

MRP Materials Reliability Program _____ **MRP 2009-090**

December 2, 2009

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U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Draft Material to Support NRC Update of NUREG 1801, "Generic Aging Lessons Learned Report" (GALL):

- 1) *EPRI DRAFT Input(12-01-09): GALL Chapter XI.M16*
- 2) *EPRI DRAFT Input(12-01-09): GALL Table IV.B2 (W)*
- 3) *EPRI DRAFT Input(12-01-09): GALL Table IV.B3 (CE)*
- 4) *EPRI DRAFT Input(12-01-09): GALL Table IV.B4 (B&W)*
- 5) *EPRI DRAFT Input (12-01-09): New Appendix A to MRP-227*

References:

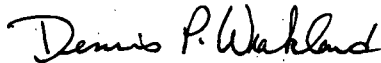
- 1) MRP Letter 2009-017, Christian B. Larsen (EPRI) to Chief Financial Officer, U.S. Nuclear Regulatory Commission, Dated February 6, 2009, Request for Exemption of NRC Review Fees
- 2) NRC Letter, J. E. Dyer (NRC) to Christian B. Larsen (EPRI), Dated February 20, 2009, Granting of Review Fee Exemption Request

To Whom It May Concern:

As provided for in References 1 and 2, one copy of each of the five draft documents listed above is being forwarded. This material is being provided as a means of exchanging information for the purpose of supporting the NRC's generic regulatory improvements related to methodologies for demonstrating pressurized water reactor internals integrity throughout the life of the plant. Specifically the staff will be using the material in updating NUREG 1801, "Generic Aging Lessons Learned Report" (GALL).

EPRI MRP works in conjunction with NEI to manage the reactor internals project for the PWR industry. If you have any questions on this project, please contact Anne Demma at EPRI (ademma@epri.com, 650-855-2026) or Victoria Anderson at NEI (vka@nei.org, 202-739-8101).

Sincerely,



Dennis P. Weakland

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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
IV.B3-1 (R-153)	IV.B3.2-e	CEA shroud assemblies (cast austenitic stainless steel items) CEA shrouds CEA shroud bases Modified CEA shroud	Cast austenitic stainless steel	Reactor coolant >250°C (>482°F) and neutron flux	Loss of fracture toughness/thermal aging embrittlement, neutron irradiation embrittlement	Chapter XI.M13, "Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS);" and Chapter XI.M16, "PWR Vessel Internals," No Additional Measures Components.	No.

EPRI DRAFT Input for GALLUP (1-09)

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
IV.B3-2 (R-149)	IV.B3.2-a	CEA shroud assemblies (wrought austenitic stainless steel items)	Stainless steel; nickel alloy	Reactor coolant	Cracking/stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XLM2, "Water Chemistry," for PWR primary water, and Chapter XLM16, "PWR Vessel Internals." Primary Components: Instrument guide tubes in the peripheral CEA shroud assemblies Expansion Components: Remaining instrument guide tubes, depending on the results of examinations of tubes in peripheral assemblies No Additional Measures: Welds in CEA shrouds, CEA shroud bases, CEA shroud extension guides, CEA shroud tie rods and nuts, internal/external spanner nuts, and snubber blocks	No.
IV.B3-5 (R-150)	IV.B3.2-b	CEA shrouds CEA shroud bases CEA shroud extension shaft guides Instrument guide tubes Internal/external spanner nuts CEA shroud bolts (screened out for SCC, IASCC, and PWSCC) CEA shroud tie rods			Cracking/stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking		

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
IV.B3-3 (R-152)	IV.B3.2-d	CEA shroud assembly CEA shroud extension shaft guides CEA shroud bolts Snubber shims	Stainless steel	Reactor coolant	Loss of material/wear	Chapter XLM16, "PWR Vessel Internals," No Additional Measures Components.	No.
IV.B3-4 (R-151)	IV.B3.2-c	CEA shroud assembly CEA shroud bolts	Stainless steel; nickel alloy	Reactor coolant	Changes in dimension/void swelling	Chapter XLM16, "PWR Vessel Internals," No Additional Measures Components.	No.
IV.B3-6 (R-154)	IV.B3.2-g	CEA shroud assembly CEA shroud bolts	Stainless steel; nickel alloy	Reactor coolant	Loss of preload/stress relaxation	Chapter XLM16, "PWR Vessel Internals," No Additional Measures Components.	No.

EPRI DRAFT REPORT FOR CALL UPDATE (60)

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
IV.B3-7 (R-165)	IV.B3.4-h	Core shroud assembly (for bolted core shroud assemblies) Core shroud bolts Barrel-shroud bolts Core shroud tie rods and nuts Guide lug insert	Stainless steel; nickel alloy	Reactor coolant	Loss of preload/stress relaxation	Chapter XI.M16, "PWR Vessel Internals." Primary Components: Core shroud bolts for cracking from IASCC and fatigue, or loss of preload/stress relaxation that eventually leads to cracking Expansion Components: Barrel-shroud bolts, depending on results of core shroud bolt examinations No Additional Measures: Core shroud tie rods and nuts, and guide lug insert bolts	No.
IV.B3-8 (R-163) IV.B3-13 (R-160)	IV.B3.4-f IV.B3.4-b	Core shroud assembly (for bolted core shroud assemblies) Shroud plates Former plates Ribs and rings Core shroud bolts	Stainless steel; nickel alloy	Reactor coolant	Changes in dimension/void swelling	Chapter XI.M16, "PWR Vessel Internals." Primary Components: Core side surfaces of shroud plates in plants with bolted core shroud assemblies No Additional Measures: Former plates, ribs and rings, core shroud bolts, and core shroud tie rods in plants with bolted core shroud assemblies for changes in dimension/void swelling	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
IV.B3-9 (R-162)	IV.B3.4-e	Core shroud assembly (for bolted core shroud assemblies)	Stainless steel	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, and Chapter XI.M16, "PWR Vessel Internals."	No.
IV.B3-11 (R-159)	IV.B3.4-a						

EPRI DRAFT Input for GALL Update (1-09)

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
IV.B3-10 (R-164)	IV.B3.4-g	Core shroud assembly (for bolted core shroud assemblies)	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness/neutron irradiation embrittlement, void swelling	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures Components.	No.
IV.B3-12 (R-161)	IV.B3.4-c	Shroud plates Former plates Ribs and rings Core shroud bolts Barrel-shroud bolts Core shroud tie					

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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
IV.B3-tt		<p>Core shroud assembly (all plants)</p> <p>Core shroud bolts</p> <p>Barrel-shroud bolts</p> <p>Core shroud tie rods and nuts</p> <p>Guide lug inserts</p>	Stainless steel	Reactor coolant	Loss of material/wear	<p>Chapter XI.M16, "PWR Vessel Internals."</p> <p>Existing Programs Components: Guide lug inserts and bolts</p> <p>No Additional Measures: Core shroud bolts, barrel-shroud bolts, and core shroud tie rods and nuts</p>	No.
IV.B3-xx		<p>Core shroud assembly (for welded core shrouds in two vertical sections)</p> <p>Shroud plate-to-former plate weld</p> <p>Remaining axial welds</p>	Stainless steel	Reactor coolant	Cracking/stress corrosion cracking, irradiation-assisted stress corrosion cracking	<p>Chapter XI.M2, "Water Chemistry," for PWR primary water, and Chapter XI.M16, "PWR Vessel Internals."</p> <p>Primary Components: Axial and horizontal weld seams at the core shroud re-entrant corners as visible from the core side of the shroud, within six inches of the central flange and the horizontal stiffeners</p> <p>Expansion Components: Remaining axial welds, depending on the results of the shroud plate-to-former plate weld examinations</p>	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
IV.B3-yy		<p>Core shroud assembly (for welded core shrouds with full-height shroud plates)</p> <p>Shroud plate-to-shroud plate welds</p> <p>Remaining axial welds, ribs, and rings</p>	Stainless steel	Reactor coolant	Cracking/stress corrosion cracking, irradiation-assisted stress corrosion cracking	<p>Chapter XLM2, "Water Chemistry," for PWR primary water, and Chapter XLM16, "PWR Vessel Internals."</p> <p>Primary Components: Enhanced visual (EVT-1) examination of the axial weld seams at the core shroud re-entrant corners, at the core mid-plane (\pm three feet in height) as visible from the core side of the shroud, is required.</p> <p>Expansion Components: Enhanced visual (EVT-1) examination of the remaining axial welds, ribs, and rings is required, depending on the results of the shroud plate-to-former plate weld.</p>	No.
IV.B3-zz		<p>Core shroud assembly (for welded core shrouds in two vertical sections)</p> <p>Upper and lower plate joint</p>	Stainless steel	Reactor coolant	Change in dimension/void swelling	<p>Chapter XLM16, "PWR Vessel Internals," Primary Components.</p> <p>Primary Component: Gap between the upper and lower plates</p>	No.
IV.B3-14 (R-158)	IV.B3.3-b	<p>Core support barrel assembly</p> <p>Upper core barrel</p>	Stainless steel	Reactor coolant	Changes in dimension/void swelling	Chapter XLM16, "PWR Vessel Internals," No Additional Measures Components.	No.

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

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Item	Link	Structure and/or Component	Material	Environment	Aging Effect/Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B3-15 (R-155)	IV.B3.3-a	Core support barrel assembly Upper cylinder Lower cylinder Upper core barrel flange Lower core barrel flange Core barrel snubber lugs Core barrel outlet nozzles	Stainless steel	Reactor coolant	Cracking/stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water, and Chapter XI.M16, "PWR Vessel Internals." Primary Components: Accessible surfaces of the upper core barrel flange weld Expansion Components: Accessible surfaces of the lower core barrel flange weld, with potential expansion to include the remaining core barrel assembly welds, depending on the results of the upper core barrel flange weld examinations No Additional Measures: All remaining core support barrel assembly components	No.
IV.B3-16 (R-157)	IV.B3.4-a	Core support barrel assembly Lower cylinder Upper core barrel flange	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/neutron irradiation embrittlement, void swelling	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures Components.	No.

EPRI DRAFT Input for GALL Update

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
IV.B3-17 (R-156)	IV.B3.3-b	Core support barrel assembly Upper core barrel flange Core barrel snubber lugs Alignment keys Core barrel outlet nozzles Thermal shield positioning	Stainless steel	Reactor coolant	Loss of material/wear	Chapter XI.M16, "PWR vessel Internals." Existing Programs Components: Upper core barrel flange No Additional Measures: Core barrel snubber lugs, alignment keys, core barrel outlet nozzles, and thermal shield positioning pins	No.
IV.B3-tt		Core support barrel assembly Lower flange weld	Stainless steel	Reactor coolant	Cracking/fatigue	Chapter XI.M16, "PWR Vessel Internals," Primary Components. Primary Component: Lower flange weld, if fatigue life cannot be demonstrated by time limited aging analysis (TLAA)	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
IV.B3-18 (R-171)	IV.B3.5-f	Lower support structure (components fabricated from cast austenitic stainless steel) Core support columns	Cast austenitic stainless steel	Reactor coolant >250°C (>482°F) and neutron flux	Loss of fracture toughness/ thermal aging embrittlement, neutron irradiation embrittlement	Chapter XI.M13, "Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS);" and Chapter XI.M16, "PWR Vessel Internals," No Additional Measures Components.	No.

