1-a IV.D1.1-b	Steam generator components	Steel	Secondary feedwater/steam	Cumulative fatigue damage/ fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3, "Metal Fatigue" for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes, TLAA
R-39	IV.D1.1-e	Steam generator feedwater impingement plate and support	Steel	Secondary feedwater	Loss of material/ erosion	A plant-specific aging management program is to be evaluated.	Yes, plant specific
R-34	IV.D1.1-c	Steam generator shell assembly (for OTSG), upper and lower shell, and transition cone (for recirculating steam generator)	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	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
R-40	IV.D1.2-i IV.D1.2-j	Tube plugs	Nickel alloy	Reactor coolant	Cracking/ 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	No
R-41	IV.D1.2-h	Tube support lattice bars	Steel	Secondary feedwater/steam	Loss of material/ flow-accelerated corrosion	Applicant must provide a commitment to submit, for NRC review and approval, an inspection plan for tube support lattice bars as based upon staff approved NEI 97-06 guidelines, or other alternative regulatory basis for steam generator degradation management, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed.
R-42	IV.D1.2-k	Tube support plates	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	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D1 Steam Generator (Recirculating)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-43	IV.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.	No
R-44	IV.D1.2-a	Tubes and sleeves	Nickel alloy	Reactor coolant	Cracking/ 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 .	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D1 Steam Generator (Recirculating)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-45	IV.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
R-48	IV.D1.2-c	Tubes and sleeves	Nickel alloy	Secondary feedwater/steam	Cracking/ intergranular attack	Chapter XI.M19, “Steam Generator Tubing Integrity” and Chapter XI.M2, “Water Chemistry,” for PWR secondary water in EPRI TR-102134	no
R-47	IV.D1.2-b	Tubes and sleeves	Nickel alloy	Secondary feedwater/steam	Cracking/ 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	no

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D1 Steam Generator (Recirculating)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-49	IV.D1.2-e	Tubes and sleeves	Nickel alloy	Secondary feedwater/steam	Loss of material/ fretting and wear	Chapter XI.M19, "Steam Generator Tubing Integrity" and Chapter XI.M2, "Water Chemistry," for PWR secondary water in EPRI TR-102134	No
R-50	IV.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	No
R-51	IV.D1.3-a	Upper assembly and separators Feedwater inlet ring and support	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

D2. STEAM GENERATOR (ONCE-THROUGH)

Systems, Structures, and Components

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

System Interfaces

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

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D2 Steam Generator (Once-Through)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-17	IV.D2.1-j IV.D2.1-b	External surfaces	Steel	Air with borated water leakage	Loss of material/ boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No
R-01	IV.D2.1-h	Instrument penetrations and primary side nozzles	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 and Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714 and plant specific AMP consistent with applicant commitments to NRC Orders, Bulletins and Generic Letters associated with nickel alloys.	Yes, plant specific
R-04	IV.D2.1-c	Piping, piping components, and piping elements	Steel, stainless steel, cast austenitic stainless steel, carbon steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	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 D2	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Steam Generator (Once-Through)						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
R-38	IV.D2.1-f	Pressure boundary and structural FW and AFW nozzles and safe ends Steam nozzles and safe ends	Steel	Secondary feedwater/steam	Wall thinning/ flow-accelerated corrosion	Chapter XI.M17, "Flow-Accelerated Corrosion"	No
R-31	IV.D2.1-l	Secondary manways and handholes (cover only)	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	IV.D2.1-k	Steam generator closure bolting	Steel	System temperature up to 340°C (644°F)	Loss of preload/ stress relaxation	Chapter XI.M18, "Bolting Integrity"	No
R-33	IV.D2.1-g IV.D2.1-d	Steam generator components	Steel	Secondary feedwater/steam	Cumulative fatigue damage/ fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3, "Metal Fatigue" for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes, TLAA

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D2 Steam Generator (Once-Through)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-36	IV.D2.1-i	Steam generator components Such as, secondary side nozzles (vent, drain, and instrumentation)	Nickel alloy	Secondary feedwater/steam	Cracking/ stress corrosion cracking	A plant-specific aging management program is to be evaluated.	Yes, plant specific
R-35	IV.D2.1-a	Steam generator components Upper and lower heads Tube sheets	Steel with stainless steel or nickel-alloy 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
D2 Steam Generator (Once-Through)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-34	IV.D2.1-e	Steam generator shell assembly (for OTSG), upper and lower shell, and transition cone (for recirculating steam generator)	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	IV.D2.2-f IV.D2.2-g	Tube plugs	Nickel alloy	Reactor coolant	Cracking/ 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	No
R-44	IV.D2.2-a	Tubes and sleeves	Nickel alloy	Reactor coolant	Cracking/ 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 .	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D2 Steam Generator (Once-Through)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
R-46	IV.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-48	IV.D2.2-c	Tubes and sleeves	Nickel alloy	Secondary feedwater/steam	Cracking/ intergranular attack	Chapter XI.M19, “Steam Generator Tubing Integrity” and Chapter XI.M2, “Water Chemistry,” for PWR secondary water in EPRI TR-102134	no
R-47	IV.D2.2-b	Tubes and sleeves	Nickel alloy	Secondary feedwater/steam	Cracking/ 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	no
R-49	IV.D2.2-d	Tubes and sleeves	Nickel alloy	Secondary feedwater/steam	Loss of material/ fretting and wear	Chapter XI.M19, “Steam Generator Tubing Integrity” and Chapter XI.M2, “Water Chemistry,” for PWR secondary water in EPRI TR-102134	No

F. COMMON MISCELLANEOUS MATERIAL ENVIRONMENT COMBINATIONS

Systems, Structures, and Components

This section includes the aging management programs for miscellaneous material environment combinations which may be found throughout the reactor vessel, internals and reactor coolant system's structures and components. For the material-environment combinations in this part, there are no aging effects which are expected to degrade the ability of the structure or component from performing its intended function for the extended period of operation, and, therefore, no resulting aging management programs for these structures and components are required.

System Interfaces

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

IV E	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Common Miscellaneous Material Environment Combinations						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
RP-02	RP-02	Piping, piping components, and piping elements	Cast austenitic stainless steel	Air – indoor uncontrolled (External)	None	None	No
RP-03	RP-03	Piping, piping components, and piping elements	Nickel alloy	Air – indoor uncontrolled (External)	None	None	No
RP-04	RP-04	Piping, piping components, and piping elements	Stainless steel	Air – indoor uncontrolled (External)	None	None	No
RP-05	RP-05	Piping, piping components, and piping elements	Stainless steel	Air with borated water leakage	None	None	No
RP-06	RP-06	Piping, piping components, and piping elements	Stainless steel	Concrete	None	None	No
RP-07	RP-07	Piping, piping components, and piping elements	Stainless steel	Gas	None	None	No
RP-08	RP-08	Piping, piping components, and piping elements	Stainless steel	Treated borated water	None	None	No

IV	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM						
E	Common Miscellaneous Material Environment Combinations						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
RP-01	RP-01	Piping, piping components, and piping elements	Steel	Concrete	None	None	No

End of Chapter IV

This Page Intentionally Left Blank