CHAPTER IV

(August 2000)

REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

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A1. Reactor Vessel (Boiling Water Reactor)

System, Structures, and Components

The system, structures, and components included in this table comprise the boiling water reactor (BWR) pressure vessel and consist of vessel shell and flanges, attachment welds, top and bottom heads, nozzles (including safe ends) for the reactor coolant system (recirculating system) and connected systems such as (high- and low-pressure core spray, high- and low-pressure coolant injection, main steam and feedwater systems), penetrations for instrument lines and drains, and control rod drive mechanism housing. Support skirt and attachment welds for vessel support are also included in the table. All structures and components in the reactor vessel are classified as Group A Quality Standards.

System Interfaces

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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A1.1.1, A1.1.2	Top Head Enclosure (without cladding)	Top Head, Nozzles (Vent, Top Head Spray or RCIC, and Spare)	SA302-Gr B, SA533-Gr B, SA336	288°C (550°F) Steam	Loss of Material	General, Pitting, and Crevice Corrosion	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. BWRVIP-29 (EPRI TR-103515).
A1.1.3	Top Head Enclosure	Head Flange	SA302-Gr B, SA533-Gr B, SA336, with or without SS cladding	288°C (550°F) Steam	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
A1.1.4	Top Head Enclosure	Closure Studs and Nuts	SA193-Gr B7, SA540-Gr B23/24, SA320-Gr L43 (AISI 4340), SA194-Gr 7 Maximum tensile strength <1172 MPa (<170 ksi)	Air, Leaking Reactor Coolant Water and/or Steam at 288°C (550°F)	Crack Initiation and Growth	SCC, IGSCC	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. RICSIL 055R1. *Supporting Documents:* Reg. Guide 1.65. ASME Code Case N-307-1.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Inservice inspection in conformance with ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1. Coolant water chemistry is monitored and maintained in accordance with the guidelines in BWRVIP-29 (TR-103515) to minimize corrosion *Supporting Documents*: BWRVIP-03 for reactor pressure vessel internals examination guidelines; and BWRVIP-62 for technical basis for inspection relief for internal components with hydrogen injection.	***(1) Scope of Program:*** The program includes preventive measures to mitigate corrosion and inservice inspection (ISI) to manage the effects of corrosion on the intended function of top head enclosure. ***(2) Preventive Actions:*** Coolant water chemistry is monitored and maintained in accordance with the guidelines in BWRVIP-29 (TR-103515) to minimize corrosion ***(3) Parameters Monitored/Inspected:*** The AMP monitors degradation due to general, pitting, and crevice corrosion by ISI of top head enclosure interior in accordance with Table IWB 2500-1. Requirements of examination category B-N-1 specify visual VT-3 of top head enclosure interior surface made accessible for examination during normal refueling outages. ***(4) Detection of Aging Effects:*** The extent and schedule of ISI assure detection of degradation before the loss of intended function of the top head enclosure. ***(5) Monitoring and Trending:*** Inspection schedule of IWB-2400 should provide for timely detection of degradation by corrosion. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with IWB-3000 by comparing ISI results with the acceptance standards of subsection IWB-3500. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWA-4000. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** The present program is effective in managing the effects of corrosion on the intended function of top head enclosure.	No
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii), and for addressing environmental effects on fatigue. See Chapter X of this report, for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes TLAA
Inservice inspection in conformance with ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a, Subsection IWB, Table IWB 2500-1, and additional recommendations of GE Rapid Information Communication Service Information Letter (RICSIL) 055 Revision 1, Supplement 1. *Supporting Documents*: Regulatory Guide 1.65 for prevention and replacement. ASME Code Case N-307-1 for ultrasonic examination.	***(1) Scope of Program:*** The program includes inservice inspection (ISI) to manage the effects of IGSCC on the intended function of reactor vessel closure stud bolting. ***(2) Preventive Actions:*** Guidance of Regulatory Guide (RG) 1.65 on material selection and protection against corrosion, prevent or mitigate IGSCC. High-strength low-alloy steels with controlled tempering procedures are used. Maximum tensile strength is limited to <1172 MPa (<170 ksi) to provide resistance to SCC, and Charpy V energy requirements of Appendix G to 10 CFR Part 50 provide adequate toughness to provide resistance to crack growth in the stud threads. Metal-plated stud bolting is avoided to prevent degradation due to corrosion or hydrogen embrittlement. Manganese phosphate or other acceptable surface treatment, or stable lubricants are permissible. During refueling and while the head is removed, the stud bolts and holes are protected from	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

A1. REACTOR VESSEL (Boiling Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A1.1.5	Top Head Enclosure	Vessel Flange Leak Detection Line	Stainless Steel, Ni–Alloys	Leaking Reactor Coolant Water and/or Steam up to 288°C (550°F)	Crack Initiation and Growth	SCC, IGSCC	–

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) corrosion and contamination in accordance with RICSIL 055 R1 S1. ***(3) Parameters Monitored/Inspected:*** The AMP detects and sizes cracks and detects leakage by ISI. Requirements of ASME Section XI, Table IWB 2500-1 examination category B-G-1, specify the following for all closure stud bolting: volumetric examination of studs in place, from top of the nut to bottom of the flange hole, and surface and volumetric examination of studs when removed; volumetric examination of flange threads; and visual VT-1 examination of surfaces of nuts, washers, and bushings. RICSIL Rev. 1 and its Supplement 1 provide additional recommendations regarding inspection and evaluation of the data. Inspection requirements of testing category B-P conducted according to IWA–5000 specify visual VT-2 (IWA-5240) examination of all pressure retaining components during system leakage test (IWB-5221) and system hydrostatic test (IWB-5222). ***(4) Detection of Aging Effects:*** Aging effects degradation of the closure stud bolting can not occur without crack initiation; the extent and schedule of ISI assure detection of cracks before the loss of intended function of closure stud bolting. ***(5) Monitoring and Trending:*** Inspection schedule of IWB-2400 and additional recommendations of RICSIL 055 Rev. 1, are effective and adequate for timely detection of cracks. Recommendations of RICSIL 055 include expansion of sample size and ultrasonic examination from the center drilled hole of studs in compliance with ASME Code Case N-307-1. ***(6) Acceptance Criteria:*** Any cracks in closure stud bolting are evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3500. ***(7) Corrective Actions:*** Repair and replacement is in conformance with IWB-4000 and material and inspection guidance of RG 1.65. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** SCC has occurred in BWR pressure vessel head studs. The AMP provides recommendations regarding inspection techniques and evaluation, material specifications, corrosion prevention, and other aspects of reactor pressure vessel head stud cracking, and is effective in managing the effects of SCC to maintain the intended function of closure studs and nuts during the period of license renewal.	
Plant-specific aging management program; existing programs may not be capable of mitigating or detecting SCC of vessel flange leak detection line.	Plant-specific aging management program is to be evaluated.	Yes, plant specific AMP

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

A1. REACTOR VESSEL (Boiling Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A1.2.1, A1.2.2	Vessel Shell	Vessel Flange, Upper Shell	SA302-Gr B, SA533-Gr B, SA336 with SS cladding	288°C (550°F) Steam	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
A1.2.3 - A1.2.6	Vessel Shell	Intermediate Nozzle Shell, Intermediate Beltline Shell, Lower Shell, Beltline Welds	SA302-Gr B, SA533-Gr B with 308, 309, 308L, 309L cladding	288°C, Reactor Coolant Water, max $5x10^9$ $n/cm^2 \cdot s$	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
A1.2.4, A1.2.6	Vessel Shell	Intermediate Beltline Shell, Beltline Welds	SA302-Gr B, SA533-Gr B with 308, 309, 308L, 309L Cladding; and Low–alloy steel weldments	288°C (550°F), Reactor Coolant Water, $5x10^8$ - $5x10^9$ $n/cm^2 \cdot s$	Loss of Fracture Toughness	Neutron Irradiation Embrittle-ment	Appendices G & H to 10 CFR Part 50, Reg. Guide 1.99, NRC GL 88-11, NRC GL 92-01, Rev. 1, S 1. NRC GL 98-05. *Supporting BWRVIP:* BWRVIP-05, BWRVIP-74, BWRVIP-78.
A1.2.4, A1.2.6	Vessel Shell	Intermediate Beltline Shell, Beltline Welds	SA302-Gr B, SA533-Gr B with 308, 309, 308L, 309L Cladding; and Low–alloy steel weldments	288°C (550°F), Reactor Coolant Water, $5x10^8$ - $5x10^9$ $n/cm^2 \cdot s$	Loss of Fracture Toughness	Neutron Irradiation Embrittle-ment	Appendices G & H to 10 CFR Part 50, ASTM E 185.
A1.2.7	Vessel Shell	Attachment Welds	SS, Inconel 182	288°C (550°F), Reactor Coolant Water	Crack Initiation and Growth	SCC, IGSCC	BWRVIP-48, ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. NUREG-0313, R. 2. NRC GL 88-01. NRC GL 88-01 S. 1. BWRVIP-29 (EPRI TR-103515). BWRVIP-14. BWRVIP-59. BWRVIP-62.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	*Same as for the effect of Fatigue on Item A1.1.3 head flange.*	Yes TLAA
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	*Same as for the effect of Fatigue on Item A1.1.3 head flange.*	Yes TLAA
For a 40 y design life, pressure vessel integrity is assured by fracture toughness and material surveillance program requirements set forth in Appendices G and H to 10 CFR Part 50, and methodology of Regulatory Guide 1.99, Rev. 2, and recommendations of Generic Letters (GLs) 88-11 and 92-01, Rev. 1, Supplement 1, to predict effects of neutron irradiation on reactor vessel materials. Upper shelf energy (USE) is maintained at acceptable levels for 54 effective full power years (EFPY) per BWRVIP-74.	Neutron irradiation embrittlement is a time-limited aging analysis (TLAA) to be evaluated for the period of license renewal for all ferritic materials that have a neutron fluence of greater than 10^{17} n/cm^2 (E>1 MeV) at the end of the license renewal term. In accordance with approved BWRVIP-74, the TLAA should evaluate the impact of neutron embrittlement on: (a) the adjusted reference temperature, the plant's pressure temperature limits, and the need for inservice inspection of circumferential and axial reactor vessel welds, (b) the Charpy upper shelf energy, (c) the equivalent margins analyses performed in accordance with 10 CFR 50, Appendix G, and (d) probability of failure of circumferential and axial welds." See the Standard Review Plan, Section 4.2 “Reactor Vessel Neutron Embrittlement” for details.	Yes TLAA
Appendix H to 10 CFR Part 50 requires the reactor vessel materials surveillance program to meet the American Society for Testing and Materials (ASTM) E 185 Standard. However, the surveillance program in ASTM E 185 is based on plant operation during the current license term, and additional surveillance capsules may be needed for the period of extended operation. Alternatively, an integrated surveillance program for the period of extended operation may be considered for a set of reactors that have similar design and operating features in accordance with Paragraph II.C of Appendix H to 10 CFR Part 50.	For evaluation and technical basis of the 10 elements of the AMP, see Chapter XI.M13 "Reactor Vessel Surveillance."	Yes
Inservice inspection in conformance with the guidelines of BWRVIP-48 and ASME Section XI (edition specified in 10CFR50.55a), Subsection IWB, Table IWB 2500-1. Coolant water chemistry is monitored and maintained in accordance with the guidelines in BWRVIP-29 (TR-103515) to minimize the potential of crack initiation and growth. *Supporting Documents*: BWRVIP-14 and -59 for evaluation of crack growth in stainless steels and nickel alloys, respectively.	***(1) Scope of Program:*** The program includes vessel ID attachment weld inspection and flaw evaluation guidelines of BWRVIP-48, as approved by the NRC, and ASME Section XI, to monitor the effects of SCC on the intended function of the component. ***(2) Preventive Actions:*** Mitigation is by selection of materials resistant to IGSCC and control of coolant water chemistry in accordance with the guidelines in BWRVIP-29 (TR-103515) including stringent control of conductivity (many BWRs now operate at <0.15 $\mu S/cm^2$). ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of IGSCC on the intended function of the component by detection and sizing of cracks by inservice inspection (ISI) in accordance with BWRVIP-48 and ASME Section XI Table IWB 2500-1.	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A1.3.1	Nozzles	Main Steam	SA508-Cl2 with or without SS Cladding	288°C (550°F) Steam	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
A1.3.2	Nozzles	Feedwater	SA508-Cl2 with or without SS cladding	Up to 288°C (550°F), Reactor Coolant Water	Crack Initiation and Growth	Cyclic Loading	GE NE-523-A71-0594. NUREG-0619. NRC GL 81-11. ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
(continued from previous page) BWRVIP-62 for technical basis for inspection relief for internal components with hydrogen injection.	*(continued from previous page)* ***(4) Detection of Aging Effects:*** Degradation due to SCC can not occur without crack initiation and growth. Attachment weld inspection and flaw evaluation guidelines are provided in BWRVIP-48. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 and BWRVIP-48 is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with IWB-3000 and BWRVIP-48. ***(7) Corrective Actions:*** Repair and replacement is in conformance with IWB-4000. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** IGSCC has occurred BWR components. The program addresses improvements in all three of the elements, viz., a susceptible (sensitized) material, significant tensile stress, and an aggressive environment, that cause IGSCC.	No
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	*Same as for the effect of Fatigue on Item A1.1.3 head flange.*	Yes TLAA
Inservice inspection in conformance with ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a, Subsection IWB, Table IWB 2500-1, as modified by the recommendations of NUREG-0619 and NRC Generic Letter 81-11, and alternative recommendation of GE NE-523-A71-0594.	***(1) Scope of Program:*** The program includes systems modifications and enhanced inservice inspection (ISI) to monitor the effects of crack initiation and growth on the intended function of the component. ***(2) Preventive Actions:*** Mitigation is by systems modifications such as removal of stainless steel cladding and installation of improved spargers, and changes to plant-operating procedures such as improved feedwater controllers and rerouting of the reactor water cleanup system, to decrease the magnitude and frequency of temperature fluctuations. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of cracking on the intended function of the component by detection and sizing of cracks by ISI in accordance with the recommendations of NUREG-0619 and alternative recommendation of GE NE-523-A71-0594. NUREG-0619 specifies ultrasonic testing (UT) of the entire nozzle and penetration testing (PT) of varying portions of the blend radius and bore. GE NE-523-A71-0594 specifies UT of specific regions of the blend radius and bore. UT examination techniques and personnel qualification is according to the guidelines of GE NE-523-A71-0594. ***(4) Detection of Aging Effects:*** Degradation of the component can not occur without crack initiation and growth. Based on inspection method/techniques and plant-specific fracture mechanics assessments, inspection schedule is in accordance with Table 6-1 of GE NE-523-A71-0594 or Table 2 of NUREG-0619. Leakage monitoring may be used to modify the inspection interval. ***(5) Monitoring and Trending:*** Inspection schedule of GE NE-523-A71-0594 or NUREG-0619 are effective and	No

Item	Structure and Component	Region of Interest	Material	Environ- ment	Aging Effect	Aging Mechanism	References
A1.3.3	Nozzles	CRDRL	SA508-Cl2 with or without SS cladding	Up to 288°C (550°F), Reactor Coolant Water	Crack Initiation and Growth	Cyclic Loading	NUREG-0619. ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with IWB-3000. ***(7) Corrective Actions:*** Repair and replacement is in conformance with IWB-4000. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Cracking has occurred in several BWR plants (NUREG-0619). The present AMP has been effective in managing the effect of cracking on the intended function of feedwater nozzles.	
Inservice inspection in conformance with ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a, Subsection IWB, Table IWB 2500-1, as modified by the recommendations of NUREG-0619.	***(1) Scope of Program:*** The program includes systems modifications, inservice inspection (ISI), and maintenance program to monitor the effects of crack initiation and growth on the intended function of control rod drive return line (CRDRL) nozzles. ***(2) Preventive Actions:*** System modifications include rerouting the CRDRL to a system that connects to the reactor vessel and, for some classes of BWRs or those that can prove satisfactory system operation, return flow capability, and two-pump operation, cut and cap the CRDRL nozzle without rerouting. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of cracking on the intended function of the component by detection and sizing of cracks by ISI in accordance with ASME Table IWB 2500-1 and NUREG-0619. Inspection recommendations include dye penetration testing (PT) of the CRDRL nozzle blend radius and bore regions and the reactor vessel wall area beneath the nozzle, and return-flow-capacity demonstration and CRD-system-performance test. Also, ultrasonic inspection (UT) inspection of welded connections in the rerouted line. The inspection must include base metal to a distance of one-pipe-wall thickness or 0.5 in. whichever is greater, on both sides of the weld. ***(4) Detection of Aging Effects:*** Degradation of the component can not occur without crack initiation and growth; extent and schedule of inspection as delineated in NUREG 0619 will assure detection of cracks before the loss of intended function of the component. ***(5) Monitoring and Trending:*** Inspection schedule of NUREG-0619 is effective and adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with IWB-3000. All cracks found in the CRDRL nozzles must be removed by grinding. ***(7) Corrective Actions:*** Repair and replacement is in conformance with IWB-4000. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Cracking has occurred in several BWR plants (NUREG-0619, GL 88-11). The present AMP has been effective in managing the effect of cracking on the intended function of CRDRL nozzles.	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
A1. REACTOR VESSEL (Boiling Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A1.3.2, A1.3.3	Nozzles	Feedwater, CRDRL	SA508-Cl2 with or without SS cladding	Up to 288°C (550°F), Reactor Coolant Water	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
A1.3.4	Nozzles	LPCI	SA508-Cl2	Up to 288°C, Reactor Coolant Water, $5x10^8$ - $5x10^9$ $n/cm^2 \cdot s$	Loss of Fracture Toughness	Neutron Irradiation Embrittle-ment	*Same as for the TLAA for Neutron Irradiation Embrittlement of Items A1.2.4 and A1.2.6.*
A1.4.1 - A1.4.5	Nozzle Safe Ends	HPCS, LPCS, CRDRL, Recirculating Water, LPCI or RHR Injection	SS, SB-166 (Inconel 182 butter, and Inconel 82 or 182 weld)	Up to 288°C (550°F), Reactor Coolant Water	Crack Initiation and Growth	SCC, IGSCC	NRC GL 88-01. NRC GL 88-01 S1. NUREG-0313, Rev. 2. ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. GE SIL 455, R 1, BWRVIP-29 (EPRI TR-103515). *Supporting Documents:* Reg. Guide 1.45. BWRVIP-75. *Operating Experience:* NRC IN 82-39. NRC IN 84-41.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	*Same as for the effect of Fatigue on Item A1.1.3 head flange.*	Yes TLAA
Same as for the TLAA for Neutron Irradiation Embrittlement of Items A1.2.4 intermediate (beltline) shell and A1.2.6 beltline welds.	*Same as for the TLAA for Neutron Irradiation Embrittlement of Items A1.2.4 intermediate (beltline) shell and A1.2.6 beltline welds.*	Yes TLAA
Program delineated in NRC Generic letter (GL) 88-01 and technical basis document NUREG-0313, Rev. 2, or in BWRVIP-75 as approved by the NRC SER, and inservice inspection in conformance with ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1. Coolant water chemistry is monitored and maintained in accordance with the guidelines in BWRVIP-29 (TR-103515) to minimize the potential of crack initiation and growth. *Supporting documents*: BWRVIP-03 for reactor pressure vessel internals examination guidelines; BWRVIP-14, -59, and -60 for evaluation of crack growth; BWRVIP-61 for BWR vessel and internals induction heating stress improvement effectiveness on crack growth in operating plants; and BWRVIP-62 for technical basis for inspection relief for internal components with hydrogen injection.	***(1) Scope of Program:*** The program includes inspection and flaw evaluation guidelines delineated in NRC GL 88-01 and its Supplement 1, or BWRVIP-75 as approved by the NRC SER, to manage the effects of IGSCC on the intended function of austenitic stainless steel (SS) piping 4 in. or larger in diameter, and reactor vessel attachments and appurtenances. Although these guidelines primarily address austenitic SS components, they are also applied to nickel alloys. ***(2) Preventive Actions:*** Coolant water chemistry is monitored and maintained in accordance with the guidelines in BWRVIP-29 (TR-103515). BWRVIP-62 provides technical basis for inspection relief for internal components with hydrogen injection. A means of mitigation of IGSCC is by selection of material considered resistant to sensitization and IGSCC, e.g., low-carbon grades of austenitic SSs and weld metal, with a maximum carbon of 0.035% and minimum 7.5% ferrite in weld metal, and by special processing such as solution heat treatment, heat sink welding, and induction heating or mechanical stress improvement. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of IGSCC on the intended function of reactor vessel nozzle safe ends by detection and sizing of cracks and detection of leakage by inservice inspection (ISI). Examination requirements of ASME Section XI, Subsection IWB, Table IWB 2500-1. specifies for all nozzle-to-safe end butt welds NPS 4 or larger, volumetric and surface examination of ID region extending 1/4 in. on either side specifies for all nozzle-to-safe end butt welds NPS 4 or larger, volumetric and surface examination of ID region extending 1/4 in. on either side of the weld and 1/3 wall thickness deep, and surface examination of OD surface extending 1/2 in. on either side; surface examination for all butt welds less than NPS 4; and for all nozzle-to-safe end socket welds, surface examination of OD surface extending 1 in. on the buttered side and 1/2 in. on the other. BWRVIP-75 contain specific recommendations regarding ultrasonic testing (UT) methods for dissimilar metal welds, i.e., the use of 45-degree and 60-degree refracted longitudinal wave transducers for detecting IGSCC cracks in alloy 182 and low-alloy materials. Examination requirements of testing are in accordance with IWA–5000 and guidelines of GL 88-01 or BWRVIP-75. ***(4) Detection of Aging Effects:*** Aging effects degradation of the nozzle safe ends can not occur without crack initiation; extent and schedule of ISI	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

A1. REACTOR VESSEL (Boiling Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A1.4.3	Nozzle Safe Ends	CRDRL	SS, SB-166 (Inconel 182 butter, and Inconel 82 or 182 weld)	Up to 288°C (550°F), Reactor Coolant Water	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
A1.5.1 - A1.5.6	Penetrations	CRD Stub Tubes, Instrumenta-tion, Jet Pump Instrument, Standby Liquid Control, Flux Monitor, Drain Line	SS, SB-167	Up to 288°C (550°F), Reactor Coolant Water	Crack Initiation and Growth	SCC, IGSCC, Unantici-pated Cyclic Loading	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. NUREG-0313, Rev. 2. NRC GL 88-01. NRC GL 88-01, Sup. 1. BWRVIP-27, BWRVIP-49, BWRVIP-29 (EPRI TR-103515).

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) assure detection of cracks before the loss of intended function of the reactor vessel nozzle safe ends. ***(5) Monitoring and Trending:*** Inspection schedule and sample size of GL 88-01 or BWRVIP-75 are based on the condition of each weld and should provide for timely detection of cracks. Welds of resistant material are as a minimum examined according to an extent and frequency comparable to those of ASME Section XI. Inspection extent and schedule are enhanced for welds of non-resistant materials, or welds that have been treated by stress improvement or reinforced by weld overlay. ***(6) Acceptance Criteria:*** Any IGSCC degradation is evaluated in accordance with IWB-3000 and guidelines of GL 88-01 or BWRVIP-75. ***(7) Corrective Actions:*** Repair, replacement, and reexaminations are in conformance with IWB-4000. Continued operation without repair requires that crack growth calculation be performed according to the guidance of GL 88-01 or BWRVIP-75 ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** IGSCC has occurred in small- and large-diameter BWR piping safe end-to-nozzle welds (IN 82-39 & IN 84-41). The present AMP has provided effective means of ensuring structural integrity of the primary coolant pressure boundary.	
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	*Same as for the effect of Fatigue on Item A1.1.3 head flange.*	Yes TLAA
Inservice inspection in conformance with ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, and for those components 4 in. or larger, program delineated in NRC Generic letter (GL) 88-01 and its Supplement 1 and technical basis document NUREG-0313, Rev. 2. Inspection and flaw evaluation guidelines for instrument penetration are in accordance with BWRVIP-49 and for standby liquid control system/core plate ΔP are in accordance with BWRVIP-27. Coolant water chemistry is monitored and maintained in accordance with the guidelines in BWRVIP-29 (TR-103515) to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes inservice inspection in accordance with ASME Section XI, and for components 4 in. or larger, NUREG-0313 and GL 88-01 provide additional guidance to manage IGSCC in BWRs. Inspection and flaw evaluation guidelines are in accordance with BWRVIP-49 and BWRVIP-27, as approved by the NRC SER, for instrument penetration and standby liquid control system, respectively. ***(2) Preventive Actions:*** Mitigation of IGSCC is by selection of material considered resistant to sensitization and IGSCC, e.g., low-carbon grades of austenitic SSs and weld metal, with a maximum carbon of 0.035% and minimum 7.5% ferrite in weld metal, and by special processing such as solution heat treatment, heat sink welding, and induction heating or mechanical stress improvement. Inconel 82 is the only nickel base weld metal considered to be resistant to IGSCC. Coolant water chemistry is monitored and maintained in accordance with the guidelines in BWRVIP-29 (TR-103515). Also, hydrogen	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
							(continued from previous page) *Supporting BWRVIP:* BWRVIP-14, BWRVIP-53, BWRVIP-57, BWRVIP-59, BWRVIP-60, BWRVIP-62, BWRVIP-75. *Operating Experience* NRC IN 82-39. NRC IN 84-41.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
(continued from previous page) *Supporting documents*: Supporting documents for repair design criteria BWRVIP-57 for instrumentation penetrations and BWRVIP-53 for standby liquid control line; BWRVIP-14, -59, and -60 for evaluation of crack growth; BWRVIP-62 for technical basis for inspection relief for internal components with hydrogen injection; and BWRVIP-75 for technical basis for revisions to GL 88-01 inspection schedule.	*(continued from previous page)* water chemistry and stringent control of conductivity is used to inhibit IGSCC. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of IGSCC on the intended function of reactor vessel penetrations by detection and sizing of cracks by inservice inspection (ISI). System leakage test, IWB-5221, is conducted prior to plant startup following each refueling outage and visual VT-2 (IWA-5240) examination performed for all pressure retaining components extending to and including the second closed valve at the boundary extremity. Leakage detection is in conformance with Position C of Regulatory Guide 1.45 and additional guidelines of GL 88-01, Suppl. 1. System hydrostatic test, IWB-5222, is conducted at or near the end of each inspection interval and visual VT-2 examination performed for all class 1 components within boundary. Inspection requirements of examination category B-E focus on visual VT-2 examination of partial penetration welds during the hydrostatic test. ***(4) Detection of Aging Effects:*** Aging effects degradation of the reactor vessel penetrations can not occur without crack initiation; extent and schedule of inspection assure detection of cracks before loss of intended function of the reactor vessel penetrations. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI should provide for timely detection of cracks. Inspection schedule and sample size specified in Table 1 of GL 88-01 are based on the condition of each weld and are adequate for timely detection of cracks. Welds of resistant material are as a minimum examined according to an extent and frequency comparable to those of ASME Section XI. Inspection extent and schedule are enhanced for welds of non-resistant materials. ***(6) Acceptance Criteria:*** Any IGSCC degradation is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3522. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWA-4000 and IWB-4000, and reexamination in accordance with requirements of IWA-2200. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** The program addresses improvements in all three of the elements, viz., a susceptible (sensitized) material, significant tensile stress, and an aggressive environment, that cause IGSCC, and has provided effective means of ensuring structural integrity of the primary coolant pressure boundary.	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

A1. REACTOR VESSEL (Boiling Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A1.5.1 - A1.5.6	Penetrations	CRD Stub Tubes, Instrumenta-tion, Jet Pump Inst., Standby Liquid Control, Flux Monitor, Drain Line	SS, SB-167	Up to 288°C (550°F), Reactor Coolant Water	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
A1.6	Bottom Head	-	SA302-Gr B, SA533-Gr B with 308, 309, 308L, 309L cladding	Up to 288°C (550°F), Reactor Coolant Water	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
A1.7	Support Skirt & Attachment Welds	-	SA533-Gr B (Welds SS or Inconel 182)	Ambient Temperature Air	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	*Same as for the effect of Fatigue on Item A1.1.3 head flange.*	Yes TLAA
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	*Same as for the effect of Fatigue on Item A1.1.3 head flange.*	Yes TLAA
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal; check Code limits for allowable cycles (less than 7000 cycles) of thermal stress range. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes TLAA

A2. Reactor Vessel (Pressurized Water Reactor)

A2.1 Closure Head

A2.1.1 Dome

A2.1.2 Head Flange

A2.1.3 Stud Assembly

A2.1.4 Vessel Flange Leak Detection Line

A2.2 Control Rod Drive (CRD) Mechanism

A2.2.1 Nozzle

A2.2.2 Pressure Housing

A2.3 Nozzles

A2.3.1 Inlet

A2.3.2 Outlet

A2.3.3 Safety Injection (on some)

A2.4 Nozzle Safe Ends

A2.4.1 Inlet

A2.4.2 Outlet

A2.4.3 Safety Injection (on some)

A2.5 Shell

A2.5.1 Upper (Nozzle) Shell

A2.5.2 Intermediate & Lower Shell

A2.5.3 Vessel Flange

A2.6 Core Support Pads

A2.7 Penetrations

A2.7.1 Instrumentation Tubes (Bottom Head)

A2.7.2 Head Vent Pipe (Top Head)

A2.8 Pressure Vessel Support

A2.8.1 Skirt Support

A2.8.2 Cantilever/Column Support

A2.8.3 Neutron Shield Tank

A2. Reactor Vessel (Pressurized Water Reactor)

System, Structures, and Components

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

System Interfaces

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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A2.1.1 - A2.1.3	Closure Head	Dome, Head Flange, Stud Assembly (External Surfaces)	Dome and Flange: SA302-Gr B, SA533-Gr B, SA508-64; Stud Assembly: SA540-Gr B23/24, SA320-Gr L43 (Alloy 4340)	Air, Leaking Chemically Treated Borated Water or Steam up to 340°C (644°F)	Loss of Material	Boric Acid Corrosion of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 or later Edition as approved in 10 CFR 50 55a. *Operating Experience* NRC IN 86-108 S 3.
A2.1.1	Closure Head	Dome	SA302-Gr B, SA533-Gr B, SA508-64 Class 2 with Stainless Steel Cladding	Chemically Treated Borated Water or Steam up to 340°C (644°F)	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
A2.1.3	Closure Head	Stud Assembly	SA540-Gr B23/24, SA320-Gr L43 (Alloy 4340), SA193-6 Maximum tensile strength <1172 MPa (<170 ksi)	Air, Leaking Chemically Treated Borated Water or Steam up to 340°C (644°F)	Crack Initiation and Growth	Stress Corrosion Cracking (SCC)	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. *Supporting Documents*: Reg. Guide 1.65.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Implementation of NRC Generic Letter 88-05 and inservice inspection (ISI) in conformance with ASME Section XI (1989 edition or later edition as approved in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, to monitor the condition of the reactor coolant pressure boundary for occurrences of borated water leakage.	For evaluation and technical basis of the 10 elements of the AMP, see Chapter XI.M5, "Boric Acid Corrosion."	No
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii), and for addressing environmental effects on fatigue. See Chapter 10 of this report, for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes TLAA
Inservice inspection in conformance with ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1. *Supporting Documents*: Regulatory Guide 1.65 for prevention and replacement.	***(1) Scope of Program:*** The program includes inservice inspection (ISI) to manage the effects of SCC on the intended function of closure head stud assemblies. ***(2) Preventive Actions:*** Guidance of Regulatory Guide (RG) 1.65 on material selection and protection against corrosion, prevent or mitigate SCC. High-strength low-alloy steels with controlled tempering procedures are used. Maximum tensile strength is limited to <1172 MPa (<170 ksi) to provide resistance to SCC, and Charpy V energy requirements of Appendix G to 10 CFR Part 50 provide adequate toughness to provide resistance to crack growth in the stud threads. Metal-plated stud bolting is avoided to prevent degradation due to corrosion or hydrogen embrittlement. Manganese phosphate or other acceptable surface treatment, or stable lubricants are permissible. During venting or filling of pressure vessel and while the head is removed, the stud bolts and holes are adequately protected from corrosion and contamination. ***(3) Parameters Monitored/Inspected:*** The AMP detects and sizes cracks and detects leakage by ISI. Examination requirements of ASME Section XI, Table IWB 2500-1, examination category B-G-1, specify the following for all closure stud assemblies: volumetric examination of studs in place, from top of the nut to bottom of the flange hole, and surface and volumetric examination of studs when removed; volumetric examination of flange threads; and visual VT-1 examination of surfaces of nuts, washers, and bushings. Testing category B-P specify visual VT-2 (IWA-5240) examination of all pressure retaining components during system leakage test (IWB-5221) and system hydrostatic test (IWB-5222). ***(4) Detection of Aging Effects:*** Aging degradation of the closure stud assembly can not occur without crack initiation, the extent and schedule of inspection assure detection of cracks before the loss of intended function of closure stud assemblies. ***(5) Monitoring and Trending:*** Inspection schedule of IWB-2400 is effective and adequate for timely detection of	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A2.1.3	Closure Head	Stud Assembly	SA540-Gr B23/24, SA320-Gr L43 (Alloy 4340) SA193-6 Maximum tensile strength <1172 MPa (<170 ksi)	Air, Leaking Chemically Treated Borated Water or Steam up to 340°C (644°F)	Loss of material	Wear	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. *Supporting Documents*: Reg. Guide 1.65.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) cracks. System leakage test is conducted prior to plant startup following each refueling outage, and hydrostatic test at or near the end of each inspection interval. ***(6) Acceptance Criteria:*** Any cracks in closure stud assemblies are evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3500. ***(7) Corrective Actions:*** Repair and replacement is in conformance with IWB-4000 and material and inspection guidance of RG 1.65. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** The AMP based on ASME Section XI inspection requirements and additional guidelines of Reg. Guide 1.65 has been effective in managing the effects of SCC to maintain the intended function of the closure head stud assembly.	
Inservice inspection in conformance with ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1. *Supporting Documents*: Regulatory Guide 1.65 for prevention and replacement.	***(1) Scope of Program:*** The program is focused on managing the effects of wear on the intended function of closure head stud assemblies. ***(2) Preventive Actions:*** Design requirements and guidelines of Regulatory Guide (RG) 1.65, frequent performance monitoring, and timely corrective action prevent or mitigate attrition due to wear of closure stud assemblies. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of wear on the intended function of closure stud assembly by performance monitoring and by detection of attrition by inservice inspection (ISI). Inspection requirements of ASME Section XI, Table IWB 2500-1, examination category B-G-1, visual VT-1 examination of surfaces of nuts, washers, and bushings. Testing category B-P specify visual VT-2 (IWA-5240) examination of all pressure retaining components during system leakage test (IWB-5221) and system hydrostatic test (IWB-5222). ***(4) Detection of Aging Effects:*** Degradation due to wear of the closure stud assembly can not occur without loss of material, the extent and schedule of inspection assure detection of attrition before the loss of intended function of closure stud assemblies. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI is effective and adequate for timely detection of attrition. System leakage test is conducted prior to plant startup following each refueling outage, and hydrostatic test at or near the end of each inspection interval. ***(6) Acceptance Criteria:*** Any wear damage in closure stud assemblies are evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3500. ***(7) Corrective Actions:*** Repair is in conformance with IWB-4000, repair according to IWB-7000, and material and inspection guidance of RG 1.65. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
A2. REACTOR VESSEL (Pressurized Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A2.1.3	Closure Head	Stud Assembly	SA540-B23 & -B24, SA320-L43, SA193-6	Air, Leaking Chemically Treated Borated Water or Steam up to 340°C (644°F)	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
A2.1.4	Closure Head	Vessel Flange Leak Detection Line	Stainless Steel (SS)	Leaking Chemically Treated Borated Water or Steam up to 340°C (644°F)	Crack Initiation and Growth	SCC	Plant Technical Specifications.
A2.2.1	Control Rod Drive (CRD) Mechanism	Nozzle	SB-166 (Alloy 600)	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation and Growth	Primary Water Stress Corrosion Cracking (PWSCC)	NRC GL 97-01. ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. EPRI TR-105714 Rev. 3 or later revisions/update. *Operating Experience*: NRC IN 90-10. NRC IN 96-11.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** The present AMP has been effective in managing the effects of wear to maintain the intended function of closure head stud assembly.	
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes TLAA
Plant-specific aging management program; existing programs may not be capable of mitigating or detecting SCC of vessel flange leak detection line.	Plant-specific aging management program is to be evaluated.	Yes, plant specific AMP
The program includes inservice inspection in accordance with ASME Subsection IWB, Table IWB 2500-1 or for susceptible components and locations an integrated, long-term inspection program based on the guidelines of NRC Generic Letter (GL) 97-01 to detect cracks or coolant leakage. Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 (Rev. 3 or later revisions or update) to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes inservice inspection (ISI) in accordance with ASME Subsection IWB, Table IWB 2500-1, or for susceptible components and locations implementation of an integrated, long-term inspection program based on the guidelines of NRC Generic Letter (GL) 97-01 to detect cracks or coolant leakage. Preventive measures are in accordance with EPRI guidelines in TR-105714 to mitigate primary water stress corrosion cracking (PWSCC). The applicant performs a susceptibility assessment in accordance with the most current industry susceptibility model and inspection results, to define the most susceptible components and locations to be included in a periodic inspection program. These locations include those that have currently been identified as susceptible to PWSCC, and those that will become susceptible during the period of license renewal. ***(2) Preventive Actions:*** Control of halogens, sulfates, and oxygen in the primary water to less than 0.05, 0.05, and 0.005 ppm, respectively, during operation, and monitor and control of water chemistry during shut down, mitigate potential of PWSCC. However, ingress of demineralizer resins has occurred in operating plants [NRC Information Notices (IN) 96-11]; the program must therefore rely upon monitoring and control of primary water chemistry to manage the effects of such excursions. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of PWSCC on the intended function of the CRD mechanism nozzle by detection and sizing of cracks and coolant leakage by ISI. Susceptibility assessment is performed in accordance with the guidelines of GL 97-01 and the most current industry susceptibility models that are based on material and operating parameters and inspection results, to identify the most susceptible locations, and develop a plant-specific long-term inspection program including schedule, scope, and determination whether an augmented inspection program of nozzle welds, including a	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A2.2.2	Control Rod Drive (CRD) Mechanism	Pressure Housing	Types 403 and 316 SS; Type 304 SS or CF-8; SA 508 Class 2 with Alloy 82/182 Cladding	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation and Growth	SCC	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. EPRI TR-105714 Rev. 3 or later revisions/update. *Operating Experience*: NRC IN 84-18. NRC IN 84-89. NRC Report 50-255/99012.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) combination of surface and volumetric examination, is necessary, ***(4) Detection of Aging Effects:*** Aging degradation of the CRD mechanism housing can not occur without crack initiation and growth. Based on GL 97-01, the applicant should review the scope and schedule of inspection, including leakage detection system, to assure detection of cracks before the loss of intended function of the components. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with the integrated inspection program based on GL 97-01 susceptibility assessment should provides timely detection of cracks. Inspection results are used to update the susceptibility models. The frequency of subsequent inspections are based on the finding of the initial inspections and crack growth rate models for Ni alloys. ***(6) Acceptance Criteria:*** Any SCC degradation is evaluated in accordance with IWB-3000 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3500. However, the applicant should provide information on crack initiation and crack growth models and the data used to validate these models to verify adequacy of the inspection program and acceptance criteria. ***(7) Corrective Actions:*** Repair and replacement is in conformance with IWB-4000. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** SCC of Alloy 600 and austenitic SSs has occurred in domestic and foreign PWRs (IN 90-10).	
Inservice inspection in conformance with ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1. Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 (Rev. 3 or later revisions or update) to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes preventive measures to mitigate stress corrosion cracking (SCC) and inservice inspection (ISI) to manage the effects of SCC on the intended function of the CRD mechanism housing. ***(2) Preventive Actions:*** Control of halogens, sulfates, and oxygen in the primary water to less than 0.05, 0.05, and 0.005 ppm, respectively, during operation, and monitor and control of water chemistry during shut down, mitigate potential of SCC. However, inadvertent introduction of contaminants into the primary coolant system can occur (IN 84–18). The AMP must therefore rely upon ISI in accordance with ASME Section XI to detect possible degradation. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of SCC on the intended function of the CRD mechanism housing by detection and sizing of cracks by inservice inspection (ISI). Inspection requirements of Table IWB 2500-1, examination category B-O specifies volumetric or surface examination extending 1/2 in. each side of the CRD housing welds, including weld buttering. Testing category B-P specify visual VT-2 (IWA-5240) examination of all pressure retaining components during system leakage test (IWB-5221) and system hydrostatic test (IWB-5222). ***(4) Detection of Aging Effects:*** Aging degradation of the CRD mechanism housing	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A2.2.1, A2.2.2	Control Rod Drive (CRD) Mechanism	Nozzle, Pressure Housing	Types 403 and 316 SS; Type 304 SS or CF-8	Chemically Treated Borated Water up to 340°C (644°F)	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
A2.2.2	Control Rod Drive (CRD) Mechanism	Pressure Housing	Cast Austenitic Stainless Steel CF-8	Chemically Treated Borated Water up to 340°C (644°F)	Loss of Fracture Toughness	Thermal Aging Embrittle-ment	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. Letter from Christopher I. Grimes (NRC) to Douglas J. Walters (NEI) dated 5/19/2000.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) can not occur without crack initiation and growth; the extent and schedule of inspection assure detection of cracks before the loss of intended function of the CRD housing. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 should provide timely detection of cracks. System leakage test is conducted prior to plant startup following each refueling outage, and hydrostatic test at or near the end of each inspection interval. ***(6) Acceptance Criteria:*** Any SCC degradation is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3500. ***(7) Corrective Actions:*** Repair is in conformance with IWA-4000 and IWB-4000, replacement according to IWB-7000 and IWA-7000, and reexamination with the requirements of IWA-2200. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Although the primary pressure boundary components of PWRs are generally not affected by SCC because of low dissolved oxygen levels and control of water chemistry, cracking has occurred in CRD seal housing (NRC Inspection Report 50-255/99012). Also, SCC could occur at creviced/cold worked locations (IN 84-89).	
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	*Same as for the effect of Fatigue of Item A2.1.1 closure head dome.*	Yes TLAA
The reactor coolant system components are inspected in accordance with ASME Section XI, Subsection IWB. This inspection is not sufficient to detect the effects of loss of fracture toughness due to thermal aging embrittlement of cast austenitic stainless steel (CASS) piping. An acceptable AMP consists of the following: Determination of the susceptibility of CASS piping to thermal aging embrittlement based on casting method, Mo content, and percent ferrite. For “potentially susceptible” piping, aging management is accomplished either through enhanced volumetric examination or plant/component– specific flaw tolerance evaluation. Additional inspection or evaluations are not required for “not susceptible” piping to demonstrate that the material has adequate fracture toughness.	For evaluation and technical basis of the 10 elements of the AMP, see Chapter XI.M1 "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)".	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A2.3.1 - A2.3.3	Nozzles	Inlet, Outlet, Safety Injection	SA336, SA508 with SS Cladding	Chemically Treated Borated Water up to 340°C (644°F) Neutron Fluence of greater than 10^{17} n/cm^2 (E>1 MeV)	Loss of Fracture Toughness	Neutron Irradiation Embrittle-ment	Appendices G & H to 10 CFR Part 50. Reg. Guide 1.99. NRC GL 88-11. NRC GL 92-01, Rev. 1, S 1.
A2.3.1 - A2.3.3	Nozzles	Inlet, Outlet, Safety Injection	SA336, SA508 with SS Cladding	Chemically Treated Borated Water up to 340°C (644°F) Neutron Fluence of greater than 10^{17} n/cm^2 (E>1 MeV)	Loss of Fracture Toughness	Neutron Irradiation Embrittle-ment	Appendix H to 10 CFR Part 50, ASTM E 185.
A2.3.1 - A2.3.3	Nozzles	Inlet, Outlet, Safety Injection	SA336, SA508 with SS Cladding	Chemically Treated Borated Water up to 340°C (644°F)	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
A2.4.1 - A2.4.3	Nozzle Safe Ends	Inlet, Outlet, Safety Injection	SS, CASS (NiCrFe buttering, and SS or NiCrFe weld)	Chemically Treated Borated Water up to 340°C (644°F)	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
A2.4.1 - A2.4.3	Nozzle Safe Ends	Inlet, Outlet, Safety Injection	SS, CASS (NiCrFe buttering, and SS or NiCrFe weld)	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation and Growth	SCC, Cyclic Loading	-