EPRI DRAFT Input for GALL UP

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
IV.B3-19 (R-168)	IV.B3.5-c	Lower support structure	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Changes in dimension/void swelling	Chapter XI.M16, "PWR vessel Internals," No Additional Measures. No Additional Measures: All of the lower support structure components for changes in dimension/void swelling and for loss of fracture toughness/neutron irradiation embrittlement	No.
IV.B3-20 (R-169)	IV.B3.5-d	Core support plate			Loss of fracture toughness/ neutron irradiation embrittlement, void swelling		
		Core support plate bolts and dowel pins					
		Anchor block bolts and dowel pins					
		Fuel alignment pins					
		Core support columns					
		Core support deep beams					
		Core support column bolts					
		Core support barrel snubber assemblies					

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
IV.B3-21 (R-166)	IV.B3.5-a	Lower support structure	Stainless steel; nickel alloy	Reactor coolant	Cracking/stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water, and Chapter XI.M16, "PWR Vessel Internals." Expansion Components: Core support column bolts for bolted core shroud designs, depending on the results of core shroud bolt examinations; core support column welds for all plants except those with core shrouds assembled with full-height shroud plates, depending on the results core support barrel upper flange weld examinations Existing Program Components: A286 fuel alignment pins (all plants with core shrouds assembled with full-height shroud plates) No Additional Measures: Welds in core support plates, core support beams, core support deep beams, and the bottom plate for SCC and IASCC	No.
IV.B3-23 (R-167)	IV.B3.5-b	Core support plate welds Core support plate bolts A286 fuel alignment pins Core support column welds Core support column bolts Core support beam welds Core support deep beams Bottom plate welds			Cracking/stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking		

EPRI DRAFT Input for (99) Update

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
IV.B3-22 (R-170)	IV.B3.5-e	Lower support structure Core support plates and bolts Anchor block bolts A286 fuel Alignment pins Core support column bolts	Stainless steel; nickel alloy	Reactor coolant	Loss of material/wear	Chapter XI.M16, "PWR Vessel Internals." Existing Programs Components: A286 fuel alignment pins (all plants with core shrouds assembled in two vertical sections) No Additional Measures: Core support plates and bolts, anchor block bolts, and core support column bolts for loss of material/wear	No.
IV.B3-ss (R-sss)		Lower support structure	Stainless steel	Reactor coolant	Cracking/fatigue	Chapter XI.M16, "PWR Vessel Internals," Primary Components. Primary Component: Beam-to-beam lower support structure deep beams (all plants with core shrouds assembled with full-height shroud plates), at the axial elevation of the beam top surface down to four inches below the top surface, unless adequacy of remaining fatigue life can be demonstrated	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
IV.B3-24 (R-53)	IV.B3.4-d IV.B3.5-g IV.B3.2-f	Reactor vessel internals components	Stainless steel; nickel alloy	Reactor coolant	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.B3-25 (R-24)	IV.B3.	Reactor vessel internals components	Stainless steel; nickel alloy	Reactor coolant	Loss of material/pitting, crevice corrosion	Chapter XIM2, "Water Chemistry," for PWR primary water.	No.

EPRI DRAFT Input for GALL UP

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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
IV.B3-26 (R-148)	IV.B3.1-c	Upper internals assembly Upper guide structure support flange – upper Flange blocks RVLMS support structure tubes Fuel alignment plate Fuel bundle guide pins and nuts Hold-down ring	Stainless steel	Reactor coolant	Loss of material/wear	Chapter XLM16, "PWR Vessel Internals," No Additional Measures Components.	No.

EPRI DRAFT Input for GALL Update (601-09)

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
IV.B3-27 (R-147)	IV.B3.1-b	Upper internals assembly Upper guide structure support flange - upper Fuel alignment plate Fuel bundle guide pins and	Stainless steel	Reactor coolant	Changes in dimension/void swelling	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures Components.	No.
IV.B3-rr (R-rrr)		Upper internals assembly Fuel alignment plate	Stainless steel	Reactor coolant	Cracking/fatigue	Chapter XI.M16, "PWR Vessel Internals," Primary Component. Primary Component: Fuel alignment plate, if fatigue life cannot be demonstrated by time limited aging analysis (TLAA)	No.

EPRI DRAFT Input for GALL Update (1-09)

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
IV.B3-28 (R-146)	IV.B3.1-a	Upper internals assembly Upper guide structure support flange - upper Fuel alignment plate Fuel bundle guide pins and nuts	Stainless steel	Reactor coolant	Cracking/stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water, and Chapter XI.M16, "PWR Vessel Internals," No Additional Measures Components.	No.

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EPRI DRAFT Input for GALL Update

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
IV.B2-1 (R-124)	IV.B2.4-b	Baffle/former assembly Baffle and former plates	Stainless steel	Reactor coolant	Changes in dimensions/void swelling	Chapter XI.M16, "PWR Vessel Internals," Primary Components.	No.
IV.B2-2 (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, and Chapter XI.M16, "PWR Vessel Internals," No Additional Measures Components.	No.
IV.B2-3 (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	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures Components.	No.
IV.B2-4 (R-126)	IV.B2.4-d	Baffle/former assembly	Stainless steel	Reactor coolant	Changes in dimensions/void swelling	Chapter XI.M16, "PWR Vessel Internals," Primary Components.	No.
IV.B2-5 (R-129)	IV.B2.4-h	Baffle/former bolts			Loss of preload/stress relaxation		

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
IV.B2-6 (R-128)	IV.B2.4-f	Core barrel assembly	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/neutron irradiation embrittlement, void swelling Cracking/stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water, and Chapter XI.M16, "PWR Vessel Internals," Primary Components.	No.
IV.B2-10 (R-125)	IV.B2.4-c	Baffle/former assembly Baffle/former bolts and screws					
IV.B2-7 (R-121)	IV.B2.3-b	Core barrel	Stainless steel	Reactor coolant	Changes in dimensions/void swelling Loss of fracture toughness/neutron irradiation embrittlement, void swelling	Chapter XI.M16, "PWR Vessel Internals." Expansion Components: Core barrel axial welds for cracking caused by SCC, IASCC, or neutron irradiation embrittlement. All components were screened out for changes in dimensions caused by void swelling, and all other components for loss of fracture toughness caused by neutron irradiation embrittlement.	No.
IV.B2-9 (R-122)	IV.B2.3-c	Core barrel upper flange Core barrel outlet nozzles Thermal shield					

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

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Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2-8 (R-120)	IV.B2.3-a	Core barrel Core barrel upper flange Core barrel outlet nozzles Thermal shield	Stainless steel	Reactor coolant	Cracking/stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XLM2, "Water Chemistry," for PWR primary water, and Chapter XLM16, "PWR Vessel Internals." Primary Components: Core barrel upper flange weld Expansion Components: Core barrel outlet nozzles and core barrel axial welds are linked to the examination results for the core barrel upper flange weld Existing Programs Component: Core barrel flange No Additional Measures: Thermal shield	No.

EPRI DRAFT Input for GALL Update

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
IV.B2-11 (R-144)	IV.B2.6-b	Instrumentation support structures	Stainless steel	Reactor coolant	Changes in dimensions/void swelling	Chapter XI.M2, "Water Chemistry," for PWR primary water; Chapter XI.M37, "Flux Thimble Tube Inspection;" and Chapter XI.M16, "PWR Vessel Internals."	No.
IV.B2-12 (R-143)	IV.B2.6-a	Flux thimble guide tubes			Cracking/stress corrosion cracking, irradiation-assisted stress corrosion cracking	Existing Program Component: Flux thimble guide tubes for wear	
IV.B2-13 (R-145)	IV.B2.6-c				Loss of material/wear	No Additional Measures Component: Flux thimble guide tubes for all other aging effects/mechanisms	
IV.B2-14 (R137)	IV.B2.5i	Lower internal assembly Clevis insert bolts	Stainless steel; nickel alloy	Reactor coolant	Loss of preload/stress relaxation	Chapter XI.M16, "PWR Vessel Internals," Existing Programs Component.	No.

1-09)

EPRI DRAFT Input for GALE Update

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

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Item	Link	Structure and/or Component	Material	Environment	Aging Effect/Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2-15 (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	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures Components.	No.
IV.B2-16 (R-133)	IV.B2.5-e	Lower internal assembly Fuel alignment pins Lower support plate column bolts	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking/stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water, and Chapter XI.M16, "PWR Vessel Internals." Expansion Components: Lower support plate column bolts are linked to baffle-former bolt examinations, dependent on results of the baffle-former bolt examinations	No.
IV.B2-17 (R-135)	IV.B2.5-g	Clevis insert bolts			Loss of fracture toughness/neutron irradiation embrittlement, void swelling	No Additional Measures: Fuel alignment pins and clevis insert bolts for these aging effects/mechanisms.	

EPRI DRAFT INPUT FOR GALL Update

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
IV.B2-18 (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	Chapter XI.M16, "PWR Vessel Internals," Existing Programs Components.	No.
IV.B2-19 (R-131)	IV.B2.5-b	Lower internal assembly Lower core plate	Stainless steel; nickel alloy	Reactor coolant	Changes in dimensions/void swelling	Chapter XI.M2, "Water Chemistry," for PWR primary water, and Chapter XI.M16, "PWR Vessel Internals."	No.
IV.B2-20 (R-130)	IV.B2.5-a	Radial keys and clevis inserts			Cracking/stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	Existing Programs Components: Lower core plate and extra-long (XL) lower core plate for cracking No Additional Measures: Lower core plate, XL lower core plate, and radial keys and clevis inserts for changes in dimensions caused by void swelling No Additional Measures: Radial keys and clevis inserts for cracking caused by SCC, PWSCC, or IASCC	

EPRI DRAFT Input for CALL Update (1-09)

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
IV.B2-21 (R-140)	IV.B2.5-m	Lower internal assembly	Cast austenitic stainless steel; stainless steel forgings	Reactor coolant >250°C (>482°F) and neutron flux	Loss of fracture toughness/thermal aging and neutron irradiation embrittlement	Chapter XLM2, "Water Chemistry," for PWR primary water; Chapter XLM13, "Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS);" and Chapter XLM16, "PWR Vessel Internals." Expansion Components: Both cast and forged lower support column bodies are linked to either the control rod guide tube (CRGT) lower flanges (for the castings) or the upper core barrel flange weld (for the forgings), depending on results of the CRGT lower flange examinations	No.
IV.B2-22 (R-141)	IV.B2.5-n	Lower support casting			Loss of fracture toughness/neutron irradiation embrittlement, void swelling		
IV.B2-24 (R-138)	IV.B2.5-k	Lower support forging			Cracking/stress corrosion cracking, irradiation-assisted stress corrosion cracking		
IV.B2-23 (R-139)	IV.B2.5-l	Lower internal assembly	Stainless steel	Reactor coolant	Changes in dimensions/void swelling	Chapter XLM16, "PWR Vessel Internals," No Additional Measures.	No.
		Lower support forging or casting					
		Lower support plate columns					