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
For a 40 y design life, pressure vessel integrity is assured by fracture toughness and material surveillance program requirements set forth in Appendices G and H to 10 CFR Part 50, and methodology of Regulatory Guide 1.99, Rev. 2, and recommendations of NRC Generic Letters (GLs) 88-11 and 92-01 Rev. 1 and Supplement 1, to predict effects of neutron irradiation on reactor vessel materials.	Neutron irradiation embrittlement is a time-limited aging analysis (TLAA) to be evaluated for the period of license renewal for all ferritic materials that have a neutron fluence of greater than 10^{17} n/cm^2 (E>1 MeV) at the end of the license renewal term. The TLAA should evaluate the impact of neutron embrittlement on: (a) the RTPS value based on the requirements in 10 CFR50.6, (b) the adjusted reference temperature, the plant's pressure temperature limits, (c) the Charpy upper shelf energy, and (d) the equivalent margins analyses performed in accordance with 10 CFR 50, Appendix G. Applicants are to determine whether the pressure-temperature operating window at 60 years is sufficient to conduct heatups and cooldowns.	Yes TLAA
Appendix H to 10 CFR Part 50 requires the reactor vessel materials surveillance program to meet the American Society for Testing and Materials (ASTM) E 185 Standard. However, the surveillance program in ASTM E 185 is based on plant operation during the current license term, and additional surveillance capsules may be needed for the period of extended operation. Alternatively, an integrated surveillance program for the period of extended operation may be considered for a set of reactors that have similar design and operating features in accordance with Paragraph II.C of Appendix H to 10 CFR Part 50.	For evaluation and technical basis of the 10 elements of the AMP, see Chapter XI.M13 "Reactor Vessel Surveillance."	Yes
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	*Same as for the effect of Fatigue of Item A2.1.1 closure head dome.*	Yes TLAA
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	*Same as for the effect of Fatigue of Item A2.1.1 closure head dome.*	Yes TLAA
Inservice inspection in conformance with ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1. Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 (Rev. 3 or later revisions or update) to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes preventive measures to mitigate stress corrosion cracking (SCC) and inservice inspection (ISI) to manage the effects of crack initiation and growth on the intended function of the component. ***(2) Preventive Actions:*** Control of halogens, sulfates, and oxygen in the primary water to less than 0.05, 0.05, and 0.005 ppm, respectively, during operation, and monitor and control of water chemistry during shut down, mitigate potential of SCC. However, inadvertent introduction of contaminants into the primary coolant system can occur (IN 84–18). The AMP must therefore rely upon ISI in accordance with ASME Section XI to detect possible degradation. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of cracking on the intended function of the component by detection and sizing of cracks by inservice inspection (ISI). Inspection	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A2.5.1, A2.5.2	Vessel Shell	Upper Shell, Intermediate and Lower Shell (Including Beltline Welds)	SA302-Gr B, SA533-Gr B, SA336, SA508-Cl 2 or Cl 3 with Types 308 or 309 Cladding	Chemically Treated Borated Water up to 340°C (644°F) Neutron Fluence of greater than 10^{17} n/cm^2 (E>1 MeV)	Loss of Fracture Toughness	Neutron Irradiation Embrittle-ment	Appendices G & H to 10 CFR Part 50. Reg. Guide 1.99. NRC GL 88-11. NRC GL 92-01, Rev. 1, S 1.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) requirements of Table IWB 2500-1, examination category B-F specifies for all nozzle-to-safe end butt welds NPS 4 or larger, volumetric or surface examination of ID region and surface examination of OD surface, only surface examination is conducted for all butt welds less than NPS 4 and for all nozzle-to-safe end socket welds. Testing category B-P specify visual VT-2 (IWA-5240) examination of all pressure retaining components during system leakage test (IWB-5221) and system hydrostatic test (IWB-5222). ***(4) Detection of Aging Effects:*** Aging degradation of the component can not occur without crack initiation and growth; the extent and schedule of inspection assure detection of cracks before the loss of intended function of the component. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 should provide timely detection of cracks. System leakage test is conducted prior to plant startup following each refueling outage, and hydrostatic test at or near the end of each inspection interval. ***(6) Acceptance Criteria:*** Any cracks are evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3500. ***(7) Corrective Actions:*** Repair is in conformance with IWA-4000 and IWB-4000, replacement according to IWB-7000 and IWA-7000, and reexamination with the requirements of IWA-2200. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Although the primary pressure boundary components of PWRs are generally not affected by SCC because of low dissolved oxygen levels and control of water chemistry, cracking could occur at creviced/cold worked locations (IN 84-89).	
For a 40 y design life, pressure vessel integrity is assured by fracture toughness and material surveillance program requirements set forth in Appendices G and H to 10 CFR Part 50, and methodology of Regulatory Guide 1.99, Rev. 2, and recommendations of NRC Generic Letters (GLs) 88-11 and 92-01 Rev. 1 and Supplement 1, to predict effects of neutron irradiation on reactor vessel materials.	Neutron irradiation embrittlement is a time-limited aging analysis (TLAA) to be evaluated for the period of license renewal for all ferritic materials that have a neutron fluence of greater than 10^{17} n/cm^2 (E>1 MeV) at the end of the license renewal term. The TLAA should evaluate the impact of neutron embrittlement on: (a) the RTPS value based on the requirements in 10 CFR50.6, (b) the adjusted reference temperature, the plant's pressure temperature limits, (c) the Charpy upper shelf energy, and (d) the equivalent margins analyses performed in accordance with 10 CFR 50, Appendix G. Applicants are to determine whether the pressure-temperature operating window at 60 years is sufficient to conduct heatups and cooldowns.	Yes TLAA

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

A2. REACTOR VESSEL (Pressurized Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A2.5.1, A2.5.2	Vessel Shell	Upper Shell, Intermediate and Lower Shell (Including Beltline Welds)	SA302-Gr B, SA533-Gr B, SA336, SA508-Cl 2 or Cl 3 with Types 308 or 309 Cladding	Chemically Treated Borated Water up to 340°C (644°F) Neutron Fluence of greater than 10^{17} n/cm^2 (E>1 MeV)	Loss of Fracture Toughness	Neutron Irradiation Embrittle-ment	Appendix H to 10 CFR Part 50, ASTM E 185.
A2.5.1 - A2.5.3	Vessel Shell	Upper (Nozzle) Shell, Intermediate and Lower Shell, Vessel Flange	SA302-Gr B, SA533-Gr B, SA336, SA508 with SS Cladding	Chemically Treated Borated Water up to 340°C (644°F)	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
A2.5.3	Vessel Shell	Vessel Flange (External Surface)	SA336, SA508	Air, Leaking Chemically Treated Borated Water or Steam up to 340°C (644°F)	Loss of Material	Boric Acid Corrosion of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.
A2.5.3	Vessel Shell	Vessel Flange (External Surface)	SA336, SA508	Leaking Chemically Treated Borated Water or Steam up to 340°C (644°F)	Loss of Material	Wear	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Appendix H to 10 CFR Part 50 requires the reactor vessel materials surveillance program to meet the American Society for Testing and Materials (ASTM) E 185 Standard. However, the surveillance program in ASTM E 185 is based on plant operation during the current license term, and additional surveillance capsules may be needed for the period of extended operation. Alternatively, an integrated surveillance program for the period of extended operation may be considered for a set of reactors that have similar design and operating features in accordance with Paragraph II.C of Appendix H to 10 CFR Part 50.	For evaluation and technical basis of the 10 elements of the AMP, see Chapter XI.M13 "Reactor Vessel Surveillance."	Yes
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	*Same as for the effect of Fatigue of Item A2.1.1 closure head dome.*	Yes TLAA
Implementation of NRC Generic Letter 88-05, and inservice inspection in conformance with ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, testing category B-P for system leakage.	*Same as Boric Acid Corrosion of external surfaces of Item A2.1.1 Closure Head Dome.*	No
Inservice inspection in conformance with ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1.	***(1) Scope of Program:*** The program relies on inservice inspection (ISI) to manage the effects of wear on the intended function of core support pads. ***(2) Preventive Actions:*** Design requirements, frequent inspection, and timely corrective action prevent or mitigate attrition due to wear of core support pads. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of wear on the intended function of core support pads by detection of loss of material by ISI. Inspection requirements of Table IWB 2500-1, examination category B-N-1 specify visual VT-3 of interior surfaces made accessible for examination during normal refueling outages. ***(4) Detection of Aging Effects:*** Aging degradation due to wear of the core support pads can not occur without attrition; extent and schedule of inspection assure detection of attrition before the loss of intended function of the core support pads. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI should provide for timely detection of attrition. All accessible welds of core support pads are inspected each inspection period. ***(6) Acceptance Criteria:*** Any wear is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and 35200. Visual examinations that reveal relevant conditions	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A2.6	Core Support Pads (Lugs)	-	SB-166 (Alloy 600), SB-168	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation and Growth	SCC	-
A2.6	Core Support Pads (Lugs)	-	SB-166 (Alloy 600), SB-168	Chemically Treated Borated Water up to 340°C (644°F)	Loss of Material	Wear	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) are supplemented by surface and volumetric examinations (IWB-3200) for flaw characterization, analytical evaluation, corrective measures, and repairs. Acceptance of the component for continued service is based on volumetric and surface examination, repair, replacement, or analytical evaluation (IWB-3132). ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWA-4000 and IWB-4000, and reexamination in accordance with requirements of IWA-2200. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** The AMP has been effective in managing effects of wear to maintain the intended function of core support pads.	
Plant specific aging management program.	Plant specific aging management program is to be evaluated.	Yes, No AMP
Inservice inspection in conformance with ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1.	***(1) Scope of Program:*** The program relies on inservice inspection (ISI) to manage the effects of wear on the intended function of core support pads. ***(2) Preventive Actions:*** Design requirements, frequent inspection, and timely corrective action prevent or mitigate attrition due to wear of core support pads. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of wear on the intended function of core support pads by detection of loss of material by ISI. Inspection requirements of Table IWB 2500-1, examination category B-N-2 specify visual VT-3 of accessible welds of core support pads. ***(4) Detection of Aging Effects:*** Aging degradation due to wear of the core support pads can not occur without attrition; extent and schedule of inspection assure detection of attrition before the loss of intended function of the core support pads. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI should provide for timely detection of attrition. All accessible welds of core support pads are inspected each inspection period. ***(6) Acceptance Criteria:*** Any wear is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and 35200. Visual examinations that reveal relevant conditions are supplemented by surface and volumetric examinations (IWB-3200) for flaw characterization, analytical evaluation, corrective measures, and repairs. Acceptance of the component for continued service is based on volumetric and surface examination, repair, replacement, or analytical evaluation (IWB-3132). ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWA-4000 and IWB-4000, and reexamination in accordance with requirements of IWA-2200. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

A2. REACTOR VESSEL (Pressurized Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A2.7.1	Penetrations	Instrumenta-tion Tubes (Bottom Head)	SB-166 (Alloy 600), SB-167	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation and Growth	SCC, PWSCC	-
A2.7.2	Penetrations	Head Vent Pipe (Top Head)	SB-166 (Alloy 600), SB-167 with SA312, Type 316 Extensions	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation and Growth	SCC, PWSCC	-
A2.8.1	Pressure Vessel Support	Skirt Support	SA302-Gr B, SA533-Gr B, SA516-Gr70, SA-36	Air	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
A2.8.1 - A2.8.3	Pressure Vessel Support	Skirt Support, Cantilever/ Column Support, Neutron Shield Tank,	SA302-Gr B, SA533-Gr B, SA516-Gr70, SA-36	Air, Leaking Chemically Treated Borated Water	Loss of Material	Boric Acid Corrosion of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** The AMP has been effective in managing effects of wear to maintain the intended function of core support pads.	
Plant specific aging management program.	Plant specific aging management program is to be evaluated.	Yes, No AMP
Plant specific aging management program.	Plant specific aging management program is to be evaluated.	Yes, No AMP
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	*Same as for the effect of Fatigue of Item A2.1.3 closure head stud assembly.*	Yes TLAA
Same as Boric Acid Corrosion of external surfaces of Item A2.1.1 Closure Head Dome.	*Same as Boric Acid Corrosion of external surfaces of Item A2.1.1 Closure Head Dome.*	No

B1. Reactor Vessel Internals (Boiling Water Reactor)

B1.1 Core Shroud, Shroud Head, and Core Plate

B1.1.1 Core Shroud (Upper, Central, Lower)

B1.1.2 Core Plate

B1.1.3 Core Plate Bolts

B1.1.4 Access Hole Cover

B1.1.5 Shroud Support Structure

B1.1.6 Standby Liquid Control Line

B1.1.7 LPCI Coupling

B1.2 Top Guide

B1.3 Core Spray Lines and Spargers

B1.3.1 Core Spray Lines (Headers)

B1.3.2 Spray Ring

B1.3.3 Spray Nozzles

B1.3.4 Thermal Sleeve

B1.4 Jet Pump Assemblies

B1.4.1 Thermal Sleeve

B1.4.2 Inlet Header

B1.4.3 Riser Brace Arm

B1.4.4 Holddown Beams

B1.4.5 Inlet Elbow

B1.4.6 Mixing Assembly

B1.4.7 Diffuser

B1.4.8 Castings

B1. Reactor Vessel Internals (Boiling Water Reactor)

System, Structures, and Components

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

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

System Interfaces

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

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

B1. REACTOR VESSEL INTERNALS (Boiling Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B1.1.1	Core Shroud, Shroud Head and Core Plate	Core Shroud (Upper, Central, Lower)	SS	288°C (550°F) High-Purity Water	Crack Initiation and Growth	Stress Corrosion Cracking (SCC), Intergra-nular Stress Corrosion Cracking (IGSCC), Irradiation Assisted Stress Corrosion Cracking (IASCC)	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. BWRVIP-63, BWRVIP-76. BWRVIP-07. BWRVIP-29 (EPRI TR-103515). *Supporting BWRVIP*: BWRVIP-03. BWRVIP-14. BWRVIP-44. BWRVIP-45. BWRVIP-59. BWRVIP-60. BWRVIP-62. *Operating Experience* NRC IN 94–42. NRC IN 97–17. NRC GL 94–03. NUREG–1544.
B1.1.2, B1.1.3	Core Shroud, Shroud Head and Core Plate	Core Plate, Core Plate Bolts (used in early BWRs)	SS	288°C (550°F) High-Purity Water	Crack Initiation and Growth	SCC, IGSCC, IASCC	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. BWRVIP-03. BWRVIP-25. BWRVIP-29 (EPRI TR-103515). *Supporting BWRVIP*: BWRVIP-07. BWRVIP-14.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Visual inspection (VT–3) is performed according to ASME Section XI, IWB-2500, category B–N–2. Inspection and evaluation are in accordance with BWRVIP-63 and -76 and reinspection of core shrouds are according to approved BWRVIP-07. Coolant water chemistry is monitored and maintained in accordance with the guidelines in BWRVIP-29 (TR-103515) to minimize the potential of crack initiation and growth. *Supporting Documents*: BWRVIP-03 for reactor pressure vessel internals examination guidelines; BWRVIP-14, -59, and -60 for evaluation of crack growth; BWRVIP-44 for weld repair of Ni-alloys; BWRVIP-45 for weldability of irradiated structural components; and BWRVIP-62 for technical basis for inspection relief for internal components with hydrogen injection.	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC, inservice inspection (ISI) to monitor the effects of SCC on the intended function of the components, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** Maintaining high water purity (many BWRs now operate at <0.15 μS/cm^2) reduces susceptibility to SCC. Hydrogen additions are effective in reducing electrochemical potentials in the recirculation piping system, but are less effective in the core region. Noble metal additions through a catalytic action increase the effectiveness of hydrogen additions in the core region. ***(3) Parameters Monitored/Inspected:*** Inspection and flaw evaluation are to be performed in accordance with referenced BWRVIP guideline, as approved by the NRC staff. ***(4) Detection of Aging Effects:*** Degradation due to SCC can not occur without crack initiation and growth; extent and schedule of inspection as delineated in approved BWRVIP guideline will assure detection of cracks before the loss of the intended function of the component. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with applicable, approved BWRVIP guideline is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with applicable, approved BWRVIP guideline. ***(7) Corrective Actions:*** The corrective action proposed by the BWRVIP is as described in staff approved topical report. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Extensive cracking has been observed at both horizontal [NRC Generic Letter (GL) 94–03] and vertical [NRC Information Notice (IN) 97–17] welds. It has affected shrouds fabricated from Type 304 and Type 304L SS, which is generally considered to be more resistant to SCC. Weld regions are most susceptible, although it is not clear whether this is due to sensitization and/or impurities associated with the welds or the high residual stresses in the weld regions. This experience is reviewed in GL 94–03 and NUREG–1544; some experiences with visual inspections are discussed in IN 94–42.	Yes, BWRVIP guidelines*
Visual inspection (VT-3) is performed according to ASME Section XI, IWB-2500, category B–N–2 or enhanced VT-1 and UT inspection guidelines of BWRVIP-03 for reactor pressure vessel internals examination. Inspection and flaw evaluation guidelines for core plate are in accordance with BWRVIP-25. Coolant water chemistry is monitored and maintained in accordance with the guidelines in BWRVIP-29 (TR-103515) to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC, inservice inspection (ISI) to monitor the effects of SCC on the intended function of the components, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** Maintaining high water purity (many BWRs now operate at <0.15 μS/cm^2) reduces susceptibility to SCC. Hydrogen additions are effective in reducing electrochemical potentials in the recirculation piping system, but are less effective in the core region. Noble metal additions through a catalytic action increase the effectiveness of hydrogen additions in the core region. ***(3) Parameters Monitored/ Inspected:*** Inspection and flaw evaluation are to be performed in accordance with	No

*The staff is currently reviewing this program. If the program is approved, no further evaluation will be required.

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
							(continued from previous page) BWRVIP-44. BWRVIP-45. BWRVIP-50. BWRVIP-59. BWRVIP-60. BWRVIP-62. BWRVIP-63. BWRVIP-76. *Operating Experience* NRC GL 94–03. NRC IN 95–17. NUREG–1544.
B1.1.2	Core Shroud, Shroud Head and Core Plate	Core Plate	SS	288°C (550°F) High-Purity Water	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
B1.1.4	Core Shroud, Shroud Head and Core Plate	Access Hole Cover (Welded Covers)	Alloy 600, Alloy 182 Welds	288°C (550°F) High-Purity Water	Crack Initiation and Growth	SCC, IGSCC, IASCC	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. GE SIL 462 S3. BWRVIP-29 (EPRI TR-103515). *Supporting BWRVIP*: BWRVIP-03. BWRVIP-14. BWRVIP-44. BWRVIP-45. BWRVIP-59. BWRVIP-60. BWRVIP-62. *Operating Experience* NRC IN 88-03. NRC IN 92-57.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Supporting Documents: BWRVIP-03 for reactor pressure vessel internals examination guidelines; BWRVIP-63 and -76 for inspection and evaluation of core shrouds; BWRVIP-07 for re inspection of BWR core shrouds; BWRVIP-14, -59, and -60 for evaluation of crack growth; BWRVIP-44 for weld repair of Ni-alloys; BWRVIP-45 for weldability of irradiated structural components; and BWRVIP-62 for technical basis for inspection relief for internal components with hydrogen injection.	*(continued from previous page)* BWRVIP guideline, as approved by the NRC staff. ***(4) Detection of Aging Effects:*** Degradation due to SCC can not occur without crack initiation and growth extent and schedule of inspection as delineated in approved BWRVIP guideline will assure detection of cracks before the loss of the intended function of the component. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with applicable, approved BWRVIP guideline is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with applicable, approved BWRVIP guideline. ***(7) Corrective Actions:*** The corrective action proposed by the BWRVIP is as described in staff approved topical report. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Cracking of the core plate has not been reported, but the creviced regions beneath the plate are difficult to inspect. NRC Information Notice (IN) 95–17 discusses cracking in top guides of the U.S. and overseas BWRs. Related experience in other components is reviewed in NRC GL 94–03 and NUREG–1544.	
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal; check Code limits for allowable cycles (less than 7000 cycles) of thermal stress range. See the Standard Review Plan, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes TLAA
Visual inspection (VT–3) is performed according to ASME Section XI, IWB-2500, category B–N–2. GE Services Information Letter (SIL) 462 Sup. 3 recommends other inspection techniques; implementation of inspection program is plant specific. Coolant water chemistry is monitored and maintained in accordance with the guidelines in BWRVIP-29 (TR-103515) to minimize the potential of crack initiation and growth. *Supporting Documents*: BWRVIP-03 for reactor pressure vessel internals examination guidelines; BWRVIP-14, -59, and -60 for evaluation of crack growth; BWRVIP-44 for weld repair of Ni-alloys; BWRVIP-45 for weldability of irradiated structural components; and BWRVIP-62 for technical basis for inspection relief for internal components with hydrogen injection.	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC, inservice inspection (ISI) to monitor the effects of SCC on the intended function of the components, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** Maintaining high water purity (many BWRs now operate at <0.15 $\mu S/cm^2$) reduces susceptibility to SCC. Hydrogen additions are effective in reducing electrochemical potentials in the recirculation piping system, but are less effective in the core region. Noble metal additions through a catalytic action increase the effectiveness of hydrogen additions in the core region. Also, the susceptibility of Ni-alloys to SCC is evaluated. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of SCC on the intended function by detection and sizing of cracks by inservice inspection (ISI). Table IWB-2500, category B-N-2 specifies visual VT-3 examination of all accessible surfaces of core support structure. Cracking initiates in crevice regions not amenable to visual inspection. GE Services Information Letter (SIL) 462 Sup. 3 recommends other inspection techniques. ***(4) Detection of Aging Effects:*** Degradation due to SCC can not occur without crack initiation and growth. Analysis may be required to assess the impact of	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B1. REACTOR VESSEL INTERNALS (Boiling Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B1.1.4	Core Shroud, Shroud Head and Core Plate	Access Hole Cover (Mechanical Covers)	Alloy 600	288°C (550°F) High-Purity Water	Crack Initiation and Growth	SCC, IGSCC IASCC	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. BWRVIP-29 (EPRI TR-103515). *Supporting BWRVIP*: BWRVIP-03. BWRVIP-14. BWRVIP-44. BWRVIP-45. BWRVIP-59. BWRVIP-60. BWRVIP-62. *Operating Experience* NRC IN 88-03. NRC IN 92-57.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) observed cracking on the function and integrity of the shroud. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with IWB-3520. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWB-3140. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Jet pump BWRs are designed with access holes in the shroud support plate at the bottom of the annulus between the core shroud and the reactor vessel wall. These holes are used for access during construction and are subsequently closed by welding a plate over the hole. Both circumferential (IN 88-03) and radial cracking (IN 92-57) has been observed in the access hole cover.	
Visual inspection (VT–3) is performed according to ASME Section XI, IWB-2500, category B–N–2. Coolant water chemistry is monitored and maintained in accordance with the guidelines in BWRVIP-29 (TR-103515) to minimize the potential of crack initiation and growth. *Supporting Documents*: BWRVIP-03 for reactor pressure vessel internals examination guidelines; BWRVIP-14, -59, and -60 for evaluation of crack growth; BWRVIP-44 for weld repair of Ni-alloys; BWRVIP-45 for weldability of irradiated structural components; and BWRVIP-62 for technical basis for inspection relief for internal components with hydrogen injection.	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC, inservice inspection (ISI) to monitor the effects of SCC on the intended function of the components, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** Maintaining high water purity (many BWRs now operate at <0.15 $\mu S/cm^2$) reduces susceptibility to SCC. Hydrogen additions are effective in reducing electrochemical potentials in the recirculation piping system, but are less effective in the core region. Noble metal additions through a catalytic action increase the effectiveness of hydrogen additions in the core region. Also, the susceptibility of Ni-alloys to SCC is evaluated. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of SCC on the intended function by detection and sizing of cracks by inservice inspection (ISI). Table IWB-2500, category B-N-2 specifies visual VT-3 examination of all accessible surfaces of core support structure. ***(4) Detection of Aging Effects:*** Degradation due to SCC can not occur without crack initiation and growth. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with IWB-3520. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWB-3140. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Access holes are used during construction and are subsequently closed, in some plants, by mechanical covers. Both circumferential (IN 88-03) and radial cracking (IN 92-57) has been observed in the access hole cover.	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B1.1.5	Core Shroud, Shroud Head and Core Plate	Shroud Support Structure (Shroud Support Cylinder, Shroud Support Plate, Shroud Support Legs)	Alloy 600, Alloy 182 Welds	288°C (550°F) High-Purity Water	Crack Initiation and Growth	SCC, IGSCC IASCC	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. BWRVIP--79 (EPRI TR-103515). BWRVIP-38. BWRVIP-76. *Supporting BWRVIP*: BWRVIP-03. BWRVIP-14. BWRVIP-44. BWRVIP-45. BWRVIP-52. BWRVIP-59. BWRVIP-60. BWRVIP-62. *Operating Experience* NRC IN 88-03. NRC IN 92-57.
B1.1.6	Core Shroud, Shroud Head and Core Plate	Standby Liquid Control Line	SS	288°C (550°F) High-Purity Water	Crack Initiation and Growth	SCC, IGSCC IASCC	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. BWRVIP-03. BWRVIP-27. BWRVIP-29 (EPRI TR-103515). *Supporting BWRVIP*: BWRVIP-03. BWRVIP-14. BWRVIP-44. BWRVIP-45. BWRVIP-53. BWRVIP-59. BWRVIP-60. BWRVIP-62.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Visual inspection (VT–3) is performed according to ASME Section XI, IWB-2500, category B–N–2. Inspection and flaw evaluation guidelines are in accordance with BWRVIP-38 for shroud support and, as applicable, BWRVIP-76 for core shroud. Coolant water chemistry is monitored and maintained in accordance with the guidelines in BWRVIP-29 (TR-103515) to minimize the potential of crack initiation and growth. *Supporting Documents*: BWRVIP-03 for reactor pressure vessel internals examination guidelines; BWRVIP-52 for shroud support and vessel bracket repair design criteria; BWRVIP-14, -59, and -60 for evaluation of crack growth; BWRVIP-44 for weld repair of Ni-alloys; BWRVIP-45 for weldability of irradiated structural components; and BWRVIP-62 for technical basis for inspection relief for internal components with hydrogen injection.	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC, inservice inspection (ISI) to monitor the effects of SCC on the intended function of the components, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** Maintaining high water purity (many BWRs now operate at <0.15 $\mu S/cm^2$) reduces susceptibility to SCC. Hydrogen additions are effective in reducing electrochemical potentials in the recirculation piping system, but are less effective in the core region. Noble metal additions through a catalytic action increase the effectiveness of hydrogen additions in the core region. Also, the susceptibility of Ni-alloys to SCC is evaluated. ***(3) Parameters Monitored/ Inspected:*** Inspection and flaw evaluation are to be performed in accordance with referenced BWRVIP guideline, as approved by the NRC staff. ***(4) Detection of Aging Effects:*** Degradation due to SCC can not occur without crack initiation and growth; extent and schedule of inspection as delineated in approved BWRVIP guideline will assure detection of cracks before the loss of the intended function of the component. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with applicable, approved BWRVIP guideline is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with applicable, approved BWRVIP guideline. ***(7) Corrective Actions:*** The corrective action proposed by the BWRVIP is as described in staff approved topical report. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Both circumferential (IN 88-03) and radial cracking (IN 92-57) has been observed in the Ni-alloy components.	Yes, BWRVIP Guideline
Visual inspection (VT–3) is performed according to ASME Section XI, IWB-2500, category B–N–2 or enhanced VT–1 and UT inspection guidelines of BWRVIP-03 for reactor pressure vessel internals examination. Inspection and flaw evaluation guidelines are in accordance with BWRVIP-27 for standby liquid control system/core plate ΔP. Coolant water chemistry is monitored and maintained in accordance with the guidelines in BWRVIP-29 (TR-103515) to minimize the potential of crack initiation and growth. *Supporting Documents*: BWRVIP-03 for reactor pressure vessel internals examination guidelines; BWRVIP-53 for standby liquid control line repair design criteria;	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC, inservice inspection (ISI) to monitor the effects of SCC on the intended function of the components, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** Maintaining high water purity (many BWRs now operate at <0.15 $\mu S/cm^2$) reduces susceptibility to SCC. Hydrogen additions are effective in reducing electrochemical potentials in the recirculation piping system, but are less effective in the core region. Noble metal additions through a catalytic action increase the effectiveness of hydrogen additions in the core region. ***(3) Parameters Monitored/Inspected:*** Inspection and flaw evaluation are to be performed in accordance with referenced BWRVIP guideline, as approved by the NRC staff. ***(4) Detection of Aging Effects:*** Degradation due to SCC can not occur without crack initiation and growth; extent and schedule of inspection as delineated in approved BWRVIP guideline will assure detection of cracks before the loss of the intended function of the component. ***(5) Monitoring and Trending:*** Inspection schedule in	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B1.1.7	Core Shroud, Shroud Head and Core Plate	LPCI Coupling	SS	288°C (550°F) High-Purity Water	Crack Initiation and Growth	SCC, IGSCC IASCC	BWRVIP-42. BWRVIP-29 (EPRI TR-103515). *Supporting BWRVIP*: BWRVIP-03. BWRVIP-14. BWRVIP-44. BWRVIP-45. BWRVIP-56. BWRVIP-59. BWRVIP-60. BWRVIP-62.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Supporting Documents: BWRVIP-14, -59, and -60 for evaluation of crack growth; BWRVIP-44 for weld repair of Ni-alloys; BWRVIP-45 for weldability of irradiated structural components; and BWRVIP-62 for technical basis for inspection relief for internal components with hydrogen injection.	accordance with applicable, approved BWRVIP guideline is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with applicable, approved BWRVIP guideline. ***(7) Corrective Actions:*** The corrective action proposed by the BWRVIP is as described in staff approved topical report. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Cracking has occurred in a number of vessel internal components. Weld regions are most susceptible, although it is not clear whether this is due to sensitization and/or impurities associated with the welds or the high residual stresses in the weld regions.	
Inspection and flaw evaluation guidelines are in accordance with BWRVIP-42 for LPCI coupling. Coolant water chemistry is monitored and maintained in accordance with the guidelines in BWRVIP-29 (TR-103515) to minimize the potential of crack initiation and growth. *Supporting Documents*: BWRVIP-03 for reactor pressure vessel internals examination guidelines; BWRVIP-56 for LPCI coupling repair design criteria; BWRVIP-14, -59, and -60 for evaluation of crack growth; BWRVIP-44 for weld repair of Ni-alloys; BWRVIP-45 for weldability of irradiated structural components; and BWRVIP-62 for technical basis for inspection relief for internal components with hydrogen injection.	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC, inspection and evaluation of BWRVIP-42 as approved by the NRC to monitor the effects of SCC on the intended function of the components, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** Maintaining high water purity (many BWRs now operate at <0.15 $\mu S/cm^2$) reduces susceptibility to SCC. Hydrogen additions are effective in reducing electrochemical potentials in the recirculation piping system, but are less effective in the core region. Noble metal additions through a catalytic action increase the effectiveness of hydrogen additions in the core region. ***(3) Parameters Monitored/Inspected:*** Inspection and flaw evaluation are to be performed in accordance with referenced BWRVIP guideline, as approved by the NRC staff. ***(4) Detection of Aging Effects:*** Degradation due to SCC can not occur without crack initiation and growth; extent and schedule of inspection as delineated in approved BWRVIP guideline will assure detection of cracks before the loss of the intended function of the component. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with applicable, approved BWRVIP guideline is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with applicable, approved BWRVIP guideline. ***(7) Corrective Actions:*** The corrective action proposed by the BWRVIP is as described in staff approved topical report. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Cracking has occurred in a number of vessel internal components. Weld regions are most susceptible, although it is not clear whether this is due to sensitization and/or impurities associated with the welds or the high residual stresses in the weld regions.	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

B1. REACTOR VESSEL INTERNALS (Boiling Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B1.2	Top Guide	Top Guide	SS	288°C (550°F) High-Purity Water	Crack Initiation and Growth	SCC, IGSCC IASCC	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. BWRVIP-03. BWRVIP-26. BWRVIP--79 (EPRI TR-103515). *Supporting BWRVIP*: BWRVIP-14. BWRVIP-44. BWRVIP-45. BWRVIP-50. BWRVIP-59. BWRVIP-60. BWRVIP-62. *Operating Experience* NRC GL 94–03. NRC IN 95–17. NUREG–1544.
B1.2	Top Guide	Top Guide	SS	288°C (550°F) High-Purity Water	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
B1.3.1 - B1.3.4	Core Spray Lines and Spargers	Core Spray Lines (Headers), Spray Rings, Spray Nozzles, Thermal Sleeves	SS	288°C (550°F) High-Purity Water	Crack Initiation and Growth	SCC, IGSCC IASCC	BWRVIP-18. BWRVIP-29 (EPRI TR-103515). *Supporting BWRVIP*: BWRVIP-03. BWRVIP-14. BWRVIP-16. BWRVIP-19. BWRVIP-44. BWRVIP-45. BWRVIP-59. BWRVIP-60. BWRVIP-62.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Visual inspection (VT–3) is performed according to ASME Section XI, IWB-2500, category B–N–2 or enhanced VT–1 and UT inspection guidelines of BWRVIP-03 for reactor pressure vessel internals examination. Inspection and flaw evaluation guidelines are in accordance with BWRVIP-26 for top guide. Coolant water chemistry is monitored and maintained in accordance with the guidelines in BWRVIP-29 (TR-103515) to minimize the potential of crack initiation and growth. *Supporting Documents*: BWRVIP-50 for top guide/core plate repair design criteria; BWRVIP-14, -59, and -60 for evaluation of crack growth; BWRVIP-44 for weld repair of Ni-alloys; BWRVIP-45 for weldability of irradiated structural components; and BWRVIP-62 for technical basis for inspection relief for internal components with hydrogen injection.	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC, inservice inspection (ISI) to monitor the effects of SCC on the intended function of the components, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** Maintaining high water purity (many BWRs now operate at <0.15 $\mu S/cm^2$) reduces susceptibility to SCC. Hydrogen additions are effective in reducing electrochemical potentials in the recirculation piping system, but are less effective in the core region. Noble metal additions through a catalytic action increase the effectiveness of hydrogen additions in the core region. ***(3) Parameters Monitored/Inspected:*** Inspection and flaw evaluation are to be performed in accordance with referenced BWRVIP guideline, as approved by the NRC staff. ***(4) Detection of Aging Effects:*** Degradation due to SCC can not occur without crack initiation and growth; extent and schedule of inspection as delineated in approved BWRVIP guideline will assure detection of cracks before the loss of the intended function of the component. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with applicable, approved BWRVIP guideline is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with applicable, approved BWRVIP guideline. ***(7) Corrective Actions:*** The corrective action proposed by the BWRVIP is as described in staff approved topical report. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** The NRC Information Notice (IN) 95–17 discusses cracking in top guides of US and overseas BWRs. Related experience in other components is reviewed in NRC Generic Letter (GL) 94–03 and NUREG–1544. Cracking has also been observed in the top guide of a Swedish BWR.	No
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	*Same as for the effect of Fatigue on Item B1.1.2 core plate.*	Yes TLAA
Inspection and flaw evaluation guidelines are in accordance with BWRVIP-18 for core spray internals. Coolant water chemistry is monitored and maintained in accordance with the guidelines in BWRVIP-29 (TR-103515) to minimize the potential of crack initiation and growth. *Supporting Documents*: BWRVIP-03 for reactor pressure vessel internals examination guidelines; BWRVIP-16 and -19 for internal core spray piping and sparger replacement and repair design criteria;	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC, inspection and evaluation guidelines of BWRVIP-18 as approved by the NRC staff to monitor the effects of SCC on the intended function of the components, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** Maintaining high water purity (many BWRs now operate at <0.15 $\mu S/cm^2$) reduces susceptibility to SCC. Hydrogen additions are effective in reducing electrochemical potentials in the recirculation piping system, but are less effective in the core region. ***(3) Parameters Monitored/Inspected:*** Inspection and flaw evaluation are to be performed in accordance with referenced BWRVIP guideline, as approved by the NRC staff. ***(4) Detection of Aging Effects:*** Degradation due to	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
							Supporting BWRVIP: BWRVIP-03. BWRVIP-14. BWRVIP-16. BWRVIP-19. BWRVIP-44. BWRVIP-45. BWRVIP-59. BWRVIP-60. BWRVIP-62.
B1.3.1 - B1.3.4	Core Spray Lines and Spargers	Core Spray Lines (Headers), Spray Rings, Spray Nozzles, Thermal Sleeves	SS	288°C (550°F) High-Purity Water	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
B1.4.1, - B1.4.8	Jet Pump Assemblies	Thermal Sleeve, Inlet Header, Riser Brace Arm, Holddown Beams, Inlet Elbow, Mixing Assembly, Diffuser, Castings	Holddown Beams: Ni Alloy (X–750), Castings: Cast Austenitic Stainless Steel (CASS), Others: SS	288°C (550°F) High-Purity Water	Crack Initiation and Growth	SCC, IGSCC IASCC	NRC IEB 80–07. GE SIL 330. GE SIL 605, Rev. 1. BWRVIP-29 (EPRI TR-103515). BWRVIP-41. BWRVIP-28. *Supporting BWRVIP*: BWRVIP-03. BWRVIP-14. BWRVIP-44. BWRVIP-45. BWRVIP-51. BWRVIP-59. BWRVIP-60. BWRVIP-62. *Operating Experience* NRC IN 93-101. NRC IN 97-02.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
(continued from previous page) BWRVIP-14, -59, and -60 for evaluation of crack growth; BWRVIP-44 for weld repair of Ni-alloys; BWRVIP-45 for weldability of irradiated structural components; and BWRVIP-62 for technical basis for inspection relief for internal components with hydrogen injection.	*(continued from previous page)* SCC can not occur without crack initiation and growth; extent and schedule of inspection as delineated in approved BWRVIP guideline will assure detection of cracks before the loss of the intended function of the component. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with applicable, approved BWRVIP guideline is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** In the event cracks are identified, an evaluation is performed in accordance with applicable, approved BWRVIP guideline. ***(7) Corrective Actions:*** Corrective actions in accordance with applicable, approved BWRVIP-16 and BWRVIP-19 guidelines are adequate. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** IEB 80–13 reviews instances of cracking in core spray spargers.	
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	*Same as for the effect of Fatigue on Item B1.1.2 core plate.*	Yes TLAA
Inspection and flaw evaluation guidelines are in accordance with BWRVIP-41 for jet pump assembly. Coolant water chemistry is monitored and maintained in accordance with the guidelines in BWRVIP-29 (TR-103515) to minimize the potential of crack initiation and growth. *Supporting Documents*: BWRVIP-03 for reactor pressure vessel internals examination guidelines; BWRVIP-28 for assessment of jet pump riser elbow to thermal sleeve weld cracking; BWRVIP-51 for jet pump repair design criteria; BWRVIP-14, -59, and -60 for evaluation of crack growth; BWRVIP-44 for weld repair of Ni-alloys; BWRVIP-45 for weldability of irradiated structural components; and BWRVIP-62 for technical basis for inspection relief for internal components with hydrogen injection.	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC, inspection and evaluation guidelines of BWRVIP-41 to monitor the effects of SCC on the intended function of jet pump assembly, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** Maintaining high water purity (many BWRs now operate at <0.15 $\mu S/cm^2$) reduces susceptibility to SCC. Hydrogen additions are effective in reducing electrochemical potentials in the recirculation piping system, but are less effective in the core region. ***(3) Parameters Monitored/Inspected:*** Inspection and flaw evaluation are to be performed in accordance with referenced BWRVIP guideline, as approved by the NRC staff. ***(4) Detection of Aging Effects:*** Degradation due to SCC can not occur without crack initiation and growth; extent and schedule of inspection as delineated in approved BWRVIP guideline will assure detection of cracks before the loss of the intended function of the component. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with applicable, approved BWRVIP guideline is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation in jet pump operation is evaluated in accordance with applicable, approved BWRVIP guideline. ***(7) Corrective Actions:*** The corrective action proposed by the BWRVIP is as described in staff approved topical report. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for	Yes, BWRVIP guidelines*

*The staff is currently reviewing this program. If the program is approved, no further evaluation will be required.

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B1.4.1 - B1.4.8	Jet Pump Assemblies	Thermal Sleeve, Inlet Header, Riser Brace Arm, Holddown Beams, Inlet Elbow, Mixing Assembly, Diffuser, Castings	Holddown Beams: Ni Alloy (X–750), Others: SS	288°C (550°F) High-Purity Water	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
B1.4.8	Jet Pump Assemblies	Castings	CASS	288°C (550°F) High-Purity Water	Loss of Fracture Toughness	Thermal Aging and Neutron Irradiation Embrittle–ment	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. Letter from Christopher I. Grimes (NRC) to Douglas J. Walters (NEI) dated 5/19/2000.
B1.4.9	Jet Pump Assemblies	Jet Pump Sensing Line	SS	288°C (550°F) High-Purity Water	Crack Initiation and Growth	Unantici-pated Cyclic Loading	-
B1.5.1	Fuel Supports & Control Rod Drive (CRD) Assemblies	Orificed Fuel Support	CASS	288°C (550°F) High-Purity Water	Loss of Fracture Toughness	Thermal Aging and Neutron Irradiation Embrittle–ment	*Same as for the effect of Thermal Aging and Neutron Irradiation Embrittlement on Item B1.4.8 jet pump assemblies castings.*
B1.5.1	Fuel Supports & CRD Assemblies	Orificed Fuel Support	SS, CASS	288°C (550°F) High-Purity Water	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) the period of license renewal. ***(10) Operating Experience:*** Instances of cracking have occurred in jet pump assembly (NRC IEB 80-07), hold-down beam [NRC Information Notice (IN) 93-101), and jet pump riser pipe elbows (NRC IN 97-02).	
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	*Same as for the effect of Fatigue on Item B1.1.2 core plate.*	Yes TLAA
The reactor vessel internals receive a visual inspection in accordance with Category B-N-3 of Subsection IWB, ASME Section XI. This inspection is not sufficient to detect the effects of loss of fracture toughness due to thermal aging and neutron irradiation embrittlement of cast austenitic stainless steel (CASS) reactor vessel internals. An acceptable AMP consists of the following: Determination of the susceptibility of CASS components to thermal aging embrittlement based on casting method, Mo content, and percent ferrite. For “potentially susceptible” components, based on the neutron fluence of the component, implement either a supplemental examination of the affected components as part of the applicant's 10-year inservice inspection (ISI) program during the license renewal term or a component-specific evaluation to determine the susceptibility to loss of fracture toughness.	For evaluation and technical basis of the 10 elements of the AMP, see Chapter XI.M2 "Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)".	No
Plant specific aging management program should be implemented.	Plant specific aging management program is to be evaluated.	Yes, plant specific AMP
Same as for the effect of Thermal Aging and Neutron Irradiation Embrittlement on Item B1.4.8 jet pump assemblies castings.	*Same as for the effect of Thermal Aging and Neutron Irradiation Embrittlement on Item B1.4.8 jet pump assemblies castings.*	No
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	*Same as for the effect of Fatigue on Item B1.1.2 core plate.*	Yes TLAA

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B1.5.2	Fuel Supports & CRD Assemblies	CRD Housing	SS	Up to 288°C, (550°F) Reactor Coolant Water	Crack Initiation and Growth	SCC, IGSCC	BWRVIP-49. BWRVIP-29 (EPRI TR-103515). *Supporting BWRVIP:* BWRVIP-58. BWRVIP-14. BWRVIP-27. BWRVIP-53. BWRVIP-59. BWRVIP-60. BWRVIP-62.
B1.6.1 - B1.6.3	Instrument Housings	Intermediate Range Monitor (IRM) Dry Tubes, Low Power Range Monitor (LPRM) Dry Tubes, Source Range Monitor (SRM) Dry Tubes	SS	288°C (550°F) High-Purity Water	Crack Initiation and Growth	SCC, IGSCC IASCC	BWRVIP-49. BWRVIP-29 (EPRI TR-103515). *Supporting BWRVIP*: BWRVIP-03. BWRVIP-14. BWRVIP-44. BWRVIP-45. BWRVIP-57. BWRVIP-59. BWRVIP-60. BWRVIP-62. GE SIL 409 R1.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Inspection and flaw evaluation guidelines are in accordance with BWRVIP-49 for instrument penetration. Coolant water chemistry is monitored and maintained in accordance with the guidelines in BWRVIP-29 (TR-103515) to minimize the potential of crack initiation and growth. *Supporting Documents*: BWRVIP-58 for CRD internal access weld repair; BWRVIP-14, -59, and -60 for evaluation of crack growth; BWRVIP-62 for technical basis for inspection relief for internal components with hydrogen injection; and BWRVIP-53 for standby liquid control line repair design criteria.	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC, and inspection and flaw evaluation guidelines of BWRVIP-49 as approved by the NRC staff for instrument penetration, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** Maintaining high water purity (many BWRs now operate at <0.15 $\mu S/cm^2$) reduces susceptibility to SCC. Hydrogen additions are effective in reducing electrochemical potentials in the recirculation piping system, but are less effective in the core region. ***(3) Parameters Monitored/ Inspected:*** Inspection and flaw evaluation are to be performed in accordance with referenced BWRVIP guideline, as approved by the NRC staff. ***(4) Detection of Aging Effects:*** Degradation due to SCC can not occur without crack initiation and growth; extent and schedule of inspection as delineated in approved BWRVIP guideline will assure detection of cracks before the loss of the intended function of the component. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with applicable, approved BWRVIP guideline is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation in jet pump operation is evaluated in accordance with applicable, approved BWRVIP guideline. ***(7) Corrective Actions:*** The corrective action proposed by the BWRVIP is as described in staff approved topical report. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Cracking has been observed in CRD mechanism housing.	No
Inspection and flaw evaluation guidelines are in accordance with BWRVIP-49 for instrument penetration. Coolant water chemistry is monitored and maintained in accordance with the guidelines in BWRVIP-29 (TR-103515) to minimize the potential of crack initiation and growth. *Supporting Documents*: BWRVIP-03 for reactor pressure vessel internals examination guidelines; BWRVIP-57 for instrument penetration repair design criteria; BWRVIP-14, -59, and -60 for evaluation of crack growth; BWRVIP-44 for weld repair of Ni-alloys; BWRVIP-45 for weldability of irradiated structural components; and BWRVIP-62 for technical basis for inspection relief for internal components with hydrogen injection.	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC, and inspection and flaw evaluation guidelines of BWRVIP-49 as approved by the NRC staff for instrument penetration, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** Based on GE SIL 409 Rev. 1 replacement of existing tubes with those fabricated from more IASCC-resistant materials and crevice free design. Maintaining high water purity (many BWRs now operate at <0.15 $\mu S/cm^2$) reduces susceptibility to SCC. Hydrogen additions are effective in reducing electrochemical potentials in the recirculation piping system, but are less effective in the core region. ***(3) Parameters Monitored/Inspected:*** Inspection and flaw evaluation are to be performed in accordance with referenced BWRVIP guideline, as approved by the NRC staff. ***(4) Detection of Aging Effects:*** Degradation due to SCC can not occur without crack initiation and growth; extent and schedule of inspection as delineated in approved BWRVIP guideline will assure detection of cracks before the loss of the intended function of the component. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with applicable, approved BWRVIP guideline is	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B1. REACTOR VESSEL INTERNALS (Boiling Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B1.6.1 - B1.6.3	Instrument Housings	IRM Dry Tubes, LPRM Dry Tubes, SRM Dry Tubes	SS	288°C (550°F) High-Purity Water	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
B1.7	Separator Support Ring	-	SS	288°C (550°F) High-Purity Water	Crack Initiation and Growth	SCC, IGSCC	-

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Crack indications are evaluated in accordance with applicable, approved BWRVIP guideline. ***(7) Corrective Actions:*** Corrective actions in accordance with applicable, approved BWRVIP-57 guidelines are adequate. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Cracking of dry tubes has been observed at 14 or more BWRs. The cracking is intergranular and has been observed in dry tubes without apparent sensitization suggesting that irradiation assisted SCC (IASCC) may also play a role in the cracking.	
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	*Same as for the effect of Fatigue on Item B1.1.2 core plate.*	Yes TLAA
Plant specific aging management program should be implemented.	Plant specific aging management program is to be evaluated.	Yes, plant specific AMP

B2. Reactor Vessel Internals (PWR) - Westinghouse

B2.1 Upper Internals Assembly

B2.1.1 Upper Support Plate

B2.1.2 Upper Support Column

B2.1.3 Upper Support Column Bolts

B2.1.4 Upper Core Plate

B2.1.5 Upper Core Plate Alignment Pins

B2.1.6 Fuel Alignment Pins

B2.1.7 Hold-Down Spring

B2.2 RCCA Guide Tube Assemblies

B2.2.1 RCCA Guide Tubes

B2.2.2 RCCA Guide Tube Bolts

B2.2.3 RCCA Guide Tube Support Pins

B2.3 Core Barrel

B2.3.1 Core Barrel

B2.3.2 Core Barrel Flange

B2.3.3 Core Barrel Outlet Nozzles

B2.3.4 Thermal Shield

B2.4 Baffle/Former Assembly

B2.4.1 Baffle and Former Plates

B2.4.2 Baffle/Former Bolts

B2.5 Lower Internal Assembly

B2.5.1 Lower Core Plate

B2.5.2 Fuel Alignment Pins

B2.5.3 Lower Support Forging or Casting

B2.5.4 Lower Support Plate Columns

B2.5.5 Lower Support Plate Columns Bolts

B2.5.6 Radial Support Keys and Clevis Inserts

B2.5.7 Clevis Insert Bolts

B2.6 Instrumentation Support Structure

B2.6.1 Flux Thimble Guide Tubes

B2.6.2 Flux Thimbles

B2. Reactor Vessel Internals (PWR) - Westinghouse

System, Structures, and Components

The system, structures, and components included in this table comprise the Westinghouse pressurized water reactor (PWR) reactor vessel internals and consist of the Upper Internals Assembly, Rod Control Cluster Assemblies (RCCA) Guide Tube Assemblies, Core Barrel, Baffle/Former Assembly, Lower Internal Assembly, and Instrument Support Structure. Based on the Nuclear Regulatory Commission Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all structures and components in the reactor vessel are classified as Group A or B Quality Standards.

System Interfaces

The systems that interface with the reactor vessel internals include the reactor pressure vessel (Table IV.A2) and reactor coolant system and connected lines (Table IV.C2).

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B2.1.1, B2.1.4, B2.1.7	Upper Internals Assembly	Upper Support Plate, Upper Core Plate, Hold-Down Spring	Stainless Steel (SS)	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation and Growth	Stress Corrosion Cracking (SCC), Irradiation Assisted Stress Corrosion Cracking (IASCC)	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. EPRI TR-105714 Rev. 3 or later revisions/update. *Operating Experience* NRC IN 84-18. NRC IN 98-11.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Visual inspection (VT–3) is performed according to Category B–N–3 of Subsection IWB, ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a). Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 (Rev. 3 or later revisions or update) to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC or IASCC, inservice inspection (ISI) to detect cracks, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** PWR operating chemistry limits the halogens, sulfates, and oxygen in the primary water to less than 0.05, 0.05, and 0.005 ppm, respectively, during operation, which greatly reduces susceptibility to SCC or IASCC. However, introduction of oxygen can occur during shutdown and potential exists for the formation of more aggressive chemistry conditions by radiolysis in creviced regions or in low–flow stagnant regions. Also, for sufficiently high fluence levels, IASCC can occur even in low oxygen environments. The AMP must rely upon ISI in accordance with ASME Section XI to detect possible degradation. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of SCC or IASCC on the intended function of the component by detection and sizing of cracks by ISI. Table IWB-2500, examination category B-N-3 specifies visual VT-3 examination of all accessible surfaces of reactor core support structures that can be removed from the reactor vessel. ***(4) Detection of Aging Effects:*** Degradation due to SCC or IASCC can not occur without crack initiation and growth. VT–3 is not adequate to detect tight cracks. The inspection technique, including the reliability in detecting the features of interest (crack appearance and size), should be specified. Also, creviced regions are difficult to inspect visually and supplementary UT or other nondestructive examinations may be necessary As an alternate to enhanced inspection, perform a component-specific evaluation including a mechanical loading assessment to determine the maximum tensile loading on the component during ASME Code Level A, B, C, and D conditions. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with IWB-3520. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWB-3140. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Although SS components in PWRs have generally not been found to be affected by SCC because of low dissolved oxygen levels and control of primary water chemistry, potential for SCC exists from inadvertent introduction of contaminants into the primary coolant system (IN 84–18), from the introduction of relatively high levels of dissolved oxygen during shutdown, or from aggressive chemistries that may develop in creviced regions. Cracking has occurred in SS baffle former bolts in a number of foreign plants (IN 98-11) and has now been observed in US plants. The mechanism of this particular cracking has not yet been resolved.	Yes, detection of aging effects should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B2.1.1, B2.1.4, B2.1.7	Upper Internals Assembly	Upper Support Plate, Upper Core Plate, Hold-Down Spring	SS	Chemically Treated Borated Water up to 340°C (644°F)	Changes in Dimensions	Void Swelling	-
B2.1.1, B2.1.4, B2.1.7	Upper Internals Assembly	Upper Support Plate, Upper Core Plate, Hold-Down Spring	SS	Chemically Treated Borated Water up to 340°C (644°F)	Cumula-tive fatigue damage	Fatigue	Design Code of record or later approved Codes.
B2.1.7	Upper Internals Assembly	Hold-Down Spring	SS	Chemically Treated Borated Water up to 340°C (644°F)	Loss of Preload	Stress Relaxation	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
The reactor vessel internals receive a visual inspection (VT-3) according to Category B-N-3 of Subsection IXB, ASME Section XI. This inspection is not sufficient to detect the effects of changes in dimension due to void swelling. An acceptable alternative AMP consists of the following: 1. Participation in industry programs to address the significance of change in dimensions due to void swelling. 2. Implementation of an inspection program should the results of the industry programs indicate the need for such inspections.	Plant specific aging management program is to be evaluated. The applicant must provide the basis for concluding that void swelling is not an issue for the component or must provide an AMP. The applicant should address loss of ductility associated with swelling.	Yes, plant specific AMP
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes TLAA
Visual inspection (VT–3) is performed according to Category B–N–3 of Subsection IWB, ASME Section XI to monitor the relevant conditions of degradation, and loose part monitoring and/or neutron noise monitoring (excore detectors) to detect core barrel motion.	***(1) Scope of Program:*** The program includes inservice inspection (ISI) to monitor the relevant conditions of degradation, loose part monitoring and/or neutron noise monitoring to detect core barrel motion, and repair and/or replacement as needed to maintain the capability of the component to perform its intended function. ***(2) Preventive Actions:*** No practical preventative actions are possible. ***(3) Parameters Monitored/Inspected:*** The program includes ISI to monitor the relevant conditions of degradation such as loose, cracked, or missing bolts or fasteners, and wear, and neutron noise monitoring (excore detectors) to detect core barrel motion. Table IWB-2500, category B-N-3 specifies visual VT-3 examination of all accessible surfaces of reactor core support structures that can be removed from the reactor vessel. ***(4) Detection of Aging Effects:*** Visual VT-3 inspection can reveal indications of degradation due to stress relaxation such as loose or missing parts, wear, debris, or the loss of integrity at bolted or welded connections, and neutron noise monitoring can detect physical displacement and motion of reactor internals. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 is adequate for timely detection of degradation. ***(6) Acceptance Criteria:*** Any relevant condition such as loose, missing, cracked, or fractured parts, bolting, or fasteners, is evaluated in accordance with IWB-3500. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWB-3140; the options include supplemental surface and/or volumetric examination to further characterize the condition, corrective measures (e.g., re-establish the preload) or repairs,	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

B2. REACTOR VESSEL INTERNALS (PWR) - Westinghouse

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B2.1.2	Upper Internals Assembly	Upper Support Column	SS, Cast Austenitic SS (CASS)	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation and Growth	SCC, IASCC	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. EPRI TR-105714 Rev. 3 or later revisions/update.
B2.1.2	Upper Internals Assembly	Upper Support Column	SS, CASS	Chemically Treated Borated Water up to 340°C (644°F)	Changes in Dimensions	Void Swelling	-
B2.1.2	Upper Internals Assembly	Upper Support Column (only CASS portions)	CASS	Chemically Treated Borated Water up to 340°C (644°F) Neutron Fluence of greater than 10^{17} n/cm^2 (E>1 MeV)	Loss of fracture toughness	Thermal Aging and Neutron Irradiation Embrittle–ment	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. Letter from Christopher I. Grimes (NRC) to Douglas J. Walters (NEI) dated 5/19/2000.
B2.1.2	Upper Internals Assembly	Upper Support Column	SS, CASS	Chemically Treated Borated Water up to 340°C (644°F)	Cumula-tive fatigue damage	Fatigue	Design Code of record or later approved Codes.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) or replacement. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** There are no reports of stress relaxation producing damage in reactor vessel internals.	
Same as for the effect of SCC or IASCC on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.	*Same as for the effect of SCC or IASCC on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.*	Yes, detection of aging effects should be further evaluated
Same as for the effect of Void Swelling on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down ring.	*Same as for the effect of Void Swelling on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down ring.*	Yes, plant specific AMP
The reactor vessel internals receive a visual inspection in accordance with Category B-N-3 of Subsection IXB, ASME Section XI. This inspection is not sufficient to detect the effects of loss of fracture toughness due to thermal aging and neutron irradiation embrittlement of cast austenitic stainless steel (CASS) reactor vessel internals. An acceptable AMP consists of the following: Determination of the susceptibility of CASS components to thermal aging embrittlement based on casting method, Mo content, and percent ferrite. For “potentially susceptible” components, based on the neutron fluence of the component, implement either a supplemental examination of the affected components as part of the applicant's 10-year inservice inspection (ISI) program during the license renewal term or a component-specific evaluation to determine the susceptibility to loss of fracture toughness.	For evaluation and technical basis of the 10 elements of the AMP, see Chapter XI.M2 "Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)".	No
Same as for the effect of Fatigue on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.	*Same as for the effect of Fatigue on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.*	Yes TLAA

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B2.1.3, B2.1.5, B2.1.6	Upper Internals Assembly	Upper Support Column Bolts, Upper Core Plate Alignment Pins, Fuel Alignment Pins	SS, Ni Alloy	Chemically Treated Borated Water at temperatures up to 340°C (644°F)	Crack Initiation and Growth	SCC, IASCC	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. EPRI TR-105714 Rev. 3 or later revisions/update. Operating Experience NRC IN 84-18.
B2.1.3, B2.1.5, B2.1.6	Upper Internals Assembly	Upper Support Column Bolts, Upper Core Plate Alignment Pins, Fuel Alignment Pins	SS, Ni Alloy	Chemically Treated Borated Water up to 340°C (644°F)	Changes in Dimensions	Void Swelling	-

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Visual inspection (VT–3) is performed according to Category B–N–3 of Subsection IWB, ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a). Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 (Rev. 3 or later revisions or update) to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC and IASCC, inservice inspection (ISI) to detect cracks, and repair and/or replacement as needed to maintain the capability of the component to perform its intended function. ***(2) Preventive Actions:*** PWR operating chemistry limits the halogens, sulfates, and oxygen in the primary water to less than 0.05, 0.05, and 0.005 ppm, respectively, during operation. However, introduction of oxygen can occur during shutdown and potential exists for the formation of more aggressive chemistry conditions by radiolysis in creviced regions or in low–flow stagnant regions. Also, for sufficiently high fluence levels, IASCC can occur even in low oxygen environments. The AMP must rely upon ISI in accordance with ASME Section XI to detect possible degradation. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of SCC and IASCC on the intended function of the component by detection and sizing of cracks by inservice inspection (ISI). Table IWB-2500, category B-N-3 specifies visual VT-3 examination of all accessible surfaces of reactor internals. ***(4) Detection of Aging Effects:*** Degradation due to SCC can not occur without crack initiation and growth. Historically the VT-3 visual examinations have not identified bolt cracking because cracking occurs at the juncture of the bolt head and shank, which is not accessible for visual inspection. Creviced and other inaccessible regions are difficult to inspect visually. Supplementary UT examinations may be necessary. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with IWB-3520. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWB-3140. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Although stainless steel components in PWRs have generally not been found to be affected by SCC because of low dissolved oxygen levels and control of primary water chemistry, potential for SCC exists from inadvertent introduction of contaminants into the primary coolant system (IN 84–18), from the introduction of relatively high levels of dissolved oxygen during shutdown, or from aggressive chemistries that may develop in creviced regions.	Yes, detection of aging effects should be further evaluated
Same as for the effect of Void Swelling on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.	*Same as for the effect of Void Swelling on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.*	Yes, plant specific AMP

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B2.1.3	Upper Internals Assembly	Upper Support Column Bolts	SS, Ni Alloy	Chemically Treated Borated Water up to 340°C (644°F)	Loss of Preload	Stress Relaxation	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Visual inspection (VT–3) is performed according to Category B–N–3 of Subsection IWB, ASME Section XI to monitor the relevant conditions of degradation including loose part monitoring.	***(1) Scope of Program:*** The program includes inservice inspection (ISI) to monitor the relevant conditions of degradation including loose part monitoring, and repair and/or replacement as needed to maintain the capability of the component to perform its intended function. ***(2) Preventive Actions:*** No practical preventative actions are possible. ***(3) Parameters Monitored/Inspected:*** The program includes ISI to monitor the relevant conditions of degradation such as loose, cracked, or missing bolts or fasteners, and wear. Table IWB-2500, category B-N-3 specifies visual VT-3 examination of all accessible surfaces of reactor core support structures that can be removed from the reactor vessel. ***(4) Detection of Aging Effects:*** Visual VT-3 inspection can reveal indications of degradation due to stress relaxation such as loose or missing parts, wear, debris, or the loss of integrity at bolted or welded connections. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 is adequate for timely detection of degradation. ***(6) Acceptance Criteria:*** Any relevant condition such as loose, missing, cracked, or fractured parts, bolting, or fasteners, is evaluated in accordance with IWB-3500. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWB-3140; the options include supplemental surface and/or volumetric examination to further characterize the condition, corrective measures (e.g., re-establish the preload) or repairs, or replacement. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** There are no reports of stress relaxation producing damage in reactor vessel internals.	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B2.1.5	Upper Internals Assembly	Upper Core Plate Alignment Pins	SS, Ni Alloy	Chemically Treated Borated Water up to 340°C (644°F)	Loss of Material	Wear	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.
B2.1.6	Upper Internals Assembly	Fuel Alignment Pins	SS, Ni Alloy	Chemically Treated Borated Water up to 340°C (644°F)	Cumula-tive fatigue damage	Fatigue	Design Code of record or later approved Codes.
B2.2.1	RCCA Guide Tube Assemblies	RCCA Guide Tubes	SS	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation and Growth	SCC, IASCC	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. EPRI TR-105714 Rev. 3 or later revisions/update.
B2.2.1	RCCA Guide Tube Assemblies	RCCA Guide Tubes	SS	Chemically Treated Borated Water up to 340°C (644°F)	Changes in Dimensions	Void Swelling	-

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Visual inspection (VT–3) is performed according to Category B–N–3 of Subsection IWB, ASME Section XI.	***(1) Scope of Program:*** The program includes inservice inspection (ISI) to monitor the condition of components and assure that they can perform their intended function. ***(2) Preventive Actions:*** No practical preventative actions are possible. ***(3) Parameters Monitored/Inspected:*** The AMP program monitors the effects of wear on the intended function by inservice inspection (ISI) to determine the general mechanical and structural condition of components. ***(4) Detection of Aging Effects:*** Inspection can reveal indications of degradation due to wear such as the verification of clearances, settings, physical displacements, loose or missing parts, debris, corrosion, wear, erosion, or the loss of integrity at bolted or welded connections. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 is adequate for timely detection of wear. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with IWB-3520. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWB-3140. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** To date aging management program has been effective in managing the effects of wear on reactor internals.	No
Same as for the effect of Fatigue on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.	*Same as for the effect of Fatigue on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.*	Yes TLAA
Same as for the effect of Stress Corrosion Cracking on Items B2.1.1, upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.	*Same as for the effect of Stress Corrosion Cracking on Items B2.1.1, upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.*	Yes, detection of aging effects should be further evaluated
Same as for the effect of Void Swelling on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.	*Same as for the effect of Void Swelling on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.*	Yes, plant specific AMP

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B2.2.1	RCCA Guide Tube Assemblies	RCCA Guide Tubes	SS	Chemically Treated Borated Water up to 340°C (644°F)	Loss of Material	Wear	-
B2.2.1	RCCA Guide Tube Assemblies	RCCA Guide Tubes	SS	Chemically Treated Borated Water up to 340°C (644°F)	Cumula-tive fatigue damage	Fatigue	Design Code of record or later approved Codes.
B2.2.2, B2.2.3	RCCA Guide Tube Assemblies	RCCA Guide Tube Bolts, RCCA Guide Tube Support Pins	SS, Ni Alloy	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation and Growth	SCC, IASCC	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. EPRI TR-105714 Rev. 3 or later revisions/update.
B2.2.2, B2.2.3	RCCA Guide Tube Assemblies	RCCA Guide Tube Bolts, RCCA Guide Tube Support Pins	SS, Ni Alloy	Chemically Treated Borated Water up to 340°C (644°F)	Changes in Dimensions	Void Swelling	-
B2.2.2, B2.2.3	RCCA Guide Tube Assemblies	RCCA Guide Tube Bolts, RCCA Guide Tube Support Pins	SS, Ni Alloy	Chemically Treated Borated Water up to 340°C (644°F)	Cumula-tive fatigue damage	Fatigue	Design Code of record or later approved Codes.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Performance monitoring by rod drop time testing to detect wear of the guide tube cards.	***(1) Scope of Program:*** The program relies on performance monitoring by rod drop time testing to detect wear of the guide tube cards and replacement as needed to ensure that the intended function will be maintained during the extended period of operation. ***(2) Preventive Actions:*** Wear between control rods and guide tubes results from the axial sliding that occurs during insertion and withdrawals, and also from the transverse motions caused by flow-induced vibrations. Recommendations for plant operation minimize the rate of wear and extend the life of the component. ***(3) Parameters Monitored/Inspected:*** Current performance monitoring programs, i.e., rod drop time testing, provide indications of any wear of the guide tube cards. ***(4) Detection of Aging Effects:*** Performance monitoring program provides indications of degradation due to wear. ***(5) Monitoring and Trending:*** Rod drop testing is performed each cycle. ***(6) Acceptance Criteria:*** Any indication of wear is evaluated . ***(7) Corrective Actions:*** Replacement as needed; guide tube assembly is easily replaceable. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** To date aging management program has been effective in managing the effects of wear on the guide tubes.	No
Same as for the effect of Fatigue on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.	*Same as for the effect of Fatigue on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.*	Yes TLAA
Same as for the effect of SCC or IASCC on Items B2.1.3 upper support column bolts, B2.1.5 upper core plate alignment pins, and B2.1.6 fuel alignment pins.	*Same as for the effect of SCC or IASCC on Items B2.1.3 upper support column bolts, B2.1.5 upper core plate alignment pins, and B2.1.6 fuel alignment pins.*	Yes, detection of aging effects should be further evaluated
Same as for the effect of Void Swelling on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.	*Same as for the effect of Void Swelling on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.*	Yes, plant specific AMP
Same as for the effect of Fatigue on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.	*Same as for the effect of Fatigue on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.*	Yes TLAA

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B2.3.1 - B2.3.4	Core Barrel	Core Barrel (CB), Upper CB Flange, CB Outlet Nozzles, Thermal Shield	SS	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation and Growth	SCC, IASCC	*Same as for the effect of Stress Corrosion Cracking on Items B2.1.1-B2.1.7.*
B2.3.1 - B2.3.4	Core Barrel	Core Barrel (CB), Upper CB Flange, CB Outlet Nozzles, Thermal Shield	SS	Chemically Treated Borated Water up to 340°C (644°F)	Changes in Dimensions	Void Swelling	-
B2.3.1 - B2.3.4	Core Barrel	Core Barrel (CB), Upper CB Flange, CB Outlet Nozzles, Thermal Shield	SS	Chemically Treated Borated Water up to 340°C (644°F) Neutron Fluence of greater than 10^{17} n/cm^2 (E>1 MeV)	Loss of Fracture Toughness	Neutron Irradiation Embrittle–ment	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.
B2.3.2	Core Barrel	CB Flange (Upper Flange)	SS	Chemically Treated Borated Water up to 340°C (644°F)	Loss of Material	Wear	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Same as for the effect of Stress Corrosion Cracking on Items B2.1.1, upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.	*Same as for the effect of Stress Corrosion Cracking on Items B2.1.1, upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.*	Yes, detection of aging effects should be further evaluated
Same as for the effect of Void Swelling on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.	*Same as for the effect of Void Swelling on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.*	Yes, plant specific AMP
Visual inspection (VT–3) is performed according to Category B-N-2/B–N–3 of Subsection IWB, ASME Section XI.	***(1) Scope of Program:*** The program includes inservice inspection (ISI) to detect cracking and/or failure, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** No practical preventative actions are possible. Stainless steels are susceptible to embrittlement under neutron irradiation. Fracture toughness will depend strongly on the fluence on a particular component. Components can be screened out if the maximum tensile loading on the component under ASME Code Level A, B, C, and D conditions is sufficiently low. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of neutron irradiation embrittlement on the intended function of the component by detection and sizing of cracks by ISI. Table IWB-2500 specifies visual VT-3 examination according to categories B-N-2 for reactor internals and B-N-3 for all accessible surfaces of reactor core support structure that can be removed from the reactor vessel. ***(4) Detection of Aging Effects:*** Loss of fracture toughness is of consequence only if cracks exist. Cracking is expected to initiate at the surface and should be detectable by ISI except for some crevice regions. VT–3 is not adequate to detect tight cracks. Also, creviced regions are difficult to inspect visually and supplementary UT or other nondestructive examinations may be needed. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with IWB-3520. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWB-3140. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** No instances of internals degradation have been recorded that have been attributed to irradiation embrittlement.	Yes, parameters monitored/inspected and detection of aging effects should be further evaluated
Same as for the effect of Wear on Item B2.1.5 upper core plate alignment pins.	*Same as for the effect of Wear on Item B2.1.5 upper core plate alignment pins.*	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B2.3.1 - B2.3.4	Core Barrel	Core Barrel (CB), Upper CB Flange, CB Outlet Nozzles, Thermal Shield	SS	Chemically Treated Borated Water up to 340°C (644°F)	Cumula-tive fatigue damage	Fatigue	Design Code of record or later approved Codes.
B2.4.1	Baffle/Former Assembly	Baffle and Former Plates	SS	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation and Growth	SCC, IASCC	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. EPRI TR-105714 Rev. 3 or later revisions/update.
B2.4.1	Baffle/Former Assembly	Baffle and Former Plates	SS	Chemically Treated Borated Water up to 340°C (644°F)	Changes in Dimensions	Void Swelling	-
B2.4.2	Baffle/Former Assembly	Baffle/Former Bolts	SS (Type 347 & cold-worked Type 316)	Chemically Treated Borated Water up to 340°C (644°F) and high fluence (>10 dpa or 7 x 10^{21} n/cm^2 E>1 MeV)	Crack Initiation and Growth	SCC, IASCC	-
B2.4.2	Baffle/Former Assembly	Baffle/Former Bolts	SS (Type 347 & cold-worked Type 316)	Chemically Treated Borated Water up to 340°C (644°F) and high fluence	Changes in Dimensions	Void Swelling	-
B2.4.1	Baffle/Former Assembly	Baffle and Former Plates	SS	Chemically Treated Borated Water up to 340°C Fluence > 10^{17} n/cm^2 (E>1 MeV)	Loss of Fracture Toughness	Neutron Irradiation Embrittle–ment	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Same as for the effect of Fatigue on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.	*Same as for the effect of Fatigue on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.*	Yes TLAA
Same as for the effect of Stress Corrosion Cracking on Items B2.1.1, upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.	*Same as for the effect of Stress Corrosion Cracking on Items B2.1.1, upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.*	Yes, detection of aging effects should be further evaluated
Same as for the effect of Void Swelling on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.	*Same as for the effect of Void Swelling on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.*	Yes, plant specific AMP
Historically the VT-3 visual examinations have not identified baffle/former bolt cracking because cracking occurs at the juncture of the bolt head and shank, which is not accessible for visual inspection. However, recent UT examinations of the baffle/former bolts have identified cracking in several plants. The industry is currently addressing the issue of baffle bolt cracking in the PWR Materials Reliability Project, Issues Task Group (ITG) activities to determine, develop, and implement the necessary steps and plans to manage the applicable aging effects on a plant-specific basis.	Plant-specific aging management program is to be evaluated.	Yes, plant specific AMP
Same as for the effect of Void Swelling on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.	Plant specific aging management program is to be evaluated. The applicant must provide the basis for concluding that void swelling is not an issue for the component or must provide an AMP. The applicant must address the effects of material dimensional changes due to void swelling with regard to the potential of overloading the baffle bolts. Address the effects of the growth of the baffle plate material that is captured between the baffle bolt head and the edge of the former plate.	Yes, plant specific AMP
Visual inspection (VT–3) is performed according to Category B-N-2/B–N–3 of Subsection IWB, ASME Section XI.	*Same as for the effect of Neutron Irradiation Embrittlement on Items B2.3.1 core barrel, B2.3.2 upper core barrel flange, B2.3.3 core barrel nozzles, B2.3.4 and thermal shield.*	Yes, parameters monitored/ inspected and detection of aging effects should be further evaluated

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2. REACTOR VESSEL INTERNALS (PWR) - Westinghouse

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B2.4.2	Baffle/Former Assembly	Baffle/Former Bolts	SS (Type 347 & cold-worked Type 316)	Treated Borated Water up to 340°C, Fluence > 10^{17} n/cm^2 (E>1 MeV)	Loss of Fracture Toughness	Neutron Irradiation Embrittle–ment	-
B2.4.1, B2.4.2	Baffle/Former Assembly	Baffle and Former Plates, Baffle/Former Bolts	SS, Ni Alloy (bolts)	Chemically Treated Borated Water up to 340°C (644°F)	Cumula-tive fatigue damage	Fatigue	Design Code of record or later approved Codes.
B2.4.2	Baffle/Former Assembly	Baffle/Former Bolts	SS, Ni Alloy	Chemically Treated Borated Water up to 340°C (644°F)	Loss of Preload	Stress Relaxation	-
B2.5.1, B2.5.6	Lower Internal Assembly	Lower Core Plate, Radial Keys and Clevis Inserts	SS	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation and Growth	SCC, IASCC	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. EPRI TR-105714 Rev. 3 or later revisions/update.
B2.5.1, B2.5.6	Lower Internal Assembly	Lower Core Plate, Radial Keys and Clevis Inserts	SS	Chemically Treated Borated Water up to 340°C (644°F)	Changes in Dimensions	Void Swelling	-
B2.5.1	Lower Internal Assembly	Lower Core Plate	SS	Treated Borated Water up to 340°C, Fluence > 10^{17} n/cm^2 (E>1 MeV)	Loss of Fracture Toughness	Neutron Irradiation Embrittle–ment	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.
B2.5.1, B2.5.4	Lower Internal Assembly	Lower Core Plate, Lower Support Plate Columns	SS	Chemically Treated Borated Water up to 340°C (644°F)	Cumula-tive fatigue damage	Fatigue	Design Code of record or later approved Codes.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Plant-specific aging management program is to be evaluated.	Plant-specific aging management program is to be evaluated.	Yes, plant specific AMP
Same as for the effect of Fatigue on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.	*Same as for the effect of Fatigue on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.*	Yes TLAA
Visual inspection (VT–3) is inadequate to detect relevant conditions of stress relaxation because only the heads of the baffle/former bolts are visible, and a plant-specific aging management program is required.	Plant-specific aging management program is to be evaluated.	Yes, plant specific AMP
Same as for the effect of Stress Corrosion Cracking on Items B2.1.1, upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.	*Same as for the effect of Stress Corrosion Cracking on Items B2.1.1, upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.*	Yes, detection of aging effects should be further evaluated
Same as for the effect of Void Swelling on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.	*Same as for the effect of Void Swelling on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.*	Yes, plant specific AMP
Visual inspection (VT–3) is performed according to Category B-N-2/B–N–3 of Subsection IWB, ASME Section XI.	*Same as for the effect of Neutron Irradiation Embrittlement on Items B2.3.1 core barrel, B2.3.2 upper core barrel flange, B2.3.3 core barrel nozzles, B2.3.4 and thermal shield.*	Yes, parameters monitored/ inspected and detection of aging effects should be further evaluated
Same as for the effect of Fatigue on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.	*Same as for the effect of Fatigue on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.*	Yes TLAA

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B2.5.2, B2.5.5, B2.5.7	Lower Internal Assembly	Fuel Alignment Pins, Lower Support Plate Column Bolts, Clevis Insert Bolts	SS, Ni Alloy	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation and Growth	SCC, IASCC	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. EPRI TR-105714 Rev. 3 or later revisions/update.
B2.5.2, B2.5.5, B2.5.7	Lower Internal Assembly	Fuel Alignment Pins, Lower Support Plate Column Bolts, Clevis Insert Bolts	SS, Ni Alloy	Chemically Treated Borated Water up to 340°C (644°F)	Changes in Dimensions	Void Swelling	-
B2.5.2, B2.5.5	Lower Internal Assembly	Fuel Alignment Pins, Lower Support Plate Column Bolts	SS, Ni Alloy	Treated Borated Water up to 340°C, Fluence > 10^{17} n/cm^2 (E>1 MeV)	Loss of Fracture Toughness	Neutron Irradiation Embrittle–ment	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.
B2.5.5	Lower Internal Assembly	Lower Support Plate Column Bolts	SS, Ni Alloy	Chemically Treated Borated Water up to 340°C (644°F)	Loss of Preload	Stress Relaxation	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.
B2.5.7	Lower Internal Assembly	Clevis Insert Bolts	SS, Ni Alloy	Chemically Treated Borated Water up to 340°C (644°F)	Loss of Preload	Stress Relaxation	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.
B2.5.2, B2.5.5	Lower Internal Assembly	Fuel Alignment Pins, Lower Support Plate Column Bolts	SS, Ni Alloy	Chemically Treated Borated Water up to 340°C (644°F)	Cumula-tive fatigue damage	Fatigue	Design Code of record or later approved Codes.
B2.5.3, B2.5.4	Lower Internal Assembly	Lower Support Forging or Casting, Lower Support Plate Columns	SS, CASS	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation and Growth	SCC, IASCC	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. EPRI TR-105714 Rev. 3 or later revisions/update.
B2.5.3, B2.5.4	Lower Internal Assembly	Lower Support Forging or Casting, Lower Support Plate Columns	SS, CASS	Chemically Treated Borated Water up to 340°C (644°F)	Changes in Dimensions	Void Swelling	-

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Same as for the effect of Void Swelling on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.	*Same as for the effect of Void Swelling on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.*	Yes, plant specific AMP
Visual inspection (VT–3) is performed according to Category B-N-2/B–N–3 of Subsection IWB, ASME Section XI.	*Same as for the effect of Neutron Irradiation Embrittlement on Items B2.3.1 core barrel, B2.3.2 upper core barrel flange, B2.3.3 core barrel nozzles, B2.3.4 and thermal shield.*	Yes, parameters monitored/ inspected and detection of aging effects should be further evaluated
Same as for the effect of Stress Relaxation on Item B2.1.3 upper support column bolts.	*Same as for the effect of Stress Relaxation on Item B2.1.3 upper support column bolts.*	No
Same as for the effect of Stress Relaxation on Item B2.1.7 hold-down spring.	*Same as for the effect of Stress Relaxation on Item B2.1.7 hold-down spring.*	No
Same as for the effect of Fatigue on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.	*Same as for the effect of Fatigue on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.*	Yes TLAA
Same as for the effect of Stress Corrosion Cracking on Items B2.1.1, upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.	*Same as for the effect of Stress Corrosion Cracking on Items B2.1.1, upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.*	Yes, detection of aging effects should be further evaluated
Same as for the effect of Void Swelling on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.	*Same as for the effect of Void Swelling on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.*	Yes, plant specific AMP
Same as for the effect of SCC or IASCC on Items B2.1.3 upper support column bolts, B2.1.5 upper core plate alignment pins, and B2.1.6 fuel alignment pins.	*Same as for the effect of SCC or IASCC on Items B2.1.3 upper support column bolts, B2.1.5 upper core plate alignment pins, and B2.1.6 fuel alignment pins.*	Yes, detection of aging effects should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B2.5.3, B2.5.4	Lower Internal Assembly	Lower Support Forging or Casting, Lower Support Plate Columns	SS, CASS	Chemically Treated Borated Water up to 340°C (644°F) Fluence > 10^{17} n/cm^2 (E>1 MeV)	Loss of fracture toughness	Thermal Aging and Neutron Irradiation Embrittle–ment	*Same as for the effect of Thermal Aging and Neutron Irradiation Embrittlement on Item B2.1.2 upper support column constructed of CASS.*
B2.5.6	Lower Internal Assembly	Radial Keys and Clevis Inserts	SS	Chemically Treated Borated Water up to 340°C (644°F)	Loss of Material	Wear	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a
B2.5.6, B2.5.7	Lower Internal Assembly	Radial Keys and Clevis Inserts, Clevis Insert Bolts	SS	Chemically Treated Borated Water up to 340°C (644°F)	Cumula-tive fatigue damage	Fatigue	Design Code of record or later approved Codes.
B2.6.1	Instru-mentation Support Structure	Flux Thimble Guide Tubes	SS	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation and Growth	SCC, IASCC	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. EPRI TR-105714 Rev. 3 or later revisions/update.
B2.6.1	Instru-mentation Support Structure	Flux Thimble Guide Tubes	SS	Chemically Treated Borated Water up to 340°C (644°F)	Changes in Dimensions	Void Swelling	-
B2.6.2	Instru-mentation Support Structure	Flux Thimbles	SS	Chemically Treated Borated Water up to 340°C (644°F)	Loss of Material	Wear	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Same as for the effect of Thermal Aging and Neutron Irradiation Embrittlement on Item B2.1.2 upper support column constructed of CASS.	*Same as for the effect of Thermal Aging and Neutron Irradiation Embrittlement on Item B2.1.2 upper support column constructed of CASS.*	No
Same as for the effect of Wear on Item B2.1.5 upper core plate alignment pins.	*Same as for the effect of Wear on Item B2.1.5 upper core plate alignment pins.*	No
Same as for the effect of Fatigue on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.	*Same as for the effect of Fatigue on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.*	Yes TLAA
Same as for the effect of Stress Corrosion Cracking on Items B2.1.1, upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.	*Same as for the effect of Stress Corrosion Cracking on Items B2.1.1, upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.*	Yes, detection of aging effects should be further evaluated
Same as for the effect of Void Swelling on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.	*Same as for the effect of Void Swelling on Items B2.1.1 upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.*	Yes, plant specific AMP
Same as for the effect of Wear on Item B2.1.5 upper core plate alignment pins.	*Same as for the effect of Wear on Item B2.1.5 upper core plate alignment pins.*	No

B3. Reactor Vessel Internals (PWR) - Combustion Engineering

B3.1 Upper Internals Assembly

B3.1.1 Upper Guide Structure Support Plate

B3.1.2 Fuel Alignment Plate

B3.1.3 Fuel Alignment Plate Guide Lugs and Guide Lug Inserts

B3.1.4 Hold-Down Ring

B3.2 CEA Shroud Assemblies

B3.2.1 CEA Shrouds

B3.2.2 CEA Shrouds Bolts

B3.2.3 CEA Shrouds Extension Shaft Guides

B3.3 Core Support Barrel

B3.3.1 Core Support Barrel

B3.3.2 Core Support Barrel Upper Flange

B3.3.3 Core Support Barrel Alignment Keys

B3.4 Core Shroud Assembly

B3.4.1 Core Shroud Assembly

B3.4.2 Core Shroud Assembly Bolts

B3.4.3 Core Shroud Tie Rods

B3.5 Lower Internal Assembly

B3.5.1 Core Support Plate

B3.5.2 Fuel Alignment Pins

B3.5.3 Lower Support Structure Beam Assemblies

B3.5.4 Core Support Column

B3.5.5 Core Support Column Bolts

B3.5.6 Core Support Barrel Snubber Assemblies

B3. Reactor Vessel Internals (PWR) - Combustion Engineering

System, Structures, and Components

The system, structures, and components included in this table comprise the ABB/Combustion Engineering pressurized water reactor (PWR) reactor vessel internals and consist of the Upper Internals Assembly, CEA Shroud Assemblies, Core Support Barrel, Core Shroud Assembly, and Lower Internal Assembly. Based on the Nuclear Regulatory Commission Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all structures and components in the reactor vessel are classified as Group A or B Quality Standards.

System Interfaces

The systems that interface with the reactor vessel internals include the reactor pressure vessel (Table IV.A2) and reactor coolant system and connected lines (Table IV.C2).

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B3.1.1 - B3.1.3	Upper Internals Assembly	Upper Guide Structure Support Plate, Fuel Alignment Plate, Fuel Alignment Plate Guide Lugs	Stainless Steel (SS)	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation & Growth	Stress Corrosion Cracking (SCC), Irradiation Assisted Stress Corrosion Cracking (IASCC)	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. EPRI TR-105714 Rev. 3 or later revisions/update. *Operating Experience* NRC IN 84-18. NRC IN 98-11.