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
IV.B2-25 (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	Chapter XI.M16, "PWR Vessel Internals," Expansion Components. Expansion Components: Lower support plate column bolts are linked to baffle-former bolts, depending on results of the baffle-former bolt examinations.	No.
IV.B2-26 (R-142)	IV.B2.5-o	Lower internal assembly Radial keys and clevis inserts	Stainless steel	Reactor coolant	Loss of material/ wear	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures Components.	No.
IV.B2-27 (R-119)	IV.B2.2-e	Rod control cluster assemblies (RCCA) or control rod guide tube (CRGT) assemblies RCCA guide tube bolts RCCA guide tube support pins	Stainless steel; nickel alloy	Reactor coolant	Changes in dimensions/void swelling	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures 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
IV.B2-28 (R-118)	IV.B2.2-d	Rod control cluster assemblies (RCCA) or control rod guide tube (CRGT) 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, and Chapter XI.M16, "PWR Vessel Internals," Existing Programs Components.	No.
IV.B2-29 (R-117)	IV.B2.2-b	Rod control cluster assemblies (RCCA) or control rod guide tube (CRGT) assemblies	Stainless steel	Reactor coolant	Changes in dimensions/void swelling	Chapter XI.M2, "Water Chemistry," for PWR primary water, and Chapter XI.M16, "PWR Vessel Internals."	No.
IV.B2-30 (R-116)	IV.B2.2a	RCCA guide tubes			Cracking/stress corrosion cracking, irradiation-assisted stress corrosion cracking	Primary Components: CRGT guide plates (cards) for wear No Additional Measures: CRGT guide tubes, including guide plates (cards), C-tubes, and sheaths, for changes in dimension caused by void swelling, and for cracking caused by SCC or IASCC	

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
IV.B2-31 (R-53)	IV.B2.1-m IV.B2.2-f IV.B2.1-c IV.B2.2-c IV.B2.3-d IV.B2.4-g IV.B2.5-p IV.B2.5-j IV.B2.5-d IV.B2.1-h	Reactor vessel internals components	Stainless steel; nickel alloy	Reactor coolant	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.B2-32 (R-24)	IV.B2.	Reactor vessel internals components	Stainless steel; nickel alloy	Reactor coolant	Loss of material/ pitting and crevice corrosion	Chapter XI.M2, "Water Chemistry," for PWR primary water.	No.
IV.B2-33 (R-108)	IV.B2.1-d	Upper internals assembly Hold-down spring	Stainless steel	Reactor coolant	Loss of preload/ stress relaxation	Chapter XI.M16, "PWR Vessel Internals," Primary Component.	No.
IV.B2-34 (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.M16, "PWR Vessel Internals," Existing Programs Component.	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
IV.B2-35 (R-110)	IV.B2.1-f	Upper internals assembly	Stainless steel	Reactor coolant	Changes in dimensions/void swelling	Chapter XLM2, "Water Chemistry," for PWR primary water, and Chapter XLM16, "PWR Vessel Internals," No Additional Measures Component.	No.
IV.B2-36 (R-109)	IV.B2.1-e	Upper support columns			Cracking/stress corrosion cracking, irradiation-assisted stress corrosion cracking	No Additional Measures: Upper support columns for changes in dimension caused by void swelling, and for cracking caused by SCC or IASCC.	
IV.B2-37 (R-111)	IV.B2.1-g	Upper internals assembly Upper support columns (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	Chapter XLM16, "PWR Vessel Internals," No Additional Measures Component.	No.
IV.B2-38 (R-114)	IV.B2.1-k	Upper internals assembly Upper support column bolts	Stainless steel; nickel alloy	Reactor coolant	Loss of preload/stress relaxation	Chapter XLM16, "PWR Vessel Internals," No Additional Measures Component.	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	
IV.B2-39 (R-113)	IV.B2.1-j	Upper internals assembly	Stainless steel; nickel alloy	Reactor coolant	Changes in dimensions/void swelling	Chapter XI.M2, "Water Chemistry," for PWR primary water, and Chapter XI.M16, "PWR Vessel Internals." Existing Programs Components: Upper core plate alignment pins for cracking	No.	
IV.B2-40 (R-112)	IV.B2.1-i	Upper support column bolts			Cracking/stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking			No Additional Measures: All components for changes in dimensions caused by void swelling, and upper support column bolts and the fuel alignment pins for cracking caused by SCC, PWSCC, and IASCC
		Upper core plate alignment pins Fuel alignment pins						
IV.B2-41 (R-107)	IV.B2.1-b	Upper internals assembly	Stainless steel	Reactor coolant	Changes in dimensions/void swelling	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures Components. No Additional Measures: Upper support plate, upper core plate, and internals hold down spring for changes in dimensions caused by void swelling, and for cracking caused by SCC and IASCC	No.	
IV.B2-42 (R-106)	IV.B2.1-a	Upper support Plate			Upper core Plate			Cracking/stress corrosion cracking, irradiation-assisted stress corrosion cracking
		Hold-down spring						

New Appendix A (to MRP-227): Operating Experience

While relatively few incidents of PWR internals aging degradation have been reported in operating U.S. commercial PWR plants, a summary of the current operating experience is useful for licensees developing aging management programs. This summary is organized by age-related degradation mechanism (effect). This compilation does not replace efforts by licensees to review and document plant-specific operating experience for impact on its program, or participate in industry initiatives that perform this function.

IASCC. A considerable amount of PWR internals IASCC has been observed in European PWRs, with emphasis on cracking of baffle-former bolting. Bolt failure rates during ultrasonic (UT) testing of baffle-former bolts in six French PWRs were found to range from 1.2% to 11%. For this reason, the U.S. PWR owners and operators began a program a decade ago to inspect the baffle-former bolting in order to determine whether similar problems might be expected in U.S. plants. A benefit of this decision was the experience gained with the UT examination techniques used in the inspections. In addition, the industry began substantial laboratory testing projects in order to gather the materials data necessary to support future inspections and evaluations. At one U.S. domestic reactor having type 347 bolting, all 728 baffle-former bolts were inspected by UT in 1998, with 55 bolts (7.5%) having indications that exceed the UT criteria. At another reactor, 639 out of the 728 baffle-former bolts were examined in 1999, with 59 bolts (9.2%) having indications failing to meet the UT acceptance criteria. On-site underwater mechanical testing at the first reactor of the removed baffle-former bolts indicated that the actual defective bolt rate was lower than suggested by the UT inspection. However, these known European or domestic baffle-former bolt IASCC indications are not necessarily applicable to all PWR designs. The incidents are generally associated with cold-worked Type 316 stainless steel or Type 347 stainless steel. Bolts fabricated from solution-annealed Type 304 stainless steel appear to be less affected.

IGSCC. In the early 1980's, inspections at several B&W units revealed that lower thermal shield bolts were missing. The majority (~80%) of the remaining bolts was loose, and several bolt locking cups were also missing. The bolts were fabricated from high-strength grade Alloy A-286 stainless steel. The failures were attributed to IGSCC at the bolt-head-to-bolt-shank transition. The replacement bolts were redesigned to reduce the tensile stress level in the bolt, and this was accomplished by redesigning the shank region, peening the surface of the bolt, and reducing the preload used to install the bolts. The material of construction was also changed from Alloy A-286 stainless steel to Alloy X-750.

In 1983, ultrasonic inspections at two B&W-designed units showed indications of cracking in a number of the upper core barrel bolts. The results were verified when the bolt heads became separated from the bolt shanks when the locking clips were removed. The bolts were fabricated from Alloy A-286. Failures were attributed to IGSCC and were not detected by visual examinations. The cracked bolts were replaced by bolts made of the same material, but manufactured by machining rather than a hot-heading operation. In addition, the torque applied to the replacement bolts during installation was significantly reduced.

In 2005, cracking of replacement core barrel bolts fabricated from cold-worked Type 316Ti were observed in a German PWR by visual inspection. These bolts had replaced the original Alloy X-750 core barrel bolts in the late 1980s, which had exhibited failure due to PWSCC. Subsequent UT inspection and failure analysis confirmed that the cracking was confined to the bolt head, initiating from the bolt fillet transition. The bolt threads and shank were free from cracking. The failure mechanism of the cold-worked Type 316Ti replacement core barrel bolts has been identified as IGSCC. So far, all known failures of core barrel bolts have been limited to the original Alloy X-750 and the replacement cold-worked Type 316 in German PWRs.

Wear. Wear of the in-core instrumentation thimble tubes was observed in the top part of the Zircaloy-4 thimble tubes at three PWR CE-designed Units. These tubes experienced through-wall tube degradation as a result of flow-induced vibration in the vicinity of the fuel alignment plate. This particular wear phenomenon was addressed by making modifications to the fuel alignment plate to alter the flow conditions in the vicinity of the entry point of the thimble tubes into the plate. Wear as a result of flow-induced vibration has not been observed in these components after implementing the modifications to the fuel alignment plate. Accordingly, this type of wear is not expected to challenge the integrity of these components in the future.

PWSCC. The few historical occurrences of SCC have been limited to components fabricated from specific age-hardenable alloys (i.e., Alloy X-750 and Alloy A-286) and fabrication-induced conditions. The SCC observed in the Alloy X-750 split pins used in Westinghouse-designed PWR internals has been attributed to the very susceptible AH heat treatment and high applied loads. Replacement split pins have been fabricated from Alloy X-750 in the HTH heat treat condition, which is very resistant to SCC in the PWR environment. SCC has occurred in Alloy A-286 internals bolting in B&W units. The Alloy A-286 bolt failures in B&W PWR internals were subjected to a comprehensive failure analysis. It was concluded that this material would exhibit SCC when the applied stress approaches its yield strength. Decreasing the initial stress levels on replacement Alloy A-286 bolts has removed this concern. Also, replacement bolts fabricated from Alloy X-750 HTH have been installed in the locations where significant bolting failures were detected.

Information Notice (IN) 90-68 provides information about SCC cracking in Alloy A-286 bolts used to hold the turning vanes to reactor coolant pumps at a foreign plant. The IN 90-68 includes a general discussion of the problems experienced with cracking of Alloy A-286 bolting materials, including the problems identified with respect to B&W PWR internals bolting. A review of Licensee Event Reports (LERs) identified several events of direct relevance to the cracking of high-strength bolting material.

Guide tube support pins supply lateral restraint to the bottom ends of the upper internals guide tubes, and were originally designed with Alloy X-750 material. However, these original equipment guide tube support pins were found to be susceptible to PWSCC. In the 1980s and 1990s, Westinghouse issued a letter to inform utilities of the emerging technical issue and also several customer status advisory reports on this topic. It should be recognized that cracked guide tube support pins do not challenge safe plant operation. Even when pins are cracked, the design of the guide tube and the geometry of the pins maintain control rod functionality. However, failure of guide tube support pins can result in a loose parts issue for the plant. After an extensive worldwide industry program to develop a material heat treatment for Alloy X-750 that would have maximum resistance to stress corrosion cracking, Westinghouse and utility customers conducted a campaign during the 1980s to replace guide tube support pins. Ultimately, Westinghouse developed a cold-worked Type 316 stainless steel support pin as a replacement and a number of utilities have performed replacements with this

design. A few utilities have opted to perform ultrasonic inspections rather than initiate wholesale replacements. Still other utilities have preferred to take no action at this time.

Not identified to Date. Visual examinations at one B&W designed reactor in 2005 indicated that three or four internal baffle-to-baffle bolts were found protruded. The bolt heads extended beyond the baffle plate surface. This was an indication that the locking devices, and potentially the bolts as well, had failed. A UT inspection of 100% of the baffle-former bolts was performed, with no detected indications of broken bolts. No UT inspection was performed on the internal baffle-to-baffle bolts. The suspect baffle-to-baffle bolts have yet to be removed to confirm failure and, if failed, the mechanism of failure. As a result of the observations, AREVA NP performed a plant-specific evaluation to assess the operational and safety functions for continued operation. That evaluation included thermal hydraulic evaluation, structural evaluation, fuel evaluation, and loose parts evaluation.