Aging Management Activity (AMA)	Evaluation and Technical Basis	Further Evaluation
Visual inspection (VT–3) is performed according to Category B–N–3 of Subsection IWB, ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a). Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 (Rev. 3 or later revisions or update) to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC or IASCC, inservice inspection (ISI) to detect cracks, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** PWR operating chemistry limits the halogens, sulfates, and oxygen in the primary water to less than 0.05, 0.05, and 0.005 ppm, respectively, during operation, which greatly reduces susceptibility to SCC or IASCC. However, introduction of oxygen can occur during shutdown and potential exists for the formation of more aggressive chemistry conditions by radiolysis in creviced regions or in low–flow stagnant regions. Also, for sufficiently high fluence levels, IASCC can occur even in low oxygen environments. The AMP must rely upon ISI in accordance with ASME Section XI to detect possible degradation. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of SCC or IASCC on the intended function of the component by detection and sizing of cracks by ISI. Table IWB-2500, examination category B-N-3 specifies visual VT-3 examination of all accessible surfaces of reactor core support structures that can be removed from the reactor vessel. ***(4) Detection of Aging Effects:*** Degradation due to SCC or IASCC can not occur without crack initiation and growth. VT–3 is not adequate to detect tight cracks. The inspection technique, including the reliability in detecting the features of interest (crack appearance and size), should be specified. Also, creviced regions are difficult to inspect visually and supplementary UT or other nondestructive examinations may be necessary As an alternate to enhanced inspection, perform a component-specific evaluation including a mechanical loading assessment to determine the maximum tensile loading on the component during ASME Code Level A, B, C, and D conditions. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with IWB-3520. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWB-3140. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Although SS components in PWRs have generally not been found to be affected by SCC because of low dissolved oxygen levels and control of primary water chemistry, potential for SCC exists from inadvertent introduction of contaminants into the primary coolant system (IN 84–18), from the introduction of relatively high levels of dissolved oxygen during shutdown, or from aggressive chemistries that may develop in creviced regions. Cracking has occurred in SS baffle former bolts in a number of foreign plants (IN 98-11) and has now been observed in US plants. The mechanism of this particular cracking has not yet been resolved.	Yes, detection of aging effects should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B3.1.1 - B3.1.3	Upper Internals Assembly	Upper Guide Structure Support Plate, Fuel Alignment Plate, Fuel Alignment Plate Guide Lugs	SS	Chemically Treated Borated Water up to 340°C (644°F)	Changes in Dimensions	Void Swelling	-
B3.1.2 - B3.1.4	Upper Internals Assembly	Fuel Alignment Plate, Fuel Alignment Plate Guide Lugs, Hold-Down Ring	SS	Chemically Treated Borated Water up to 340°C (644°F)	Loss of Material	Wear	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a
B3.2.1	CEA Shroud Assemblies	CEA Shroud	SS, Cast Austenitic SS (CASS)	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation & Growth	SCC, IASCC	*Same as for the effect of SCC or IASCC on Items B3.1.1 - B3.1.3.*
B3.2.2	CEA Shroud Assemblies	CEA Shrouds Bolts	SS, Ni alloy	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation & Growth	SCC, IASCC	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. EPRI TR-105714 Rev. 3 or later revisions/update.

Aging Management Activity (AMA)	Evaluation and Technical Basis	Further Evaluation
The reactor vessel internals receive a visual inspection (VT-3) according to Category B-N-3 of Subsection IXB, ASME Section XI. This inspection is not sufficient to detect the effects of changes in dimension due to void swelling. An acceptable alternative AMP consists of the following: 1. Participation in industry programs to address the significance of change in dimensions due to void swelling. 2. Implementation of an inspection program should the results of the industry programs indicate the need for such inspections.	Plant specific aging management program is to be evaluated. The applicant must provide the basis for concluding that void swelling is not an issue for the component or must provide an AMP. The applicant should address loss of ductility associated with swelling.	Yes, plant specific AMP
Visual inspection (VT–3) is performed according to Category B–N–3 of Subsection IWB, ASME Section XI.	***(1) Scope of Program:*** The program includes inservice inspection (ISI) to monitor the condition of components and assure that they can perform their intended function. ***(2) Preventive Actions:*** No practical preventative actions are possible. ***(3) Parameters Monitored/Inspected:*** The AMP program monitors the effects of wear on the intended function by inservice inspection (ISI) to determine the general mechanical and structural condition of components. ***(4) Detection of Aging Effects:*** Inspection can reveal indications of degradation due to wear such as the verification of clearances, settings, physical displacements, loose or missing parts, debris, corrosion, wear, erosion, or the loss of integrity at bolted or welded connections. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 is adequate for timely detection of wear. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with IWB-3520. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWB-3140. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** To date aging management program has been effective in managing the effects of wear on reactor internals.	No
Same as for the effect of SCC or IASCC on Items B3.1.1 upper guide structure support plate, B3.1.2 fuel alignment plate, and B3.1.3 fuel alignment plate guide lugs.	*Same as for the effect of SCC or IASCC on Items B3.1.1 upper guide structure support plate, B3.1.2 fuel alignment plate, and B3.1.3 fuel alignment plate guide lugs.*	Yes, detection of aging effects should be further evaluated
Visual inspection (VT–3) is performed according to Category B–N–3 of Subsection IWB, ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a). Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 (Rev. 3 or later revisions or update) to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC and IASCC, inservice inspection (ISI) to detect cracks, and repair and/or replacement as needed to maintain the capability of the component to perform its intended function. ***(2) Preventive Actions:*** PWR operating chemistry limits the halogens, sulfates, and oxygen in the primary water to less than 0.05, 0.05, and 0.005 ppm, respectively, during operation. However, introduction of oxygen can occur during shutdown and potential exists for the formation of more aggressive	Yes, detection of aging effects should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
							Operating Experience NRC IN 84-18.
B3.2.1, B3.2.2	CEA Shroud Assemblies	CEA Shroud, CEA Shrouds Bolts	SS, CASS, Ni alloy	Chemically Treated Borated Water up to 340°C (644°F)	Changes in Dimensions	Void Swelling	-
B3.2.3	CEA Shroud Assemblies	CEA Shroud Extension Shaft Guides	SS	Chemically Treated Borated Water up to 340°C (644°F)	Loss of Material	Wear	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.

Aging Management Activity (AMA)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) chemistry conditions by radiolysis in creviced regions or in low–flow stagnant regions. Also, for sufficiently high fluence levels, IASCC can occur even in low oxygen environments. The AMP must rely upon ISI in accordance with ASME Section XI to detect possible degradation. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of SCC and IASCC on the intended function of the component by detection and sizing of cracks by inservice inspection (ISI). Table IWB-2500, category B-N-3 specifies visual VT-3 examination of all accessible surfaces of reactor internals. ***(4) Detection of Aging Effects:*** Degradation due to SCC can not occur without crack initiation and growth. Historically the VT-3 visual examinations have not identified bolt cracking because cracking occurs at the juncture of the bolt head and shank, which is not accessible for visual inspection. Creviced and other inaccessible regions are difficult to inspect visually. Supplementary UT examinations may be necessary. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with IWB-3520. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWB-3140. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Although stainless steel components in PWRs have generally not been found to be affected by SCC because of low dissolved oxygen levels and control of primary water chemistry, potential for SCC exists from inadvertent introduction of contaminants into the primary coolant system (IN 84–18), from the introduction of relatively high levels of dissolved oxygen during shutdown, or from aggressive chemistries that may develop in creviced regions.	
Same as for the effect of Void Swelling on Items B3.1.1 upper guide structure support plate, B3.1.2 fuel alignment plate, and B3.1.3 fuel alignment plate guide lugs.	*Same as for the effect of Void Swelling on Items B3.1.1 upper guide structure support plate, B3.1.2 fuel alignment plate, and B3.1.3 fuel alignment plate guide lugs.*	Yes, plant specific AMP
Same as for the effect of Wear on Items B3.1.2 fuel alignment plate, B3.1.3 fuel alignment plate guide lugs., and B3.1.4 hold-down ring.	*Same as for the effect of Wear on Items B3.1.2 fuel alignment plate, B3.1.3 fuel alignment plate guide lugs., and B3.1.4 hold-down ring.*	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B3.2.1	CEA Shroud Assemblies	CEA Shroud	CASS	Chemically Treated Borated Water up to 340°C (644°F) Neutron Fluence of greater than 10^{17} n/cm^2 (E>1 MeV)	Loss of Fracture Toughness	Thermal Aging and Neutron Irradiation Embrittle–ment	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. Letter from Christopher I. Grimes (NRC) to Douglas J. Walters (NEI) dated 5/19/2000.
B3.2.1, B3.2.2	CEA Shroud Assemblies	CEA Shroud, CEA Shrouds Bolts	SS, CASS, Ni Alloy	Chemically Treated Borated Water up to 340°C (644°F)	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
B3.2.2	CEA Shroud Assemblies	CEA Shrouds Bolts	SS, Ni Alloy	Chemically Treated Borated Water up to 340°C (644°F)	Loss of Preload	Stress Relaxation	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.

Aging Management Activity (AMA)	Evaluation and Technical Basis	Further Evaluation
The reactor vessel internals receive a visual inspection in accordance with Category B-N-3 of Subsection IXB, ASME Section XI. This inspection is not sufficient to detect the effects of loss of fracture toughness due to thermal aging and neutron irradiation embrittlement of cast austenitic stainless steel (CASS) reactor vessel internals. An acceptable AMP consists of the following: Determination of the susceptibility of CASS components to thermal aging embrittlement based on casting method, Mo content, and percent ferrite. For “potentially susceptible” components, based on the neutron fluence of the component, implement either a supplemental examination of the affected components as part of the applicant's 10-year inservice inspection (ISI) program during the license renewal term or a component-specific evaluation to determine the susceptibility to loss of fracture toughness.	For evaluation and technical basis of the 10 elements of the AMP, see Chapter XI.M2 "Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)".	No
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes TLAA
Visual inspection (VT–3) is performed according to Category B–N–3 of Subsection IWB, ASME Section XI to monitor the relevant conditions of degradation and loose part monitoring.	***(1) Scope of Program:*** The program includes inservice inspection (ISI) to monitor the relevant conditions of degradation, loose part monitoring, and repair and/or replacement as needed to maintain the capability of the component to perform its intended function. ***(2) Preventive Actions:*** No practical preventative actions are possible. ***(3) Parameters Monitored/Inspected:*** The program includes ISI to monitor the relevant conditions of degradation such as loose, cracked, or missing bolts or fasteners, and wear, and neutron noise monitoring (excore detectors) to detect core barrel motion. Table IWB-2500, category B-N-3 specifies visual VT-3 examination of all accessible surfaces of reactor core support structures that can be removed from the reactor vessel. ***(4) Detection of Aging Effects:*** Visual VT-3 inspection can reveal indications of degradation due to stress relaxation such as loose or missing parts, wear, debris, or the loss of integrity at bolted or welded connections, and neutron noise monitoring can detect physical displacement and motion of reactor internals. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 is adequate for timely detection of degradation.	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B3.3.1, B3.3.2	Core Support Barrel	Core Support Barrel, Core Support Barrel Upper Flange	SS	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation & Growth	SCC, IASCC	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. EPRI TR-105714 Rev. 3 or later revisions/update.
B3.3.1, B3.3.2	Core Support Barrel	Core Support Barrel, Core Support Barrel Upper Flange	SS	Chemically Treated Borated Water up to 340°C (644°F)	Changes in Dimensions	Void Swelling	-
B3.3.1, B3.3.2	Core Support Barrel	Core Support Barrel, Core Support Barrel Upper Flange	SS	Chemically Treated Borated Water up to 340°C (644°F), Neutron Fluence of greater than 10^{17} n/cm^2 (E>1 MeV)	Loss of Fracture Toughness	Neutron Irradiation Embrittle–ment	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.

Aging Management Activity (AMA)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) ***(6) Acceptance Criteria:*** Any relevant condition such as loose, missing, cracked, or fractured parts, bolting, or fasteners, is evaluated in accordance with IWB-3500. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWB-3140; the options include supplemental surface and/or volumetric examination to further characterize the condition, corrective measures (e.g., re-establish the preload) or repairs, or replacement. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** There are no reports of stress relaxation producing damage in reactor vessel internals.	
Same as for the effect of SCC or IASCC on Items B3.1.1 upper guide structure support plate, B3.1.2 fuel alignment plate, and B3.1.3 fuel alignment plate guide lugs.	*Same as for the effect of SCC or IASCC on Items B3.1.1 upper guide structure support plate, B3.1.2 fuel alignment plate, and B3.1.3 fuel alignment plate guide lugs.*	Yes, detection of aging effects should be further evaluated
Same as for the effect of Void Swelling on Items B3.1.1 upper guide structure support plate, B3.1.2 fuel alignment plate, and B3.1.3 fuel alignment plate guide lugs.	*Same as for the effect of Void Swelling on Items B3.1.1 upper guide structure support plate, B3.1.2 fuel alignment plate, and B3.1.3 fuel alignment plate guide lugs.*	Yes, plant specific AMP
Visual inspection (VT–3) is performed according to Category B-N-2/B–N–3 of Subsection IWB, ASME Section XI.	***(1) Scope of Program:*** The program includes inservice inspection (ISI) to detect cracking and/or failure, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** No practical preventative actions are possible. Stainless steels are susceptible to embrittlement under neutron irradiation. Fracture toughness will depend strongly on the fluence on a particular component. Components can be screened out if the maximum tensile loading on the component under ASME Code Level A, B, C, and D conditions is sufficiently low. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of neutron irradiation embrittlement on the intended function of the component by detection and sizing of cracks by inservice inspection (ISI). Table IWB-2500 specifies visual VT-3 examination according to categories B-N-2 for reactor internals and B-N-3 for all accessible surfaces of reactor core support structure that can be removed from the reactor vessel. ***(4) Detection of Aging Effects:*** Loss of fracture toughness is of consequence only if cracks exist. Cracking is expected to initiate at the surface and should be detectable by ISI except for some crevice regions. VT–3 is not adequate to detect tight cracks. Also, creviced regions are difficult to inspect visually and supplementary UT or other nondestructive	Yes, parameters monitored/ inspected and detection of aging effects should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B3.3.2, B3.3.3	Core Support Barrel	Core Support Barrel Upper Flange, Core Support Barrel Alignment Keys	SS	Chemically Treated Borated Water up to 340°C (644°F)	Loss of Material	Wear	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.
B3.4.1, B3.4.3	Core Shroud Assembly	Core Shroud Assembly, Core Shroud Tie Rods (core support plate attached by welds in later plants)	SS	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation & Growth	SCC, IASCC	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. EPRI TR-105714 Rev. 3 or later revisions/update.
B3.4.1, B3.4.3	Core Shroud Assembly	Core Shroud Assembly, Core Shroud Tie Rods (core support plate attached by welds in later plants)	SS, CASS, Ni alloy	Chemically Treated Borated Water up to 340°C (644°F)	Changes in Dimensions	Void Swelling	-
B3.4.1, B3.4.3	Core Shroud Assembly	Core Shroud Assembly, Core Shroud Tie Rods (core support plate attached by welds in later plants)	SS	Chemically Treated Borated Water up to 340°C, Fluence > 10^{17} n/cm^2 (E>1 MeV)	Loss of Fracture Toughness	Neutron Irradiation Embrittle–ment	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.
B3.4.1 - B3.4.3	Core Shroud Assembly	Core Shroud Assembly, Core Shroud Assembly Bolts, Core Shroud Tie Rods	SS, Ni Alloy (bolts)	Chemically Treated Borated Water up to 340°C (644°F)	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.

Aging Management Activity (AMA)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) examinations may be needed. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with IWB-3520. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWB-3140. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** No instances of internals degradation have been recorded that have been definitely attributed to irradiation embrittlement.	
Same as for the effect of Wear on Items B3.1.2 fuel alignment plate, B3.1.3 fuel alignment plate guide lugs., and B3.1.4 hold-down ring.	*Same as for the effect of Wear on Items B3.1.2 fuel alignment plate, B3.1.3 fuel alignment plate guide lugs., and B3.1.4 hold-down ring.*	No
Same as for the effect of SCC or IASCC on Items B3.1.1 upper guide structure support plate, B3.1.2 fuel alignment plate, and B3.1.3 fuel alignment plate guide lugs.	*Same as for the effect of SCC or IASCC on Items B3.1.1 upper guide structure support plate, B3.1.2 fuel alignment plate, and B3.1.3 fuel alignment plate guide lugs.*	Yes, detection of aging effects should be further evaluated
Same as for the effect of Void Swelling on Items B3.1.1 upper guide structure support plate, B3.1.2 fuel alignment plate, and B3.1.3 fuel alignment plate guide lugs.	*Same as for the effect of Void Swelling on Items B3.1.1 upper guide structure support plate, B3.1.2 fuel alignment plate, and B3.1.3 fuel alignment plate guide lugs.*	Yes, plant specific AMP
Same as for the effect of Neutron Irradiation Embrittlement on Items B3.3.1 core support barrel and B3.3.2 core support barrel upper flange.	*Same as for the effect of Neutron Irradiation Embrittlement on Items B3.3.1 core support barrel and B3.3.2 core support barrel upper flange.*	Yes parameters monitored/ inspected and detection of aging effects should be further evaluated
Same as for the effect of Fatigue on Items B3.2.1 CEA shroud and B3.2.2 CEA shrouds bolts.	*Same as for the effect of Fatigue on Items B3.2.1 CEA shroud and B3.2.2 CEA shrouds bolts.*	Yes TLAA

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B3.4.2	Core Shroud Assembly	Core Shroud Assembly Bolts (later plants are welded)	SS, Ni Alloy	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation & Growth	SCC, IASCC	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. EPRI TR-105714 Rev. 3 or later revisions/update.
B3.4.2	Core Shroud Assembly	Core Shroud Assembly Bolts (later plants are welded)	SS, Ni Alloy	Chemically Treated Borated Water up to 340°C (644°F)	Changes in Dimensions	Void Swelling	-
B3.4.2	Core Shroud Assembly	Core Shroud Assembly Bolts (later plants are welded)	SS, Ni Alloy	Chemically Treated Borated Water up to 340°C, Fluence > 10^{17} n/cm^2 (E>1 MeV)	Loss of Fracture Toughness	Neutron Irradiation Embrittle–ment	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.
B3.4.2, B3.4.3	Core Shroud Assembly	Core Shroud Assembly Bolts, Core Shroud Tie Rods	SS, Ni Alloy	Chemically Treated Borated Water up to 340°C (644°F)	Loss of Preload	Stress Relaxation	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.
B3.5.1, B3.5.3, B3.5.4, B3.5.6	Lower Internal Assembly	Core Support Plate, Lower Support Structure Beam Assemblies, Core Support Column, Core Support Barrel Snubber Assemblies	SS	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation & Growth	SCC, IASCC	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. EPRI TR-105714 Rev. 3 or later revisions/update.
B3.5.2, B3.5.5	Lower Internal Assembly	Fuel Alignment Pins, Core Support Column Bolts,	SS, Ni Alloy	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation & Growth	SCC, IASCC	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. EPRI TR-105714 Rev. 3 or later revisions/update.

Aging Management Activity (AMA)	Evaluation and Technical Basis	Further Evaluation
Same as for the effect of SCC or IASCC on Item B3.2.2 CEA shrouds bolts.	*Same as for the effect of SCC or IASCC on Item B3.2.2 CEA shrouds bolts.*	Yes, detection of aging effects should be further evaluated
Same as for the effect of Void Swelling on Items B3.1.1 upper guide structure support plate, B3.1.2 fuel alignment plate, and B3.1.3 fuel alignment plate guide lugs.	*Same as for the effect of Void Swelling on Items B3.1.1 upper guide structure support plate, B3.1.2 fuel alignment plate, and B3.1.3 fuel alignment plate guide lugs.*	Yes, plant specific AMP
Same as for the effect of Neutron Irradiation Embrittlement on Items B3.3.1 core support barrel and B3.3.2 core support barrel upper flange.	*Same as for the effect of Neutron Irradiation Embrittlement on Items B3.3.1 core support barrel and B3.3.2 core support barrel upper flange.*	Yes parameters monitored/ inspected and detection of aging effects should be further evaluated
Same as for the effect of Stress Relaxation on Item B3.2.2 CEA shrouds bolts.	*Same as for the effect of Stress Relaxation on Item B3.2.2 CEA shrouds bolts.*	Yes, detection of aging effects should be further evaluated
Same as for the effect of Stress Corrosion Cracking on Items B3.1.1 upper guide structure support plate, B3.1.2 fuel alignment plate, and B3.1.3 fuel alignment plate guide lugs.	*Same as for the effect of Stress Corrosion Cracking on Items B3.1.1 upper guide structure support plate, B3.1.2 fuel alignment plate, and B3.1.3 fuel alignment plate guide lugs.*	Yes, detection of aging effects should be further evaluated
Same as for the effect of SCC or IASCC on Item B3.2.2 CEA shrouds bolts.	*Same as for the effect of SCC or IASCC on Item B3.2.2 CEA shrouds bolts.*	Yes, detection of aging effects should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B3.5.1 - B3.5.6	Lower Internal Assembly	Core Support Plate, Fuel Alignment Pins, Lower Support Structure Beam Assemblies, Core Support Column, Core Support Column Bolts, Core Support Barrel Snubber Assemblies	SS, Ni Alloy (pins/bolts), CASS (support column)	Chemically Treated Borated Water up to 340°C (644°F)	Changes in Dimensions	Void Swelling	-
B3.5.1 - B3.5.6	Lower Internal Assembly	Core Support Plate, Fuel Alignment Pins, Lower Support Structure Beam Assemblies, Core Support Column, Core Support Column Bolts, Core Support Barrel Snubber Assemblies	SS, Ni Alloy (pins/bolts), CASS (support column)	Chemically Treated Borated Water up to 340°C (644°F) Neutron Fluence of greater than 10^{17} n/cm^2 (E>1 MeV)	Loss of Fracture Toughness	Neutron Irradiation Embrittle–ment	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.
B3.5.2, B3.5.6	Lower Internal Assembly	Fuel Alignment Pins, Core Support Barrel Snubber Assemblies	SS, Ni Alloy (pins)	Chemically Treated Borated Water up to 340°C (644°F)	Loss of Material	Wear	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.
B3.5.4	Lower Internal Assembly	Core Support Column	CASS	Chemically Treated Borated Water up to 340°C (644°F)	Loss of Fracture Toughness	Thermal Aging and Neutron Irradiation Embrittle–ment	*Same as for the effect of Thermal Aging and Neutron Irradiation Embrittlement on Item B3.2.1 CEA shroud.*

Aging Management Activity (AMA)	Evaluation and Technical Basis	Further Evaluation
Same as for the effect of Void Swelling on Items B3.1.1 upper guide structure support plate, B3.1.2 fuel alignment plate, and B3.1.3 fuel alignment plate guide lugs.	*Same as for the effect of Void Swelling on Items B3.1.1 upper guide structure support plate, B3.1.2 fuel alignment plate, and B3.1.3 fuel alignment plate guide lugs.*	Yes, plant specific AMP
Same as for the effect of Neutron Irradiation Embrittlement on Items B3.3.1 core support barrel and B3.3.2 core support barrel upper flange.	*Same as for the effect of Neutron Irradiation Embrittlement on Items B3.3.1 core support barrel and B3.3.2 core support barrel upper flange.*	Yes parameters monitored/ inspected and detection of aging effects should be further evaluated
Same as for the effect of Wear on Items B3.1.2 fuel alignment plate, B3.1.3 fuel alignment plate guide lugs., and B3.1.4 hold-down ring.	*Same as for the effect of Wear on Items B3.1.2 fuel alignment plate, B3.1.3 fuel alignment plate guide lugs., and B3.1.4 hold-down ring.*	No
Same as for the effect of Thermal Aging and Neutron Irradiation Embrittlement on Item B3.2.1 CEA shrouds.	*Same as for the effect of Thermal Aging and Neutron Irradiation Embrittlement on Item B3.2.1 CEA shrouds.*	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B3.5.1 - B3.5.6	Lower Internal Assembly	Core Support Plate, Fuel Alignment Pins, Lower Support Structure Beam Assemblies, Core Support Column, Core Support Column Bolts, Core Support Barrel Snubber Assemblies	SS, Ni Alloy (pins/bolts), CASS (support column)	Chemically Treated Borated Water up to 340°C (644°F)	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.

Aging Management Activity (AMA)	Evaluation and Technical Basis	Further Evaluation
Same as for the effect of Fatigue on Items B3.2.1 CEA shroud and B3.2.2 CEA shrouds bolts.	*Same as for the effect of Fatigue on Items B3.2.1 CEA shroud and B3.2.2 CEA shrouds bolts.*	Yes TLAA

B4 Reactor Vessel Internals (PWR) - Babcock & Wilcox

B4.1 Plenum Cover and Plenum Cylinder

B4.1.1 Plenum Cover Assembly

B4.1.2 Plenum Cylinder

B4.1.3 Reinforcing Plates

B4.1.4 Top Flange-to-Cover Bolts

B4.1.5 Bottom Flange-to-Upper Grid Screws

B4.2 Upper Grid Assembly

B4.2.1 Upper Grid Rib Section

B4.2.2 Upper Grid Ring Forging

B4.2.3 Fuel Assembly Support Pads

B4.2.4 Plenum Rib Pads

B4.2.5 Rib-to-Ring Screws

B4.3 Control Rod Guide Tube (CRGT) Assembly

B4.3.1 CRGT Pipe and Flange

B4.3.2 CRGT Spacer Casting

B4.3.3 CRGT Spacer Screws

B4.3.4 Flange-to-Upper Grid Screws

B4.3.5 CRGT Rod Guide Tubes

B4.3.6 CRGT Rod Guide Sectors

B4.4 Core Support Shield Assembly

B4.4.1 Core Support Shield Cylinder (Top and Bottom Flange)

B4.4.2 Core Support Shield-to-Core Barrel Bolts

B4.4.3 Outlet and Vent Valve Nozzles

B4.4.4 Vent Valve Body and Retaining Ring

B4.4.5 Vent Valve Assembly Locking Device

B4.5 Core Barrel Assembly

B4.5.1 Core Barrel Cylinder (Top and Bottom Flange)

B4.5.2 Lower Internals Assembly-to-Core Barrel Bolts

B4.5.3 Core Barrel-to-Thermal Shield Bolts

B4.5.4 Baffle Plates & Formers

B4.5.5 Baffle/Former Bolts and Screws

B4.6 Lower Grid (LG) Assembly

B4.6.1 Lower Grid Rib Section

B4.6.2 Fuel Assembly Support Pads

B4.6.3 Lower Grid Rib-to-Shell Forging Screws

B4.6.4 Lower Grid Flow Distributor Plate

B4.6.5 Orifice Plugs

B4.6.6 Lower Grid and Shell Forgings

B4.6.7 Lower Internals Assembly-to-Thermal Shield Bolts

B4.6.8 Guide Blocks and Bolts

B4.6.9 Shock Pads and Bolts

B4.6.10 Support Post Pipes

B4.7 Flow Distributor Assembly

B4.7.1 Flow Distributor Head and Flange

B4.7.2 Shell Forging-to-Flow Distributor Bolts

B4.7.3 Incore Guide Support Plate

B4.7.4 Clamping Ring

B4.8 Thermal Shield

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

System, Structures, and Components

The system, structures, and components included in this table comprise the Babcock & Wilcox pressurized water reactor (PWR) reactor vessel internals and consist of the Plenum Cover and Plenum Cylinder, Upper Grid Assembly, Control Rod Guide Tube (CRGT) Assemblies, Core Support Shield Assembly, Vent Valve Assembly, Core Barrel Assembly, Flow Distributor Assembly, and Lower Internal Assembly. Based on the Nuclear Regulatory Commission Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all structures and components in the reactor vessel are classified as Group A or B Quality Standards.

System Interfaces

The systems that interface with the reactor vessel internals include the reactor pressure vessel (Table IV.A2) and reactor coolant system and connected lines (Table IV.C2).

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B4.1.1 - B4.1.3	Plenum Cover and Plenum Cylinder	Plenum Cover Assembly, Plenum Cylinder, Reinforcing Plates	Type 304 Stainless Steel (SS); Plenum Cylinder: Type 304 Forging	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation & Growth	Stress Corrosion Cracking (SCC), Irradiation Assisted Stress Corrosion Cracking (IASCC)	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. EPRI TR-105714 Rev. 3 or later revisions/update. *Operating Experience* NRC IN 84-18. NRC IN 98-11.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Visual inspection (VT–3) is performed according to Category B–N–3 of Subsection IWB, ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a). Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 (Rev. 3 or later revisions or update) to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC or IASCC, inservice inspection (ISI) to detect cracks, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** PWR operating chemistry limits the halogens, sulfates, and oxygen in the primary water to less than 0.05, 0.05, and 0.005 ppm, respectively, during operation, which greatly reduces susceptibility to SCC or IASCC. However, introduction of oxygen can occur during shutdown and potential exists for the formation of more aggressive chemistry conditions by radiolysis in creviced regions or in low–flow stagnant regions. Also, for sufficiently high fluence levels, IASCC can occur even in low oxygen environments. The AMP must rely upon ISI in accordance with ASME Section XI to detect possible degradation. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of SCC or IASCC on the intended function of the component by detection and sizing of cracks by ISI. Table IWB-2500, examination category B-N-3 specifies visual VT-3 examination of all accessible surfaces of reactor core support structures that can be removed from the reactor vessel. ***(4) Detection of Aging Effects:*** Degradation due to SCC or IASCC can not occur without crack initiation and growth. VT–3 is not adequate to detect tight cracks. The inspection technique, including the reliability in detecting the features of interest (crack appearance and size), should be specified. Also, creviced regions are difficult to inspect visually and supplementary UT or other nondestructive examinations may be necessary As an alternate to enhanced inspection, perform a component-specific evaluation including a mechanical loading assessment to determine the maximum tensile loading on the component during ASME Code Level A, B, C, and D conditions. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with IWB-3520. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWB-3140. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Although SS components in PWRs have generally not been found to be affected by SCC because of low dissolved oxygen levels and control of primary water chemistry, potential for SCC exists from inadvertent introduction of contaminants into the primary coolant system (IN 84–18), from the introduction of relatively high levels of dissolved oxygen during shutdown, or from aggressive chemistries that may develop in creviced regions. Cracking has occurred in SS baffle former bolts in a number of foreign plants (IN 98-11) and has now been observed in US plants. The mechanism of this particular cracking has not yet been resolved.	Yes, detection of aging effects should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B4.1.4, B4.1.5	Plenum Cover and Plenum Cylinder	Top Flange-to-Cover Bolts, Bottom Flange-to-Upper Grid Screws	Gr. B-8 SS	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation and Growth	SCC, IASCC	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. EPRI TR-105714 Rev. 3 or later revisions/update. Operating Experience NRC IN 84-18.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Visual inspection (VT–3) is performed according to Category B–N–3 of Subsection IWB, ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a). Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 (Rev. 3 or later revisions or update) to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC and IASCC, inservice inspection (ISI) to detect cracks, and repair and/or replacement as needed to maintain the capability of the component to perform its intended function. ***(2) Preventive Actions:*** PWR operating chemistry limits the halogens, sulfates, and oxygen in the primary water to less than 0.05, 0.05, and 0.005 ppm, respectively, during operation. However, introduction of oxygen can occur during shutdown and potential exists for the formation of more aggressive chemistry conditions by radiolysis in creviced regions or in low–flow stagnant regions. Also, for sufficiently high fluence levels, IASCC can occur even in low oxygen environments. The AMP must rely upon ISI in accordance with ASME Section XI to detect possible degradation. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of SCC and IASCC on the intended function of the component by detection and sizing of cracks by inservice inspection (ISI). Table IWB-2500, category B-N-3 specifies visual VT-3 examination of all accessible surfaces of reactor internals. ***(4) Detection of Aging Effects:*** Degradation due to SCC can not occur without crack initiation and growth. Historically the VT-3 visual examinations have not identified bolt cracking because cracking occurs at the juncture of the bolt head and shank, which is not accessible for visual inspection. Creviced and other inaccessible regions are difficult to inspect visually. Supplementary UT examinations may be necessary. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with IWB-3520. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWB-3140. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Although stainless steel components in PWRs have generally not been found to be affected by SCC because of low dissolved oxygen levels and control of primary water chemistry, potential for SCC exists from inadvertent introduction of contaminants into the primary coolant system (IN 84–18), from the introduction of relatively high levels of dissolved oxygen during shutdown, or from aggressive chemistries that may develop in creviced regions.	Yes, detection of aging effects should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B4.1.1 - B4.1.5	Plenum Cover and Plenum Cylinder	Plenum Cover Assembly, Plenum Cylinder, Reinforcing Plates, Top Flange-to-Cover Bolts, Bottom Flange-to-Upper Grid Screws	Type 304 SS, Bolts: Gr. B-8 SS	Chemically Treated Borated Water up to 340°C (644°F)	Changes in Dimensions	Void Swelling	-
B4.1.1 - B4.1.5	Plenum Cover and Plenum Cylinder	Plenum Cover Assembly, Plenum Cylinder, Reinforcing Plates, Top Flange-to-Cover Bolts, Bottom Flange-to-Upper Grid Screws	Type 304 SS; Plenum Cylinder: Type 304 Forging	Chemically Treated Borated Water up to 340°C (644°F)	Cumulative fatigue damage	Fatigue	Design Code of record or later approved Codes.
B4.2.1 - B4.2.4	Upper Grid Assembly	Upper Grid Rib Section, Upper Grid Ring Forging, Fuel Assembly Support Pads, Plenum Rib Pads	Type 304 SS	Chemically Treated Borated Water up to 340°C (644°F)	Crack initiation & growth	SCC, IASCC	*Same as for the effect of SCC or IASCC on Items B4.1.1 plenum cover assembly, B4.1.2 plenum cylinder, and B4.1.3 reinforcing plates.*
B4.2.5	Upper Grid Assembly	Rib-to-Ring Screws	Gr. B-8 SS	Chemically Treated Borated Water up to 340°C (644°F)	Crack initiation & growth	SCC, IASCC	*Same as for the effect of SCC or IASCC on Items B4.1.4 top flange -to-cover bolts and B4.1.5 bottom flange-to-upper grid screws.*
B4.2.1 - B4.2.5	Upper Grid Assembly	Upper Grid Rib Section, Upper Grid Ring Forging, Fuel Assembly Support Pads, Plenum Rib Pads, Rib-to-Ring Screws	Type 304 SS, Screws: Gr. B-8 SS	Chemically Treated Borated Water up to 340°C (644°F)	Changes in Dimensions	Void Swelling	-

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
The reactor vessel internals receive a visual inspection (VT-3) according to Category B-N-3 of Subsection IXB, ASME Section XI. This inspection is not sufficient to detect the effects of changes in dimension due to void swelling. An acceptable alternative AMP consists of the following: 1. Participation in industry programs to address the significance of change in dimensions due to void swelling. 2. Implementation of an inspection program should the results of the industry programs indicate the need for such inspections.	Plant specific aging management program is to be evaluated. The applicant should provide the basis for concluding that void swelling is not an issue for the component or must provide an AMP. The applicant should address loss of ductility associated with swelling.	Yes, plant specific AMP
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes TLAA
Same as for the effect of SCC or IASCC on Items B4.1.1 plenum cover assembly, B4.1.2 plenum cylinder, and B4.1.3 reinforcing plates.	*Same as for the effect of SCC or IASCC on Items B4.1.1 plenum cover assembly, B4.1.2 plenum cylinder, and B4.1.3 reinforcing plates.*	Yes, detection of aging effects should be further evaluated
Same as for the effect of SCC or IASCC on Items B4.1.4 top flange -to-cover bolts and B4.1.5 bottom flange-to-upper grid screws.	*Same as for the effect of SCC or IASCC on Items B4.1.4 top flange -to-cover bolts and B4.1.5 bottom flange-to-upper grid screws.*	Yes, detection of aging effects should be further evaluated
Same as for the effect of Void Swelling on Items B4.1.1 - B4.1.5 plenum cover and plenum cylinder components.	*Same as for the effect of Void Swelling on Items B4.1.1 - B4.1.5 plenum cover and plenum cylinder components.*	Yes, plant specific AMP

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B4.2.1 - B4.2.5	Upper Grid Assembly	Upper Grid Rib Section, Upper Grid Ring Forging, Fuel Assembly Support Pads, Plenum Rib Pads, Rib-to-Ring Screws	Type 304 SS	Chemically Treated Borated Water up to 340°C (644°F)	Cumulative fatigue damage	Fatigue	Design Code of record or later approved Codes.
B4.2.1 - B4.2.5	Upper Grid Assembly	Upper Grid Rib Section, Upper Grid Ring Forging, Fuel Assembly Support Pads, Plenum Rib Pads, Rib-to-Ring Screws	Type 304 SS, Screws: Gr. B-8 SS	Chemically Treated Borated Water up to 340°C (644°F)	Loss of Fracture Toughness	Neutron Irradiation Embrittle–ment	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Same as for the effect of Fatigue on Items B4.1.1 - B4.1.5 plenum cover and plenum cylinder components.	*Same as for the effect of Fatigue on Items B4.1.1 - B4.1.5 plenum cover and plenum cylinder components.*	Yes, TLAA
Visual inspection (VT–3) is performed according to Category B-N-2/B–N–3 of Subsection IWB, ASME Section XI.	***(1) Scope of Program:*** The program includes inservice inspection (ISI) to detect cracking and/or failure, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** No practical preventative actions are possible. Stainless steels are susceptible to embrittlement under neutron irradiation. Fracture toughness will depend strongly on the fluence on a particular component. Components can be screened out if the maximum tensile loading on the component under ASME Code Level A, B, C, and D conditions is sufficiently low. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of neutron irradiation embrittlement on the intended function of the component by detection and sizing of cracks by inservice inspection (ISI). Table IWB-2500 specifies visual VT-3 examination according to categories B-N-2 for reactor internals and B-N-3 for all accessible surfaces of reactor core support structure that can be removed from the reactor vessel. ***(4) Detection of Aging Effects:*** Loss of fracture toughness is of consequence only if cracks exist. Cracking is expected to initiate at the surface and should be detectable by ISI except for some crevice regions. VT–3 is not adequate to detect tight cracks. Also, creviced regions are difficult to inspect visually and supplementary UT or other nondestructive examinations may be needed. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with IWB-3520. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWB-3140. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** No instances of internals degradation have been recorded that have been definitely attributed to irradiation embrittlement.	Yes, parameters monitored/inspected and detection of aging effects should be further evaluated

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B4. REACTOR VESSEL INTERNALS (PWR) - Babcock & Wilcox

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B4.2.3, B4.2.4	Upper Grid Assembly	Fuel Assembly Support Pads, Plenum Rib Pads	Type 304 SS	Chemically Treated Borated Water up to 340°C (644°F)	Loss of Material	Wear	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.
B4.3.1, B4.3.2, B4.3.5 B4.3.6	Control Rod Guide Tube (CRGT) Assembly	CRGT Pipe and Flange, CRGT Spacer Casting, CRGT Rod Guide Tubes, CRGT Rod Guide Sectors	Pipe & Flange: Type 304 SS; Spacer Casting: CF-3M; Guide Tubes & Sectors: Type 304L	Chemically Treated Borated Water up to 340°C (644°F)	Crack initiation & growth	SCC, IASCC	*Same as for the effect of SCC or IASCC on Items B4.1.1 plenum cover assembly, B4.1.2 plenum cylinder, and B4.1.3 reinforcing plates.*
B4.3.3, B4.3.4	Control Rod Guide Tube (CRGT) Assembly	Spacer Casting Screws, Flange -to-Upper Grid Screws	Gr. B-8 SS	Chemically Treated Borated Water up to 340°C (644°F)	Crack initiation & growth	SCC, IASCC	*Same as for the effect of SCC or IASCC on Items B4.1.4 top flange -to-cover bolts and B4.1.5 bottom flange-to-upper grid screws.*
B4.3.1 - B4.3.6	Control Rod Guide Tube (CRGT) Assembly	CRGT Pipe and Flange, CRGT Spacer Casting, Spacer Casting Screws, Flange -to-Upper Grid Screws, CRGT Rod Guide Tubes, CRGT Rod Guide Sectors	Pipe & Flange: Type 304 SS; Spacer Casting: CF-3M; Guide Tubes & Sectors: Type 304L; Screws: Gr. B-8 SS	Chemically Treated Borated Water up to 340°C (644°F)	Changes in Dimensions	Void Swelling	-

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Visual inspection (VT–3) is performed according to Category B–N–3 of Subsection IWB, ASME Section XI.	***(1) Scope of Program:*** The program includes inservice inspection (ISI) to monitor the condition of components and assure that they can perform their intended function. ***(2) Preventive Actions:*** No practical preventative actions are possible. ***(3) Parameters Monitored/Inspected:*** The AMP program monitors the effects of wear on the intended function by inservice inspection (ISI) to determine the general mechanical and structural condition of components. ***(4) Detection of Aging Effects:*** Inspection can reveal indications of degradation due to wear such as the verification of clearances, settings, physical displacements, loose or missing parts, debris, corrosion, wear, erosion, or the loss of integrity at bolted or welded connections. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 is adequate for timely detection of wear. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with IWB-3520. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWB-3140. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** To date aging management program has been effective in managing the effects of wear on reactor internals.	No
Same as for the effect of SCC or IASCC on Items B4.1.1 plenum cover assembly, B4.1.2 plenum cylinder, and B4.1.3 reinforcing plates.	*Same as for the effect of SCC or IASCC on Items B4.1.1 plenum cover assembly, B4.1.2 plenum cylinder, and B4.1.3 reinforcing plates.*	Yes, detection of aging effects should be further evaluated
Same as for the effect of SCC or IASCC on Items B4.1.4 top flange -to-cover bolts and B4.1.5 bottom flange-to-upper grid screws.	*Same as for the effect of SCC or IASCC on Items B4.1.4 top flange -to-cover bolts and B4.1.5 bottom flange-to-upper grid screws.*	Yes, detection of aging effects should be further evaluated
Same as for the effect of Void Swelling on Items B4.1.1 - B4.1.5 plenum cover and plenum cylinder components.	*Same as for the effect of Void Swelling on Items B4.1.1 - B4.1.5 plenum cover and plenum cylinder components.*	Yes, plant specific AMP

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B4.3.2	Control Rod Guide Tube (CRGT) Assembly	CRGT Spacer Casting	Cast Austenitic Stainless Steel (CASS) CF-3M	Chemically Treated Borated Water up to 340°C (644°F) Neutron Fluence of greater than 10^{17} n/cm^2 (E>1 MeV)	Loss of Fracture Toughness	Thermal Aging and Neutron Irradiation Embrittle–ment	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. Letter from Christopher I. Grimes (NRC) to Douglas J. Walters (NEI) dated 5/19/2000.
B4.3.4	Control Rod Guide Tube (CRGT) Assembly	Flange-to-Upper Grid Screws	Gr. B-8 SS	Chemically Treated Borated Water up to 340°C (644°F)	Loss of Preload	Stress Relaxation	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
The reactor vessel internals receive a visual inspection in accordance with Category B-N-3 of Subsection IXB, ASME Section XI. This inspection is not sufficient to detect the effects of loss of fracture toughness due to thermal aging and neutron irradiation embrittlement of cast austenitic stainless steel (CASS) reactor vessel internals. An acceptable AMP consists of the following: Determination of the susceptibility of CASS components to thermal aging embrittlement based on casting method, Mo content, and percent ferrite. For "potentially susceptible" components, based on the neutron fluence of the component, implement either a supplemental examination of the affected components as part of the applicant's 10-year inservice inspection (ISI) program during the license renewal term or a component-specific evaluation to determine the susceptibility to loss of fracture toughness.	For evaluation and technical basis of the 10 elements of the AMP, see Chapter XI.M2 "Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)".	No
Visual inspection (VT–3) is performed according to Category B–N–3 of Subsection IWB, ASME Section XI.	***(1) Scope of Program:*** The program includes inservice inspection (ISI) to monitor the condition of the components that depend on preload, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** No practical preventative actions are possible. ***(3) Parameters Monitored/Inspected:*** The AMP utilizes ISI to monitor the relevant conditions of degradation such as loose, cracked, or missing bolts or fasteners, and wear. Table IWB-2500, category B-N-3 specifies visual VT-3 examination of all accessible surfaces of reactor internals. Because VT–3 inspection can only detect degradation that occurs after the loss of preload, in some cases, enhanced inspection may be required. ***(4) Detection of Aging Effects:*** Visual VT-3 inspection can reveal indications of degradation due to stress relaxation such as loose or missing parts, wear, debris, or the loss of integrity at bolted or welded connections. However, VT–3 inspection may not be adequate to detect the loss of mechanical closure integrity in components. An augmented inspection program to determine critical locations and appropriate monitoring and inspection techniques may be necessary. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 is adequate for timely detection of degradation. ***(6) Acceptance Criteria:*** Any relevant condition such as loose, missing, cracked, or fractured parts, bolting, or fasteners, is evaluated in accordance with IWB-3500. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWB-3140; the options include supplemental surface and/or volumetric examination to further characterize the condition, corrective measures (e.g., re-establish the preload) or repairs, ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and	Yes parameters monitored/ inspected and detection of aging effects should be further evaluated

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B4. REACTOR VESSEL INTERNALS (PWR) - Babcock & Wilcox

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B4.3.1 - B4.3.6	Control Rod Guide Tube (CRGT) Assembly	CRGT Pipe and Flange, CRGT Spacer Casting, Spacer Casting Screws, Flange -to-Upper Grid Screws, CRGT Rod Guide Tubes, CRGT Rod Guide Sectors	Pipe & Flange: Type 304 SS; Spacer Casting: CF-3M; Guide Tubes & Sectors: Type 304L; Screws: Gr. B-8 SS	Chemically Treated Borated Water up to 340°C (644°F)	Cumulative fatigue damage	Fatigue	Design Code of record or later approved Codes.
B4.4.1, B4.4.3, B4.4.4	Core Support Shield Assembly	Core Support Shield Cylinder (Top & Bottom Flange), Outlet and Vent Valve (VV) Nozzles, VV Body and Retaining Ring	Shield Cylinder: Type 304; Nozzles: SS Forging, CF-8; VV Body: CF-8; VV Ring: Type 15-5PH Forging	Chemically Treated Borated Water up to 340°C (644°F)	Crack initiation & growth	SCC, IASCC	*Same as for the effect of SCC or IASCC on Items B4.1.1 plenum cover assembly, B4.1.2 plenum cylinder, and B4.1.3 reinforcing plates.*
B4.4.2, B4.4.5	Core Support Shield Assembly	Core Support Shield-to-Core Barrel Bolts, VV Assembly Locking Device	Bolts: Gr. 660 (A-286), Gr. 688 (X-750); VV Locking Device: Gr. B-8 or B-8M	Chemically Treated Borated Water up to 340°C (644°F)	Crack initiation & growth	SCC, IASCC	*Same as for the effect of SCC or IASCC on Items B4.1.4 top flange -to-cover bolts and B4.1.5 bottom flange-to-upper grid screws.*
B4.4.1 - B4.4.5	Core Support Shield Assembly	Core Support Shield Cylinder (Top & Bottom Flange), Core Support Shield-to-Core Barrel Bolts, Outlet and Vent Valve (VV) Nozzles, VV Body and Retaining Ring, VV Assembly Locking Device	Shield Cylinder: Type 304; Bolts: A-286, X-750; Nozzles: SS Forging, CF-8; VV Body: CF-8; VV Ring: Type 15-5PH Forging; Locking Device: Gr. B-8 or B-8M	Chemically Treated Borated Water up to 340°C (644°F)	Changes in Dimensions	Void Swelling	-

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** There are no reports of stress relaxation producing damage in reactor vessel internals.	
Same as for the effect of Fatigue on Items B4.1.1 - B4.1.5 plenum cover and plenum cylinder components.	*Same as for the effect of Fatigue on Items B4.1.1 - B4.1.5 plenum cover and plenum cylinder components.*	Yes, TLAA
Same as for the effect of SCC or IASCC on Items B4.1.1 plenum cover assembly, B4.1.2 plenum cylinder, and B4.1.3 reinforcing plates.	*Same as for the effect of SCC or IASCC on Items B4.1.1 plenum cover assembly, B4.1.2 plenum cylinder, and B4.1.3 reinforcing plates.*	Yes, detection of aging effects should be further evaluated
Same as for the effect of SCC or IASCC on Items B4.1.4 top flange -to-cover bolts and B4.1.5 bottom flange-to-upper grid screws.	*Same as for the effect of SCC or IASCC on Items B4.1.4 top flange -to-cover bolts and B4.1.5 bottom flange-to-upper grid screws.*	Yes, detection of aging effects should be further evaluated
Same as for the effect of Void Swelling on Items B4.1.1 - B4.1.5 plenum cover and plenum cylinder components.	*Same as for the effect of Void Swelling on Items B4.1.1 - B4.1.5 plenum cover and plenum cylinder components.*	Yes, plant specific AMP

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B4.4.1 - B4.4.5	Core Support Shield Assembly	Core Support Shield Cylinder (Top & Bottom Flange), Core Support Shield-to-Core Barrel Bolts, Outlet and Vent Valve (VV) Nozzles, VV Body and Retaining Ring, VV Assembly Locking Device	Shield Cylinder: Type 304; Bolts: A-286, X-750; Nozzles: SS Forging, CF-8; VV Body: CF-8; VV Ring: Type 15-5PH Forging; Locking Device: Gr. B-8 or B-8M	Chemically Treated Borated Water up to 340°C (644°F) Neutron Fluence of greater than 10^{17} n/cm^2 (E>1 MeV)	Loss of Fracture Toughness	Neutron Irradiation Embrittle–ment	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.
B4.4.1 - B4.4.4	Core Support Shield Assembly	Core Support Shield Cylinder (Top & Bottom Flange), Core Support Shield-to-Core Barrel Bolts, Outlet and Vent Valve (VV) Nozzles, VV Body and Retaining Ring	Shield Cylinder: Type 304; Bolts: A-286, X-750; Nozzles: SS Forging, CF-8; VV Body: CF-8; VV Ring: Type 15-5PH Forging	Chemically Treated Borated Water up to 340°C (644°F)	Cumulative fatigue damage	Fatigue	Design Code of record or later approved Codes.
B4.4.1, B4.4.5	Core Support Assembly	Core Support Shield Cylinder (Top Flange), VV Assembly Locking Device	Top Flange: Type 304, VV Locking Device: Gr. B-8 or B-8M	Chemically Treated Borated Water up to 340°C (644°F)	Loss of Material	Wear	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.
B4.4.3, B4.4.4	Core Support Shield Assembly	Outlet Nozzle, VV Body and Retaining Ring	CASS CF-8	Chemically Treated Borated Water up to 340°C Fluence > 10^{17} n/cm^2 (E>1 MeV)	Loss of Fracture Toughness	Thermal Aging and Neutron Irradiation Embrittle–ment	*Same as for the effect of Thermal Aging and Neutron Embrittlement on Items B4.3.2.*
B4.4.2	Core Support Shield Assembly	Core Support Shield-to-Core Barrel Bolts	Gr. 660 (A-286), Gr. 688 (X-750)	Chemically Treated Borated Water up to 340°C (644°F)	Loss of Preload	Stress Relaxation	*Same as for the effect of Stress Relaxation on Item B4.3.4 CRGT flange-to-upper grid screws.*
B4.5.1, B4.5.4	Core Barrel Assembly	Core Barrel Cylinder (Top & Bottom Flange), Baffle Plates & Formers	CB Cylinder: Type 304 Forging, Baffle Plates & Formers: Type 304 SS	Chemically Treated Borated Water up to 340°C (644°F)	Crack initiation & growth	SCC, IASCC	*Same as for the effect of SCC or IASCC on Items B4.1.1 plenum cover assembly, B4.1.2 plenum cylinder, and B4.1.3 reinforcing plates.*

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Same as for the effect of Neutron Irradiation Embrittlement on Items B4.2.1 - B4.2.5 upper grid assembly components.	*Same as for the effect of Neutron Irradiation Embrittlement on Items B4.2.1 - B4.2.5 upper grid assembly components.*	Yes parameters monitored/ inspected and detection of aging effects should be further evaluated
Same as for the effect of Fatigue on Items B4.1.1 - B4.1.5 plenum cover and plenum cylinder components.	*Same as for the effect of Fatigue on Items B4.1.1 - B4.1.5 plenum cover and plenum cylinder components.*	Yes, TLAA
Same as for the effect of Wear on Items B4.2.3 Fuel Assembly Support Pads and B4.2.4 Plenum Rib Pads.	*Same as for the effect of Wear on Items B4.2.3 Fuel Assembly Support Pads and B4.2.4 Plenum Rib Pads.*	No
Same as for the effect of Thermal Aging and Neutron Embrittlement on Items B4.3.2 CRGT spacer casting.	*Same as for the effect of Thermal Aging and Neutron Embrittlement on Items B4.3.2 CRGT spacer casting.*	No
Same as for the effect of Stress Relaxation on Item B4.3.4 CRGT flange-to-upper grid screws.	*Same as for the effect of Stress Relaxation on Item B4.3.4 CRGT flange-to-upper grid screws.*	Yes, Element 3 and 4 should be further evaluated
Same as for the effect of SCC or IASCC on Items B4.1.1 plenum cover assembly, B4.1.2 plenum cylinder, and B4.1.3 reinforcing plates.	*Same as for the effect of SCC or IASCC on Items B4.1.1 plenum cover assembly, B4.1.2 plenum cylinder, and B4.1.3 reinforcing plates.*	Yes, detection of aging effects should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B4.5.2, B4.5.3	Core Barrel Assembly	Lower Internal Assembly-to-Core Barrel Bolts, Core Barrel-to-Thermal Shield Bolts	A-286, X-750	Chemically Treated Borated Water up to 340°C (644°F)	Crack initiation & growth	SCC, IASCC	*Same as for the effect of SCC or IASCC on Items B4.1.4 top flange -to-cover bolts and B4.1.5 bottom flange-to-upper grid screws.*
B4.5.1 - B4.5.4	Core Barrel Assembly	Core Barrel Cylinder (Top & Bottom Flange), Lower Internal Assembly-to-Core Barrel Bolts, Core Barrel-to-Thermal Shield Bolts, Baffle Plates & Formers	CB Cylinder: Type 304 Forging; CB Bolts: A-286, X-750; Baffle Plates & Formers: Type 304 SS	Chemically Treated Borated Water up to 340°C (644°F)	Changes in Dimensions	Void Swelling	-
B4.5.1 - B4.5.4	Core Barrel Assembly	Core Barrel Cylinder (Top & Bottom Flange), Lower Internal Assembly-to-Core Barrel Bolts, Core Barrel-to-Thermal Shield Bolts, Baffle Plates & Formers	CB Cylinder: Type 304 Forging; CB Bolts: A-286, X-750; Baffle Plates & Formers: Type 304 SS	Chemically Treated Borated Water up to 340°C (644°F) Neutron Fluence of greater than 10^{17} n/cm^2 (E>1 MeV)	Loss of Fracture Toughness	Neutron Irradiation Embrittle–ment	*Same as for the effect of Neutron Irradiation Embrittlement on Items B4.2.1 - B4.2.5 upper grid assembly components.*
B4.5.2, B4.5.3	Core Barrel Assembly	Lower Internal Assembly-to-Core Barrel Bolts, Core Barrel-to-Thermal Shield Bolts	A-286, X-750	Chemically Treated Borated Water up to 340°C (644°F)	Loss of Preload	Stress Relaxation	*Same as for the effect of Stress Relaxation on Item B4.3.4 CRGT flange-to-upper grid screws.*
B4.5.1 - B4.5.5	Core Barrel Assembly	Core Barrel Cylinder (Top & Bottom Flange), Lower Internal Assembly-to-Core Barrel Bolts, Core Barrel-to-Thermal Shield Bolts, Baffle Plates & Formers, Baffle/Former Bolts and Screws	CB Cylinder: Type 304 Forging; CB Bolts: A-286, X-750; Baffle Plates & Formers: Type 304 SS; Baffle/Former Bolts & Screws: Gr. B-8 SS	Chemically Treated Borated Water up to 340°C (644°F)	Cumulative fatigue damage	Fatigue	Design Code of record or later approved Codes.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Same as for the effect of SCC or IASCC on Items B4.1.4 top flange -to-cover bolts and B4.1.5 bottom flange-to-upper grid screws.	*Same as for the effect of SCC or IASCC on Items B4.1.4 top flange -to-cover bolts and B4.1.5 bottom flange-to-upper grid screws.*	Yes, detection of aging effects should be further evaluated
Same as for the effect of Void Swelling on Items B4.1.1 - B4.1.5 plenum cover and plenum cylinder components.	*Same as for the effect of Void Swelling on Items B4.1.1 - B4.1.5 plenum cover and plenum cylinder components.*	Yes, plant specific AMP
Same as for the effect of Neutron Irradiation Embrittlement on Items B4.2.1 - B4.2.5 upper grid assembly components.	*Same as for the effect of Neutron Irradiation Embrittlement on Items B4.2.1 - B4.2.5 upper grid assembly components.*	Yes parameters monitored/ inspected and detection of aging effects should be further evaluated
Same as for the effect of Stress Relaxation on Item B4.3.4 CRGT flange-to-upper grid screws.	*Same as for the effect of Stress Relaxation on Item B4.3.4 CRGT flange-to-upper grid screws.*	Yes, Element 3 and 4 should be further evaluated
Same as for the effect of Fatigue on Items B4.1.1 - B4.1.5 plenum cover and plenum cylinder components.	*Same as for the effect of Fatigue on Items B4.1.1 - B4.1.5 plenum cover and plenum cylinder components.*	Yes, TLAA

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B4.5.5	Core Barrel Assembly	Baffle/Former Bolts and Screws	Gr. B-8 SS	Chemically Treated Borated Water up to 340°C (644°F)	Crack initiation & growth	SCC, IASCC	-
B4.5.5	Core Barrel Assembly	Baffle/Former Bolts and Screws	Gr. B-8 SS	Chemically Treated Borated Water up to 340°C (644°F)	Changes in Dimensions	Void Swelling	-
B4.5.5	Core Barrel Assembly	Baffle/Former Bolts and Screws	Gr. B-8 SS	Chemically Treated Borated Water up to 340°C Fluence > 10^{17} n/cm^2 (E>1 MeV)	Loss of Fracture Toughness	Neutron Irradiation Embrittle–ment	-
B4.5.5	Core Barrel Assembly	Baffle/Former Bolts and Screws	Gr. B-8 SS	Chemically Treated Borated Water up to 340°C (644°F)	Loss of Preload	Stress Relaxation	-
B4.6.1, B4.6.2, B4.6.4 - B4.6.6, B4.6.8 - B4. 6.10	Lower Grid (LG) Assembly	LG Rib Section, Fuel Assembly Support Pads, LG Flow Dist. Plate, Orifice Plugs, LG & Shell Forgings, Guide Blocks, Shock Pads, Support Post Pipes	Type 304 SS	Chemically Treated Borated Water up to 340°C (644°F)	Crack initiation & growth	SCC, IASCC	*Same as for the effect of SCC or IASCC on Items B4.1.1 plenum cover assembly, B4.1.2 plenum cylinder, and B4.1.3 reinforcing plates.*

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Historically the VT-3 visual examinations have not identified baffle/former bolt cracking because cracking occurs at the juncture of the bolt head and shank, which is not accessible for visual inspection. However, recent UT examinations of the baffle/former bolts have identified cracking in several plants. The industry is currently addressing the issue of baffle bolt cracking in the PWR Materials Reliability Project, Issues Task Group (ITG) activities to determine, develop, and implement the necessary steps and plans to manage the applicable aging effects on a plant-specific basis.	Plant-specific aging management program is to be evaluated.	Yes, plant specific AMP
Same as for the effect of Void Swelling on Items B4.1.1 - B4.1.5 plenum cover and plenum cylinder components.	Plant specific aging management program is to be evaluated. The applicant should provide the basis for concluding that void swelling is not an issue for the component or must provide an AMP. The applicant should address the effects of material dimensional changes due to void swelling with regard to the potential of overloading the baffle bolts. Address the effects of the growth of the baffle plate material that is captured between the baffle bolt head and the edge of the former plate.	Yes, plant specific AMP
Plant-specific aging management program is to be evaluated.	Plant-specific aging management program is to be evaluated.	Yes, plant specific AMP
Visual inspection (VT–3) is inadequate to detect relevant conditions of stress relaxation because only the heads of the baffle/former bolts are visible, and a plant-specific aging management program is required.	Plant-specific aging management program is to be evaluated.	Yes, plant specific AMP
Same as for the effect of SCC or IASCC on Items B4.1.1 plenum cover assembly, B4.1.2 plenum cylinder, and B4.1.3 reinforcing plates.	*Same as for the effect of SCC or IASCC on Items B4.1.1 plenum cover assembly, B4.1.2 plenum cylinder, and B4.1.3 reinforcing plates.*	Yes, detection of aging effects should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B4.6.3, B4.6.7 - B4.6.9	Lower Grid (LG) Assembly	LG Rib-to-Shell Forging Screws, Lower Internals Assembly-to-Thermal Shield Bolts, Guide Blocks Bolts, Shock Pads Bolts	Lower Internal Assembly-to-Thermal Shield Bolts: A-286, X-750; Other Bolts and Screws: Gr. B-8 SS	Chemically Treated Borated Water up to 340°C (644°F)	Crack initiation & growth	SCC, IASCC	*Same as for the effect of SCC or IASCC on Items B4.1.4 top flange - to-cover bolts and B4.1.5 bottom flange-to-upper grid screws.*
B4.6.1 - B4. 6.10	Lower Grid (LG) Assembly	LG Rib Section, Fuel Assembly Support Pads, LG Rib-to-Shell Forging Screws, LG Flow Dist. Plate, Orifice Plugs, LG & Shell Forgings, Lower Internals Assembly-to-Thermal Shield Bolts, Guide Blocks & Bolts, Shock Pads & Bolts, Support Post Pipes	Type 304 SS, Lower Internal Assembly-to-Thermal Shield Bolts: A-286, X-750; Other Bolts and Screws: Gr. B-8 SS	Chemically Treated Borated Water up to 340°C (644°F)	Changes in Dimensions	Void Swelling	-
B4.6.1 - B4. 6.10	Lower Grid (LG) Assembly	LG Rib Section, Fuel Assembly Support Pads, LG Rib-to-Shell Forging Screws, LG Flow Dist. Plate, Orifice Plugs, LG & Shell Forgings, Lower Internals Assembly-to-Thermal Shield Bolts, Guide Blocks & Bolts, Shock Pads & Bolts, Support Post Pipes	Type 304 SS, Lower Internal Assembly-to-Thermal Shield Bolts: A-286, X-750; Other Bolts and Screws: Gr. B-8 SS	Chemically Treated Borated Water up to 340°C (644°F) Neutron Fluence of greater than 10^{17} n/cm^2 (E>1 MeV)	Loss of Fracture Toughness	Neutron Irradiation Embrittle–ment	*Same as for the effect of Neutron Irradiation Embrittlement on Items B4.2.1 - B4.2.5 upper grid assembly components.*

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Same as for the effect of SCC or IASCC on Items B4.1.4 top flange -to-cover bolts and B4.1.5 bottom flange-to-upper grid screws.	*Same as for the effect of SCC or IASCC on Items B4.1.4 top flange -to-cover bolts and B4.1.5 bottom flange-to-upper grid screws.*	Yes, detection of aging effects should be further evaluated
Same as for the effect of Void Swelling on Items B4.1.1 - B4.1.5 plenum cover and plenum cylinder components.	*Same as for the effect of Void Swelling on Items B4.1.1 - B4.1.5 plenum cover and plenum cylinder components.*	Yes, plant specific AMP
Same as for the effect of Neutron Irradiation Embrittlement on Items B4.2.1 - B4.2.5 upper grid assembly components.	*Same as for the effect of Neutron Irradiation Embrittlement on Items B4.2.1 - B4.2.5 upper grid assembly components.*	Yes parameters monitored/ inspected and detection of aging effects should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B4.6.1 - B4. 6.10	Lower Grid (LG) Assembly	LG Rib Section, Fuel Assembly Support Pads, LG Rib-to-Shell Forging Screws, LG Flow Dist. Plate, Orifice Plugs, LG & Shell Forgings, Lower Internals Assembly-to-Thermal Shield Bolts, Guide Blocks & Bolts, Shock Pads & Bolts, Support Post Pipes	Type 304 SS, Lower Internal Assembly-to-Thermal Shield Bolts: A-286, X-750; Other Bolts and Screws: Gr. B-8 SS	Chemically Treated Borated Water up to 340°C (644°F)	Cumulative fatigue damage	Fatigue	Design Code of record or later approved Codes.
B4.6.3, B4.6.7	Lower Grid (LG) Assembly	LG Rib-to-Shell Forging Screws, Lower Internals Assembly-to-Thermal Shield Bolts	Shell Forging Screws: Gr. B-8 SS; Thermal Shield Bolts: A-286, X-750	Chemically Treated Borated Water up to 340°C (644°F)	Loss of Preload	Stress Relaxation	*Same as for the effect of Stress Relaxation on Item B4.3.4 CRGT flange-to-upper grid screws.*
B4.6.2, B4.6.8	Lower Grid (LG) Assembly	Fuel Assembly Support Pads, Guide Blocks	Type 304 SS	Chemically Treated Borated Water up to 340°C (644°F)	Loss of Material	Wear	*Same as for the effect of Wear on Items B4.2.3 Fuel Assembly Support Pads and B4.2.4 Plenum Rib Pads.*
B4.7.1, B4.7.3, B4.7.4	Flow Distributor Assembly	Flow Distributor Head and Flange, Incore Guide Support Plate, Clamping Ring	Type 304 SS	Chemically Treated Borated Water up to 340°C (644°F)	Crack initiation & growth	SCC, IASCC	
B4.7.2	Flow Distributor Assembly	Shell Forging-to-Flow Distributor Bolts	A-286, X-750	Chemically Treated Borated Water up to 340°C (644°F)	Crack initiation & growth	SCC, IASCC	

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Same as for the effect of Fatigue on Items B4.1.1 - B4.1.5 plenum cover and plenum cylinder components.	*Same as for the effect of Fatigue on Items B4.1.1 - B4.1.5 plenum cover and plenum cylinder components.*	Yes, TLAA
Same as for the effect of Stress Relaxation on Item B4.3.4 CRGT flange-to-upper grid screws.	*Same as for the effect of Stress Relaxation on Item B4.3.4 CRGT flange-to-upper grid screws.*	Yes, Element 3 and 4 should be further evaluated
Same as for the effect of Wear on Items B4.2.3 Fuel Assembly Support Pads and B4.2.4 Plenum Rib Pads.	*Same as for the effect of Wear on Items B4.2.3 Fuel Assembly Support Pads and B4.2.4 Plenum Rib Pads.*	No
Same as for the effect of SCC or IASCC on Items B4.1.1 plenum cover assembly, B4.1.2 plenum cylinder, and B4.1.3 reinforcing plates.	*Same as for the effect of SCC or IASCC on Items B4.1.1 plenum cover assembly, B4.1.2 plenum cylinder, and B4.1.3 reinforcing plates.*	Yes, detection of aging effects should be further evaluated
Same as for the effect of SCC or IASCC on Items B4.1.4 top flange -to-cover bolts and B4.1.5 bottom flange-to-upper grid screws.	*Same as for the effect of SCC or IASCC on Items B4.1.4 top flange -to-cover bolts and B4.1.5 bottom flange-to-upper grid screws.*	Yes, detection of aging effects should be further evaluated

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

B4. REACTOR VESSEL INTERNALS (PWR) - Babcock & Wilcox

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B4.7.1 - B4.7.4	Flow Distributor Assembly	Flow Distributor Head and Flange, Shell Forging-to-Flow Distributor Bolts, Incore Guide Support Plate, Clamping Ring	Type 304 SS; Bolts: A-286, X-750	Chemically Treated Borated Water up to 340°C (644°F)	Changes in Dimensions	Void Swelling	-
B4.7.1 - B4.7.4	Flow Distributor Assembly	Flow Distributor Head and Flange, Shell Forging-to-Flow Distributor Bolts, Incore Guide Support Plate, Clamping Ring	Type 304 SS; Bolts: A-286, X-750	Chemically Treated Borated Water up to 340°C (644°F) Neutron Fluence of greater than 10^{17} n/cm^2 (E>1 MeV)	Loss of Fracture Toughness	Neutron Irradiation Embrittle–ment	*Same as for the effect of Neutron Irradiation Embrittlement on Items B4.2.1 - B4.2.5 upper grid assembly components.*
B4.7.2	Flow Distributor Assembly	Shell Forging-to-Flow Distributor Bolts	A-286, X-750	Chemically Treated Borated Water up to 340°C (644°F)	Loss of Preload	Stress Relaxation	*Same as for the effect of Stress Relaxation on Item B4.3.4 CRGT flange-to-upper grid screws.*
B4.8	Thermal Shield	-	SS	Chemically Treated Borated Water up to 340°C (644°F)	Crack initiation & growth	SCC, IASCC	*Same as for the effect of SCC or IASCC on Items B4.1.1 plenum cover assembly, B4.1.2 plenum cylinder, and B4.1.3 reinforcing plates.*
B4.8	Thermal Shield	-	SS	Chemically Treated Borated Water up to 340°C (644°F)	Changes in Dimensions	Void Swelling	-
B4.8	Thermal Shield	-	SS	Chemically Treated Borated Water up to 340°C Fluence > 10^{17} n/cm^2 (E>1 MeV)	Loss of Fracture Toughness	Neutron Irradiation Embrittle–ment	*Same as for the effect of Neutron Irradiation Embrittlement on Items B4.2.1 - B4.2.5 upper grid assembly components.*

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Same as for the effect of Void Swelling on Items B4.1.1 - B4.1.5 plenum cover and plenum cylinder components.	*Same as for the effect of Void Swelling on Items B4.1.1 - B4.1.5 plenum cover and plenum cylinder components.*	Yes, plant specific AMP
Same as for the effect of Neutron Irradiation Embrittlement on Items B4.2.1 - B4.2.5 upper grid assembly components.	*Same as for the effect of Neutron Irradiation Embrittlement on Items B4.2.1 - B4.2.5 upper grid assembly components.*	Yes parameters monitored/ inspected and detection of aging effects should be further evaluated
Same as for the effect of Stress Relaxation on Item B4.3.4 CRGT flange-to-upper grid screws.	*Same as for the effect of Stress Relaxation on Item B4.3.4 CRGT flange-to-upper grid screws.*	Yes, Element 3 and 4 should be further evaluated
Same as for the effect of SCC or IASCC on Items B4.1.1 plenum cover assembly, B4.1.2 plenum cylinder, and B4.1.3 reinforcing plates.	*Same as for the effect of SCC or IASCC on Items B4.1.1 plenum cover assembly, B4.1.2 plenum cylinder, and B4.1.3 reinforcing plates.*	Yes, detection of aging effects should be further evaluated
Same as for the effect of Void Swelling on Items B4.1.1 - B4.1.5 plenum cover and plenum cylinder components.	*Same as for the effect of Void Swelling on Items B4.1.1 - B4.1.5 plenum cover and plenum cylinder components.*	Yes, plant specific AMP
Same as for the effect of Neutron Irradiation Embrittlement on Items B4.2.1 - B4.2.5 upper grid assembly components.	*Same as for the effect of Neutron Irradiation Embrittlement on Items B4.2.1 - B4.2.5 upper grid assembly components.*	Yes parameters monitored/ inspected and detection of aging effects should be further evaluated

C1. Reactor Coolant Pressure Boundary (Boiling Water Reactor)

C1.1 Piping & Fittings

C1.1.1 Main Steam

C1.1.2 Feedwater

C1.1.3 High Pressure Coolant Injection (HPCI) System

C1.1.4 Reactor Core Isolation Cooling (RCIC) System

C1.1.5 Recirculation

C1.1.6 Residual Heat Removal (RHR) System

C1.1.7 Low Pressure Coolant Injection (LPCI) System

C1.1.8 Low Pressure Core Spray (LPCS) System

C1.1.9 High Pressure Core Spray (HPCS) System

C1.1.10 Lines to Isolation Condenser

C1.1.11 Lines to Reactor Water Cleanup (RWC) and Standby Liquid Control (SLC) Systems

C1.1.12 Steam Line to HPCI and RCIC Pump Turbine

C1.1.13 Small Bore Piping

C1.2 Recirculation Pump

C1.2.1 Casing

C1.2.2 Cover

C1.2.3 Seal Flange

C1.2.4 Closure Bolting

C1.3 Valves

C1.3.1 Body

C1.3.2 Bonnet

C1.3.3 Seal Flange

C1.3.4 Closure Bolting

C1.4 Isolation Condenser

C1.4.1 Tubing

C1.4.2 Tubesheet

C1.4.3 Channel Head

C1.4.4 Shell

C1. Reactor Coolant Pressure Boundary (Boiling Water Reactor)

System, Structures, and Components

The system, structures, and components included in this table comprise the boiling water reactor (BWR) primary coolant pressure boundary and consist of the reactor coolant recirculation system and portions of other systems connected to the pressure vessel extending to the second containment isolation valve or to the first anchor point, and including the containment isolation valves. The connected systems include residual heat removal (RHR), low–pressure core spray (LPCS), high–pressure core spray (HPCS), low–pressure coolant injection (LPCI), high–pressure coolant injection (HPCI), reactor core isolation cooling (RCIC), isolation condenser (IC), reactor water cleanup (RWC), feedwater (FW), and main steam (MS) systems, and steam line to HPCI and RCIC pump turbine. Based on the Nuclear Regulatory Commission Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all systems, structures, and components in the reactor coolant pressure boundary are classified as Group A Quality Standards.

The pumps and valves internals are considered to be active components. They perform their intended functions with moving parts or with a change in configuration, or are subject to replacement based on qualified life or specified time period, and are not subject to aging management review pursuant to 10 CFR 54.21(a)(1)(i and ii).

System Interfaces

The systems that interface with the reactor coolant pressure boundary include the reactor pressure vessel (Table IV A1), emergency core cooling system (Table V D2), standby liquid control system (Table VII E2), reactor water cleanup system (Table VII E3), shutdown cooling system (older plants) (Table VII E5), main steam system (Table VIII B2), and feedwater system (Table VIII D2).

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

C1. REACTOR COOLANT PRESSURE BOUNDARY (Boiling Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C1.1.1, C1. 1.12	Piping & Fittings	Main Steam, Steam Line to HPCI and RCIC Pump Turbine	Carbon Steel (CS) SA106-Gr B, SA333-Gr 6, SA155-Gr KCF70	288°C (550°F) Steam	Wall Thinning	Flow Accelerated Corrosion (FAC)	EPRI NSAC–202L–R2. NRC GL 89-08. NUREG-1344. *Supporting Documents:* BWRVIP-75. BWRVIP-29 (EPRI TR-103515). *Operating Experience* NRC Bl 87-01. NRC IN 81-28. NRC IN 89-53. NRC IN 91-18. NRC IN 91-18 S1. NRC IN 92-35. NRC IN 93-21. NRC IN 95-11. NRC IN 97-84.
C1.1.1	Piping & Fittings	Main Steam	CS SA106-Gr B, SA333-Gr 6, SA155-Gr KCF70	288°C (550°F) Steam	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
C1.1.2	Piping & Fittings	Feedwater	CS SA106-Gr B, SA333-Gr 6, SA155-Gr KCF70	Up to 225°C, (437°F) Reactor Coolant Water	Wall Thinning	FAC	*Same as for the effect of FAC on Items C1.1.1 and C1.1.12 piping and fittings.*
C1.1.2	Piping & Fittings	Feedwater	CS SA106-Gr B, SA333-Gr 6, SA155-Gr KCF70	Up to 225°C, (437°F) Reactor Coolant Water	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
C1.1.3, C1.1.4	Piping & Fittings	High Pressure Coolant Injection (HPCI), Reactor Core Isolation Cooling (RCIC)	CS SA106-Gr B, SA333-Gr 6, SA155-Gr KCF70	288°C (550°F) Reactor Coolant Water or Steam	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Implementation of EPRI guidelines of NSAC–202L–R2 for effective flow accelerated corrosion (FAC) program.	For evaluation and technical basis of the 10 elements of the AMP, see Chapter XI.M6 "Flow Accelerated Corrosion."	No
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii), and for addressing environmental effects on fatigue. See Chapter X of this report, for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes TLAA
Same as for the effect of FAC on Items C1.1.1 and C1.1.12 piping and fittings.	*Same as for the effect of FAC on Items C1.1.1 and C1.1.12 piping and fittings.*	No
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	*Same as for the effect of Fatigue on Item C1.1.1 main steam line piping and fittings.*	Yes TLAA
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	*Same as for the effect of Fatigue on Item C1.1.1 main steam line piping and fittings.*	Yes TLAA

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C1.1.5 - C1. 1.11	Piping & Fittings	Recirculation, Residual Heat Removal (RHR), Low Pressure Coolant Injection (LPCI), Low Pressure Core Spray (LPCS), High Pressure Core Spray (HPCS), Isolation Condenser (IC), Lines to Reactor Water Cleanup (RWC) and Standby Liquid Control (SLC) Systems	Stainless Steel (SS) (e.g., Types 304, 316, or 316NG); Cast Austenitic Stainless Steel (CASS); Nickel Alloys (e.g., Alloys 600, 182, and 82)	288°C (550°F) Reactor Coolant Water (RCW) or Steam	Crack Initiation and Growth	Stress Corrosion Cracking (SCC), Inter-granular Stress Corrosion Cracking (IGSCC)	NUREG-0313, Rev. 2. NRC GL 88-01. NRC GL 88-01, S 1. ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. BWRVIP-29 (EPRI TR-103515). BWRVIP-27. BWRVIP-42. BWRVIP-75. *Supporting Documents:* Code Case N504-1. BWRVIP-03. BWRVIP-14. BWRVIP-53. BWRVIP-59. BWRVIP-60. BWRVIP-61. BWRVIP-62.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Program delineated in NUREG-0313, Rev. 2 and NRC Generic letter (GL) 88-01 and its Supplement 1, and inservice inspection in conformance with ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1. BWRVIP-75 provides technical basis for revisions to GL 88-01 inspection schedule, and BWRVIP-27 provides standby liquid control/core plate ΔP inspection and flaw evaluation guidelines. Coolant water chemistry is monitored and maintained in accordance with the guidelines in BWRVIP-29 (TR-103515) to minimize the potential of crack initiation and growth. *Supporting documents*: BWRVIP-03 for reactor pressure vessel internals examination guidelines; BWRVIP-14, -59, and -60 for evaluation of crack growth; BWRVIP-53 standby liquid control line repair design criteria; BWRVIP-61 for BWR vessel and internals induction heating stress improvement effectiveness on crack growth in operating plants; and BWRVIP-62 for technical basis for inspection relief for internal components with hydrogen injection.	***(1) Scope of Program:*** The program focuses on managing and implementing countermeasures to mitigate IGSCC and inservice inspection (ISI) to monitor IGSCC and its effects on the intended function of austenitic stainless steel (SS) piping 4 in. or larger in diameter, and reactor vessel attachments and appurtenances. NUREG-0313 and GL 88-01, respectively, describe the technical basis and staff guidance regarding mitigating IGSCC in BWRs. ***(2) Preventive Actions:*** Based on NUREG-0313, mitigation of IGSCC is by selection of material considered resistant to sensitization and IGSCC, e.g., low-carbon grades of austenitic SSs and weld metal, with a maximum carbon of 0.035% and minimum 7.5% ferrite in weld metal, and by special processing such as solution heat treatment, heat sink welding, and induction heating or mechanical stress improvement. Coolant water chemistry is monitored and maintained according to EPRI guidelines in BWRVIP-29 (TR-103515) to minimize the potential of crack initiation and growth. Also, hydrogen water chemistry and stringent control of conductivity is used to inhibit IGSCC. ***(3) Parameters Monitored/Inspected:*** Inspection and flaw evaluation are to be performed in accordance with referenced BWRVIP guideline, as approved by the NRC staff. ***(4) Detection of Aging Effects:*** Aging degradation of the piping can not occur without crack initiation and growth; extent, method, and schedule of inspection as delineated in GL 88-01 and updated in BWRVIP-75 is adequate and will assure timely detection of cracks before the loss of intended function of austenitic SS piping and fittings. The inspection guidance in BWRVIP-75 is as described in staff approved topical report. The topical (BWRVIP-75) when approved by the staff may serve to replace the inspection extent and schedule in GL 88-01. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with GL 88-01 or applicable approved BWRVIP guideline. ***(6) Acceptance Criteria:*** Any IGSCC degradation is evaluated according to applicable approved BWRVIP guideline. ***(7) Corrective Actions:*** The guidance for weld overlay repair, stress improvement or replacement is provided in GL 88-01, Code Case N 504-1, or ASME Section XI. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** IGSCC has occurred in small- and large-diameter BWR piping made of austenitic SSs and Nickel-base alloys. Significant cracking has occurred in recirculation, core spray, and RHR systems and reactor water cleanup system piping welds.	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C1.1.6, - C1. 1.11	Piping & Fittings	RHR, LPCI, LPCS, HPCS, Lines to IC, Lines to RWC & SLC Systems	CASS	288°C (550°F) Reactor Coolant Water or Steam	Loss of Fracture Toughness	Thermal Aging Embrittle-ment	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. Letter from Christopher I. Grimes (NRC) to Douglas J. Walters (NEI) dated 5/19/2000.
C1.1.5 - C1. 1.11	Piping & Fittings	Recirculation, RHR, LPCI, LPCS, HPCS, IC, Lines to RWC and SLC Systems	CS, CASS, SS	288°C (550°F) Reactor Coolant Water or Steam	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
C1. 1.13	Piping & Fittings	Small Bore Piping	SS, CS	288°C (550°F) Reactor Coolant Water	Crack Initiation and Growth	Unantici-pated Thermal and Mechani-cal Loading	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. BWRVIP-29 (EPRI TR-103515). Plant Technical Specifications. NRC IN 97-46.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
The reactor coolant system components are inspected in accordance with ASME Section XI, Subsection IWB. This inspection is not sufficient to detect the effects of loss of fracture toughness due to thermal aging embrittlement of cast austenitic stainless steel (CASS) piping. An acceptable AMP consists of the following: Determination of the susceptibility of CASS piping to thermal aging embrittlement based on casting method, Mo content, and percent ferrite. For "potentially susceptible" piping, aging management is accomplished either through enhanced volumetric examination or plant/component–specific flaw tolerance evaluation. Additional inspection or evaluations are not required for "not susceptible" piping to demonstrate that the material has adequate fracture toughness. For pump casings and valve bodies, screening for susceptibility to thermal aging is not required.	For evaluation and technical basis of the 10 elements of the AMP, see Chapter XI.M1 "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)".	No
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	*Same as for the effect of Fatigue on Item C1.1.1 main steam line piping and fittings.*	Yes TLAA
Inservice inspection in conformance with ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination category B-J for pressure retaining welds in piping and testing category B-P for system leakage, and primary water chemistry is monitored and maintained in accordance with the guidelines in BWRVIP-29 (TR-103515) to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes preventive measures to inhibit cracking and inservice inspection (ISI) to monitor the effects of cracking on the intended function of small-bore piping of reactor coolant system and connected lines. ***(2) Preventive Actions:*** Coolant water chemistry is monitored and maintained according to BWRVIP-29 (TR-103515) to minimize the potential of crack initiation and growth. Also, hydrogen water chemistry and stringent control of conductivity is used to inhibit IGSCC. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of cracking on the intended function of piping and fittings by detection cracks and leakage by ISI. Inspection requirements of Table IWB 2500-1, examination category B-J specifies surface examination for circumferential and longitudinal welds in each pipe or branch run less than 4 inches nominal pipe size (NPS), and category B-P specifies visual VT-2 (IWA-5240) examination of all pressure retaining components during system leakage test (IWB-5221) and system hydrostatic test (IWB-5222). However, inspection in accordance with ASME Section XI does not require volumetric examination of pipes less than NPS 4. A plant-specific destructive examination or a nondestructive examination (NDE) that	Yes parameters monitored/inspected and detection of aging effects should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C1.2.1 - C1.2.3	Recirculation Pump	Casing, Cover, Seal Flange	CASS, SS	288°C (550°F) Reactor Coolant Water	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
C1.2.1	Recirculation Pump	Casing	CASS, SS	288°C (550°F) Reactor Coolant Water	Crack Initiation and Growth	SCC, IGSCC	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. NUREG-0313, Rev. 2. NRC GL 88-01. NRC GL 88-01, S 1. BWRVIP-29 (EPRI TR-103515).

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) permits inspection of the inside surfaces of the piping should be conducted to ensure that cracking has not occurred and the component intended function will be maintained during the extended period. ***(4) Detection of Aging Effects:*** Degradation of the piping due to cracking would result in leakage of coolant. A one-time inspection of a sample of locations most susceptible to cracking should be conducted to verify that service-induced weld cracking is not occurring in the small-bore piping less than NPS 4, including pipe, fittings, and branch connections. Actual inspection locations should be based on physical accessibility, exposure levels, and NDE examinations techniques, and locations identified in NRC Information Notice (IN) 97-46. ***(5) Monitoring and Trending:*** System leakage test is conducted prior to plant startup following each refueling outage, and hydrostatic test at or near the end of each inspection interval. The results of one-time inspection will be used to dictate the frequency of future inspections. ***(6) Acceptance Criteria:*** Any relevant conditions that may be detected during the leakage tests are evaluated in accordance with IWC-3516. ***(7) Corrective Actions:*** Repair is in conformance with IWA-4000 and IWB-4000, replacement according to IWA-7000 and IWB-7000. If destructive examination is employed, repair and replacement are in accordance with ASME Section XI rules. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Cracking has occurred in HPCI piping (IN 89-80) and instrument lines (LER 50-249/99-003-1) due to thermal and mechanical loading.	
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	*Same as for the effect of Fatigue on Item C1.1.1 main steam line piping and fittings.*	Yes TLAA
Program delineated in NUREG-0313, Rev. 2 and NRC Generic letter (GL) 88-01 and its Supplement 1, and inservice inspection in conformance with ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1. Coolant water chemistry is monitored and maintained in accordance with the guidelines in BWRVIP-29 (TR-103515) to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC and inservice inspection (ISI) to monitor the effects of SCC on intended function of the pump. NUREG-0313 and GL 88-01, respectively, describe the technical basis and staff guidance regarding the problem of IGSCC in BWRs. ***(2) Preventive Actions:*** Based on NUREG-0313, mitigation of IGSCC is by selection of material considered resistant to sensitization and IGSCC, e.g., low-carbon grades of cast SSs and weld metal, with a maximum carbon of 0.035% and minimum 7.5% ferrite. Also, hydrogen water chemistry and stringent control of conductivity is used to inhibit IGSCC. High–carbon grades of cast SS, e.g., CF–8 and CF–8M are susceptible to SCC. The aging management program	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) must therefore rely upon ISI in accordance with GL 88-01 to detect possible degradation. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of SCC on the intended function of the pump by detection and sizing of cracks by ISI. The inspection requirements of pump casing welds are delineated in Table IWB 2500-1, examination category B-L-1; alternative examination requirements for CASS pump casings are in accordance with ASME Code Case N-481. Examination category B–L–2 for casing specifies visual VT–3 examination of internal surfaces of the pump. Requirements of testing category B-P conducted according to IWA–5000 specify visual VT-2 (IWA-5240) examination of all pressure retaining boundary of the pump during system leakage test (IWB-5221) and system hydrostatic test (IWB-5222). Also, coolant water chemistry is monitored and maintained in accordance with the guidelines in BWRVIP-29 (TR-103515) to minimize the potential of crack initiation and growth. ***(4) Detection of Aging Effects:*** Degradation of the pump due to SCC can not occur without crack initiation and growth; extent and schedule of inspection as delineated in GL 88-01 will assure detection of cracks before the loss of intended function of the pump. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with GL 88-01 should provide timely detection of cracks. All welds are inspected each inspection period from at least one pump in each group performing similar functions in the system. Visual examination is required only when the pump is disassembled for maintenance, repair, or volumetric examination, but at least once during the period. System leakage test is conducted prior to plant startup following each refueling outage, and hydrostatic test is conducted at or near the end of each inspection interval. ***(6) Acceptance Criteria:*** Any SCC degradation is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3500. Supplementary surface examination may be performed on interior and/or exterior surfaces when flaws are detected in volumetric examination. ***(7) Corrective Actions:*** Repair is in conformance with IWA-4000 and IWB-4000 or GL 88-01. Continued operation without repair require that crack growth calculations be performed according to the guidance of GL 88-01 or other approved procedure. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** The comprehensive AMP outlined in NUREG-0313 and GL 88-01 addresses improvements in all elements that cause IGSCC and has provided effective means of ensuring structural integrity of the primary coolant pressure boundary.	