Irradiation-Induced Growth. Although irradiation-induced growth of zirconium alloys in CE plants was not explicitly identified in MRP-175 as an age-related degradation mechanism to be evaluated as part of the screening process, irradiation-induced growth in the axial direction of the in-core instrumentation thimble tubes has reduced the clearance between the thimble nose and the bottom of the fuel assembly. Some plants had observed that the thimble tube support plate was raised above its normal support position when the upper internals structure was replaced after fuel reload. This indicated that some of the thimbles had bottomed out in the fuel assemblies and were being loaded in compression. Ten plants affected by this issue have taken some action. Six of these plants have already replaced the thimble tube assemblies with modified designs that are shorter in length to accommodate the expected irradiation-induced growth in the future. Two additional plants have replacement designs in fabrication and have made preparations to install the replacement thimbles in an upcoming outage. The remaining two plants have not yet begun preparations for a full replacement of the thimble tubes, but one of these two has instead taken the intermediate step of raising the thimble support plate to accommodate additional axial growth. As a result of this intermediation, these plants are planning to execute a thimble assembly replacement program with a refueling outage in the future that is not currently encumbered with other large-scale replacements of major components. All affected plants will likely have replaced their thimble tubes prior to license extension.

EPRI DRAFT INPUT

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
IV.B4-1 (R-128)	IV.B4.5-i	<p>Core barrel assembly</p> <p>Baffle/former assembly</p> <p>Baffle/former bolts and screws</p> <p>Locking devices (including welds) of baffle/former bolts and internal baffle/baffle bolts</p> <p>Baffle-to-baffle bolts and their locking devices, including locking welds</p> <p>Core barrel-to-former bolts and their locking devices, including locking welds</p>	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/neutron irradiation embrittlement	<p>Chapter XI.M16, "PWR Vessel Internals."</p> <p>Primary Components: Accessible baffle/former bolts and screws; Accessible baffle-to-former and internal baffle-to-baffle bolt locking devices</p> <p>Expansion Components: Baffle-to-baffle bolts and core barrel-to-former bolts and their locking devices are inaccessible; justification for continued operation will depend on the examination results for baffle-to-former bolts and their locking devices, and on evaluation or replacement</p>	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
IV.B4-2 (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	Reactor coolant	Cracking/stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water, and Chapter XI.M16, "PWR Vessel Internals," No Additional Measures.	No.

EPRI DRAFT Input for GALL Update (60-09)

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
IV.B4-3 (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 and sectors	Stainless steel	Reactor coolant	Changes in dimension/void swelling	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures.	No.

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EPRI DRAFT Input for GALL Update

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
IV.B4-4 (R-183)	IV.B4.3-d	Control rod guide tube (CRGT) assembly CRGT spacer castings	Cast austenitic stainless steel	Reactor coolant >250°C (>482°F) and neutron flux	Loss of fracture toughness/thermal aging and neutron irradiation embrittlement	Chapter XLM13, "Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS);" and Chapter XLM16, "PWR Vessel Internals," Expansion Components. Expansion Components: Accessible surfaces at four screw locations (every 90°) for CRGT spacer castings, depending on examination results for the core support shield assembly cast outlet nozzles and vent valve discs	No.

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EPRI DRAFT Input for GALL Update

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
IV.B4-5 (R-181)	IV.B4.3-b	Control rod guide tube (CRGT) assembly CRGT spacer screws Flange-to-upper grid screws CRGT spacer screws Flange-to-upper grid screws CRGT rod guide tubes CRGT rod guide sectors	Stainless steel	Reactor coolant	Cracking/stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XLM2, "Water Chemistry," for PWR primary water, and Chapter XLM16, "PWR Vessel Internals," No Additional Measures Components.	No.

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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
IV.B4-6 (R-184)	IV.B4.3-e	Control rod guide tube (CRGT) assembly CRGT spacer screws Flange-to-upper grid screws	Stainless steel	Reactor coolant	Loss of preload/stress relaxation	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures.	No.

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EPRI DRAFT Input for GALL Update

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
IV.B4-7 (R-125)	IV.B4.5-g	<p>Core barrel assembly</p> <p>Baffle/former assembly</p> <p>Baffle/former bolts and screws</p> <p>Locking devices (including welds) of baffle/former bolts and internal baffle/baffle bolts</p> <p>Baffle-to-baffle bolts and their locking devices, including locking welds</p> <p>Core barrel-to-former bolts and their locking devices, including locking welds</p>	Stainless steel	Reactor coolant	Cracking/stress corrosion cracking, irradiation-assisted stress corrosion cracking	<p>Chapter XI.M2, "Water Chemistry," for PWR primary water, and Chapter XI.M16, "PWR Vessel Internals."</p> <p>Primary Components: Accessible baffle/former bolts and screws; accessible baffle-to-former and internal baffle-to-baffle bolt locking devices</p> <p>Expansion Components: Baffle-to-baffle bolts and core barrel-to-former bolts and their locking devices, including locking welds, are inaccessible; justification for continued operation will depend on the examination results for baffle-to-former bolts and their locking devices, and on evaluation or replacement</p>	No.

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EPR DRAFT Input for GALL Update

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
IV.B4-8 (R-199)	IV.B4.5-h	Core barrel assembly Baffle/former assembly Baffle/former bolts and screws Locking devices (including welds) of baffle/former bolts and internal baffle/baffle bolts Baffle-to-baffle bolts Core barrel-to-former bolts	Stainless steel	Reactor coolant	Changes in dimension/ void swelling	Chapter XI.M16, "PWR Vessel Internals." Primary Components: Accessible baffle/former bolts and screws; accessible baffle-to-former and internal baffle-to-baffle bolt locking devices Expansion Components: Baffle-to-baffle bolts and core barrel-to-former bolts are inaccessible; justification for continued operation will depend on the examination results for baffle-to-former bolts, and on evaluation or replacement	No.

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

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Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B4-9 (R-201)	IV.B4.5-j	Core barrel assembly Baffle/former assembly Baffle/former bolts and screws Locking devices (including welds) of baffle/former bolts and internal baffle/baffle bolts Baffle-to-baffle bolts Core barrel-to-former bolts	Stainless steel	Reactor coolant	Loss of preload/stress relaxation	Chapter XI.M16, "PWR Vessel Internals." Primary Components: Accessible baffle/former bolts and screws; accessible baffle-to-former and internal baffle-to-baffle bolt locking devices Expansion Components: Baffle-to-baffle bolts and core barrel-to-former bolts are inaccessible; justification for continued operation will depend on the examination results for baffle-to-former bolts, and on evaluation or replacement	No.

EPRI DRAFT INPUT for GALL Update

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
IV.B4-10 (R-193)	IV.B4.5-a	Core barrel assembly	Stainless steel; nickel alloy	Reactor coolant	Cracking/stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water, and Chapter XI.M16, "PWR Vessel Internals."	No.
IV.B4-13 (R-194)	IV.B4.5-b	Core barrel cylinder (top and bottom flange)					
		Lower core barrel (LCB) bolts and locking devices			Cracking/stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	Primary Components: Accessible lower core barrel bolts; accessible lower core barrel bolt locking devices	
		Core barrel-to-thermal shield bolts				Expansion Components: Accessible upper thermal shield (UTS) bolts and surveillance specimen holder tube (SSHT) bolts (Davis-Besse) or studs/nuts (Crystal River Unit 3), depending on the examination results for the upper core barrel and lower core barrel bolts	
		Surveillance specimen holder tube (SSHT) bolts or studs/nuts					
		Baffle plates					
		Former plates					

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EPRI DRAFT Input for C&I Update

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
IV.B4-11 (R-195)	IV.B4.5-c	<p>Core barrel assembly</p> <p>Core barrel cylinder (top and bottom flange)</p> <p>Lower core barrel (LCB) bolts</p> <p>Upper thermal shield (UTS) bolts</p> <p>Surveillance specimen holder tube (SSHT) bolts or studs/nuts</p> <p>Baffle plates</p> <p>Former plates</p>	Stainless steel; nickel alloy	Reactor coolant	Changes in dimension/void swelling	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures.	No.

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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
IV.B4-12 (R-196)	IV.B4.5-d	<p>Core barrel assembly</p> <p>Core barrel cylinder (top and bottom flange)</p> <p>Lower core barrel (LCB) bolts</p> <p>Core barrel-to-thermal shield bolts</p> <p>Baffle plates</p> <p>Former plates</p>	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness/ neutron irradiation embrittlement, void swelling	<p>Chapter XI.M16, "PWR Vessel Internals."</p> <p>Primary Components: Accessible surfaces within one inch around each baffle plate flow and bolt hole</p> <p>Expansion Components: The core barrel cylinder (including vertical and circumferential seam welds) and former plates are inaccessible, but are linked to the examination results for the baffle plates; justification for continued operation is by evaluation or by replacement</p>	No.

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EPRI DRAFT Input for CALL Update

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
IV.B4-14 (R-197)	IV.B4.5-e	<p>Core barrel assembly</p> <p>Core barrel cylinder (top and bottom flange)</p> <p>Lower core barrel (LCB) bolts</p> <p>Core barrel-to-thermal shield bolts</p> <p>Baffle plates</p> <p>Former plates</p>	Stainless steel; nickel alloy	Reactor coolant	Loss of preload/stress relaxation	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures.	No.
IV.B4-15 (R-190)	IV.B4.4-f	<p>Core support shield (CSS) assembly</p> <p>CSS cylinder (top and bottom flange)</p> <p>CSS vent valve assembly locking device</p>	Stainless steel	Reactor coolant	Loss of material/wear	<p>Chapter XI.M16, "PWR Vessel Internals."</p> <p>Primary Components: CSS cylinder top flange (differential height from the top of the plenum rib pads to the reactor vessel seating surface) for loss of material/wear</p> <p>No Additional Measures: CSS vent valve top and bottom retaining rings for loss of material/wear.</p>	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
IV.B4-16 (R-188)	IV.B4.4-d	Core support shield (CSS) assembly CSS cylinder (top and bottom flange) Upper core barrel (UCB) bolts Outlet and vent valve nozzles CSS vent valve assembly locking device	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness/neutron irradiation embrittlement, void swelling	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures.	No.

EPRI DRAFT INPUT for GALL Update (609)

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
IV.B4-17 (R-187)	IV.B4.4-c	Core support shield (CSS) assembly CSS cylinder (top and bottom flange) Upper core barrel (UCB) bolts Vent valve assembly retaining ring and locking device	Stainless steel; nickel alloy	Reactor coolant	Changes in dimension/void swelling	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures.	No.

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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
IV.B4-18 (R-185)	IV.B4.4-a	Core support shield (CSS) assembly	Stainless steel; nickel alloy	Reactor coolant	Cracking/stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water, and Chapter XI.M16, "PWR Vessel Internals."	No.
IV.B4-20 (R-186)	IV.B4.4-b	CSS cylinder (top and bottom flange)					
		Upper core barrel (UCB) bolts and their locking devices					
		Outlet and vent valve nozzles					
		Vent valve body and retaining ring			Cracking/stress corrosion cracking, primary water stress corrosion cracking irradiation-assisted stress corrosion cracking	Primary Components: Accessible upper core barrel bolts; accessible upper core barrel bolt locking devices No Additional Measures: All other CSS assembly components relative to SCC, IASCC, or PWSCC	
IV.B4-19 (R-192)	IV.B4.4-h	Core support shield (CSS) assembly	Stainless steel; nickel alloy	Reactor coolant	Loss of preload/stress relaxation	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures.	No.
		Upper core barrel (UCB) bolts					

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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
IV.B4-21 (R-191)	IV.B4.4-g	<p>Core support shield (CSS) assembly</p> <p>CSS cast outlet nozzles</p> <p>CSS vent valve discs</p> <p>CSS vent valve disc shaft or hinge pin</p> <p>CSS vent valve top and bottom retaining rings</p>	Cast austenitic stainless steel	Reactor coolant >250°C (>482°F) and neutron flux	Loss of fracture toughness/ thermal aging and neutron irradiation embrittlement	<p>Chapter XI.M13, "Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS);" and Chapter XI.M16, "PWR Vessel Internals," Primary Components.</p> <p>Primary Components: CSS cast outlet nozzles (Oconee Unit 3 and Davis-Besse), vent valve discs, vent valve disc shaft or hinge pin, and vent valve top and bottom retaining rings</p>	No.

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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
IV.B4-22 (R-209)	IV.B4.7-a	Flow distributor assembly	Stainless steel; nickel alloy	Reactor coolant	Cracking/stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water, and Chapter XI.M16, "PWR Vessel Internals," Expansion Components.	No.
IV.B4-25 (R-210)	IV.B4.7-b	Flow distributor head and flange Incore guide support plate Clamping ring Shell forging-to-flow distributor bolts			Cracking/stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking		

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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
IV.B4-23 (R-211)	IV.B4.7-c	Flow distributor assembly	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Changes in dimension/void swelling	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures. No Additional Measures: All of the components of the flow distributor assembly for the effects of changes in dimension/void swelling, and for the effects of loss of fracture toughness/neutron irradiation embrittlement, void swelling	No.
IV.B4-24 (R-212)	IV.B4.7-d	Flow distributor head and flange			Loss of fracture toughness/neutron irradiation embrittlement, void swelling		
		Incore guide support plate					
		Clamping ring					
		Shell forging-to-flow distributor bolts					
IV.B4-26 (R-213)	IV.B4.7-e	Flow distributor assembly	Stainless steel; nickel alloy	Reactor coolant	Loss of preload/stress relaxation	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures.	No.
		Clamping ring					
		Shell forging-to-flow distributor bolts					

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
IV.B4-27 (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.M16, "PWR Vessel Internals," No Additional Measures.	No.
IV.B4-28 (R-206)	IV.B4.6-e	Incore Monitoring Instrumentation (IMI) Guide Tube Assembly IMI Incore guide tube spider castings IMI guide tube spider-to-lower grid rib section welds	Cast austenitic stainless steel	Reactor coolant >250°C (>482°F) and neutron flux	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);" and Chapter XI.M16, "PWR Vessel Internals," Primary Components. Primary Components: Accessible top surfaces of 52 spider castings and welds to the adjacent lower grid rib section	No.

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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
IV.B4-29 (R-202)	IV.B4.6-a	Lower grid assembly Lower grid rib section Fuel assembly support pads Lower grid flow distributor plate Orifice plugs Lower grid and shell forgings Guide blocks Shock pads Support post pipes	Stainless steel	Reactor coolant	Cracking/stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water, and Chapter XI.M16, "PWR Vessel Internals," No Additional Measures. No Additional Measures: All components of the lower grid assembly for SCC and IASCC	No.

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

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Item	Link	Structure and/or Component	Material	Environment	Aging Effect/Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B4-30 (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 distributor 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	Changes in dimension/void swelling	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures. No Additional Measures: All components of the lower grid assembly for changes in dimension/void swelling	No.

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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
IV.B4-31 (R-205)	IV.B4.6-d	<p>Lower grid assembly</p> <p>Lower grid rib section</p> <p>Fuel assembly support pads</p> <p>Lower grid rib-to-shell forging screws</p> <p>Lower grid flow distributor plate</p> <p>Orifice plugs</p> <p>Lower grid and shell forgings</p> <p>Lower internals assembly-to-thermal shield bolts</p> <p>Guide blocks and bolts</p>	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness/neutron irradiation embrittlement, void swelling	<p>Chapter XI.M16, "PWR Vessel Internals."</p> <p>Expansion Components: Accessible pads, pad-to-rib section welds, Alloy X-750 dowels, cap screws, and their associated locking devices, depending on the results of the examination of the IMI guide tube spiders and the spider-to-lower grid rib section welds</p> <p>No Additional Measures: Lower grid rib-to-shell forging screws, lower grid and shell forgings, lower internals assembly-to-thermal shield bolts, guide blocks and bolts, the lower grid rib section, the lower grid distributor plate, the orifice plugs, the support post pipes, and shock pads and bolts for loss of fracture toughness/neutron irradiation embrittlement, void swelling</p>	No.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
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Item	Link	Structure and/or Component	Material	Environment	Aging Effect/Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B4-32 (R-203)	IV.B4.6-b	<p>Lower grid assembly</p> <p>Lower grid rib-to-shell forging screws</p> <p>Lower internals assembly-to-thermal shield bolts</p> <p>Guide block bolts</p> <p>Shock pad bolts</p>	Stainless steel; nickel alloy	Reactor coolant	Cracking/stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	<p>Chapter XI.M2, "Water Chemistry," for PWR primary water, and Chapter XI.M16, "PWR Vessel Internals."</p> <p>Primary Components: Accessible locking welds for 24 dowel-to-guide block welds</p> <p>Expansion Components: Accessible lower grid shock pad bolts at TMI-1 and the lower internals assembly-to-thermal shield bolts at all plants, depending on the results of the examinations of upper core barrel (UCB) and lower core barrel (LCB) bolts; Accessible Alloy X-750 dowel-to-lower fuel assembly support pad welds, depending on the results of the examinations of the Alloy X-750 dowel-to-guide block welds</p> <p>No Additional Measures: Lower grid rib-to-shell forging screws for cracking</p>	No.

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

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Item	Link	Structure and/or Component	Material	Environment	Aging Effect/Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B4-33 (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	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures.	No.
IV.B4-34 (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.M16, "PWR Vessel Internals," No Additional Measures. No Additional Measures: The top flange-to-cover bolts, the plenum cover assembly, the plenum cylinder, the reinforcing plates and the bottom flange-to-upper grid screws for both SCC and IASCC	No.
IV.B4-36 (R-173)	IV.B4.1-b	Top flange-to-cover bolts Bottom flange-to-upper grid screws			Cracking/stress corrosion cracking, irradiation-assisted stress corrosion cracking		

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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
IV.B4-35 (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 dimension/void swelling	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures.	No.
IV.B4-37 (R-53)	IV.B4.3-f IV.B4.5-f IV.B4.6-f IV.B4.2-d IV.B4.1-d IV.B4.4-e	Reactor vessel internals components	Stainless steel; nickel alloy	Reactor coolant	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.B4-38 (R-24)	IV.B4.	Reactor vessel internals components	Stainless steel; nickel alloy	Reactor coolant	Loss of material/pitting and crevice corrosion	Chapter XI.M2, "Water Chemistry," for PWR primary water.	No.

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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
IV.B-39 (R-215)	IV.B4.8-b	Core barrel assembly Thermal shield cylinder	Stainless steel	Reactor coolant	Changes in dimension/void swelling	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures.	No.
IV.B4-40 (R-214)	IV.B4.8-a	Core barrel assembly Thermal shield cylinder	Stainless steel	Reactor coolant	Cracking/stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water, and Chapter XI.M16, "PWR Vessel Internals," No Additional Measures.	No.
IV.B-41 (R-216)	IV.B4.8-c	Core barrel assembly Thermal shield cylinder	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/neutron irradiation embrittlement, void swelling	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures.	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
IV.B4-42 (R-179)	IV.B4.2-f	<p>Plenum cover and plenum cylinder assemblies</p> <p>Plenum rib pads (weldment rib pads)</p> <p>Support flange</p> <p>Lifting lug-to-base block bolts</p> <p>Top flange-to-cover bolts</p> <p>Bottom flange-to-upper grid bolts</p> <p>Upper grid assembly</p> <p>Fuel assembly support pad cap screws</p> <p>Rib-to-ring cap screws</p>	Stainless steel	Reactor coolant	Loss of material and associated loss of clamping load/wear	<p>Chapter XI.M16, "PWR Vessel Internals."</p> <p>Primary Components: Differential height between top of plenum rib pads and reactor vessel seating surface, with plenum in vessel, for wear</p> <p>Expansion Components: Accessible dowel locking welds for the Alloy X-750 dowel-to-upper fuel assembly support pad welds for all plants except Davis-Besse, depending on the results of the examination of the Alloy X-750 dowel-to-guide block welds</p> <p>No Additional Measures: Top flange-to-cover bolts, bottom flange-to-upper grid bolts, and rib-to-ring cap screws for loss of material/wear</p>	No.

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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
IV.B4-43 (R-176)	IV.B4.2-b	Plenum cover and plenum cylinder assemblies Plenum rib pads (Weldment rib pads)	Stainless steel	Reactor coolant	Cracking/stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water, and Chapter XI.M16, "PWR Vessel Internals," No Additional Measures. No Additional Measures: All upper grid assembly and plenum cover/plenum cylinder assembly components with respect to SCC and IASCC	No.
IV.B4-44 (R-175)	IV.B4.2-a	Upper grid assembly Rib section Ring forging Fuel assembly support pad cap screws Rib-to-ring cap screws			Cracking/stress corrosion cracking, irradiation-assisted stress corrosion cracking		

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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
IV.B4-45 (R-177)	IV.B4.2-c	Plenum cover and plenum cylinder assemblies	Stainless steel	Reactor coolant and neutron flux	Changes in dimension/void swelling	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures. No Additional Measures: All plenum cover, plenum cylinder assembly, and the upper grid assembly components for changes in dimension/void swelling and for loss of fracture toughness/neutron irradiation embrittlement	No.
IV.B4-46 (R-178)	IV.B4.2-e	Plenum rib pads (Weldment rib pads) Upper grid assembly Rib section Ring forging Fuel assembly support pads Rib-to-ring cap screws			Loss of fracture toughness/neutron irradiation embrittlement, void swelling		

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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
IV.B4-1 (R-128)	IV.B4.5-i	<p>Core barrel assembly</p> <p>Baffle/former assembly</p> <p>Baffle/former bolts and screws</p> <p>Locking devices (including welds) of baffle/former bolts and internal baffle/baffle bolts</p> <p>Baffle-to-baffle bolts and their locking devices, including locking welds</p> <p>Core barrel-to-former bolts and their locking devices, including locking welds</p>	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/neutron irradiation embrittlement	<p>Chapter XI.M16, "PWR Vessel Internals."</p> <p>Primary Components: Accessible baffle/former bolts and screws; Accessible baffle-to-former and internal baffle-to-baffle bolt locking devices</p> <p>Expansion Components: Baffle-to-baffle bolts and core barrel-to-former bolts and their locking devices are inaccessible; justification for continued operation will depend on the examination results for baffle-to-former bolts and their locking devices, and on evaluation or replacement</p>	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
IV.B4-2 (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	Reactor coolant	Cracking/stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water, and Chapter XI.M16, "PWR Vessel Internals," No Additional Measures.	No.

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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
IV.B4-3 (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 and sectors	Stainless steel	Reactor coolant	Changes in dimension/void swelling	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures.	No.

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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
IV.B4-4 (R-183)	IV.B4.3-d	Control rod guide tube (CRGT) assembly CRGT spacer castings	Cast austenitic stainless steel	Reactor coolant >250°C (>482°F) and neutron flux	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);” and Chapter XI.M16, “PWR Vessel Internals,” Expansion Components. Expansion Components: Accessible surfaces at four screw locations (every 90°) for CRGT spacer castings, depending on examination results for the core support shield assembly cast outlet nozzles and vent valve discs	No.