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C1.2.1	Recirculation Pump	Casing	CASS	288°C (550°F) Reactor Coolant Water	Loss of Fracture Toughness	Thermal Aging Embrittle-ment	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. Letter from Christopher I. Grimes (NRC) to Douglas J. Walters (NEI) dated 5/19/2000.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Inservice inspection in conformance with ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1. For pump casings, screening for susceptibility to thermal aging is not required.	***(1) Scope of Program:*** The existing ASME Section XI inspection requirements, including the alternative requirements of ASME Code Case N-481 for pump casings, are considered adequate for managing the effects of loss of fracture toughness due to thermal embrittlement of CASS pump casings. Based on the assessment in the NRC letter from Christopher Grimes to Douglas Walters, May 19, 2000, screening for susceptibility to thermal aging embrittlement is not required. ***(2) Preventive Actions:*** The program provides no guidance on methods to mitigate thermal aging embrittlement. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of loss of fracture toughness on the intended function of the component by timely detection and sizing of cracks by ISI. Loss of fracture toughness is of consequence only if cracks exist. Cracking is expected to initiate at the surface and should be detectable by ISI. Inspection requirements of pump casing welds are delineated in Table IWB 2500-1, alternative examination requirements for CASS pump casings are in accordance with ASME Code Case N-481. Examination category B-L–1 specifies volumetric examination of all welds extending 1/2 in. on either side of the weld and through wall thickness, and category B–L–2 specifies visual VT–3 examination of internal surfaces of the pump. Requirements of testing category B-P conducted according to IWA–5000 specify visual VT-2 (IWA-5240) examination of all pressure retaining boundary of the pump during system leakage test (IWB-5221) and system hydrostatic test (IWB-5222). ***(4) Detection of Aging Effects:*** The extent and schedule of inspection as delineated in IWB 2500-1 will assure detection of cracks before the loss of intended function of the component. ***(5) Monitoring and Trending:*** Inspection schedule IWB 2500-1 should provide timely detection of cracks. All welds are inspected each inspection period from at least one pump in each group performing similar functions in the system. Visual examination is required only when the pump is disassembled for maintenance, repair, or volumetric examination, but at least once during the period. System leakage test is conducted prior to plant startup following each refueling outage, and hydrostatic test is conducted at or near the end of each inspection interval. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3500. ***(7) Corrective Actions:*** Repair is in conformance with IWA-4000 and IWB-4000. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Operating experience has shown no significant degradation of pump casings due to thermal aging embrittlement of CASS.	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C1.2.3, C1.2.4	Recirculation Pump	Seal Flange, Closure Bolting	Flange: SS; Bolting: High Strength Low-Alloy Steel (HSLAS) SA193 GrB7	Air, Leaking Reactor Coolant Water (RCW) and/or Steam at 288°C (550°F)	Loss of Material	Wear	NRC GL 91-17. NUREG-1339. EPRI NP-5769. IEB 82-02. ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.
C1.2.4	Recirculation Pump	Closure Bolting	HSLAS SA193 GrB7	Air, Leaking Reactor Coolant Water and/or Steam at 288°C	Loss of Preload	Stress Relaxation	NRC GL 91-17. NUREG-1339. EPRI NP-5769. IEB 82-02. ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.
C1.2.4	Recirculation Pump	Closure Bolting	HSLAS SA193 GrB7	Air, Leaking RCW and/or Steam at 288°C	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
C1.3.1	Valves (Check, Control, Hand, Motor-Operated, and Relief, and Containment Isolation Valves)	Body	CS	288°C (550°F) Reactor Coolant Water	Wall Thinning	FAC	*Same as for the effect of FAC on Items C1.1.1 and C1.1.12 piping and fittings.*
C1.3.1, C1.3.2	Valves (Check, Control, Hand, Motor Operated, Relief, and Containment Isolation Valves)	Body, Bonnet	CASS	288°C (550°F) Reactor Coolant Water	Loss of Fracture Toughness	Thermal Aging Embrittle-ment	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. Letter from Christopher I. Grimes (NRC) to Douglas J. Walters (NEI) dated 5/19/2000.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Recommendations for a comprehensive bolting integrity program delineated in NUREG–1339 on resolution of Generic Safety Issue 29 and recommendations of the NRC Generic Letter 91-17; additional details on bolting integrity outlined in EPRI NP-5769; and inservice inspection in conformance with ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1.	For evaluation and technical basis of the 10 elements of the AMP, see Chapter XI.M12 "Bolting Integrity."	No
Same as for the effect of Wear on Item C1.2.4 closure bolting for recirculation pump.	*Same as for the effect of Wear on Item C1.2.4 closure bolting for recirculation pump.*	No
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal; check Code limits for allowable cycles (less than 7000 cycles) of thermal stress range. See the Standard Review Plan, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes TLAA
Same as for the effect of FAC on Items C1.1.1 and C1.1.12 piping and fittings.	*Same as for the effect of FAC on Items C1.1.1 and C1.1.12 piping and fittings.*	No
Inservice inspection in conformance with ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1. For valve bodies, screening for susceptibility to thermal aging is not required.	***(1) Scope of Program:*** The existing ASME Section XI inspection requirements are considered adequate for managing the effects of loss of fracture toughness due to thermal embrittlement of CASS valve bodies. Based on the assessment in the NRC letter from Christopher Grimes to Douglas Walters, May 19, 2000, screening for susceptibility to thermal aging embrittlement is not required. ***(2) Preventive Actions:*** The program provides no guidance on methods to mitigate thermal aging embrittlement. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of loss of fracture toughness on the intended function of the component by timely detection and sizing of cracks by ISI. Loss of fracture toughness is of consequence only if cracks exist. Cracking is expected inspection requirements of welds in valve bodies are delineated in Table IWB 2500-1. Examination category B-M–1 specifies for all welds NPS 4 or larger, volumetric examination extending 1/2 in. on either side of the weld and through wall thickness, and for welds less than NPS 4, surface examination of OD surface extending 1/2 in. on either side of the weld. Category B–M–2	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C1.3.1, C1.3.2	Valves (Check, Control, Hand, Motor-Operated, Relief, and Containment Isolation Valves)	Body, Bonnet	CASS, SS	288°C (550°F) Reactor Coolant Water	Crack Initiation and Growth	SCC, IGSCC	NUREG-0313, Rev. 2. ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. NRC GL 88-01. NRC GL 88-01, S 1. BWRVIP-29 (EPRI TR-103515).

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	specifies visual VT–3 examination of internal surfaces of the valve. Requirements of testing category B-P conducted according to IWA–5000 specify visual VT-2 (IWA-5240) examination of all pressure retaining boundary of the valve during system leakage test (IWB-5221) and system hydrostatic test (IWB-5222). ***(4) Detection of Aging Effects:*** The extent and schedule of inspection as delineated in IWB 2500-1 will assure detection of cracks before the loss of intended function of the component. ***(5) Monitoring and Trending:*** Inspection schedule IWB 2500-1 should provide timely detection of cracks. All welds are inspected each inspection period from at least one valve in each group with similar design and performing similar functions in the system. Visual examination is required only when the valve is disassembled for maintenance, repair, or volumetric examination, but at least once during the period. System leakage test is conducted prior to plant startup following each refueling outage, and hydrostatic test is conducted at or near the end of each inspection interval. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3500. ***(7) Corrective Actions:*** Repair is in conformance with IWA-4000 and IWB-4000. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented according to 10 CFR 50 Appendix B and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Operating experience has shown no significant degradation of valve bodies due to thermal aging embrittlement of CASS.	
Guidelines of NUREG-0313, Rev. 2 and NRC Generic letter (GL) 88-01 and its Supplement 1; inservice inspection in conformance with ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1. Coolant water chemistry is monitored and maintained in accordance with the guidelines in BWRVIP-29 (TR-103515) to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes preventive measures to mitigate stress corrosion cracking (SCC) and inservice inspection (ISI) to monitor the effects of SCC on intended function of the valves. NUREG-0313 and GL 88-01, respectively, describe the technical basis and staff guidance regarding the problem of IGSCC in BWRs. ***(2) Preventive Actions:*** Based on NUREG-0313, mitigation of IGSCC is by selection of material considered resistant to sensitization and IGSCC, e.g., low-carbon grades of cast SSs and weld metal, with a maximum carbon of 0.035% and minimum 7.5% ferrite. Also, hydrogen water chemistry and stringent control of conductivity is used to inhibit IGSCC. High–carbon grades of cast SS, e.g., CF–8 and CF–8M are susceptible to SCC. The aging management program must therefore rely upon ISI in accordance with GL 88-01 to detect possible degradation. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of SCC on intended function of the valves by detection and sizing of cracks by ISI. For welds NPS 4 or larger, the inspection requirements follow those delineated in Table IWB 2500-1, examination category B-M-1, and additional guidelines in GL 88-01. Examination category B–M–2 for valve bodies specifies visual VT–3 examination of internal surfaces of the valve. Inspection requirements of testing category B-P conducted according to IWA–5000 specify visual VT-2	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C1.3.1 - C1.3.3	Valves (Check, Control, Hand, Motor-Operated, Relief, and Containment Isolation Valves)	Body, Bonnet, Seal Flange	CS, CASS, SS	288°C (550°F) Reactor Coolant Water	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) (IWA-5240) examination of all pressure retaining components during system leakage test (IWB-5221) and system hydrostatic test (IWB-5222). Also, coolant water chemistry is monitored and maintained in accordance with the guidelines in BWRVIP-29 (TR-103515) to minimize the potential of crack initiation and growth. ***(4) Detection of Aging Effects:*** Degradation of the valves due to SCC can not occur without crack initiation and growth; extent and schedule of inspection as delineated in GL 88-01 will assure detection of cracks before the loss of the intended function of the valves. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with GL 88-01 should provide timely detection of cracks. All welds are inspected each inspection period from at least one valve in each group with similar design and performing similar functions in the system. Visual examination is required only when the valve is disassembled for maintenance, repair, or volumetric examination, but at least once during the period. System leakage test is conducted prior to plant startup following each refueling outage, and hydrostatic test is conducted at or near the end of each inspection interval. ***(6) Acceptance Criteria:*** Any SCC degradation is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400; IWB-3518 for volumetric examination of welds and 3519 for visual examination of valve internal surfaces. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWA-4000 and IWB-4000 or GL 88-01, and reexamination in accordance with requirements of IWA-2200. Continued operation without repair require that crack growth calculations be performed according to the guidance of GL 88-01 or other approved procedure. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** The comprehensive AMP outlined in NUREG-0313 and GL 88-01 has provided effective means of ensuring structural integrity of the primary coolant pressure boundary.	
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	*Same as for the effect of Fatigue on Item C1.1.1 main steam line piping and fittings.*	Yes TLAA

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

C1. REACTOR COOLANT PRESSURE BOUNDARY (Boiling Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C1.3.4	Valves	Closure Bolting	Flange: CS, SS Bolting: HSLAS	Air, Leaking Reactor Coolant Water (RCW) and/or Steam at 288°C (550°F)	Loss of Material	Wear	NRC GL 91-17. NUREG-1339. EPRI NP-5769. IEB 82-02. ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.
C1.3.4	Valves	Closure Bolting	HSLAS SA193 GrB7	Air, Leaking RCW and/or Steam at 288°C	Loss of Preload	Stress Relaxation	*Same as for the effect of Wear on Item C1.2.4 closure bolting for recirculation pump.*
C1.3.4	Valves	Closure Bolting	HSLAS SA193 GrB7	Air, Leaking RCW and/or Steam at 288°C	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
C1.4.1 - C1.4.4	Isolation Condenser	Tubing, Tubesheet, Channel Head, Shell	Tubes: SS; Tubesheet: CS, SS; Channel Head: CS, SS; Shell: CS	Tube side: Steam; Shell side: Demi-neralized Water	Crack Initiation and Growth	SCC, Unantici-pated Cyclic Loading	BWRVIP-29 (EPRI TR-103515). ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Same as for the effect of Wear on Item C1.2.4 closure bolting for recirculation pump.	*Same as for the effect of Wear on Item C1.2.4 closure bolting for recirculation pump.*	No
Same as for the effect of Wear on Item C1.2.4 closure bolting for recirculation pump.	*Same as for the effect of Wear on Item C1.2.4 closure bolting for recirculation pump.*	No
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	*Same as for the effect of Fatigue on Item C1.2.4 closure bolting for recirculation pump.*	Yes TLAA
The program includes monitoring and control of reactor water chemistry based on the EPRI guidelines of BWRVIP-29 (TR-103515) and inspection in accordance with ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a), Table IWC 2500-1. However, the AMP is inadequate to detect cracking due to stress corrosion cracking and cyclic loading or loss of material due to pitting and crevice corrosion, and verification of the effectiveness of the program is required to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period of operation. An acceptable verification program consists of an augmented program to include temperature and radioactivity monitoring of the shell side water, and eddy current testing of tubes	***(1) Scope of Program:*** The program includes monitoring and control of reactor water chemistry based on the EPRI guidelines of TR-103515 and inservice inspection (ISI) in accordance with ASME Section XI, augmented with temperature and radioactivity monitoring of the shell side water, and eddy current testing of the tubes. ***(2) Preventive Actions:*** Concentration of corrosive impurities are monitored to mitigate corrosion, and water quality (pH and conductivity) is maintained per the guidance outlined in the EPRI water chemistry guidelines which include specifications for chemical species, sampling and analysis frequencies, and corrective actions for control of reactor water chemistry. ***(3) Parameters Monitored/Inspected:*** The temperature monitoring is directly related to detecting leakage of the condensate return valves, the radioactivity measurement, ASME Section XI inspections, and eddy current testing to detect tube cracking or loss of material. ***(4) Detection of Aging Effects:*** Degradation of the component can not occur without crack initiation and growth and leakage. Monitoring of temperature would detect leakage; monitoring of radioactivity in shell side water and ASME inspection and eddy current testing assure detection of degradation before the loss of intended function of the component. ***(5) Monitoring and Trending:*** The results of temperature and radioactivity monitoring are monitored and trended. ***(6) Acceptance Criteria:*** The results of Section XI ISI are evaluated in accordance with IWC-3100 and acceptance standards of IWC-3400 and IWB-3500. ***(7) Corrective Actions:*** Root cause evaluation and appropriate corrective action is taken when acceptable limits are exceeded or leakage is detected. Repair is in conformance with IWA-4000 and replacement is in	Yes, plant specific augmenta-tion program

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C1.4.1 - C1.4.4	Isolation Condenser	Tubing, Tubesheet, Channel Head, Shell	Tubes: SS; Tubesheet: CS, SS; Channel Head: CS, SS; Shell: CS	Tube side: Steam; Shell side: deminera-lized water	Loss of Material	General, Pitting, and Crevice Corrosion	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. Plant Technical Specifications.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) accordance with IWA-7000. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Failure of the isolation condenser tube bundles due to thermal fatigue and transgranular stress corrosion cracking due to leaky valve has occurred in plants (LER 50-219/98-014).	
Same as for the effect of SCC or Unanticipated Cyclic Loading *on Items C1.4.1 – C1.4.4 isolation condenser components*.	*Same as for the effect of* SCC or Unanticipated Cyclic Loading *on Items C1.4.1 – C1.4.4 isolation condenser components*.	Yes, plant specific augmenta-tion program

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

C2.1 Reactor Coolant System Piping & Fittings

C2.1.1 Cold–Leg

C2.1.2 Hot–Leg

C2.1.3 Surge Line

C2.1.4 Spray Line

C2.1.5 Small-Bore RCS Piping, Fittings, and Branch Connections NPS 4 or Smaller

C2.2 Connected Systems Piping & Fittings

C2.2.1 Residual Heat Removal (RHR) or Low–Pressure Injection System (Decay Heat Removal (DHR)/ Shutdown System)

C2.2.2 Core Flood System (CFS)

C2.2.3 High-Pressure Injection System (Makeup & Letdown Functions)

C2.2.4 Chemical and Volume Control System

C2.2.5 Sampling System

C2.2.6 Drains and Instrument Lines

C2.2.7 Nozzles and Safe Ends

C2.2.8 Small-Bore Piping, Fittings, and Branch Connections NPS 4 or Smaller in Connected Systems

C2.3 Reactor Coolant Pump

C2.3.1 Casing

C2.3.2 Cover

C2.3.3 Closure Bolting

C2.4 Valves

C2.4.1 Body

C2.4.2 Bonnet

C2.4.3 Closure Bolting

C2.5 Pressurizer

C2.5.1 Shell/Heads

C2.5.2 Spray Line Nozzle

C2.5.3 Surge Line Nozzle

C2.5.4 Spray Head

C2.5.5 Thermal Sleeves

C2.5.6 Instrument penetrations

C2.5.7 Safe Ends

C2.5.8 Manway and Flanges

C2.5.9 Manway and Flange Bolting

C2.5.10 Heater Sheaths and Sleeves

C2.5.11 Support Keys, Skirt, & Shear Lugs

C2.5.12 Integral Support

C2.6 Pressurizer Relief Tank

C2.6.1 Tank Shell and Heads

C2.6.2 Flanges and Nozzles

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

System, Structures, and Components

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

The pumps and valves internals are considered to be active components. They perform their intended functions with moving parts or with a change in configuration, or are subject to replacement based on qualified life or specified time period, and are not subject to aging management review pursuant to 10 CFR 54.21(a)(1)(i and ii).

System Interfaces

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

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

C2. REACTOR COOLANT SYSTEM AND CONNECTED LINES (Pressurized Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C2.1.1, C2.1.2	Reactor Coolant System (RCS) Piping & Fittings	Cold-Leg, Hot-Leg	Stainless Steel (SS), Cast Austenitic SS (CASS), Carbon Steel (CS) with SS Cladding	Chemically Treated Borated Water up to 340°C (644°F)	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
C2.1.3, C2.1.4	Reactor Coolant System (RCS) Piping & Fittings	Surge Line, Spray Line	Surge Line: SS, CASS; Spray Line: SS	Chemically Treated Borated Water up to 340°C (644°F)	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
C2.1.1 - C2.1.4	Reactor Coolant System (RCS) Piping & Fittings	Cold-Leg, Hot-Leg, Surge Line, Spray Line	SS, CS with SS Cladding	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation and Growth	Stress Corrosion Cracking (SCC)	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. EPRI TR-105714 Rev. 3 or later revisions/update. *Supporting Documents*: Reg. Guide 1.43. Reg. Guide 1.44. *Operating Experience*: NRC IN 80-38. NRC IN 84-18. NRC IN 84-89. NRC IN 91-05. NRC IN 94-63. NRC IN 97-19.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii), and for addressing environmental effects on fatigue. See Chapter 10 of this report, for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes TLAA
Same as for the effect of Fatigue on Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.	*Same as for the effect of Fatigue on Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.*	Yes TLAA
Inservice inspection in conformance with ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1. Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 (Rev. 3 or later revisions or update) to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC of SS components and cladding, and inservice inspection (ISI) to monitor the effects of SCC on the intended function of piping and fittings of reactor coolant system and connected lines. ***(2) Preventive Actions:*** Control of halogens, sulfates, and oxygen in the primary water to less than 0.05, 0.05, and 0.005 ppm, respectively, during operation, and monitor and control of water chemistry during shut down, mitigate the potential of SCC. However, inadvertent introduction of contaminants into the coolant system can occur, e.g., introduced through the free surface of spent fuel pool which can be a natural collector of airborne contaminants or the introduction of oxygen during cool down. Also, cladding regions with low delta ferrite may be sensitized during post-weld heat treatment and thus become susceptible to SCC. The AMP must therefore rely upon water chemistry monitoring to detect possible excursions. Selection of material in compliance with the guidelines of Regulatory Guide (RG) 1.44 prevent or mitigate SCC, and guidelines of RG 1.43 prevent underclad cracking. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of SCC on the intended function of piping and fittings by detection and sizing of cracks by ISI. Requirements of Table IWB 2500-1, examination category B-J specifies for circumferential and longitudinal welds in each pipe or branch run NPS 4 or larger, volumetric and surface examination of ID region extending 1/4 in. on either side of the weld and 1/3 wall thickness deep, and surface examination of OD surface extending 1/2 in. on either side. Surface examination is conducted for circumferential and longitudinal welds in each pipe or branch run less than NPS 4. For socket welds, surface examination is specified of OD surface extending 1 in. on the buttered side and 1/2 in. on the other. Inspection requirements of testing category B-P conducted according to IWA–5000 specify visual VT-2 (IWA-5240) examination of all pressure retaining components, during system leakage test (IWB-5221) and system hydrostatic test (IWB-5222). ***(4) Detection of Aging Effects:*** Degradation of the piping due to SCC can not occur without crack initiation and growth; extent and	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C2.1.1, C2.1.2	Reactor Coolant System (RCS) Piping & Fittings	Cold-Leg, Hot-Leg (External Surfaces)	CS	Air, Leaking Chemically Treated Borated Water	Loss of Material	Boric Acid Corrosion of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. *Operating Experience* NRC IN 86-108 S 3.
C2.1.1 - C2.1.3	Reactor Coolant System (RCS) Piping & Fittings	Cold-Leg, Hot-Leg, Surge Line	CASS	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation and Growth	SCC	EPRI TR-105714 Rev. 3 or later revisions/update. NUREG-0313, Rev. 2.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) schedule of inspection assure detection of cracks before the loss of intended function of piping and fittings of reactor coolant system and connected lines. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 should provides timely detection of cracks. System leakage test is conducted prior to plant startup following each refueling outage, and hydrostatic test at or near the end of each inspection interval. ***(6) Acceptance Criteria:*** Any SCC degradation is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3500. Planar and liner flaws are sized according to IWA-3300 and IWA-3400. ***(7) Corrective Actions:*** Repair is in conformance with IWA-4000 and IWB-4000, replacement according to IWA-7000 and IWB-7000, and reexamination in accordance with IWA-2200. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Although the primary pressure boundary piping of PWRs have generally not been found to be affected by SCC because of low dissolved oxygen levels and control of primary water chemistry, potential of SCC exists due to inadvertent introduction of contaminants into the primary coolant system from (a) unacceptable levels of contaminants in the boric acid, and (b) free surface of the spent fuel pool (IN 84–18). Also, SCC could occur at creviced or cold worked locations (IN 84-89). SCC has been observed in safety injection lines (IN 97-19 and 84-18), charging pump casing cladding (INs 80-38 and 94-63), and instrument nozzles in safety injection tanks (IN 91-05).	
Implementation of NRC Generic Letter 88-05 and inservice inspection (ISI) in conformance with ASME Section XI (1989 edition or later edition as approved in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, to monitor the condition of the reactor coolant pressure boundary for occurrences of borated water leakage.	For evaluation and technical basis of the 10 elements of the AMP, see Chapter XI.M5, "Boric Acid Corrosion."	No
Monitoring and control of primary water chemistry in accordance with the EPRI guidelines in TR-105714 (Rev. 3 or later revisions or update) minimize the potential of SCC, and material selection according to the NUREG-0313, Rev. 2 guidelines of 0.035% C and 7.5% ferrite has reduced susceptibility to SCC. For CASS components that do not meet either one of the above guidelines, define plant-specific aging management program.	Plant-specific aging management program is to be evaluated for CASS components that do not meet either the water chemistry guidelines of EPRI TR-105714, or the C and ferrite content criteria of NUREG-0313, Rev. 2. The program will include (a) adequate inspection methods to ensure detection of cracks and (b) flaw evaluation methodology for CASS components that are susceptible to thermal aging embrittlement.	Yes, plant specific AMP

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C2.1.1 - C2.1.3	Reactor Coolant System (RCS) Piping & Fittings	Cold-Leg, Hot-Leg, Surge Line	CASS	Chemically Treated Borated Water up to 340°C (644°F)	Loss of Fracture Toughness	Thermal Aging Embrittle-ment	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. Letter from Christopher I. Grimes (NRC) to Douglas J. Walters (NEI) dated 5/19/2000.
C2.1.5	Reactor Coolant System (RCS) Piping & Fittings	Small-Bore RCS Piping, Fittings, and Branch Connections NPS 4 or smaller	SS, CS	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation and Growth	SCC, Unantici-pated Thermal and Mechani-cal Loading	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. EPRI TR-105714 Rev. 3 or later revisions/update. *Operating Experience* NRC IN 97-46.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
The reactor coolant system components are inspected in accordance with ASME Section XI, Subsection IWB. This inspection is not sufficient to detect the effects of loss of fracture toughness due to thermal aging embrittlement of cast austenitic stainless steel (CASS) piping. An acceptable AMP consists of the following: Determination of the susceptibility of CASS piping to thermal aging embrittlement based on casting method, Mo content, and percent ferrite. For "potentially susceptible" piping, aging management is accomplished either through enhanced volumetric examination or plant/component– specific flaw tolerance evaluation. Additional inspection or evaluations are not required for "not susceptible" piping to demonstrate that the material has adequate fracture toughness.	For evaluation and technical basis of the 10 elements of the AMP, see Chapter XI.M1 "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)".	No
Inservice inspection in conformance with ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, and primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 (Rev. 3 or later revisions or update) to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes preventive measures to inhibit cracking and inservice inspection (ISI) to monitor the effects of cracking on the intended function of small-bore piping of reactor coolant system and connected lines. ***(2) Preventive Actions:*** Control of halogens, sulfates, and oxygen in the primary water to less than 0.05, 0.05, and 0.005 ppm, respectively, during operation, and monitor and control of water chemistry during shut down, mitigate potential of SCC. However, inadvertent introduction of contaminants into the coolant system can occur, and impurities can buildup in stagnant regions and crevices. The AMP must therefore rely upon water chemistry monitoring to detect possible excursions. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of cracking on the intended function of piping and fittings by detection of cracks and leakage by ISI. Inspection requirements of Table IWB 2500-1, examination category B-J specifies surface examination for circumferential and longitudinal welds in each pipe or branch run less than 4 inches nominal pipe size (NPS), and category B-P specifies visual VT-2 (IWA-5240) examination of all pressure retaining components during system leakage test (IWB-5221) and system hydrostatic test (IWB-5222). However, inspection in accordance with ASME Section XI does not require volumetric examination of pipes less than NPS 4. A plant-specific destructive examination or a nondestructive examination (NDE) that permits inspection of the inside surfaces of the piping should be conducted to ensure that cracking has not occurred and the component intended function will be maintained during the extended period. ***(4) Detection of Aging Effects:*** Degradation of the piping due to cracking would result in leakage of coolant. A one-time inspection of a sample of locations most susceptible to cracking should be conducted to verify that service-induced weld cracking is not occurring in the small-bore piping less than NPS 4, including pipe, fittings, and branch	Yes parameters monitored/ inspected and detection of aging effects should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C2.2.1 - C2.2.4	Connected Systems Piping & Fittings	Residual Heat Removal (RHR), Core Flood System (CFS), High Pressure Injection System, Chemical & Volume Control System(CVCS)	SS	Chemically Treated Borated Water up to 340°C (644°F)	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
C2.2.5, C2.2.6	Connected Systems Piping & Fittings	Sampling System, Drains & In-strumentat Lines	CS with SS cladding, SS	Chemically Treated Borated Water up to 340°C (644°F)	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
C2.2.7	Connected Systems Piping & Fittings	Nozzles & Safe Ends	SS, CASS	Chemically Treated Borated Water up to 340°C (644°F)	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
C2.2.5, C2.2.6	Connected Systems Piping & Fittings	Sampling System, Drains & In-strumentat Lines (External Surfaces)	CS	Air, Leaking Chemically Treated Borated Water	Loss of Material	Boric Acid Corrosion of External Surfaces	*Same as for the effect of Boric Acid Corrosion of external surfaces of Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.*

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) connections. Actual inspection locations should be based on physical accessibility, exposure levels, and NDE examinations techniques, and locations identified in NRC Information Notice (IN) 97-46. ***(5) Monitoring and Trending:*** System leakage test is conducted prior to plant startup following each refueling outage, and hydrostatic test at or near the end of each inspection interval. The results of one-time inspection should be used to dictate the frequency of future inspections. ***(6) Acceptance Criteria:*** Any relevant conditions that may be detected during the leakage tests are evaluated in accordance with IWC-3500. (7) Corrective Actions: Repair is in conformance with IWA-4000 and IWB-4000, replacement according to IWA-7000 and IWB-7000. If destructive examination is employed, repair and replacement are in accordance with ASME Section XI rules. (8 & 9) Confirmation Process and Administrative Controls: Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. (10) Operating Experience: Cracking has occurred in unisolable section of combined makeup and high-pressure injection lines (IN 97-46).	
Same as for the effect of Fatigue on Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.	Same as for the effect of Fatigue on Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.	Yes TLAA
Same as for the effect of Fatigue on Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.	Same as for the effect of Fatigue on Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.	Yes TLAA
Same as for the effect of Fatigue on Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.	Same as for the effect of Fatigue on Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.	Yes TLAA
Same as for the effect of Boric Acid Corrosion of external surfaces of Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.	Same as for the effect of Boric Acid Corrosion of external surfaces of Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C2.2.7	Connected Systems Piping & Fittings	Nozzles & Safe Ends	CASS	Chemically Treated Borated Water up to 340°C (644°F)	Loss of Fracture Toughness	Thermal Aging Embrittle-ment	Same as for the effect of Thermal Aging Embrittlement on Items C2.1.1 - C2.1.3.
C2.2.1 - C2.2.6	Connected Systems Piping & Fittings	RHR, CFS, HPI System, CVCS, Sampling System, Drains & In-strumentat Lines	SS	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation and Growth	SCC	Same as for the effect of Stress Corrosion Cracking on Items C2.1.1 - C2.1.4 cold-leg, hot-leg, surge line, and spray line piping and fittings constructed of SS
C2.2.7	Connected Systems Piping & Fittings	Nozzles & Safe Ends	SS	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation and Growth	SCC	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. EPRI TR-105714 Rev. 3 or later revisions/update. Operating Experience: NRC IN 84-18. NRC IN 84-89.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Same as for the effect of Thermal Aging Embrittlement on Items C2.1.1 - C2.1.3 Cold-Leg, Hot-Leg, and Surge Line Piping and Fittings.	Same as for the effect of Thermal Aging Embrittlement on Items C2.1.1 - C2.1.3 Cold-Leg, Hot-Leg, and Surge Line Piping and Fittings.	No
Same as for the effect of Stress Corrosion Cracking on Items C2.1.1 - C2.1.4 cold-leg, hot-leg, surge line, and spray line piping and fittings constructed of SS.	Same as for the effect of Stress Corrosion Cracking on Items C2.1.1 - C2.1.4 cold-leg, hot-leg, surge line, and spray line piping and fittings constructed of SS.	No
Inservice inspection in conformance with ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1. Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 (Rev. 3 or later revisions or update) to minimize the potential of crack initiation and growth.	(1) Scope of Program: The program includes preventive measures to mitigate stress corrosion cracking (SCC) and inservice inspection (ISI) to manage the effects of crack initiation and growth on the intended function of the component. (2) Preventive Actions: Control of halogens, sulfates, and oxygen in the primary water to less than 0.05, 0.05, and 0.005 ppm, respectively, during operation, and monitor and control of water chemistry during shut down, mitigate potential of SCC. However, inadvertent introduction of contaminants into the primary coolant system can occur (IN 84–18). The AMP must therefore rely upon ISI in accordance with ASME Section XI to detect possible degradation. (3) Parameters Monitored/ Inspected: The AMP monitors the effects of cracking on the intended function of the component by detection and sizing of cracks by inservice inspection (ISI). Inspection requirements of Table IWB 2500-1, examination category B-D specifies volumetric examination of nozzle-to-vessel welds and nozzle inside radius, and B-F specifies for all nozzle-to-safe end butt welds NPS 4 or larger, volumetric or surface examination of ID region and surface examination of OD surface, only surface examination is conducted for all butt welds less than NPS 4 and for all nozzle-to-safe end socket welds. Testing category B-P specify visual VT-2 (IWA-5240) examination of all pressure retaining components during system leakage test (IWB-5221) and system hydrostatic test (IWB-5222). (4) Detection of Aging Effects: Aging degradation of the component can not occur without crack initiation and growth; the extent and schedule of inspection assure detection of cracks before the loss of intended function of the component. (5) Monitoring and Trending: Inspection schedule in accordance with IWB-2400 should provide timely detection of cracks. System leakage test is conducted prior to plant startup following each refueling outage, and hydrostatic test at or near the end of each inspection interval. (6) Acceptance Criteria: Any cracks are evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3500. (7) Corrective Actions: Repair is in conformance with IWA-4000 and IWB-4000, replacement	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C2.2.7	Connected Systems Piping & Fittings	Nozzles & Safe Ends	CASS	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation and Growth	SCC	EPRI TR-105714 Rev. 3 or later revisions/update. NUREG-0313, Rev. 2.
C2.2.8	Connected Systems Piping & Fittings	Small-Bore Piping, Fittings, and Branch Connections NPS 4 or smaller in Connected Systems	SS, CS	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation and Growth	SCC, Unantici-pated Thermal and Mechani-cal Loading	Same as for the effect of SCC or unanticipated thermal or mechanical loading on item C2.1.5 small bore RCS piping.
C2.3.1, C2.3.2	Reactor Coolant Pump	Casing, Cover	Bowl: CASS CF-8 or CF-8M; Cover: SS	Chemically Treated Borated Water up to 340°C (644°F)	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
C2.3.1	Reactor Coolant Pump	Casing	CASS CF-8 or CF-8M	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation and Growth	SCC	EPRI TR-105714 Rev. 3 or later revisions/update. NUREG-0313, Rev. 2. ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) according to IWB-7000 and IWA-7000, and reexamination with the requirements of IWA-2200. (8 & 9) Confirmation Process and Administrative Controls: Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. (10) Operating Experience: Although the primary pressure boundary components of PWRs are generally not affected by SCC because of low dissolved oxygen levels and control of water chemistry, cracking could occur at creviced/cold worked locations (IN 84-89).	
Monitoring and control of primary water chemistry in accordance with the EPRI guidelines in TR-105714 (Rev. 3 or later revisions or update) minimize the potential of SCC, and material selection according to the NUREG-0313, Rev. 2 guidelines of 0.035% C and 7.5% ferrite has reduced susceptibility to SCC. For CASS components that do not meet either one of the above guidelines, define plant-specific aging management program.	Plant-specific aging management program is to be evaluated for CASS components that do not meet either the water chemistry guidelines of EPRI TR-105714, or the C and ferrite content criteria of NUREG-0313, Rev. 2. The program will include (a) adequate inspection methods to ensure detection of cracks and (b) flaw evaluation methodology for CASS components that are susceptible to thermal aging embrittlement.	Yes, plant specific AMP
Same as for the effect of SCC or unanticipated thermal or mechanical loading on item C2.1.5 small bore RCS piping.	Same as for the effect of SCC or unanticipated thermal or mechanical loading on item C2.1.5 small bore RCS piping.	Yes parameters monitored/ inspected and detection of aging effects should be further evaluated
Same as for the effect of Fatigue on Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.	Same as for the effect of Fatigue on Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.	Yes TLAA
Monitoring and control of primary water chemistry in accordance with the EPRI guidelines in TR-105714 (Rev. 3 or later revisions or update) minimize the potential of SCC, and material selection according to the NUREG-0313, Rev. 2 guidelines of 0.035% C and 7.5% ferrite have reduced susceptibility to SCC. For CASS components that do not meet either one of the water chemistry and material selection guidelines, inservice inspection in conformance with ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1.	For components that do not meet either one of the water chemistry and material selection guidelines, inservice inspection (ISI) to monitor the effects of SCC on the intended function of the valve. For evaluation and technical basis of the 10 elements of the AMP, see Chapter XI.M14 “Inspection of Class 1 Pump Casing and Valve Body.”	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C2.3.1	Reactor Coolant Pump	Casing	CASS CF-8 or CF-8M	Chemically Treated Borated Water up to 340°C (644°F)	Loss of Fracture Toughness	Thermal Aging Embrittle-ment	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.
C2.3.3	Reactor Coolant Pump	Closure Bolting	High Strength Low-Alloy Steel (HSLAS) SA540 GrB23, SA193 GrB7	Air, Leaking Chemically Treated Borated Water &/or Steam up to 340°C (644°F)	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
C2.3.3	Reactor Coolant Pump	Closure Bolting	HSLAS SA540 GrB23, SA193 GrB7	Air, Leaking Chemically Treated Borated Water or Steam up to 340°C (644°F)	Crack Initiation and Growth	SCC	NUREG-1339 EPRI NP-5769 EPRI NP-5067 ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. Operating Experience: IEB 82-02. NRC GL 91-17
C2.3.3	Reactor Coolant Pump	Closure Bolting	High Strength Low-Alloy Steel (HSLAS) SA540 GrB23, SA193 GrB7	Air, Leaking Chemically Treated Borated Water or Steam up to 340°C (644°F)	Loss of Material	Boric Acid Corrosion of External Surfaces	Same as for the effect of Boric Acid Corrosion of external surfaces of Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.
C2.3.3	Reactor Coolant Pump	Closure Bolting	HSLAS SA540 GrB23, SA193 GrB7	Air, Leaking Chemically Treated Borated Water or Steam up to 340°C (644°F)	Loss of Prestress	Stress Relaxatin	Same as for the effect of Stress Corrosion Cracking on Item C2.3.3 closure bolting for reactor coolant pump.
C2.4.1, C2.4.2	Valves (Check Control, Hand, Motor-Operated, Relief, and Containment Isolation Valves)	Body, Bonnet	CASS CF-8M, SA182 F316, SA582 Type 416	Chemically Treated Borated Water up to 340°C (644°F)	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Inservice inspection in conformance with ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1. For valve body, screening for susceptibility to thermal aging is not required.	The existing ASME Section XI inspection requirements are considered adequate for managing the effects of loss of fracture toughness due to thermal embrittlement of CASS valve bodies. For evaluation and technical basis of the 10 elements of the AMP, see Chapter XI.M14 "Inspection of Class 1 Pump Casing and Valve Body."	No
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii). See Chapter 10 of this report, for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes TLAA
Recommendations for a comprehensive bolting integrity program delineated in NUREG–1339 and industry's recommendations delineated in EPRI NP-5769, with the exceptions noted in NUREG 1339, for safety related bolting, and EPRI NP-5067 for other bolting.	For evaluation and technical basis of the 10 elements of the AMP, see Chapter XI.M12 "Bolting Integrity."	No
Same as for the effect of Boric Acid Corrosion of external surfaces of Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.	Same as for the effect of Boric Acid Corrosion of external surfaces of Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.	No
Same as for the effect of Stress Corrosion Cracking on Item C2.3.3 closure bolting for reactor coolant pump.	Same as for the effect of Stress Corrosion Cracking on Item C2.3.3 closure bolting for reactor coolant pump	No
Same as for the effect of Fatigue on Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.	Same as for the effect of Fatigue on Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.	Yes TLAA