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B4 Reactor Vessel Internals (PWR) – Babcock & Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B4-5 (R-181)	IV.B4.3-b	Control rod guide tube (CRGT) assembly CRGT spacer screws Flange-to-upper grid screws CRGT spacer screws Flange-to-upper grid screws CRGT rod guide tubes CRGT rod guide sectors	Stainless steel	Reactor coolant	Cracking/stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XLM2, "Water Chemistry," for PWR primary water, and Chapter XLM16, "PWR Vessel Internals," No Additional Measures Components.	No.

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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
IV.B4-6 (R-184)	IV.B4.3-e	Control rod guide tube (CRGT) assembly CRGT spacer screws Flange-to-upper grid screws	Stainless steel	Reactor coolant	Loss of preload/stress relaxation	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures.	No.

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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
IV.B4-7 (R-125)	IV.B4.5-g	<p>Core barrel assembly</p> <p>Baffle/former assembly</p> <p>Baffle/former bolts and screws</p> <p>Locking devices (including welds) of baffle/former bolts and internal baffle/baffle bolts</p> <p>Baffle-to-baffle bolts and their locking devices, including locking welds</p> <p>Core barrel-to-former bolts and their locking devices, including locking welds</p>	Stainless steel	Reactor coolant	Cracking/stress corrosion cracking, irradiation-assisted stress corrosion cracking	<p>Chapter XLM2, "Water Chemistry," for PWR primary water, and Chapter XLM16, "PWR Vessel Internals."</p> <p>Primary Components: Accessible baffle/former bolts and screws; accessible baffle-to-former and internal baffle-to-baffle bolt locking devices</p> <p>Expansion Components: Baffle-to-baffle bolts and core barrel-to-former bolts and their locking devices, including locking welds, are inaccessible; justification for continued operation will depend on the examination results for baffle-to-former bolts and their locking devices, and on evaluation or replacement</p>	No.

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IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
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Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B4-8 (R-199)	IV.B4.5-h	<p>Core barrel assembly</p> <p>Baffle/former assembly</p> <p>Baffle/former bolts and screws</p> <p>Locking devices (including welds) of baffle/former bolts and internal baffle/baffle bolts</p> <p>Baffle-to-baffle bolts</p> <p>Core barrel-to-former bolts</p>	Stainless steel	Reactor coolant	Changes in dimension/ void swelling	<p>Chapter XI.M16, "PWR Vessel Internals."</p> <p>Primary Components: Accessible baffle/former bolts and screws; accessible baffle-to-former and internal baffle-to-baffle bolt locking devices</p> <p>Expansion Components: Baffle-to-baffle bolts and core barrel-to-former bolts are inaccessible; justification for continued operation will depend on the examination results for baffle-to-former bolts, and on evaluation or replacement</p>	No.

EPRI DRAFT Input for GALL Update

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
IV.B4-9 (R-201)	IV.B4.5-j	<p>Core barrel assembly</p> <p>Baffle/former assembly</p> <p>Baffle/former bolts and screws</p> <p>Locking devices (including welds) of baffle/former bolts and internal baffle/baffle bolts</p> <p>Baffle-to-baffle bolts</p> <p>Core barrel-to-former bolts</p>	Stainless steel	Reactor coolant	Loss of preload/stress relaxation	<p>Chapter XI.M16, "PWR Vessel Internals."</p> <p>Primary Components: Accessible baffle/former bolts and screws; accessible baffle-to-former and internal baffle-to-baffle bolt locking devices</p> <p>Expansion Components: Baffle-to-baffle bolts and core barrel-to-former bolts are inaccessible; justification for continued operation will depend on the examination results for baffle-to-former bolts, and on evaluation or replacement</p>	No.

EPRI DRAFT Input for GALL Update (60)

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

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Item	Link	Structure and/or Component	Material	Environment	Aging Effect/Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B4-10 (R-193)	IV.B4.5-a	Core barrel assembly	Stainless steel; nickel alloy	Reactor coolant	Cracking/stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water, and Chapter XI.M16, "PWR Vessel Internals."	No.
IV.B4-13 (R-194)	IV.B4.5-b	Core barrel cylinder (top and bottom flange)					
		Lower core barrel (LCB) bolts and locking devices			Cracking/stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	Primary Components: Accessible lower core barrel bolts; accessible lower core barrel bolt locking devices	
		Core barrel-to-thermal shield bolts				Expansion Components: Accessible upper thermal shield (UTS) bolts and surveillance specimen holder tube (SSHT) bolts (Davis-Besse) or studs/nuts (Crystal River Unit 3), depending on the examination results for the upper core barrel and lower core barrel bolts	
		Surveillance specimen holder tube (SSHT) bolts or studs/nuts					
		Baffle plates					
		Former plates					

EPRI DRAFT Input for C&I Update

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
IV.B4-11 (R-195)	IV.B4.5-c	Core barrel assembly Core barrel cylinder (top and bottom flange) Lower core barrel (LCB) bolts Upper thermal shield (UTS) bolts Surveillance specimen holder tube (SSHT) bolts or studs/nuts Baffle plates Former plates	Stainless steel; nickel alloy	Reactor coolant	Changes in dimension/void swelling	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures.	No.

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EPRI DRAFT Input for GALL Update

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
IV.B4-12 (R-196)	IV.B4.5-d	Core barrel assembly Core barrel cylinder (top and bottom flange) Lower core barrel (LCB) bolts Core barrel-to-thermal shield bolts Baffle plates Former plates	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness/ neutron irradiation embrittlement, void swelling	Chapter XI.M16, "PWR Vessel Internals." Primary Components: Accessible surfaces within one inch around each baffle plate flow and bolt hole Expansion Components: The core barrel cylinder (including vertical and circumferential seam welds) and former plates are inaccessible, but are linked to the examination results for the baffle plates; justification for continued operation is by evaluation or by replacement	No.

EPRI DRAFT Input for GALL Update (1-09)

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
IV.B4-14 (R-197)	IV.B4.5-e	<p>Core barrel assembly</p> <p>Core barrel cylinder (top and bottom flange)</p> <p>Lower core barrel (LCB) bolts</p> <p>Core barrel-to-thermal shield bolts</p> <p>Baffle plates</p> <p>Former plates</p>	Stainless steel; nickel alloy	Reactor coolant	Loss of preload/stress relaxation	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures.	No.
IV.B4-15 (R-190)	IV.B4.4-f	<p>Core support shield (CSS) assembly</p> <p>CSS cylinder (top and bottom flange)</p> <p>CSS vent valve assembly locking device</p>	Stainless steel	Reactor coolant	Loss of material/wear	<p>Chapter XI.M16, "PWR Vessel Internals."</p> <p>Primary Components: CSS cylinder top flange (differential height from the top of the plenum rib pads to the reactor vessel seating surface) for loss of material/wear</p> <p>No Additional Measures: CSS vent valve top and bottom retaining rings for loss of material/wear.</p>	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
IV.B4-16 (R-188)	IV.B4.4-d	Core support shield (CSS) assembly CSS cylinder (top and bottom flange) Upper core barrel (UCB) bolts Outlet and vent valve nozzles CSS vent valve assembly locking device	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness/neutron irradiation embrittlement, void swelling	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures.	No.

EPRI DRAFT Input for GALL Update (1-09)

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
IV.B4-17 (R-187)	IV.B4.4-c	Core support shield (CSS) assembly CSS cylinder (top and bottom flange) Upper core barrel (UCB) bolts Vent valve assembly retaining ring and locking device	Stainless steel; nickel alloy	Reactor coolant	Changes in dimension/void swelling	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures.	No.

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EPRI DRAFT Input for GALL Update

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
IV.B4-18 (R-185)	IV.B4.4-a	Core support shield (CSS) assembly	Stainless steel; nickel alloy	Reactor coolant	Cracking/stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water, and Chapter XI.M16, "PWR Vessel Internals."	No.
IV.B4-20 (R-186)	IV.B4.4-b	CSS cylinder (top and bottom flange)					
		Upper core barrel (UCB) bolts and their locking devices					
		Outlet and vent valve nozzles					
		Vent valve body and retaining ring			Cracking/stress corrosion cracking, primary water stress corrosion cracking irradiation-assisted stress corrosion cracking	Primary Components: Accessible upper core barrel bolts; accessible upper core barrel bolt locking devices No Additional Measures: All other CSS assembly components relative to SCC, IASCC, or PWSCC	
IV.B4-19 (R-192)	IV.B4.4-h	Core support shield (CSS) assembly	Stainless steel; nickel alloy	Reactor coolant	Loss of preload/stress relaxation	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures.	No.
		Upper core barrel (UCB) bolts					

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EPRI Draft Report for Client Update

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
IV.B4-21 (R-191)	IV.B4.4-g	<p>Core support shield (CSS) assembly</p> <p>CSS cast outlet nozzles</p> <p>CSS vent valve discs</p> <p>CSS vent valve disc shaft or hinge pin</p> <p>CSS vent valve top and bottom retaining rings</p>	Cast austenitic stainless steel	Reactor coolant >250°C (>482°F) and neutron flux	Loss of fracture toughness/ thermal aging and neutron irradiation embrittlement	<p>Chapter XI.M13, “Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS);” and Chapter XI.M16, “PWR Vessel Internals,” Primary Components.</p> <p>Primary Components: CSS cast outlet nozzles (Oconee Unit 3 and Davis-Besse), vent valve discs, vent valve disc shaft or hinge pin, and vent valve top and bottom retaining rings</p>	No.

EPRI DRAFT Input for GALL Update (60)

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

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Item	Link	Structure and/or Component	Material	Environment	Aging Effect/Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B4-22 (R-209)	IV.B4.7-a	Flow distributor assembly	Stainless steel; nickel alloy	Reactor coolant	Cracking/stress corrosion cracking, irradiation-assisted stress corrosion cracking.	Chapter XI.M2, "Water Chemistry," for PWR primary water, and Chapter XI.M16, "PWR Vessel Internals," Expansion Components.	No.
IV.B4-25 (R-210)	IV.B4.7-b	Incore guide support plate Clamping ring Shell forging-to-flow distributor bolts			Cracking/stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking		

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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
IV.B4-23 (R-211)	IV.B4.7-c	Flow distributor assembly	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Changes in dimension/void swelling	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures. No Additional Measures: All of the components of the flow distributor assembly for the effects of changes in dimension/void swelling, and for the effects of loss of fracture toughness/neutron irradiation embrittlement, void swelling	No.
IV.B4-24 (R-212)	IV.B4.7-d	Flow distributor head and flange			Loss of fracture toughness/neutron irradiation embrittlement, void swelling		
		Incore guide support plate					
		Clamping ring					
		Shell forging-to-flow distributor bolts					
IV.B4-26 (R-213)	IV.B4.7-e	Flow distributor assembly	Stainless steel; nickel alloy	Reactor coolant	Loss of preload/stress relaxation	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures.	No.
		Clamping ring					
		Shell forging-to-flow distributor bolts					

EPRI DRAFT INPUT FOR GAIN Update (1-09)

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
IV.B4-27 (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.M16, "PWR Vessel Internals," No Additional Measures.	No.
IV.B4-28 (R-206)	IV.B4.6-e	Incore Monitoring Instrumentation (IMI) Guide Tube Assembly IMI Incore guide tube spider castings IMI guide tube spider-to-lower grid rib section welds	Cast austenitic stainless steel	Reactor coolant >250°C (>482°F) and neutron flux	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);" and Chapter XI.M16, "PWR Vessel Internals," Primary Components. Primary Components: Accessible top surfaces of 52 spider castings and welds to the adjacent lower grid rib section	No.

EPRI DRAFT Input for (99) Update

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
IV.B4-29 (R-202)	IV.B4.6-a	Lower grid assembly Lower grid rib section Fuel assembly support pads Lower grid flow distributor plate Orifice plugs Lower grid and shell forgings Guide blocks Shock pads Support post pipes	Stainless steel	Reactor coolant	Cracking/stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water, and Chapter XI.M16, "PWR Vessel Internals," No Additional Measures. No Additional Measures: All components of the lower grid assembly for SCC and IASCC	No.

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EPRI DRAFT Input for GALL Update

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

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Item	Link	Structure and/or Component	Material	Environment	Aging Effect/Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B4-30 (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 distributor 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	Changes in dimension/void swelling	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures. No Additional Measures: All components of the lower grid assembly for changes in dimension/void swelling	No.

EMERGENCY DRAFT Input for GALL Update

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
IV.B4-31 (R-205)	IV.B4.6-d	<p>Lower grid assembly</p> <p>Lower grid rib section</p> <p>Fuel assembly support pads</p> <p>Lower grid rib-to-shell forging screws</p> <p>Lower grid flow distributor plate</p> <p>Orifice plugs</p> <p>Lower grid and shell forgings</p> <p>Lower internals assembly-to-thermal shield bolts</p> <p>Guide blocks and bolts</p>	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness/neutron irradiation embrittlement, void swelling	<p>Chapter XI.M16, "PWR Vessel Internals."</p> <p>Expansion Components: Accessible pads, pad-to-rib section welds, Alloy X-750 dowels, cap screws, and their associated locking devices, depending on the results of the examination of the IMI guide tube spiders and the spider-to-lower grid rib section welds</p> <p>No Additional Measures: Lower grid rib-to-shell forging screws, lower grid and shell forgings, lower internals assembly-to-thermal shield bolts, guide blocks and bolts, the lower grid rib section, the lower grid distributor plate, the orifice plugs, the support post pipes, and shock pads and bolts for loss of fracture toughness/neutron irradiation embrittlement, void swelling</p>	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
IV.B4-32 (R-203)	IV.B4.6-b	<p>Lower grid assembly</p> <p>Lower grid rib-to-shell forging screws</p> <p>Lower internals assembly-to-thermal shield bolts</p> <p>Guide block bolts</p> <p>Shock pad bolts</p>	Stainless steel; nickel alloy	Reactor coolant	Cracking/stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	<p>Chapter XI.M2, "Water Chemistry," for PWR primary water, and Chapter XI.M16, "PWR Vessel Internals."</p> <p>Primary Components: Accessible locking welds for 24 dowel-to-guide block welds</p> <p>Expansion Components: Accessible lower grid shock pad bolts at TMI-1 and the lower internals assembly-to-thermal shield bolts at all plants, depending on the results of the examinations of upper core barrel (UCB) and lower core barrel (LCB) bolts; Accessible Alloy X-750 dowel-to-lower fuel assembly support pad welds, depending on the results of the examinations of the Alloy X-750 dowel-to-guide block welds</p> <p>No Additional Measures: Lower grid rib-to-shell forging screws for cracking</p>	No.

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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
IV.B4-33 (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	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures.	No.
IV.B4-34 (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.M16, "PWR Vessel Internals," No Additional Measures. No Additional Measures: The top flange-to-cover bolts, the plenum cover assembly, the plenum cylinder, the reinforcing plates and the bottom flange-to-upper grid screws for both SCC and IASCC	No.
IV.B4-36 (R-173)	IV.B4.1-b	Top flange-to-cover bolts Bottom flange-to-upper grid screws			Cracking/stress corrosion cracking, irradiation-assisted stress corrosion cracking		

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
IV.B4-35 (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 dimension/void swelling	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures.	No.
IV.B4-37 (R-53)	IV.B4.3-f IV.B4.5-f IV.B4.6-f IV.B4.2-d IV.B4.1-d IV.B4.4-e	Reactor vessel internals components	Stainless steel; nickel alloy	Reactor coolant	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.B4-38 (R-24)	IV.B4.	Reactor vessel internals components	Stainless steel; nickel alloy	Reactor coolant	Loss of material/pitting and crevice corrosion	Chapter XI.M2, "Water Chemistry," for PWR primary water.	No.

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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
IV.B-39 (R-215)	IV.B4.8-b	Core barrel assembly Thermal shield cylinder	Stainless steel	Reactor coolant	Changes in dimension/void swelling	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures.	No.
IV.B4-40 (R-214)	IV.B4.8-a	Core barrel assembly Thermal shield cylinder	Stainless steel	Reactor coolant	Cracking/stress corrosion, cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water, and Chapter XI.M16, "PWR Vessel Internals," No Additional Measures.	No.
IV.B-41 (R-216)	IV.B4.8-c	Core barrel assembly Thermal shield cylinder	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/neutron irradiation embrittlement, void swelling	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures.	No.

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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
IV.B4-42 (R-179)	IV.B4.2-f	Plenum cover and plenum cylinder assemblies Plenum rib pads (weldment rib pads) Support flange Lifting lug-to-base block bolts Top flange-to-cover bolts Bottom flange-to-upper grid bolts Upper grid assembly Fuel assembly support pad cap screws Rib-to-ring cap screws	Stainless steel	Reactor coolant	Loss of material and associated loss of clamping load/wear	Chapter XI.M16, "PWR Vessel Internals." Primary Components: Differential height between top of plenum rib pads and reactor vessel seating surface, with plenum in vessel, for wear Expansion Components: Accessible dowel locking welds for the Alloy X-750 dowel-to-upper fuel assembly support pad welds for all plants except Davis-Besse, depending on the results of the examination of the Alloy X-750 dowel-to-guide block welds No Additional Measures: Top flange-to-cover bolts, bottom flange-to-upper grid bolts, and rib-to-ring cap screws for loss of material/wear	No.

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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
IV.B4-43 (R-176)	IV.B4.2-b	Plenum cover and plenum cylinder assemblies Plenum rib pads (Weldment rib pads)	Stainless steel	Reactor coolant	Cracking/stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water, and Chapter XI.M16, "PWR Vessel Internals," No Additional Measures. No Additional Measures: All upper grid assembly and plenum cover/plenum cylinder assembly components with respect to SCC and IASCC	No.
IV.B4-44 (R-175)	IV.B4.2-a	Upper grid assembly Rib section Ring forging Fuel assembly support pad cap screws Rib-to-ring cap screws			Cracking/stress corrosion cracking, irradiation-assisted stress corrosion cracking		

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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
IV.B4-45 (R-177)	IV.B4.2-c	Plenum cover and plenum cylinder assemblies	Stainless steel	Reactor coolant and neutron flux	Changes in dimension/void swelling	Chapter XI.M16, "PWR Vessel Internals," No Additional Measures. No Additional Measures: All plenum cover, plenum cylinder assembly, and the upper grid assembly components for changes in dimension/void swelling and for loss of fracture toughness/neutron irradiation embrittlement	No.
IV.B4-46 (R-178)	IV.B4.2-e	Plenum rib pads (Weldment rib pads) Upper grid assembly Rib section Ring forging Fuel assembly support pads Rib-to-ring cap screws			Loss of fracture toughness/neutron irradiation embrittlement, void swelling		

EPRI DRAFT INPUT for GALE Update

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XI.M16 PWR VESSEL INTERNALS

Program Description

This program relies on implementation of the Electric Power Research Institute (EPRI) "Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227)" to manage aging effects on the reactor vessel internals.

This program includes:

- (a) Examinations and other inspections, and comparison with examination acceptance criteria as defined in MRP-227, Revision 0 and MRP-228, Revision 0 or later revisions;
- (b) Disposition of indications that exceed examination acceptance criteria by entering them into the licensee's Corrective Action Program, and may include evaluation for continued service until the next examination; and
- (c) Monitoring and control of reactor primary coolant water chemistry, in accordance with the EPRI PWR Primary Water Chemistry guidelines (EPRI TR-1014986, or later revisions).

The program also includes future industry operating experience as incorporated in periodic revisions to MRP-227. The program thus ensures the long-term integrity and safe operation of reactor internals in all commercial operating U.S. pressurized water reactor (PWR) nuclear power plants.

Evaluation and Technical Basis

1. Scope of Program: The guidance in MRP-227 provides requirements that assure functional integrity of safety-related internals in commercial operating U.S. PWR nuclear power plants designed by Babcock & Wilcox (B&W), Combustion Engineering (CE), and Westinghouse. The scope of components considered for guidance includes core support structures (typically denoted as B-N-3 by the ASME Code Section XI) and other internals components that by failure may affect the achievement of a safety related function. This scope definition was the basis for the requirements of MRP-227, and subject to the applicability assumptions listed in Section 2.4 of the document, is an acceptable scope definition for individual applicants. The scope does not include consumable items such as fuel assemblies, reactivity control assemblies, and nuclear instrumentation. The scope also does not include welded attachments to the internal surface of the reactor vessel.

This program is focused on managing the effects of eight age-related degradation mechanisms – stress corrosion cracking (SCC), irradiation assisted stress corrosion cracking (IASCC), loss of material caused by wear, cracking caused by fatigue, loss of fracture toughness caused by either thermal aging or neutron irradiation embrittlement, dimensional changes and potential loss of fracture toughness caused by void swelling and irradiation growth, and loss of preload caused by thermal and irradiation-enhanced stress relaxation or creep. The guidance also depends on preventive measures, such as fuel loading management and primary water chemistry control, to limit the degradation.

The guidance is based on a sampling methodology as defined by the Branch Technical Position RSLB-1, with periodic examinations and other inspections of highly-affected internals locations, in order to detect the effects of the eight age-related degradation mechanisms in a timely and effective manner. The sampling program includes a requirement for expanding the sample of periodic examinations and other inspections if the extent of the degradation effects exceeds the expected levels.

The selection of highly-affected internals locations is based on a four-step process:

- Screening of reactor internals for all three (B&W, CE, and Westinghouse) designs, considering chemical composition, neutron fluence exposure, temperature history, and representative stress levels, in order to determine the susceptibility or non-susceptibility of PWR internals to the eight postulated aging mechanisms;
- Further categorization of these reactor internals, based on the screening results and the likelihood/severity of safety and economic consequences, into categories (for each degradation effect) ranging from insignificant effects (Category A) to potentially moderately significant effects (Category B) to potentially significant effects (Category C);
- Functionality assessment of components and assemblies of components based on representative plant designs using irradiated and aged material properties to determine the effects of the degradation mechanisms on functionality;
- Aging management strategy development combining the results of functionality assessment with component accessibility, operating experience, existing evaluations, and prior examination results to determine the appropriate methodologies for maintaining the long-term functions of PWR internals safely and economically.

The result of this four-step sample selection process is a set of Primary internals locations for each of the three plant designs that are expected to show the leading indications of the degradation effects, with another set of Expansion internals locations that are specified to expand the sample should the indications be more severe than anticipated. The degradation effects in a third set of internals locations are deemed to be adequately managed by Existing Programs, such as ASME Code Section XI Examination Category B-N-3 examinations of core support structures, while a fourth set of internals locations are deemed to require No Additional Measures. Typically, 5% to 15% of the internals locations were classified as Primary, with another 7% to 10% of the internals locations classified as Expansion. Another 5% to 15% of the internals locations are covered by Existing Programs, with the remainder requiring No Additional Measures. This sample selection process is adequate to assure safety function integrity of the subject safety related PWR reactor internal components.