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C2.4.1	Valves (Check, Control, Hand, Motor-Operated, Relief, and Containment Isolation Valves)	Body	CASS CF-8M	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation and Growth	SCC	EPRI TR-105714 Rev. 3 or later revisions/update. NUREG-0313, Rev. 2. ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.
C2.4.1	Valves (Check, Control, Hand, Motor-Operated, Relief, and Containment Isolation Valves)	Body	CASS CF-8M	Chemically Treated Borated Water up to 340°C (644°F)	Loss of Fracture Toughness	Thermal Aging Embrittle-ment	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.
C2.4.3	Valves	Closure Bolting	HSLAS	Air, Leaking Chemically Treated Borated Water or Steam	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
C2.4.3	Valves	Closure Bolting	HSLAS	Air, Leaking Chemically Treated Borated Water or Steam	Crack Initiation and Growth	SCC	Same as for the effect of Stress Corrosion Cracking on Item C2.3.3 closure bolting for reactor coolant pump.
C2.4.3	Valves	Closure Bolting	HSLAS	Air, Leaking Chemically Treated Borated Water or Steam	Loss of Material	Boric Acid Corrosion of External Surfaces	Same as for the effect of Boric Acid Corrosion on Item C2.3.3 closure bolting for reactor coolant pump.
C2.4.3	Valves	Closure Bolting	HSLAS	Air, Leaking Chemically Treated Borated Water or Steam	Loss of Prestress	Stress Relaxation	Same as for the effect of Stress Corrosion Cracking on Item C2.3.3 closure bolting for reactor coolant pump.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Monitoring and control of primary water chemistry in accordance with the EPRI guidelines in TR-105714 (Rev. 3 or later revisions or update) minimize potential of SCC, and material selection according to NUREG-0313, Rev. 2 guidelines of 0.035% C and 7.5% ferrite have reduced susceptibility to SCC. For CASS components that do not meet either one of the water chemistry and material selection guidelines, inservice inspection in conformance with ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1.	For components that do not meet either one of the water chemistry and material selection guidelines, inservice inspection (ISI) to monitor the effects of SCC on the intended function of the valve. For evaluation and technical basis of the 10 elements of the AMP, see Chapter XI.M14 “Inspection of Class 1 Pump Casing and Valve Body.”	No
Inservice inspection in conformance with ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1. For valve body, screening for susceptibility to thermal aging is not required.	The existing ASME Section XI inspection requirements are considered adequate for managing the effects of loss of fracture toughness due to thermal embrittlement of CASS valve bodies. For evaluation and technical basis of the 10 elements of the AMP, see Chapter XI.M14 “Inspection of Class 1 Pump Casing and Valve Body.”	No
Same as for the effect of Fatigue on Item C2.3.3 closure bolting for reactor coolant pump.	Same as for the effect of Fatigue on Item C2.3.3 closure bolting for reactor coolant pump.	Yes TLAA
Same as for the effect of Stress Corrosion Cracking on Item C2.3.3 closure bolting for reactor coolant pump.	Same as for the effect of Stress Corrosion Cracking on Item C2.3.3 closure bolting for reactor coolant pump.	No
Same as for the effect of Fatigue on Item C2.3.3 closure bolting for reactor coolant pump.	Same as for the effect of Fatigue on Item C2.3.3 closure bolting for reactor coolant pump.	No
Same as for the effect of Stress Corrosion Cracking on Item C2.3.3 closure bolting for reactor coolant pump.	Same as for the effect of Stress Corrosion Cracking on Item C2.3.3 closure bolting for reactor coolant pump.	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C2.5.1	Pressurizer	Shell/Heads	Low-Alloy Steel (LAS) with SS or Alloy 600 Cladding	Chemically Treated Borated Water or Saturated Steam 290-343°C (554-650°F)	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
C2.5.1	Pressurizer	Shell/Heads (Outer surfaces)	LAS	Air, Leaking Chemically Treated Borated Water or Steam at up to 340°C (644°F)	Loss of Material	Boric Acid Corrosion of External Surfaces	*Same as for the effect of Boric Acid Corrosion of external surfaces of Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.*
C2.5.1	Pressurizer	Shell/Heads	LAS with Types 308, 308L or 309 SS or Alloy 600 Cladding	Chemically Treated Borated Water or Saturated Steam 290-343°C 554-650°F)	Crack Initiation and Growth	SCC, Cyclic Loading	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. EPRI TR-105714 Rev. 3 or later revisions/update. *Supporting Documents*: Reg. Guide 1.43. Reg. Guide 1.44. *Operating Experience:* NRC IN 84-18. NRC IN 84-89.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Same as for the effect of Fatigue on Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.	*Same as for the effect of Fatigue on Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.*	Yes TLAA
Same as for the effect of Boric Acid Corrosion of external surfaces of Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.	*Same as for the effect of Boric Acid Corrosion of external surfaces of Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.*	No
Inservice inspection in conformance with ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1. Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 (Rev. 3 or later revisions or update) to minimize the potential of crack initiation and growth.	***(1) Scope of Program*:** The program includes preventive measures to mitigate SCC and inservice inspection (ISI) to monitor the effects of cracking on the intended function of SS components and cladding. Cracks in the pressurizer cladding could propagate from cyclic loading into the ferrite base metal and weld metal. However, because the weld metal between the surge nozzle and the vessel lower head is subjected to the maximum stress cycles and the area is periodically inspected as part of the ISI program, the existing AMP is adequate for managing the effect of pressurizer clad cracking. ***(2) Preventive Actions*:** Control of halogens, sulfates, and oxygen in the primary water to less than 0.05, 0.05, and 0.005 ppm, respectively, during operation, and monitor and control of water chemistry during shut down, mitigate potential of SCC. However, inadvertent introduction of contaminants into the coolant system can occur through the free surface of spent fuel pool or the introduction of oxygen during cool down. The AMP must therefore rely upon water chemistry monitoring to detect possible excursions. Selection of material in compliance with the guidelines of Regulatory Guide (RG) 1.44 prevent or mitigate SCC, and RG 1.43 prevents underclad cracking. ***(3) Parameters Monitored/Inspected*:** The AMP monitors the effects of crack initiation and growth on the intended function of pressurizer components by detection and sizing of cracks by ISI. Inspection requirements of Table IWB 2500-1, examination categories B-B for pressure-retaining welds, B-D for full-penetration welded nozzles, B-E for pressure retaining partial retaining welds, and B-F for pressure-retaining dissimilar metal welds in nozzles, specify the extent and schedule for inspection of the components making up the pressurizer pressure boundary. . Requirements of testing category B-P conducted according to IWA–5000 specify visual VT-2 (IWA-5240) examination of all pressure retaining components during system leakage test (IWB-5221) and system hydrostatic test (IWB-5222). ***(4) Detection of Aging Effects*:** Degradation of the pressurizer pressure boundary due to SCC can not occur without crack initiation and growth; extent and schedule of inspection assure detection of cracks before the loss of	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C2.5.2, C2.5.4	Pressurizer	Spray Line Nozzle, Spray Head	Nozzle: CS or LAS with SS Cladding; Spray Head: Alloy 600, SS, CASS	Chemically Treated Borated Water or Saturated Steam 290-343°C (554-650°F)	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
C2.5.3	Pressurizer	Surge Line Nozzle	CS or LAS with SS Cladding, CASS	Chemically Treated Borated Water up to 340°C (644°F)	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
C2.5.5 - C2.5.7	Pressurizer	Thermal Sleeves, Instrument Penetrations, Safe Ends	Thermal Sleeves: Alloy 600; Penetrations: A 600, SS; Safe Ends: SS	Chemically Treated Borated Water or Saturated Steam 290-343°C (554-650°F)	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) intended function of the pressurizer components. ***(5) Monitoring and Trending*:** Inspection schedule in accordance with IWB-2500 should provide timely detection of cracks. System leakage test is conducted prior to plant startup following each refueling outage, and hydrostatic test at or near the end of each inspection interval. ***(6) Acceptance Criteria*:** Any SCC degradation is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3514. Planar and liner flaws are sized according to IWA-3300 and IWA-3400. ***(7) Corrective Actions*:** Repair is in conformance with IWA-4000 and IWB-4000, replacement according to IWA-7000 and IWB-7000, and reexamination in accordance with requirements of IWA-2200. Continued operation without repair requires that crack growth calculation be performed according to the guidance of ASME Section XI or other approved procedures. ***(8 & 9) Confirmation Process and Administrative Controls*:** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience*:** Although the pressurizer primary pressure boundary has generally not been found to be affected by SCC because of low dissolved oxygen levels potential of SCC exists from inadvertent introduction of contaminants into the primary coolant system (IN 84–18). Also, SCC could occur at creviced or cold worked locations (IN 84-89).	
Same as for the effect of Fatigue on Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.	*Same as for the effect of Fatigue on Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.*	Yes TLAA
Same as for the effect of Fatigue on Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.	*Same as for the effect of Fatigue on Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.*	Yes TLAA
Same as for the effect of Fatigue on Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.	*Same as for the effect of Fatigue on Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.*	Yes TLAA

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C2.5.2, C2.5.3, C2.5.6	Pressurizer	Spray Line Nozzle, Surge Line Nozzle, Instrument Penetrations	CS or LAS with SS Cladding; or SS	Chemically Treated Borated Water or Saturated Steam 290-343°C (554-650°F)	Crack Initiation and Growth	SCC	*Same as for the effect of SCC on C2.5.1 pressurizer shell/heads*
C2.5.7	Pressurizer	Safe Ends	SS	Chemically Treated Borated Water or Saturated Steam 290-343°C (554-650°F)	Crack Initiation and Growth	SCC	*Same as for the effect of Stress Corrosion Cracking on Item C2.2.7 nozzles and safe ends in connected systems.*
C2.5.3	Pressurizer	Surge Line Nozzle	CASS	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation and Growth	SCC	EPRI TR-105714 Rev. 3 or later revisions/update. NUREG-0313, Rev. 2.
C2.5.4	Pressurizer	Spray Head	Alloy 600, SS, CASS	Chemically Treated Borated Water or Saturated Steam 290-343°C (554-650°F)	Crack Initiation and Growth	Primary Water Stress Corrosion Cracking (PWSCC), SCC	-
C2.5.6	Pressurizer	Instrument Penetrations	Alloy 600	Chemically Treated Borated Water or Saturated Steam 290-343°C (554-650°F)	Crack Initiation and Growth	PWSCC	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. EPRI TR-105714 Rev. 3 or later revisions/update. *Operating Experience* NRC GL 97-01. NRC IN 90-10. NRC IN 96-11.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Same as for the effect of SCC on C2.5.1 pressurizer shell/heads.	*Same as for the effect of SCC on C2.5.1 pressurizer shell/heads.*	No
Same as for the effect of Stress Corrosion Cracking on Item C2.2.7 nozzles and safe ends in connected systems.	*Same as for the effect of Stress Corrosion Cracking on Item C2.2.7 nozzles and safe ends in connected systems.*	No
Monitoring and control of primary water chemistry in accordance with the EPRI guidelines in TR-105714 (Rev. 3 or later revisions or update) minimize the potential of SCC, and material selection according to the NUREG-0313, Rev. 2 guidelines of 0.035% C and 7.5% ferrite has reduced susceptibility to SCC. For CASS components that do not meet either one of the above guidelines, define plant-specific aging management program.	Plant-specific aging management program is to be evaluated for CASS components that do not meet either the water chemistry guidelines of EPRI TR-105714, or the C and ferrite content criteria of NUREG-0313, Rev. 2. The program will include (a) adequate inspection methods to ensure detection of cracks and (b) flaw evaluation methodology for CASS components that are susceptible to thermal aging embrittlement.	Yes, plant specific AMP
Plant specific aging management program.	Plant specific aging management program is to be evaluated.	Yes, No AMP
Inservice inspection in conformance with ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1. Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 (Rev. 3 or later revisions or update) to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes preventive measures to mitigate primary water stress corrosion cracking (PWSCC) and inservice inspection (ISI) to detect cracks or leakage. However, the program is inadequate to manage the effects of SCC on the intended function of Ni-alloy components. The applicant should evaluate the susceptibility of Ni-alloys to PWSCC. ***(2) Preventive Actions:*** Control of halogens, sulfates, and oxygen in the primary water to less than 0.05, 0.05, and 0.005 ppm, respectively, during operation, and monitor and control of water chemistry during shut down, mitigate potential of SCC. However, based on the experiences described in NRC Information Notices (IN) 90-10 and 96-11, and Generic Letter (GL) 97-01, the susceptibility of Ni-alloys and welds to primary water PWSCC has not been addressed adequately, particularly when demineralizer resins contaminate the reactor coolant system. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of PWSCC on the intended function of instrument nozzles and penetrations by detection of cracks and leakage by ISI. Testing category B-P specify	Yes, Elements 2 to 6 should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C2.5.3, C2.5.4	Pressurizer	Surge Line Nozzle, Spray head	CASS	Chemically Treated Borated Water or Saturated Steam 290-343°C (554-650°F)	Loss of Fracture Toughness	Thermal Aging Embrittle-ment	*Same as for the effect of Thermal Aging Embrittlement on Items C2.1.1 - C2.1.3 Cold-Leg, Hot-Leg, and Surge Line Piping and Fittings.*

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) visual VT-2 (IWA-5240) examination of all pressure retaining components during system leakage test (IWB-5221) and system hydrostatic test (IWB-5222). Examination category B-E specifies visual VT-2 examination of partial penetration welds during the hydrostatic test. However, the applicant should perform a susceptibility study of all Ni-alloy components to identify the most susceptible locations and to determine whether an augmented inspection program, including a combination of surface and volumetric examination, is necessary. ***(4) Detection of Aging Effects:*** Aging effects degradation of the vessel penetrations can not occur without crack initiation. Based on GL 97-01, the applicant should review the scope and schedule of inspection, including leakage detection system, to assure detection of cracks before the loss of intended function of the penetrations. ***(5) Monitoring and Trending:*** System leakage test is conducted prior to plant startup following each refueling outage, and hydrostatic test at or near the end of each inspection interval. Applicant should either provide the technical basis that justifies the adequacy of the program or develop an integrated long-term program which includes periodic inspection of the most susceptible locations to detect the occurrence of PWSCC. The frequency of subsequent inspections should be based on the finding of the initial inspections and crack growth rate models for Ni alloys. ***(6) Acceptance Criteria:*** Any SCC degradation is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and 3500. However, applicant should provide information on crack initiation and growth models and the data used to validate these models to verify adequacy of the inspection program and acceptance criteria. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWA-4000 and IWB-4000. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** SCC of Alloy 600 and austenitic SSs has occurred in domestic and foreign PWRs (IN 90-10). Very little attention has been given to the inspection for SCC in Alloy 600 applications other than that associated with steam generator tubes.	
Same as for the effect of Thermal Aging Embrittlement on Items C2.1.1 - C2.1.3 Cold-Leg, Hot-Leg, and Surge Line Piping and Fittings.	*Same as for the effect of Thermal Aging Embrittlement on Items C2.1.1 - C2.1.3 Cold-Leg, Hot-Leg, and Surge Line Piping and Fittings.*	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C2.5.8	Pressurizer	Manway and Flange	LAS with Types 308, 308L, or 309 SS; or Alloy 600 Cladding	Chemically Treated Borated Water or Saturated Steam 290-343°C (554-650°F)	Crack Initiation and Growth	SCC, PWSCC	*Same as for the effect of SCC on Item C2.5.1 pressurizer shell/heads*
C2.5.9	Pressurizer	Manway and Flange Bolting	HSLAS	Air, Leaking Chemically Treated Borated Water or Steam up to 340°C (644°F)	Crack Initiation and Growth	SCC	*Same as for the effect of Stress Corrosion Cracking on Item C2.3.3 closure bolting for reactor coolant pump.*
C2.5.9	Pressurizer	Manway and Flange Bolting	HSLAS	Air, Leaking Chemically Treated Borated Water or Steam up to 340°C (644°F)	Loss of Material	Boric Acid Corrosion of External Surfaces	*Same as for the effect of Boric Acid Corrosion on Item C2.3.3 closure bolting for reactor coolant pump.*
C2.5.9	Pressurizer	Manway and Flange Bolting	HSLAS	Air, Leaking Chemically Treated Borated Water or Steam up to 340°C (644°F)	Loss of Prestress	Stress Relaxation	*Same as for the effect of Stress Corrosion Cracking on Item C2.3.3 closure bolting for reactor coolant pump.*
C2. 5.10	Pressurizer	Heater Sheaths and Sleeves	Alloy 600 or Austenitic SS	Chemically Treated Borated Water up to 340°C (644°F)	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
C2. 5.10	Pressurizer	Heater Sheaths and Sleeves	Austenitic SS	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation and Growth	SCC	*Same as for the effect of SCC on Item C2.5.1 pressurizer shell/heads*
C2. 5.10	Pressurizer	Heater Sheaths and Sleeves	Alloy 600	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation and Growth	PWSCC	*Same as for the effect of SCC on Item C2.5.6 pressurizer instrumentation penetrations constructed of Alloy 600*

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Same as for the effect of Stress Corrosion Cracking on Item C2.5.1 pressurizer shell/heads.	*Same as for the effect of Stress Corrosion Cracking on Item C2.5.1 pressurizer shell/heads.*	No
Same as for the effect of Stress Corrosion Cracking on Item C2.3.3 closure bolting for reactor coolant pump.	*Same as for the effect of Stress Corrosion Cracking on Item C2.3.3 closure bolting for reactor coolant pump.*	No
Same as for the effect of Boric Acid Corrosion on Item C2.3.3 closure bolting for reactor coolant pump.	*Same as for the effect of Boric Acid Corrosion on Item C2.3.3 closure bolting for reactor coolant pump.*	No
Same as for the effect of Stress Corrosion Cracking on Item C2.3.3 closure bolting for reactor coolant pump.	*Same as for the effect of Stress Corrosion Cracking on Item C2.3.3 closure bolting for reactor coolant pump.*	No
Same as for the effect of Fatigue on Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.	*Same as for the effect of Fatigue on Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.*	Yes TLAA
Same as for the effect of Stress Corrosion Cracking on Item C2.5.1 pressurizer shell/heads.	*Same as for the effect of Stress Corrosion Cracking on Item C2.5.1 pressurizer shell/heads.*	No
Same as for the effect of SCC on Item C2.5.6 pressurizer instrument penetrations constructed of Alloy 600.	*Same as for the effect of SCC on Item C2.5.6 pressurizer instrument penetrations constructed of Alloy 600.*	Yes, Elements 2 to 6 should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C2. 5.11	Pressurizer	Support Keys, Skirts, and Shear Lugs	CS, LAS	Air, with metal temperatures up to 340°C (644°F)	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
C2. 5.12	Pressurizer	Integral Support	CS	Air, Leaking Chemically Treated Borated Water	Loss of Material	Boric Acid Corrosion of External Surfaces	*Same as for the effect of Boric Acid Corrosion of external surfaces of Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.*
C2. 5.12	Pressurizer	Integral Support	CS, SS	Air	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
C2.6.1, C2.6.2	Pressurizer Relief Tank	Tank Shell and Heads, Flanges and Nozzles	CS with Type 304 SS Cladding	Chemically Treated Borated Water at 93°C (200°F)	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
C2.6.1	Pressurizer Relief Tank	Tank Shells and Heads (External Surfaces)	CS	Air, Chemically Treated Borated Water at 93°C (200°F	Loss of Material	Boric Acid Corrosion of External Surfaces	*Same as for the effect of Boric Acid Corrosion of external surfaces of Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.*
C2.6.1, C2.6.2	Pressurizer Relief Tank	Tank Shell and Heads, Flanges and Nozzles	CS with Type 304 SS Cladding	Chemically Treated Borated Water at 93°C (200°F	Crack Initiation and Growth	SCC	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. EPRI TR-105714 Rev. 3 or later revisions/update. *Supporting Documents*: Reg. Guide 1.43. Reg. Guide 1.44.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Same as for the effect of Fatigue on Item C2.3.3 closure bolting for reactor coolant pump.	*Same as for the effect of Fatigue on Item C2.3.3 closure bolting for reactor coolant pump.*	Yes TLAA
Same as for the effect of Boric Acid Corrosion of external surfaces of Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.	*Same as for the effect of Boric Acid Corrosion of external surfaces of Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.*	No
Same as for the effect of Fatigue on Item C2.3.3 closure bolting for reactor coolant pump.	*Same as for the effect of Fatigue on Item C2.3.3 closure bolting for reactor coolant pump.*	Yes TLAA
Same as for the effect of Fatigue on Item C2.3.3 closure bolting for reactor coolant pump.	*Same as for the effect of Fatigue on Item C2.3.3 closure bolting for reactor coolant pump.*	Yes TLAA
Same as for the effect of Boric Acid Corrosion of external surfaces of Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.	*Same as for the effect of Boric Acid Corrosion of external surfaces of Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.*	No
Inservice inspection in conformance with ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a), Subsection IWC, Table IWC 2500-1. Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 (Rev. 3 or later revisions or update) to minimize the potential of crack initiation and growth.	**(1) Scope of Program:** The program relies on inservice inspection (ISI) to monitor the effects of SCC on the intended function of clad components. **(2) Preventive Actions:** Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 (Rev. 3 or later revisions or update) to minimize the potential of crack initiation and growth. Selection of material in compliance with the guidelines of Regulatory Guide (RG) 1.44 prevent or mitigate SCC, and guidelines of RG 1.43 prevent underclad cracking. **(3) Parameters Monitored/Inspected:** The AMP monitors the effects of SCC on the intended function of clad components by detection and sizing of cracks by ISI in accordance with examination categories C-A for pressure retaining welds in vessels and C-B for pressure retaining nozzle welds in vessels, and testing category C-H for all pressure retaining components. ***(4) Detection of Aging Effects:*** Degradation of the pressurizer relief tank pressure boundary due to SCC can not occur without	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) crack initiation and growth; extent and schedule of inspection assure detection of cracks before the loss of intended function of the pressurizer relief tank. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWC-2500 should provide timely detection of cracks. ***(6) Acceptance Criteria:*** Any SCC degradation is evaluated in accordance with IWC-3100 by comparing ISI results with the acceptance standards of IWC-3400. ***(7) Corrective Actions:*** Repair is in conformance with IWA-4000, replacement according to IWA-7000, and reexamination in accordance with requirements of IWA-2200. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** The pressurizer relief tank pressure boundary has generally not been found to be affected by SCC because of the low operating temperatures.	

D1 Steam Generator (Recirculating)

D1.1 Pressure Boundary and Structural

D1.1.1 Top Head

D1.1.2 Steam Nozzle and Safe End

D1.1.3 Upper and Lower Shell

D1.1.4 Transition Cone

D1.1.5 Feedwater Nozzle and Safe End

D1.1.6 Feedwater Impingement Plate and Support

D1.1.7 Secondary Manway and Handhole Bolting

D1.1.8 Lower Head

D1.1.9 Primary Nozzles and Safe Ends

D1.1.10 Instrument Nozzles

D1.1.11 Primary Manway Bolting

D1.2 Tube Bundle

D1.2.1 Tubes and Sleeves

D1.2.2 Tube Support Lattice Bars (Combustion Engineering)

D1.2.3 Tube Plugs

D1.3 Upper Assembly and Separators

D1.3.1 Feedwater Inlet Ring and Support

D1. Steam Generator (Recirculating)

System, Structures, and Components

The system, structures, and components included in this table consist of the recirculating-type steam generators, as found in Westinghouse and Combustion Engineering pressurized water reactors (PWRs), including all internal components and water/steam nozzles and safe ends. Based on the Nuclear Regulatory Commission Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” the primary water side (tube side) of the steam generator is classified as Group A Quality Standards and secondary water side is classified as Group B Quality Standard.

System Interfaces

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

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

D1. STEAM GENERATOR (Recirculating)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.1.1, D1.1.2	Pressure Boundary and Structural	Top Head, Steam Nozzle & Safe End	Low-alloy Steel (LAS)	Up to 300°C (572°F) Steam	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
D1.1.3 - D1.1.6	Pressure Boundary and Structural	Upper & Lower Shell, Transition Cone, FW Nozzle & Safe End, FW Impingement Plate Support	Carbon steel (CS), LAS	Up to 300°C (572°F) Secondary-side Water Chemistry at 5.3-7.2 MPa	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
D1.1.3, D1.1.4	Pressure Boundary and Structural	Upper & Lower Shell, Transition Cone	CS, LAS	Up to 300°C (572°F) Secondary-side Water Chemistry at 5.3-7.2 MPa	Loss of Material	Pitting and Crevice Corrosion	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. EPRI TR-102134 Rev. 3 or later revisions/update. *Operating Experience* NRC IN 82-37. NRC IN 85-65. NRC IN 90-04.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)	Yes TLAA
Same as for the effect of Fatigue on Items D1.1.1 top head and D1.1.2 steam nozzle and safe end.	*Same as for the effect of Fatigue on Items D1.1.1 top head and D1.1.2 steam nozzle and safe end.*	Yes TLAA
Inservice inspection in conformance with ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a) Table IWC 2500-1. Secondary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-102134 (Rev. 3 or later revisions or update) to minimize the potential of pitting and crevice corrosion.	***(1) Scope of Program:*** The program relies on preventive measures to mitigate crevice or pitting corrosion and inservice inspection (ISI) to monitor the effects of corrosion on the intended function of the steam generator shell. ***(2) Preventive Actions:*** Stringent control of secondary water chemistry in accordance with the guidance of EPRI TR-102134, frequent monitoring, and timely corrective action when specified impurity levels are exceeded, prevent or mitigate corrosion. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of corrosion by detection and sizing of flaws in pressure retaining welds by ISI and by -detecting coolant leakage. Inspection requirements of Table IWC 2500-1, examination category C-A specify volumetric examination, extending 1/2 in. each side, of all circumferential and tubesheet-to-shell welds, and category C-H specifies visual VT-2 (IWA-5240) examination during system leakage and hydrostatic tests of all pressure retaining Class 2 components. ***(4) Detection of Aging Effects:*** The extent and schedule of the inspections prescribed by the program are designed to ensure that flaws cannot attain a depth sufficient to threaten the integrity of the welds. However, based on NRC Information notice 90-04 where general corrosion pitting of the shell exists, the program recommendations may not be sufficient to detect pitting and corrosion and additional inspection procedures may be required. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI should provide for timely detection of leakage. System leakage test is typically conducted at 40-month intervals, and hydrostatic test at or near the end of each inspection interval. ***(6) Acceptance Criteria:*** Any defect detected is compared with the requirements of Section XI, IWC 3500. Any relevant conditions that may be detected during the leakage and hydrostatic tests are evaluated in accordance with IWC-3400 and IWC-3500. ***(7) Corrective Actions:*** Welds containing flaws that exceed the maximum permissible size must be repaired. Repair and replacement are in conformance with IWA-4000. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls, in conjunction with NRC oversight, are implemented in accordance with the	Yes, detection of aging effects should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.1.2, D1.1.5	Pressure Boundary and Structural	Steam Nozzle and Safe End, FW Nozzle & Safe End	CS, LAS	Up to 300°C (572°F) Steam or Secondary-side Water Chemistry at 5.3-7.2 MPa	Wall Thinning	Flow Accelerated Corrosion (FAC)	EPRI NSAC–202L–R2. NRC GL 89-08. NUREG-1344. *Supporting Documents:* BWRVIP-75. BWRVIP-79 (EPRI TR-103515). *Operating Experience* NRC Bl 87-01. NRC IN 81-28. NRC IN 89-53. NRC IN 91-18. NRC IN 91-18 S1. NRC IN 92-35. NRC IN 93-21. NRC IN 95-11. NRC IN 97-84.
D1.1.6	Upper Assembly and Separators	Feedwater Impingement Plate Support	CS	Up to 300°C (572°F) Secondary-side Water Chemistry	Loss of section thickness	Erosion	-
D1.1.7	Pressure Boundary and Structural	Secondary Manway and Handhole Bolting	LAS	Air, with metal tempera-ture up to 340°C (644°F)	Loss of Preload	Stress Relaxation	NUREG-1339. EPRI NP-5769. EPRI NP-5067. ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. *Operating Experience*: IEB 82-02. NRC GL 91-17.
D1.1.8	Pressure Boundary and Structural	Lower Head (External Surfaces)	LAS	Air, Leaking Chemically Treated Borated Water or Steam up to 340°C (644°F)	Loss of Material	Boric Acid Corrosion of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.
D1.1.8, D1.1.9	Pressure Boundary and Structural	Lower Head, Primary Nozzles and Safe Ends	Carbon Steel (CS) with SS cladding, Safe Ends: SS	Chemically Treated Borated Water up to 340°C (644°F) & 15.2 MPa	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) requirements of Appendix B of 10 CFR 50. ***(10) Operating Experience:*** This AMP has resulted in the timely detection and correction of corrosion in several steam generators in the US. (NRC Information Notices 82-37 & 85-65).	
Implementation of EPRI guidelines of NSAC–202L–R2 for effective flow accelerated corrosion (FAC) program.	For evaluation and technical basis of the 10 elements of the AMP, see Chapter XI.M6 "Flow Accelerated Corrosion".	No
Plant-specific aging management program	Plant-specific aging management program will be evaluated	Yes plant specific AMP
Recommendations for a comprehensive bolting integrity program delineated in NUREG–1339 and industry's recommendations delineated in EPRI NP-5769, with the exceptions noted in NUREG 1339, for safety related bolting, and EPRI NP-5067 for other bolting.	For evaluation and technical basis of the 10 elements of the AMP, see Chapter XI.M12 “Bolting Integrity.”	No
Implementation of NRC Generic Letter 88-05 and inservice inspection (ISI) in conformance with ASME Section XI (1989 edition or later edition as approved in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, to monitor the condition of the reactor coolant pressure boundary for occurrences of borated water leakage.	For evaluation and technical basis of the 10 elements of the AMP, see Chapter XI.M5, “Boric Acid Corrosion.”	No
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal. See the Standard Review Plan, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes TLAA

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.1.9	Pressure Boundary and Structural	Primary Nozzles and Safe Ends	Carbon Steel (CS) with SS cladding, Safe Ends: SS (NiCrFe buttering, and SS or NiCrFe weld)	Chemically Treated Borated Water at temperatures up to 340°C (644°F) and 15.2 MPa	Crack Initiation and Growth	SCC	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. EPRI TR-105714 Rev. 3 or later revisions/update. *Supporting Documents*: Reg. Guide 1.43. Reg. Guide 1.44. *Operating Experience:* NRC IN 84-18. NRC IN 90-10. NRC IN 90-30.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Inservice inspection in conformance with ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1. Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 (Rev. 3 or later revisions or update) to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes preventive measures to mitigate stress corrosion cracking (SCC) of SS cladding and inservice inspection (ISI) to monitor the effects of SCC of the cladding on the intended function of the component. ***(2) Preventive Actions:*** Control of halogens, sulfates, and oxygen in the primary water to less than 0.05, 0.05, and 0.005 ppm, respectively, during operation, and monitor and control of water chemistry during shut down, mitigate potential of SCC. However, inadvertent introduction of contaminants into the coolant system can occur, and cladding regions with low delta ferrite may be sensitized during post-weld heat treatment and thus become susceptible to SCC. The AMP must therefore rely upon water chemistry monitoring to detect possible excursions. Selection of material in compliance with the recommendations of Regulatory Guide (RG) 1.44 prevent or mitigate SCC, and RG 1.43 prevents underclad cracking. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of crack initiation and growth on the intended function of the components by detection and sizing of cracks by ISI. Requirements of Table IWB 2500-1, examination categories B-D specify volumetric examination of nozzle to vessel welds and nozzle inside radius, and B-F specifies for dissimilar metal welds, volumetric and surface examination of nozzle-to-safe end butt welds NPS 4 or larger and surface examination of those less than NPS 4 or nozzle-to-safe end socket welds Requirements of testing category B-P conducted according to IWA–5000 specify visual VT-2 (IWA-5240) examination of all pressure retaining components during system leakage and hydrostatic test. ***(4) Detection of Aging Effects:*** Degradation of the component pressure boundary due to SCC can not occur without crack initiation and growth; extent and schedule of inspection assure detection of cracks before the loss of intended function of the components. However, the extent and schedule of inspection does not assure detection of cracks because ASME Section XI inspection requires examination of only the welds and weld regions, the potential of cracking in cladding remote from welds is not addressed. Also, based on NRC Information Notices (INs) 90-10 and 90-30 applicant should review Ni-alloy applications in primary coolant and implement an augmented inspection program and evaluate choice of transducers for ultrasonic examination of dissimilar metal welds. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2500 should provide timely detection of cracks. System leakage test is conducted prior to plant startup following each refueling outage, and hydrostatic test at or near the end of each inspection interval. ***(6) Acceptance Criteria:*** Any SCC degradation is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3514. Planar and liner laws are sized according to IWA-3300 and IWA-3400. ***(7) Corrective Actions:*** Repair is in conformance with IWA-4000 and IWB-4000, replacement according to IWA-7000 and IWB-7000, and reexamination in accordance	Yes detection of aging effects should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1. 1.10	Pressure Boundary and Structural	Instrument Nozzles	Alloy 600	Chemically Treated Borated Water up to 340°C (644°F) and 15.5 MPa	Crack Initiation and Growth	Primary Water Stress Corrosion Cracking (PWSCC)	-
D1. 1.11	Pressure Boundary and Structural	Primary Manway Bolting	CS, LAS	Air, Leaking Chemically Treated Borated Water and/or Steam up to 340°C (644°F)	Loss of Material	Boric Acid Corrosion of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.
D1. 1.11	Pressure Boundary and Structural	Primary Manway Bolting	CS, LAS	Air, Leaking Chemically Treated Borated Water and/or Steam up to 340°C (644°F)	Crack Initiation and Growth	Stress Corrosion Cracking (SCC)	*Same as for the effect of Stress Relaxation on Items D1.1.7 secondary manway and handhole bolting.*

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) with requirements of IWA-2200. Continued operation without repair requires that crack growth calculation be performed according to the guidance of ASME Section XI or other approved procedures. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls, in conjunction with NRC over-sight, are implemented in accordance with the require-ments of Appendix B of 10 CFR 50. ***(10) Operating Experience:*** Although PWR primary pressure boundary has generally not been found to be affected by SCC because of low dissolved oxygen levels potential of SCC exists from inadvertent introduction of contaminants into the primary system (IN 84–18). of all pressure retaining Class 2 components.	
Plant-specific aging management program	Plant-specific aging management program will be evaluated	Yes plant specific AMP
Implementation of NRC Generic Letter 88-05 and inservice inspection (ISI) in conformance with ASME Section XI (1989 edition or later edition as approved in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, to monitor the condition of the reactor coolant pressure boundary for occurrences of borated water leakage.	For evaluation and technical basis of the 10 elements of the AMP, see Chapter XI.M5, “Boric Acid Corrosion.”	No
Same as for the effect of Stress Relaxation on Items D1.1.7 secondary manway and handhole bolting.	*Same as for the effect of Stress Relaxation on Items D1.1.7 secondary manway and handhole bolting.*	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.2.1	Tube Bundle	Tubes and Sleeves	Alloy 600	Chemically Treated Borated Water up to 340°C (644°F) and 15.5 MPa	Crack Initiation and Growth	PWSCC	Reg. Guide 1.83. EPRI document "PWR Steam Generator Examination Guidelines Rev. 5." NEI 97-06. Reg. Guide 1.121. Plant Technical Specifications. EPRI TR-105714 Rev. 3 or later revisions/update. *Operating Experience*: NRC IN 97-88.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Inservice inspection of steam generator tubes in conformance with NRC Regulatory Guide (RG) 1.83 and specific guidance of EPRI document "PWR Steam Generator Examination Guidelines, Revision 5," and NEI 97-06. Assessment of tube integrity and plugging or repair criteria of flawed tubes is governed by NRC RG 1.121 and NEI 97-06. Supplemental inspection guidelines, plugging and repair criteria, and limits on allowable leakage, may be contained in plant-specific Technical Specifications. Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 (Rev. 3 or later revisions or update) to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes inservice inspection (ISI) of steam generator tubes to detect degradation, and evaluation plugging/repair as needed to maintain the structural integrity of the pressure boundary. Inservice inspection is in accordance with RG 1.83 and EPRI steam generator examination guidelines, and evaluation of tube integrity, plugging or repair are in accordance with RG 1.121 and NEI 97–06. These guidelines provide criteria for the qualification of personnel, specific techniques and the associated acquisition and analysis of data including procedures, probe selection, analysis protocols, and reporting criteria. ***(2) Preventive Actions:*** Primary water chemistry guidelines given in EPRI TR-105714 provide guidance on the reduction of susceptibility to PWSCC. ***(3) Parameters Monitored/Inspected:*** The inspection activity in the program detects the presence of cracks and may, in some cases, monitor flaw size and depth, or, alternatively, remaining sound wall thickness. ***(4) Detection of Aging Effects:*** The extent and schedule of the inspections and test techniques prescribed by the program are designed to ensure continued tube integrity and that aging effects will be discovered and repaired before the loss of intended function of the tubes. Problems with tube inspection (IN 97-88), e.g., failures to detect some flaws, uncertainties in flaw sizing, inaccuracies in flaw locations, and inability to detect some cracks at locations with dents are addressed by current inspection guidelines given in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5" and NEI 97-06. ***(5) Monitoring and Trending:*** Required inspection intervals, typically 18 months or each refueling or maintenance/repair outage, provide for timely detection of PWSCC. ***(6) Acceptance Criteria:*** Any cracking detected is compared with the guidelines of the Plant Technical Specifications, RG 1.121, and the structural integrity recommendations described in NEI 97–06. ***(7) Corrective Actions:*** Tubes containing flaws that do not meet the acceptance criteria must be plugged or repaired. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls, in conjunction with NRC oversight, are implemented in accordance with the requirements of Appendix B of 10 CFR 50. Recent experience indicates the importance of performing a complete inspection using the best available techniques and equipment for the reliable detection of tube degradation and to ensure that new forms of degradation are detected. ***(10) Operating Experience:*** The present AMP has been effective in ensuring timely detection and correction of PWSCC degradation in steam generator tubes.	No, provided Plant Technical Specifi-cations conform to EPRI inspection guidelines and NEI 97-06 and primary water chemistry limits are adhered to.

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.2.1	Tube Bundle	Tubes and Sleeves	Alloy 600	Up to 300°C (572°F) Secondary-side Water Chemistry at 5.3-7.2 MPa	Crack Initiation and Growth	Outer Diameter Stress Corrosion Cracking (ODSCC)	Reg. Guide 1.83. EPRI document "PWR Steam Generator Examination Guidelines Rev. 5." NEI 97-06. Reg. Guide 1.121. NRC GL 95-05. Plant Technical Specifications. EPRI TR-102134 Rev. 3 or later revisions/update. *Operating Experience*: NRC IN 97-88.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Inservice inspection of steam generator tubes in conformance with NRC Regulatory Guide (RG) 1.83 and specific guidance of EPRI document “PWR Steam Generator Examination Guidelines, Revision 5,” and NEI 97-06. Assessment of tube integrity and plugging or repair criteria of flawed tubes is governed by NRC RG 1.121 and NEI 97-06. Alternative criteria applicable under certain circumstances to cracking at tube support plates in Westinghouse-designed steam generators are provided in NRC Generic Letter (GL) 95-05. Supplemental inspection guidelines, plugging and repair criteria, and limits on allowable leakage, may be contained in plant-specific Technical Specifications. Secondary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-102134 (Rev. 3 or later revisions or update) to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes inservice inspection (ISI) of steam generator tubes to detect degradation, and evaluation plugging/repair as needed to maintain the structural integrity of the pressure boundary. Inservice inspection is in accordance with RG 1.83 and EPRI steam generator examination guidelines, and evaluation of tube integrity, plugging or repair are in accordance with RG 1.121, GL 95-05, and NEI 97–06. These guidelines provide criteria for the qualification of personnel, specific techniques and the associated acquisition and analysis of data including procedures, probe selection, analysis protocols, and reporting criteria. ***(2) Preventive Actions:*** Secondary water chemistry guidelines given in EPRI TR-102134 provide guidance on the reduction of susceptibility to ODSCC. ***(3) Parameters Monitored/Inspected:*** The inspection activity in the program detects the presence of cracks and may, in some cases, monitor flaw size and depth, or, alternatively, remaining sound wall thickness, or other parameters such as voltage that may be used to assess structural integrity. ***(4) Detection of Aging Effects:*** The extent and schedule of the inspections and test techniques prescribed by the program are designed to ensure continued tube integrity and that aging effects will be discovered and repaired before the loss of intended function of the tubes. Problems with tube inspection (IN 97-88), e.g., failures to detect some flaws, uncertainties in flaw sizing, inaccuracies in flaw locations, and inability to detect some cracks at locations with dents are addressed by current inspection guidelines given in the EPRI document “PWR Steam Generator Examination Guidelines: Revision 5” and NEI 97-06. ***(5) Monitoring and Trending:*** Required inspection intervals, typically 18 months or each refueling or maintenance/repair outage, provide for timely detection of PWSCC. ***(6) Acceptance Criteria:*** Any cracking detected is compared with the guidelines of the Plant Technical Specifications, RG 1.121, and the structural integrity guidelines described in NEI 97–06 and GL 95-05. ***(7) Corrective Actions:*** Tubes containing flaws that do not meet the acceptance criteria must be plugged or repaired. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls, in conjunction with NRC oversight, are implemented in accordance with the requirements of Appendix B of 10 CFR 50. Recent experience indicates the importance of performing a complete inspection using the best available techniques and equipment for the reliable detection of tube degradation and to ensure that new forms of degradation are detected. ***(10) Operating Experience:*** The present AMP has been effective in ensuring timely detection and correction of ODSCC degradation in steam generator tubes.	No, provided Plant Technical Specifi-cations conform to EPRI inspection guidelines and NEI 97-06 and secondary water chemistry limits are adhered to.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D1. STEAM GENERATOR (Recirculating)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.2.1	Tube Bundle	Tubes and Sleeves	Alloy 600	Up to 300°C (572°F) Secondary-side Water Chemistry at 5.3-7.2 MPa	Crack Initiation and Growth	Intergranu-lar Attack (IGA)	Reg. Guide 1.83. EPRI document "PWR Steam Generator Examination Guidelines Rev. 5." NEI 97-06. Reg. Guide 1.121. NRC GL 95-05. Plant Technical Specifications. EPRI TR-102134 Rev. 3 or later revisions/update.
D1.2.1	Tube Bundle	Tubes and Sleeves	Alloy 600	ID Chemically Treated Borated Water up to 340°C (644°F); OD up to 300°C (572°F) Secondary-side Water Chemistry	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Inservice inspection of steam generator tubes in conformance with NRC Regulatory Guide (RG) 1.83 and specific guidance of EPRI document "PWR Steam Generator Examination Guidelines, Revision 5," and NEI 97-06. Assessment of tube integrity and plugging or repair criteria of flawed tubes is governed by NRC RG 1.121 and NEI 97-06; defects at the tube sheet may be shown to satisfy RG 1.121 guidelines using the P*, F*, or L* criterion. Alternative criteria applicable under certain circumstances to cracking at tube support plates in Westinghouse-designed steam generators are provided in NRC Generic Letter (GL) 95-05. Supplemental inspection guidelines, plugging and repair criteria, and limits on allowable leakage, may be contained in plant-specific Technical Specifications. Secondary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-102134 (Rev. 3 or later revisions or update) to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes inservice inspection (ISI) of steam generator tubes to detect degradation, and evaluation plugging/repair as needed to maintain the structural integrity of the pressure boundary. Inservice inspection is in accordance with RG 1.83 and EPRI steam generator examination guidelines, and evaluation of tube integrity, plugging or repair are in accordance with RG 1.121, GL 95-05, and NEI 97–06. These guidelines provide criteria for the qualification of personnel, specific techniques and the associated acquisition and analysis of data including procedures, probe selection, analysis protocols, and reporting criteria. ***(2) Preventive Actions:*** Secondary water chemistry guidelines given in EPRI TR-102134 provide guidance on the reduction of susceptibility to IGA. ***(3) Parameters Monitored/Inspected:*** The inspection activity in the program monitors the extent of degradation or, alternatively, remaining sound wall thickness. ***(4) Detection of Aging Effects:*** The extent and schedule of the inspections prescribed by the program are designed to ensure that flaws cannot attain a depth sufficient to threaten the integrity of the tubes. Problems with tube inspection, e.g., failures to detect and properly characterize IGA, are addressed in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5" and NEI 97-06. ***(5) Monitoring and Trending:*** Required inspection intervals, typically 18 months or each refueling or maintenance/repair outage, provide for timely detection of PWSCC. ***(6) Acceptance Criteria:*** Any cracking detected is compared with the guidelines of the Plant Technical Specifications, RG 1.121, and the structural integrity guidelines described in NEI 97–06 and GL 95-05. ***(7) Corrective Actions:*** Tubes containing flaws that do not meet the acceptance criteria must be plugged or repaired. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls, in conjunction with NRC oversight, are implemented in accordance with the requirements of Appendix B of 10 CFR 50. Recent experience indicates the importance of performing a complete inspection using the best available techniques and equipment for the reliable detection of tube degradation and to ensure that new forms of degradation are detected. ***(10) Operating Experience:*** The present AMP has been effective in ensuring timely detection and correction of IGA degradation in SG tubes.	No, provided Plant Technical Specifications conform to EPRI inspection guidelines and NEI 97-06 and secondary water chemistry limits are adhered to.
Same as for the effect of Fatigue on Items D1.1.1 top head and D1.1.2 steam nozzle and safe end.	*Same as for the effect of Fatigue on Items D1.1.1 top head and D1.1.2 steam nozzle and safe end.*	Yes TLAA

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

D1. STEAM GENERATOR (Recirculating)