The guidance in MRP-227 includes information on component description and function (Section 3); requirements for methods, extent, and frequency of the examinations (Section 4); examination acceptance criteria and requirements for expanding the scope of the examinations as needed (Section 5); information on acceptable methods for evaluating the structural integrity significance of flaws detected during these examinations that exceed examination acceptance criteria (Section 6); and general information on component repair and replacement procedures (Section 6). The guidance also contains provisions for reporting to the EPRI MRP by the individual utilities on the results of the examinations, with the intent that the sampling program extends beyond an individual plant to include all other PWRs. In this way, the combined results from many sets of internals examinations are used to determine the need for program adjustments.

2. Preventive Actions: The guidance in MRP-227 does not specify any preventive actions other than the applicability limitations to base-loaded plants and the fuel loading management assumptions upon which the functionality analyses were based. These limitations and assumptions require a determination of applicability by the licensee for each reactor, and are covered in Section 2.4 of MRP-227.

In addition, the guidance in MRP-227 relies on PWR water chemistry control to manage SCC and reduce the impact of IASCC. Therefore, an important adjunct to the aging management methodologies described by the guidance in MRP-227 is PWR water chemistry control. The water chemistry program for PWRs relies on

monitoring and control of reactor water chemistry as presented in Chapter XI.M2, "Water Chemistry," of NUREG-1801, Volume 2.

3. Parameters Monitored/Inspected: The program monitors the effects of eight aging degradation mechanisms on the intended function of PWR internals through a set of periodic examinations and other inspections using well-established visual examination, volumetric examination, and physical measurement techniques in accordance with the requirements of MRP-227. Cracking caused by SCC, IASCC, and fatigue is monitored/inspected by either visual (VT-1 or EVT-1) examination (for internals other than bolting) or by volumetric (UT) examination (bolting). Visual (VT-3) examination is used to monitor/inspect for the gross effects of SCC, IASCC, and fatigue cracking, and for loss of material caused by wear. The VT-3 detection of gross cracking effects is used only when the tolerance of the component or affected assembly is known or has been shown to be tolerant of easily detected large flaws. In addition, VT-3 examinations are used to monitor/inspect for general aging conditions, such as gross distortion caused by void swelling and irradiation growth or by gross effects of loss of preload caused by thermal and irradiation-enhanced stress relaxation and creep. The loss of fracture toughness, whether caused by either thermal aging or neutron irradiation embrittlement, or by void swelling and irradiation growth, is not directly measured; instead, the consequential effects of loss of fracture toughness are monitored/inspected using visual or volumetric examinations. In addition, physical measurements are used to monitor/inspect for the gross effects of wear in a few representative cases.

The parameters monitored/inspected for Primary components are described specifically for B&W designs in Table 4-1, for CE designs in Table 4-2, and for Westinghouse designs in Table 4-3. The parameters monitored/inspected for Expansion components are described specifically for B&W designs in Table 4-4, for CE designs in Table 4-5, and for Westinghouse designs in Table 4-6. These tables provide detailed descriptions of the relevant conditions that require disposition within the Corrective Action Program. The parameters monitored/inspected for Existing Program components follow the requirements of the referenced existing programs, such as the ASME Code Section XI Table IWB-2500-1 descriptions or the GALL AMP XI.M37 Flux Thimble Tube Inspection. In several Existing Programs, additional descriptive information is provided to supplement the existing program relevant conditions, on the basis that more precise information on the degradation mechanism and its effects is known. The relevant conditions constitute the important characteristics of the parameters monitored/inspected.

4. Detection of Aging Effects: The detection of aging effects is covered in two places: (1) the guidance in Section 4 of MRP-227 provides an introductory discussion and justification of the examination methods selected for detecting the aging effects of interest; and (2) standards for examination methods, procedures, and personnel are provided in a companion document, MRP-228. In all cases, well-established methods were selected. These methods include volumetric (UT) examination methods for detecting flaws in bolting and various visual (VT-3, VT-1, and EVT-1) examinations for detecting effects ranging from general conditions to detection and sizing of surface-breaking discontinuities.

The capability of volumetric examination by ultrasonic testing (UT) to detect loss of integrity of PWR internals bolts, pins, and fasteners, such as baffle-former bolting in B&W and Westinghouse units has been well demonstrated by operating experience. In addition, MRP-228 requires that Technical Justifications that are needed for volumetric examination method demonstrations, based on the requirements of the ASME Code, Section V. Based upon this supporting documentation, the methods, coverage, and schedule of the inspection and test techniques prescribed by MRP-227 are capable of maintaining structural integrity and ensuring the detection and correction of aging effects before the loss of intended function of PWR internals.

For some components MRP-227 specifies a focused visual (VT-3) examination, similar to the current ASME Code Section XI Examination Category B-N-3 examinations, in order to determine the general mechanical and structural condition of the internals by: (a) verifying parameters, such as clearances, settings, and physical displacements; and (b) detecting discontinuities and imperfections, such as loss of integrity at bolted or welded connections, loose or missing parts, debris, corrosion, wear, or erosion. When more rigorous detection of cracking is required, PWR internals will be examined by visual (VT-1) examination, in order to detect discontinuities and imperfections, such as cracks, corrosion, wear, or erosion, on the surfaces of components. In some cases, where even more stringent examinations are required, enhanced visual (EVT-1) examinations or ultrasonic methods of volumetric inspection, are specified for certain selected components and locations.

5. Monitoring and Trending: The examinations and re-examinations required by the MRP-227 guidance, together with the requirements specified in MRP-228 for inspection methodologies, inspection procedures, and inspection personnel, provide timely detection, reporting, and corrective actions with respect to the effects of the eight age-related degradation mechanisms within the scope of the program. The extent of the examinations, beginning with the sample of susceptible PWR internals component locations identified as Primary locations, with the potential for inclusion of Expansion locations if the effects are greater than anticipated, plus the continuation of the Existing Programs activities, such as the ASME Code Section XI Examination Category B-N-3 examinations for core support structures, provides a high degree of confidence in the total program.

6. Acceptance Criteria: Section 5 of MRP-227 provides specific examination acceptance criteria for the Primary and Expansion component examinations. For Existing Programs components referenced to ASME Section XI, the IWB-3500 acceptance criteria apply. For other Existing Programs, the examination acceptance criteria is described within the existing program reference document.

The guidance in MRP-227 contains three types of examination acceptance criteria:

- For visual examination (and surface examination as an alternative to visual examination), the examination acceptance criterion is the absence of any of the specific, descriptive relevant conditions; in addition, there are requirements to record and disposition surface breaking indications that are detected and sized for length by the visual (VT-1/EVT-1) examinations;
- For volumetric examination, the examination acceptance criterion is the capability for reliable detection of indications in bolting, as demonstrated in the examination Technical Justification; in addition, there are requirements for system-level assessment of bolted or pinned assemblies with unacceptable volumetric (UT) examination indications that exceed specified limits; and
- For physical measurements, the examination acceptance criterion for the acceptable tolerance in the measured differential height from the top of the plenum rib pads to the vessel seating surface in B&W plants has been generically established and is given in Table 5-1, while the Westinghouse plant internals hold-down spring height limit will be established on a plant-specific basis.

The use of visual examination relevant conditions (and the absence of relevant conditions as a visual examination acceptance criterion) is consistent with the ASME Code Section XI rules for visual examination. MRP-227 has added specificity to the visual examination relevant conditions by providing descriptions that are more applicable to the components and degradation effects, so that the absence of these specific degradation effect conditions gives improved confidence in the examination results.

The technical basis for volumetric examination relevant conditions can be found in MRP-228, where the review of existing bolting ultrasonic examination technical justifications has demonstrated the indication detection capability of at least two vendors, and where vendor technical justification is a requirement prior to any additional bolting examinations.

In addition to examination acceptance criteria for Primary components, MRP-227 also defines expansion criteria to be used for expanding the examinations to include the Expansion components. This implements the sampling basis inspection approach to adequately determine the significant extent of the observed condition.

Any detected conditions that do not satisfy the examination acceptance criteria are required to be dispositioned through the plant corrective action program, which may require repair, replacement, or analytical evaluation for continued service until the next inspection. The disposition will ensure that design basis functions of the reactor internals components will continue to be fulfilled for all licensing basis loads and events. Example methodologies that can be used to analytically disposition unacceptable conditions are discussed or referenced in Section 6 of MRP-227. However, other alternatives to the Section 6 methodologies may also be used, such as the methodologies in WCAP-17096.

7. Corrective Actions: Corrective actions following the detection of unacceptable conditions are fundamentally provided for in each plant's corrective action program. Additional guidance for disposition of unacceptable conditions for PWR internals may be found in the ASME Code, Section XI, and in Section 6 of MRP-227. Section 6 of MRP-227 describes the options that are available for disposition of detected conditions that exceed the examination acceptance criteria of Section 5. These include engineering evaluation methods, as well as supplementary examinations to further characterize the detected condition, or the alternative of component repair and replacement procedures. The latter are subject to the requirements of the ASME Code Section XI. The implementation of the guidance in MRP-227, plus the implementation of any ASME Code requirements, provides an acceptable level of aging management of safety-related components addressed in accordance with the corrective actions of 10 CFR Part 50, Appendix B or its equivalent, as applicable.

8. Confirmation Process: Site quality assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B or their equivalent, as applicable. It is expected that the implementation of the guidance in MRP-227 will provide an acceptable level of quality for inspection, flaw evaluation, and other elements of aging management of the PWR internals that are addressed in accordance with the 10 CFR Part 50, Appendix B or their equivalent (as applicable), confirmation process, and administrative controls.

9. Administrative Controls: The administrative controls for such programs, including their implementing procedures and review and approval processes are under existing site 10 CFR 50 Appendix B Quality Assurance Programs or their equivalent, as applicable. Such a program is thus expected to be established with a sufficient level of documentation and administrative controls to ensure effective long term implementation.

10. Operating Experience: Relatively few incidents of PWR internals aging degradation have been reported in operating U.S. commercial PWR plants. A summary of observations to date is provided in Appendix A of MRP-227-A. The licensee is expected to review subsequent operating experience for impact on its program, or participate in industry initiatives that perform this function.

The application of the MRP-227 guidance will establish a considerable amount of operating experience over the next few years. Section 7 of MRP-227 describes the reporting requirements for these applications, and the plan for evaluating the accumulated additional operating experience.

References

- MRP-227-Rev. 0, *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines*. Electric Power Research Institute, Palo Alto, CA: 2008. 1016596.
- MRP-228, *Materials Reliability Program: Inspection Standard for PWR Internals*. Electric Power Research Institute, Palo Alto, CA: 2009. 1016609.
- WCAP-17096, *Reactor Internals Acceptance Criteria Methodology and Data Requirements*, PWROG, July 2009.
- EPRI TR-1014986, *PWR Primary Water Chemistry Guidelines*, Revision 6, Electric Power Research Institute, Palo Alto, CA, December 2007.
- NUREG-1801, Volume 2, *Generic Aging Lessons Learned (GALL) Report*, Volume 2, "Tabulation of Results," Revision 1, U.S. Nuclear Regulatory Commission, Washington, DC, September 2005.
- ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, ASME Boiler and Pressure Vessel Code, 2001 Edition, including the 2002 and 2003 Addenda, American Society of Mechanical Engineers, New York, NY.
- ASME Boiler & Pressure Vessel Code, Section V, Nondestructive Examination*, American Society of Mechanical Engineers, New York, NY, 2004 Edition, July 1, 2004.
- 10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2005.
- EPRI TR-102134, *PWR Secondary Water Chemistry Guideline-Revision 3*, Electric Power Research Institute, Palo Alto, CA, May 1993.
- 10 CFR 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2005.