Item	Structure and Component	Region of Interest	Material	Environment	Aging Effect	Aging Mechanism	References
D1.2.1	Tube Bundle	Tubes and Sleeves	Alloy 600	Up to 300°C (572°F) Secondary-side Water Chemistry at 5.3-7.2 MPa	Loss of section thickness	Fretting and Wear	Reg. Guide 1.83. EPRI document "PWR Steam Generator Examination Guidelines Rev. 5." NEI 97-06. Reg. Guide 1.121. Plant Technical Specifications.
D1.2.1	Tube Bundle	Tubes and Sleeves (Exposed to Phosphate Chemistry)	Alloy 600	Up to 300°C (572°F) Secondary-side Water Chemistry at 5.3-7.2 MPa	Loss of material	General and Pitting corrosion (OD)	Reg. Guide 1.83. EPRI document "PWR Steam Generator Examination Guidelines Rev. 5." NEI 97-06. Reg. Guide 1.121. Plant Technical Specifications. EPRI TR-102134 Rev. 3 or later revisions/update.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Inservice inspection of steam generator tubes in conformance with NRC Regulatory Guide (RG) 1.83 and specific guidance of EPRI document “PWR Steam Generator Examination Guidelines, Revision 5,” and NEI 97-06. Assessment of tube integrity and plugging or repair criteria of flawed tubes is governed by NRC RG 1.121 and NEI 97-06. Supplemental inspection guidelines, plugging and repair criteria, and limits on allowable leakage, may be contained in plant-specific Technical Specifications.	***(1) Scope of Program:*** The program includes inservice inspection (ISI) of steam generator tubes to detect degradation, and evaluation plugging/repair as needed to maintain the structural integrity of the pressure boundary. Inservice inspection is in accordance with RG 1.83 and EPRI steam generator examination guidelines, and evaluation of tube integrity, plugging or repair are in accordance with RG 1.121 and NEI 97–06. These guidelines provide criteria for the qualification of personnel, specific techniques and the associated acquisition and analysis of data including procedures, probe selection, analysis protocols, and reporting criteria. ***(2) Preventive Actions:*** The program provides no guidance or recommendations on measures preventing fretting and wear. ***(3) Parameters Monitored/Inspected:*** The inspection activity in the program monitors flaw size and depth, or, alternatively, remaining sound wall thickness. ***(4) Detection of Aging Effects:*** The extent and schedule of the inspections and test techniques prescribed by the program are designed to ensure continued tube integrity and that aging effects will be discovered and repaired before the loss of intended function of the tubes. Guidance on tube inspection is given in the EPRI docu-ment “PWR Steam Generator Examination Guidelines: Revision 5” and NEI 97-06. ***(5) Monitoring and Trending:*** Required inspection intervals, typically 18 months or each refueling or maintenance/repair outage, provide for timely detection of PWSCC. ***(6) Acceptance Criteria:*** Any cracking detected is compared with the guidelines of the Plant Technical Specifications, RG 1.121, and the structural integrity guidelines described in NEI 97–06. ***(7) Corrective Actions:*** Tubes containing flaws that do not meet the acceptance criteria must be plugged or repaired. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls, in conjunction with NRC oversight, are implemented in accordance with the requirements of Appendix B of 10 CFR 50. ***(10) Operating Experience:*** The present program has been effective in ensuring timely detection and correction of wear in SG tubes.	No, provided Plant Technical Specifi-cations conform to EPRI inspection guidelines and NEI 97-06.
Inservice inspection of steam generator tubes in conformance with NRC Regulatory Guide (RG) 1.83 and specific guidance of EPRI document “PWR Steam Generator Examination Guidelines, Revision 5,” and NEI 97-06. Assessment of tube integrity and plugging or repair criteria of flawed tubes is governed by NRC RG 1.121 and NEI 97-06. Supplemental inspection guidelines, plugging and repair criteria, and limits on allowable leakage, may be contained in plant-specific Technical Specifications. Secondary water chemistry is monitored and maintained in accordance with EPRI	***(1) Scope of Program:*** The program includes inservice inspection (ISI) of steam generator tubes to detect degradation, and evaluation plugging/repair as needed to maintain the structural integrity of the pressure boundary. Inservice inspection is in accordance with RG 1.83 and EPRI steam generator examination guidelines, and evaluation of tube integrity, plugging or repair are in accordance with RG 1.121 and NEI 97–06. ***(2) Preventive Actions:*** Secondary water chemistry guidelines given in EPRI TR-102134 can significantly reduce secondary-side pitting. ***(3) Parameters Monitored/Inspected:*** The inspection activity in the program monitors flaw size and depth, or, alternatively, remaining sound wall thickness. ***(4) Detection of Aging Effects:*** The extent and schedule of the inspections prescribed by the program are designed to ensure that flaws cannot attain a depth sufficient to	No, provided Plant TS conform to EPRI inspection guidelines & NEI 97-06, and secondary water chemistry limits are adhered to.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

D1. STEAM GENERATOR (Recirculating)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.2.1	Tube Bundle	Tubes	Alloy 600	Up to 300°C (572°F) Secondary-side Water Chemistry at 5.3-7.2 MPa	Deformation (Denting)	Corrosion of tube support plate at intersec-tions with tubes	Reg. Guide 1.83. EPRI document “PWR Steam Generator Examination Guidelines Rev. 5.” NEI 97-06. Reg. Guide 1.121. Plant Technical Specifications. EPRI TR-102134 Rev. 3 or later revisions/update.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
(continued from previous page) guidelines in TR-102134 (Rev. 3 or later revisions or update) to minimize the potential of crack initiation and growth.	*(continued from previous page)* threaten the integrity of the tubes. Specific guidance on the inservice inspection of tubes is provided in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5" and NEI 97-06. ***(5) Monitoring and Trending:*** Required inspection intervals (typically 18 months or each refueling or maintenance/repair outage) should provide for timely detection of virtually all forms of pitting. ***(6) Acceptance Criteria:*** Any flaws detected are compared with the guidelines of the Plant Technical Specifications and Reg. Guide 1.121. ***(7) Corrective Actions:*** Tubes containing flaws that exceed the maximum permissible size must be plugged or repaired (e.g., by sleeving). ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls, in conjunction with NRC oversight, are implemented in accordance with the requirements of Appendix B of 10 CFR 50. ***(10) Operating Experience:*** The present program has been effective in ensuring the timely detection and correction of pitting in SG tubes.	
Inservice inspection of steam generator tubes in conformance with NRC Regulatory Guide (RG) 1.83 and specific guidance of EPRI document "PWR Steam Generator Examination Guidelines, Revision 5," and NEI 97-06. Assessment of tube integrity and plugging or repair criteria of flawed tubes is governed by NRC RG 1.121 and NEI 97-06. Supplemental inspection guidelines, plugging and repair criteria, and limits on allowable leakage, may be contained in plant-specific Technical Specifications. Secondary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-102134 (Rev. 3 or later revisions or update) to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes inservice inspection (ISI) of steam generator tubes to detect degradation, and evaluation plugging/repair as needed to maintain the structural integrity of the pressure boundary. Inservice inspection is in accordance with RG 1.83 and EPRI steam generator examination guidelines, and evaluation of tube integrity, plugging or repair are in accordance with RG 1.121 and NEI 97–06. ***(2) Preventive Actions:*** Secondary water chemistry guidelines given in EPRI TR-102134 can significantly reduce denting. ***(3) Parameters Monitored/Inspected:*** The inspection activity in the program monitors extent of degradation. ***(4) Detection of Aging Effects:*** The extent and schedule of the inspections prescribed by the program are designed to ensure that degradation cannot proceed to an extent sufficient to threaten the integrity of the tubes. ***(5) Monitoring and Trending:*** Required inspection intervals (typically 18 months or each refueling or maintenance/repair outage) should provide for timely detection of denting. ***(6) Acceptance Criteria:*** Any denting detected is compared with the guidelines of the plant Technical Specifications. ***(7) Corrective Actions:*** Tubes containing denting that exceed the maximum permissible size must be plugged or repaired. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls, in conjunction with NRC oversight, are implemented in accordance with the requirements of Appendix B of 10 CFR 50. ***(10) Operating Experience:*** The present AMP has been effective in ensuring timely detection and control of denting in SG tubes.	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.2.2	Tube Bundle	Tube Support Lattice Bars	CS	Up to 300°C (572°F) Secondary-side Water Chemistry at 5.3-7.2 MPa	Loss of section thickness	Flow Accelerated Corrosion	NRC GL 97-06.
D1.2.3	Tube Bundle	Tube Plugs (Mechanical) (Westing-house)	Alloy 600. Alloy 690	Chemically Treated Borated Water at temperatures up to 340°C (644°F) and 15.5 MPa	Crack Initiation and Growth	PWSCC	Reg. Guide 1.83. EPRI document "PWR Steam Generator Examination Guidelines Rev. 5." NEI 97-06. WCAP-12244. WCAP-12245. Plant Technical Specifications. EPRI TR-105714 Rev. 3 or later revisions/update. *Operating Experience*: NRC IN 89-33. NRC BL 89-01. 89-01, S 1 and S 2. NRC IN 94-87.
D1.2.3	Tube Bundle	Tube Plugs (Mechanical) (Babcock & Wilcox)	Alloy 600, Alloy 690	Chemically Treated Borated Water at temperatures up to 340°C (644°F) and 15.5 MPa	Crack Initiation and Growth	PWSCC	*Same as for the effect of PWSCC on Item D1.2.3 SG tube mechanical plugs made by Westinghouse.*

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Plant-specific aging management program will be evaluated and redaction of NRC Generic Letter 97-06 for creation of an inspection program for steam generator internals is to be considered.	Plant-specific aging management program will be evaluated and redaction of NRC Generic Letter 97-06 for creation of an inspection program for steam generator internals is to be considered.	Yes plant specific AMP
Inservice inspection of steam generator tubes in conformance with NRC Regulatory Guide (RG) 1.83 and specific guidance of EPRI document "PWR Steam Generator Examination Guidelines, Revision 5," and NEI 97-06. Correlations for estimating the life of Westinghouse plugs are contained in WCAP-12244 and WCAP 12245. Supplemental inspection guidelines, plugging and repair criteria, and limits on allowable leakage, may be contained in plant-specific Technical Specifications. Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 (Rev. 3 or later revisions or update) to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program deals with the periodic inspection of steam generator tube plugs. ***(2) Preventive Actions:*** Guidance on primary water chemistry provided in EPRI TR-105714 can significantly reduce PWSCC. The program also recommends that certain susceptible heats of material be avoided. ***(3) Parameters Monitored/Inspected:*** The inspection activity in the program monitors flaw size and depth. ***(4) Detection of Aging Effects:*** The extent and schedule of the inspections prescribed by the program are designed to ensure that flaws cannot attain a depth sufficient to threaten the integrity of the plugs. Past problems with failure to detect flaws in plugs have led to leaking or failed plugs in several plants ((BL 89-01; 89-01, S. 1; 89-01, S. 2; IN 89-33, 94-87). However, improved detection procedures are provided in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5" and NEI 97-06. ***(5) Monitoring and Trending:*** Required inspection intervals (typically 18 months or each refueling or maintenance/repair outage) are intended to provide for timely detection of SCC. ***(6) Acceptance Criteria:*** Any cracking detected requires replacement. The lives of Westinghouse plugs can be estimated using procedures in WCAP-12244 and 12245 and compared with the limits given in those reports. ***(7) Corrective Actions:*** Tube plugs containing flaws or having insufficient estimated lives must be replaced. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls, in conjunction with NRC oversight, are implemented in accordance with the requirements of Appendix B of 10 CFR 50. This should ensure the timely detection and correction of cracking. ***(10) Operating Experience:*** Problems appear to have been related to susceptible heats of material and improper heat treatment. Any Alloy 600 mechanical plugs remaining in service must be considered susceptible (BL 89-01; 89-01, S 1; 89-01, S 2; IN 89-33, 94-87).	No, provided Plant Technical Specifi-cations conform to EPRI inspection guidelines and NEI 97-06.
Same as for the effect of PWSCC on Item D1.2.3 SG tube mechanical plugs made by Westinghouse.	*Same as for the effect of PWSCC on Item D1.2.3 SG tube mechanical plugs made by Westinghouse.*	No, provided Plant Technical Specifi-cations conform to EPRI inspection guidelines and NEI 97-06.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

D1. STEAM GENERATOR (Recirculating)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.3.1	Upper Assembly and Separators	Feedwater Inlet Ring and Supports	CS	Up to 300°C (572°F) Secondary-side Water Chemistry at 5.3-7.2 MPa	Loss of Material	Flow Accelerated Corrosion	Combustion Engineering Info-bulletin 90-04. ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. *Operating Experience*: NRC IN 91-19. NRC LER 50-362/90-05-01.

Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Periodic inservice inspections in conformance with the recommendations of Combustion Engineering Info-bulletin 90-04, and supplemental inspection guidelines contained in plant-specific Technical Specifications.	***(1) Scope of Program:*** The program deals with the periodic inspection of pressure retaining components and integral attachments (including feedwater distribution system components). ***(2) Preventive Actions:*** The program provides no guidance or recommendations on measures to prevent degradation. ***(3) Parameters Monitored/Inspected:*** The program monitors wall thinning and degradation of supports and attachments. ***(4) Detection of Aging Effects:*** The extent and schedule of the inspections prescribed by the program should ensure that thinning and degradation cannot proceed sufficiently to threaten the integrity of the welds. ***(5) Monitoring and Trending:*** Required inspection intervals should provide for timely detection of all forms of degradation. ***(6) Acceptance Criteria:*** Any defect detected is compared with the guidelines of the Plant Technical Specifications and with the recommendations of Combustion Engineering Info-bulletin 90-04. These recommendations should ensure structural integrity. ***(7) Corrective Actions:*** Excessively degraded components must be repaired or replaced ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls, in conjunction with NRC oversight, are implemented in accordance with the requirements of Appendix B of 10 CFR 50. These recommendations should continue to ensure the timely detection and correction of all forms of degradation. ***(10) Operating Experience:*** This form of degradation has been detected only in certain Combustion Engineering System 80 steam generators, where it has been successfully dealt with (NRC Information Notice I91-19; NRC LER 50-362/90-05-01).	No

D2. Steam Generator (Once–Through)

D2.1 Pressure Boundary and Structural

D2.1.1 Upper & Lower Heads

D2.1.2 Tube Sheets

D2.1.3 Primary Nozzles & Safe Ends

D2.1.4 Shell Assembly

D2.1.5 Feed Water and Auxiliary Feed Water Nozzles & Safe Ends

D2.1.6 Steam Nozzles & Safe Ends

D2.1.7 Primary Side Drain Nozzles

D2.1.8 Secondary Side Nozzles (Vent, Drain, and Instrumentation)

D2.1.9 Primary Manways Bolting

D2.1.10 Secondary Manways Handholes Bolting

D2.2 Tube Bundle

D2.2.1 Tubes and Sleeves

D2.2.2 Tube Plugs

D2. Steam Generator (Once-Through)

System, Structures, and Components

The system, structures, and components included in this table consist of the once-through type steam generators, as found in Babcock & Wilcox pressurized water reactors (PWRs), including all internal components and water/steam nozzles and safe ends. Based on US Nuclear Regulatory Commission Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," the primary water side (tube side) of the steam generator is classified as Group A Quality Standards and secondary water side is classified as Group B Quality Standard.

System Interfaces

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

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

D2. STEAM GENERATOR (Once-Through)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D2.1.1, D2.1.2	Pressure Boundary and Structural	Upper & Lower Heads, Tube Sheets	Low-Alloy Steel (LAS) with Stainless Steel (SS) (Head) and Alloy 82/182 (Tubesheet) Cladding	Chemically Treated Borated Water up to 340°C (644°F)	Crack Initiation and Growth	Stress Corrosion Cracking (SCC)	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. EPRI TR-105714 Rev. 3 or later revisions/update. *Supporting Documents*: Reg. Guide 1.43. Reg. Guide 1.44. *Operating Experience:* NRC IN 84-18.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Inservice inspection in conformance with ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1. Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 (Rev. 3 or later revisions or update) to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes preventive measures to mitigate stress corrosion cracking (SCC) of SS cladding and inservice inspection (ISI) to monitor the effects of SCC on the cladding on the intended function of the component. ***(2) Preventive Actions:*** Control of halogens, sulfates, and oxygen in the primary water to less than 0.05, 0.05, and 0.005 ppm, respectively, during operation, and monitor and control of water chemistry during shut down, mitigate potential of SCC. However, inadvertent introduction of contaminants into the coolant system can occur, and cladding regions with low delta ferrite may be sensitized during post-weld heat treatment and thus become susceptible to SCC. The AMP must therefore rely upon water chemistry monitoring to detect possible excursions. Selection of material in compliance with the recommendations of Regulatory Guide (RG) 1.44 prevent or mitigate SCC, and RG 1.43 prevents underclad cracking. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of crack initiation and growth on the intended function of the components by detection and sizing of cracks by ISI. Requirements of Table IWB 2500-1, examination category B-B specify the extent and schedule for inspection for all pressure-retaining welds. Requirements of testing category B-P conducted according to IWA–5000 specify visual VT-2 (IWA-5240) examination of all pressure retaining components during system leakage and hydrostatic test. ***(4) Detection of Aging Effects:*** Degradation of the component pressure boundary due to SCC can not occur without crack initiation and growth; extent and schedule of inspection assure detection of cracks before the loss of intended function of the components. However, the extent and schedule of inspection does not assure detection of cracks because ASME Section XI inspection requires examination of only the welds and weld regions, the potential of cracking in cladding remote from welds is not addressed. ***(5) Monitoring and Trending*****:** Inspection schedule in accordance with IWB-2500 should provide timely detection of cracks. System leakage test is conducted prior to plant startup following each refueling outage, and hydrostatic test at or near the end of each inspection interval. ***(6) Acceptance Criteria*****:** Any SCC degradation is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3514. Planar and liner flaws are sized according to IWA-3300 and IWA-3400. ***(7) Corrective Actions*****:** Repair is in conformance with IWA-4000 and IWB-4000, replacement according to IWA-7000 and IWB-7000, and reexamination in accordance with requirements of IWA-2200. Continued operation without repair requires that crack growth calculation be performed according to the guidance of ASME Section XI or other approved procedures. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls, in conjunction with NRC over-sight, are implemented in accordance with the require-ments of Appendix B of 10 CFR 50. ***(10) Operating Experience:*** Although the PWR primary pressure	Yes detection of aging effects should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D2.1.1	Pressure Boundary and Structural	Upper & Lower Heads (External Surfaces)	LAS	Air, Leaking Chemically Treated Borated Water and/or Steam up to 340°C (644°F)	Loss of Material	Boric Acid Corrosion of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.
D2.1.3	Pressure Boundary and Structural	Primary Nozzles and Safe Ends	Carbon Steel (CS) with SS cladding, Safe Ends: SS	Chemically Treated Borated Water up to 340°C (644°F) and 15.2 MPa	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
D2.1.3	Pressure Boundary and Structural	Primary Nozzles and Safe Ends	Carbon Steel (CS) with SS cladding, Safe Ends: SS (NiCrFe buttering, and SS or NiCrFe weld)	Chemically Treated Borated Water at temperatures up to 340°C (644°F) and 15.2 MPa	Crack Initiation and Growth	SCC	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. EPRI TR-105714 Rev. 3 or later revisions/update. *Supporting Documents*: Reg. Guide 1.43. Reg. Guide 1.44. *Operating Experience:* NRC IN 84-18. NRC IN 90-10. NRC IN 90-30.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) boundary has generally not been found to be affected by SCC because of low dissolved oxygen levels potential of SCC exists from inadvertent introduction of contaminants into the primary coolant system (IN 84–18).	
Implementation of NRC Generic Letter 88-05 and inservice inspection (ISI) in conformance with ASME Section XI (1989 edition or later edition as approved in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, to monitor the condition of the reactor coolant pressure boundary for occurrences of borated water leakage.	For evaluation and technical basis of the 10 elements of the AMP, see Chapter XI.M5, “Boric Acid Corrosion.”	No
Components have been designed or evaluated for fatigue for a 40 y design life based on postulated cycles, according to the requirements of the Code of record or later approved Codes.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal. See the Standard Review Plan, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes TLAA
Inservice inspection in conformance with ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1. Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 (Rev. 3 or later revisions or update) to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes preventive measures to mitigate stress corrosion cracking (SCC) of SS cladding and inservice inspection (ISI) to monitor the effects of SCC of the cladding on the intended function of the component. ***(2) Preventive Actions:*** Control of halogens, sulfates, and oxygen in the primary water to less than 0.05, 0.05, and 0.005 ppm, respectively, during operation, and monitor and control of water chemistry during shut down, mitigate potential of SCC. However, inadvertent introduction of contaminants into the coolant system can occur, and cladding regions with low delta ferrite may be sensitized during post-weld heat treatment and thus become susceptible to SCC. The AMP must therefore rely upon water chemistry monitoring to detect possible excursions. Selection of material in compliance with the recommendations of Regulatory Guide (RG) 1.44 prevent or mitigate SCC, and RG 1.43 prevents underclad cracking. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of crack initiation and growth on the intended function of the components by detection and sizing of cracks by ISI. Requirements of Table IWB 2500-1, examination categories B-D specify volumetric examination of nozzle to vessel welds and nozzle inside radius, and B-F specifies for dissimilar metal welds, volumetric and surface examination of nozzle-to-safe end butt welds NPS 4 or larger and surface examination of those less than NPS 4 or nozzle-to-safe end socket welds Requirements of testing category B-P conducted according to IWA–5000 specify visual VT-2 (IWA-5240) examination of all pressure retaining components during system leakage and hydrostatic test. ***(4) Detection of Aging Effects:*** Degradation of the component pressure boundary due to SCC can not occur without crack initiation and growth; extent and schedule of inspection assure detection of cracks before the loss of intended function of the	Yes detection of aging effects should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D2.1.4, D2.1.5	Pressure Boundary and Structural	Shell Assembly, Feedwater (FW) and Auxiliary FW (AFW) Nozzles & Safe Ends	CS	Up to 300°C Secondary-side Water Chemistry at 5.3-7.2 MPa	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
D2.1.4	Pressure Boundary and Structural	Shell Assembly	CS	Up to 300°C (572°F) Secondary -side Water Chemistry at 5.3-7.2 MPa	Loss of Material	Crevice and Pitting Corrosion	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. EPRI TR-102134 Rev. 3 or later revisions/update. *Operating Experience* NRC IN 82-37. NRC IN 85-65. NRC IN 90-04.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) components. However, the extent and schedule of inspection does not assure detection of cracks because ASME Section XI inspection requires examination of only the welds and weld regions, the potential of cracking in cladding remote from welds is not addressed. Also, based on NRC Information Notices (INs) 90-10 and 90-30 applicant should review Ni-alloy applications in primary coolant and implement an augmented inspection program and evaluate choice of transducers for ultrasonic examination of dissimilar metal welds. ***(5) Monitoring and Trending*:** Inspection schedule in accordance with IWB-2500 should provide timely detection of cracks. System leakage test is conducted prior to plant startup following each refueling outage, and hydrostatic test at or near the end of each inspection interval. ***(6) Acceptance Criteria*:** Any SCC degradation is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3514. Planar and liner flaws are sized according to IWA-3300 and IWA-3400. ***(7) Corrective Actions*:** Repair is in conformance with IWA-4000 and IWB-4000, replacement according to IWA-7000 and IWB-7000, and reexamination in accordance with requirements of IWA-2200. Continued operation without repair requires that crack growth calculation be performed according to the guidance of ASME Section XI or other approved procedures. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls, in conjunction with NRC oversight, are implemented in accordance with the requirements of Appendix B of 10 CFR 50. ***(10) Operating Experience:*** Although PWR primary pressure boundary has generally not been found to be affected by SCC because of low dissolved oxygen levels potential of SCC exists from inadvertent introduction of contaminants into the primary system (IN 84–18). of all pressure retaining Class 2 components.	
Same as for the effect of Fatigue on Item D2.1.3 primary nozzle and safe end.	*Same as for the effect of Fatigue on Item D2.1.3 primary nozzle and safe end.*	Yes TLAA
Inservice inspection in conformance with ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a) Table IWC 2500-1. Secondary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-102134 (Rev. 3 or later revisions or update) to minimize the potential of pitting and crevice corrosion.	***(1) Scope of Program:*** The program relies on preventive measures to mitigate crevice or pitting corrosion and inservice inspection (ISI) to monitor the effects of corrosion on the intended function of the steam generator shell. ***(2) Preventive Actions:*** Stringent control of secondary water chemistry in accordance with the guidance of EPRI TR-102134, frequent monitoring, and timely corrective action when specified impurity levels are exceeded, prevent or mitigate corrosion. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of corrosion by detection and sizing of flaws in pressure retaining welds by ISI and by detecting coolant leakage. Inspection requirements of Table IWC 2500-1, examination category C-A specify volumetric examination, extending 1/2 in.	Yes, detection of aging effects should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D2.1.5, D2.1.6	Pressure Boundary and Structural	FW and AFW Nozzles & Safe Ends, Steam Nozzles & Safe Ends	CS	Up to 300°C (572°F) Steam or Secondary -side Water Chemistry at 5.3-7.2 MPa	Wall Thinning	Flow Accelerated Corrosion (FAC)	EPRI NSAC–202L–R2. NRC GL 89-08. NUREG-1344. *Supporting Documents:* BWRVIP-75. BWRVIP-79 (EPRI TR-103515). *Operating Experience* NRC Bl 87-01. NRC IN 81-28. NRC IN 89-53. NRC IN 91-18. NRC IN 91-18 S1. NRC IN 92-35. NRC IN 93-21. NRC IN 95-11. NRC IN 97-84.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) each side, of all circumferential and tubesheet-to-shell welds, and category C-H specifies visual VT-2 (IWA-5240) examination during system leakage and hydrostatic tests of all pressure retaining Class 2 components. ***(4) Detection of Aging Effects:*** The extent and schedule of the inspections prescribed by the program are designed to ensure that flaws cannot attain a depth sufficient to threaten the integrity of the welds. However, based on NRC Information notice 90-04 where general corrosion pitting of the shell exists, the program guidelines may not be sufficient to detect pitting and corrosion and additional inspection procedures may be required. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI should provide for timely detection of leakage. System leakage test is typically conducted at 40-month intervals, and hydrostatic test at or near the end of each inspection interval. ***(6) Acceptance Criteria:*** Any defect detected is compared with the requirements of Section XI, IWC 3500. Any relevant conditions that may be detected during the leakage and hydrostatic tests are evaluated in accordance with IWC-3400 and IWC-3500. ***(7) Corrective Actions:*** Welds containing flaws that exceed the maximum permissible size must be repaired. Repair and replacement are in conformance with IWA-4000. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls, in conjunction with NRC oversight, are implemented in accordance with the requirements of Appendix B of 10 CFR 50. ***(10) Operating Experience:*** This AMP has resulted in the timely detection and correction of corrosion in several steam generators in the US. (NRC Information Notices 82-37 & 85-65).	
Implementation of EPRI guidelines of NSAC–202L–R2 for effective flow accelerated corrosion (FAC) program.	For evaluation and technical basis of the 10 elements of the AMP, see Chapter XI.M6 "Flow Accelerated Corrosion".	No

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D2. STEAM GENERATOR (Once-Through)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D2.1.6	Pressure Boundary and Structural	Steam Nozzles & Safe Ends	CS	Up to 300°C (572°F) Steam	Cumulative Fatigue Damage	Fatigue	Design Code of record or later approved Codes.
D2.1.7	Pressure Boundary and Structural	Primary Side Drain Nozzles	Alloy 600	Chemically Treated Borated Water up to 340°C (644°F) and 15.2 MPa	Crack Initiation and Growth	SCC, Primary Water Stress Corrosion Cracking (PWSCC)	-
D2.1.8	Pressure Boundary and Structural	Secondary Side Nozzles (Vent, Drain, and Instru-mentation)	Alloy 600	Up to 300°C (572°F) Secondary -side Water Chemistry at 5.3-7.2 MPa	Crack Initiation and Growth	SCC, Primary Water Stress Corrosion Cracking (PWSCC)	-
D2.1.4 - D2.1.6, D2.1.9, D2.1.10	Pressure Boundary and Structural	External Surfaces of Shell Assembly, FW and AFW Nozzles & Safe Ends, Steam Nozzles & Safe Ends; Primary Manways Bolting, Secondary Manways Handholes Bolting	CS, LAS	Air, Leaking Chemically Treated Borated Water and/or Steam at temperatures up to 340°C (644°F)	Loss of Material	Boric Acid Corrosion of External Surfaces	*Same as for the effect of Boric Acid Corrosion on Item D2.1.1 upper and lower heads.*
D2.1.9, D2. 1.10	Pressure Boundary and Structural	Primary Manways Bolting, Secondary Manways Handholes Bolting	LAS	Air, with metal temperatures up to 340°C (644°F)	Loss of Preload	Stress Relaxation	NUREG-1339. EPRI NP-5769. EPRI NP-5067. ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a. *Operating Experience*: IEB 82-02. NRC GL 91-17.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Same as for the effect of Fatigue on Item D2.1.3 primary nozzle and safe end.	*Same as for the effect of Fatigue on Item D2.1.3 primary nozzle and safe end.*	Yes TLAA
Plant-specific aging management program.	Plant-specific aging management program is to be evaluated.	Yes, plant specific AMP
Plant-specific aging management program.	Plant-specific aging management program is to be evaluated.	Yes, plant specific AMP
Same as for the effect of Boric Acid Corrosion on Item D2.1.1 upper and lower heads.	*Same as for the effect of Boric Acid Corrosion on Item D2.1.1 upper and lower heads.*	No
Recommendations for a comprehensive bolting integrity program delineated in NUREG–1339 and industry's recommendations delineated in EPRI NP-5769, with the exceptions noted in NUREG 1339, for safety related bolting, and EPRI NP-5067 for other bolting.	For evaluation and technical basis of the 10 elements of the AMP, see Chapter XI.M12 "Bolting Integrity."	No

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D2. STEAM GENERATOR (Once-Through)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D2. 1.10	Pressure Boundary and Structural	Secondary Manways and Handholes	CS	Air, Leaking Secondary-side Water and/or Steam at temperatures up to 300°C (572°F)	Wall Thinning	Erosion	ASME Section XI, 1989 or later edition as approved in 10 CFR 50.55a.
D2.2.1	Tube Bundle (Babcock & Wilcox)	Tubes and Sleeves	Alloy 600	Chemi-cally Treated Borated Water up to 340°C (644°F) and 15.2 MPa	Crack Initiation and Growth	PWSCC	Reg. Guide 1.83. EPRI document "PWR Steam Generator Examination Guidelines Rev. 5." NEI 97-06. Reg. Guide 1.121. Plant Technical Specifications. EPRI TR-105714 Rev. 3 or later revisions/update. *Operating Experience*: NRC IN 97-88.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Inservice inspection in conformance with ASME Section XI (1989 or later edition as approved in 10 CFR 50.55a), Subsection IWC, Table IWC 2500-1.	***(1) Scope of Program:*** The program is focused on managing the effects of erosion damage to the sealing surfaces of manway and handhole covers due to localized steam leakage. ***(2) Preventive Actions:*** Verification of the sealing function by frequent monitoring and timely corrective action prevent or mitigate erosion damage. ***(3) Parameter Monitored/ Inspected:*** The AMP monitors the effects of erosion on the sealing function of manway and handhole covers by detection of leakage or erosion damage by inservice inspection (ISI). Inspection requirements of ASME Section XI, Table IWC 2500-1, testing category C-H specify visual VT-2 (IWA-5240) examination of all pressure retaining components. ***(4) Detection of Aging Effects:*** Aging degradation of manway and handhole covers can not occur without erosion damage and steam leakage; extent and schedule of inspection assure detection of leakage or erosion damage before the loss of sealing function of the flange. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI should provide for timely detection of leakage. System leakage test is typically conducted at 40-month intervals. ***(6) Acceptance Criteria:*** Any relevant conditions that may be detected during the leakage and hydrostatic tests are evaluated in accordance with IWC-3100 and acceptance standards of IWC-3400 and IWC-3500. ***(7) Corrective Actions:*** Prior to service, corrective measures are needed to meet the requirements of IWC-3100. The leakage source and areas of erosion damage are located. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** The present AMP has been effective in managing the effects of erosion due to localized steam leakage on the sealing function of secondary manways and handholes.	No
Inservice inspection of steam generator tubes in conformance with NRC Regulatory Guide (RG) 1.83 and specific guidance of EPRI document "PWR Steam Generator Examination Guidelines, Revision 5," and NEI 97-06. Assessment of tube integrity and plugging or repair criteria of flawed tubes is governed by NRC RG 1.121 and NEI 97-06. Supplemental inspection guidelines, plugging and repair criteria, and limits on allowable leakage, may be contained in plant-specific Technical Specifications. Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 (Rev. 3 or later revisions or update) to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes inservice inspection (ISI) of steam generator tubes to detect degradation, and evaluation plugging/repair as needed to maintain the structural integrity of the pressure boundary. Inservice inspection is in accordance with RG 1.83 and EPRI steam generator examination guidelines, and evaluation of tube integrity, plugging or repair are in accordance with RG 1.121 and NEI 97–06. These guidelines provide criteria for the qualification of personnel, specific techniques and the associated acquisition and analysis of data including procedures, probe selection, analysis protocols, and reporting criteria. ***(2) Preventive Actions:*** Primary water chemistry guidelines given in EPRI TR-105714 provide guidance on the reduction of susceptibility to PWSCC. ***(3) Parameters Monitored/Inspected:*** The inspection activity in the program detects the presence of cracks and may, in some cases, monitor flaw size and depth, or, alternatively, remaining sound wall thickness. ***(4) Detection of Aging Effects:*** The extent and schedule of the inspections and test techniques prescribed by the program are designed to	No, provided Plant Technical Specifi-cations conform to EPRI inspection guidelines and NEI 97-06 and primary water chemistry limits are adhered to.

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D2.2.1	Tube Bundle (Babcock & Wilcox)	Tubes and Sleeves	Alloy 600	Up to 300°C (572°F) Secondary-side Water Chemistry at 5.3-7.2 MPa	Crack Initiation and Growth	Outer Diameter Stress Corrosion Cracking (ODSCC)	Reg. Guide 1.83. EPRI document "PWR Steam Generator Examination Guidelines Rev. 5." NEI 97-06. Reg. Guide 1.121. NRC GL 95-05. Plant Technical Specifications. EPRI TR-102134 Rev. 3 or later revisions/update. *Operating Experience*: NRC IN 97-88.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) ensure continued tube integrity and that aging effects will be discovered and repaired before the loss of intended function of the tubes. Problems with tube inspection (IN 97-88), e.g., failures to detect some flaws, uncertainties in flaw sizing, inaccuracies in flaw locations, and inability to detect some cracks at locations with dents are addressed by current inspection guidelines given in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5" and NEI 97-06. ***(5) Monitoring and Trending:*** Required inspection intervals, typically 18 months or each refueling or maintenance/repair outage, provide for timely detection of PWSCC. ***(6) Acceptance Criteria:*** Any cracking detected is compared with the guidelines of the Plant Technical Specifications, RG 1.121, and the structural integrity guidelines described in NEI 97–06. ***(7) Corrective Actions:*** Tubes containing flaws that do not meet the acceptance criteria must be plugged or repaired. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls, in conjunction with NRC oversight, are implemented in accordance with the requirements of Appendix B of 10 CFR 50. Recent experience indicates the importance of performing a complete inspection using the best available techniques and equipment for the reliable detection of tube degrada-tion and to ensure that new forms of degradation are detected. ***(10) Operating Experience:*** The present AMP has been effective in ensuring timely detection and correc-tion of PWSCC degradation in steam generator tubes.	
Inservice inspection of steam generator tubes in conformance with NRC Regulatory Guide (RG) 1.83 and specific guidance of EPRI document "PWR Steam Generator Examination Guidelines, Revision 5," and NEI 97-06. Assessment of tube integrity and plugging or repair criteria of flawed tubes is governed by NRC RG 1.121 and NEI 97-06. Alternative criteria applicable under certain circumstances to cracking at tube support plates in Westinghouse-designed steam generators are provided in NRC Generic Letter (GL) 95-05. Supplemental inspection guidelines, plugging and repair criteria, and limits on allowable leakage, may be contained in plant-specific Technical Specifications. Secondary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-102134 (Rev. 3 or later revisions or update) to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes inservice inspection (ISI) of steam generator tubes to detect degradation, and evaluation plugging/repair as needed to maintain the structural integrity of the pressure boundary. Inservice inspection is in accordance with RG 1.83 and EPRI steam generator examination guidelines, and evaluation of tube integrity, plugging or repair are in accordance with RG 1.121, GL 95-05, and NEI 97–06. These guidelines provide criteria for the qualification of personnel, specific techniques and the associated acquisition and analysis of data including procedures, probe selection, analysis protocols, and reporting criteria. ***(2) Preventive Actions:*** Secondary water chemistry guidelines given in EPRI TR-102134 provide guidance on the reduction of susceptibility to ODSCC. ***(3) Parameters Monitored/Inspected:*** The inspection activity in the program detects the presence of cracks and may, in some cases, monitor flaw size and depth, or, alternatively, remaining sound wall thickness, or other parameters such as voltage that may be used to assess structural integrity. ***(4) Detection of Aging Effects:*** The extent and schedule of the inspections and test techniques prescribed by the program are designed to ensure continued tube integrity and that aging effects will be discovered and repaired before the loss of intended function of the tubes. Problems with tube inspection (IN 97-88), e.g., failures to detect some flaws, uncertainties in flaw sizing, inaccuracies in	No, provided Plant Technical Specifi-cations conform to EPRI inspection guidelines and NEI 97-06 and secondary water chemistry limits are adhered to.

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D2. STEAM GENERATOR (Once-Through)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D2.2.1	Tube Bundle (Babcock & Wilcox)	Tubes and Sleeves	Alloy 600	Up to 300°C (572°F) Secondary-side Water Chemistry at 5.3-7.2 MPa	Crack Initiation and Growth	Intergranu-lar Attack (IGA)	Reg. Guide 1.83. EPRI document "PWR Steam Generator Examination Guidelines Rev. 5." NEI 97-06. Reg. Guide 1.121. NRC GL 95-05. Plant Technical Specifications. EPRI TR-102134 Rev. 3 or later revisions/update.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) flaw locations, and inability to detect some cracks at locations with dents are addressed by current inspection guidelines given in the EPRI document “PWR Steam Generator Examination Guidelines: Revision 5” and NEI 97-06. ***(5) Monitoring and Trending:*** Required inspection intervals, typically 18 months or each refueling or maintenance/repair outage, provide for timely detection of ODSCC. ***(6) Acceptance Criteria:*** Any cracking detected is compared with the guidelines of the Plant Technical Specifications, RG 1.121, and the structural integrity guidelines described in NEI 97–06 and GL 95-05. ***(7) Corrective Actions:*** Tubes containing flaws that do not meet the acceptance criteria must be plugged or repaired. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls, in conjunction with NRC oversight, are implemented in accordance with the requirements of Appendix B of 10 CFR 50. Recent experience indicates the importance of performing a complete inspection using the best available techniques and equipment for the reliable detection of tube degradation and to ensure that new forms of degradation are detected. ***(10) Operating Experience:*** The present AMP has been effective in ensuring timely detection and correction of ODSCC degradation in steam generator tubes.	
Inservice inspection of steam generator tubes in conformance with NRC Regulatory Guide (RG) 1.83 and specific guidance of EPRI document “PWR Steam Generator Examination Guidelines, Revision 5,” and NEI 97-06. Assessment of tube integrity and plugging or repair criteria of flawed tubes is governed by NRC RG 1.121 and NEI 97-06; defects at the tube sheet may be shown to satisfy RG 1.121 guidelines using the P*, F*, or L* criterion. Alternative criteria applicable under certain circumstances to cracking at tube support plates in Westinghouse-designed steam generators are provided in NRC Generic Letter (GL) 95-05. Supplemental inspection guidelines, plugging and repair criteria, and limits on allowable leakage, may be contained in plant-specific Technical Specifications. Secondary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-102134 (Rev. 3 or later revisions or update) to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes inservice inspection (ISI) of steam generator tubes to detect degradation, and evaluation plugging/repair as needed to maintain the structural integrity of the pressure boundary. Inservice inspection is in accordance with RG 1.83 and EPRI steam generator examination guidelines, and evaluation of tube integrity, plugging or repair are in accordance with RG 1.121, GL 95-05, and NEI 97–06. These guidelines provide criteria for the qualification of personnel, specific techniques and the associated acquisition and analysis of data including procedures, probe selection, analysis protocols, and reporting criteria. ***(2) Preventive Actions:*** Secondary water chemistry guidelines given in EPRI TR-102134 provide guidance on the reduction of susceptibility to IGA. ***(3) Parameters Monitored/Inspected:*** The inspection activity in the program monitors the extent of degradation or, alternatively, remaining sound wall thickness. ***(4) Detection of Aging Effects:*** The extent and schedule of the inspections prescribed by the program are designed to ensure that flaws cannot attain a depth sufficient to threaten the integrity of the tubes. Problems with tube inspection, e.g., failures to detect and properly characterize IGA, are addressed in the EPRI document “PWR Steam Generator Examination Guidelines: Revision 5” and NEI 97-06. ***(5) Monitoring and Trending:*** Required inspection intervals, typically 18 months or each refueling or maintenance/repair outage, provide for timely detection of PWSCC. ***(6) Acceptance Criteria:*** Any cracking detected is compared with the requirements of	No, provided Plant Technical Specifi-cations conform to EPRI inspection guidelines and NEI 97-06 and secondary water chemistry limits are adhered to.

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D2.2.1	Tube Bundle (Babcock & Wilcox)	Tubes and Sleeves	Alloy 600	Up to 300°C (572°F) Secondary-side Water Chemistry at 5.3-7.2 MPa	Crack Initiation and Growth	Fatigue	Plant Technical Specifications,
D2.2.2	Tube Bundle	Tube Plugs (Mechanical) (Westing-house)	Alloy 600. Alloy 690	Chemically Treated Borated Water at temperatures up to 340°C (644°F) and 15.5 MPa	Crack Initiation and Growth	PWSCC	Reg. Guide 1.83. EPRI document “PWR Steam Generator Examination Guidelines Rev. 5.” NEI 97-06. WCAP-12244. WCAP-12245. Plant Technical Specifications. EPRI TR-105714 Rev. 3 or later revisions/update. *Operating Experience*: NRC IN 89-33. NRC BL 89-01. 89-01, S 1 and S 2. NRC IN 94-87.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) the Plant Technical Specifications, RG 1.121, and the structural integrity guidelines described in NEI 97–06 and GL 95-05. ***(7) Corrective Actions:*** Tubes containing flaws that do not meet the acceptance criteria must be plugged or repaired. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls, in conjunction with NRC oversight, are implemented in accordance with the requirements of Appendix B of 10 CFR 50. Recent experience indicates the importance of performing a complete inspection using the best available techniques and equipment for the reliable detection of tube degradation and to ensure that new forms of degradation are detected. ***(10) Operating Experience:*** The present AMP has been effective in ensuring timely detection and correction of IGA degradation in SG tubes.	
Same as for the effect of Fatigue on Item D2.1.3 primary nozzle and safe end.	*Same as for the effect of Fatigue on Item D2.1.3 primary nozzle and safe end.*	Yes TLAA
Inservice inspection of steam generator tubes in conformance with NRC Regulatory Guide (RG) 1.83 and specific guidance of EPRI document "PWR Steam Generator Examination Guidelines, Revision 5," and NEI 97-06. Correlations for estimating the life of Westinghouse plugs are contained in WCAP-12244 and WCAP 12245. Supplemental inspection guidelines, plugging and repair criteria, and limits on allowable leakage, may be contained in plant-specific Technical Specifications. Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 (Rev. 3 or later revisions or update) to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program deals with the periodic inspection of steam generator tube plugs. ***(2) Preventive Actions:*** Guidance on primary water chemistry provided in EPRI TR-105714 can significantly reduce PWSCC. The program also recommends that certain susceptible heats of material be avoided. ***(3) Parameters Monitored/ Inspected:*** The inspection activity in the program monitors flaw size and depth. ***(4) Detection of Aging Effects:*** The extent and schedule of the inspections prescribed by the program are designed to ensure that flaws cannot attain a depth sufficient to threaten the integrity of the plugs. Past problems with failure to detect flaws in plugs have led to leaking or failed plugs in several plants ((BL 89-01; 89-01, S. 1; 89-01, S. 2; IN 89-33, 94-87). However, improved detection procedures are provided in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5" and NEI 97-06. ***(5) Monitoring and Trending:*** Required inspection intervals (typically 18 months or each refueling or maintenance/repair outage) are intended to provide for timely detection of SCC. ***(6) Acceptance Criteria:*** Any cracking detected requires replacement. The lives of Westinghouse plugs can be estimated using procedures in WCAP-12244 and 12245 and compared with the limits given in those reports. ***(7) Corrective Actions:*** Tube plugs containing flaws or having insufficient estimated lives must be replaced. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls, in conjunction with NRC oversight, are implemented in accordance with the requirements of Appendix B of 10	No, provided Plant Technical Specifi-cations conform to EPRI inspection guidelines and NEI 97-06.

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D2.2.2	Tube Bundle	Tube Plugs (Mechanical) (Babcock & Wilcox)	Alloy 600, Alloy 690	Chemically Treated Borated Water at temperatures up to 340°C (644°F) and 15.5 MPa	Crack Initiation and Growth	PWSCC	*Same as for the effect of PWSCC on Item D1.2.3 SG tube mechanical plugs made by Westinghouse.*

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) CFR 50. This should ensure the timely detection and correction of cracking. ***(10) Operating Experience:*** Problems appear to have been related to susceptible heats of material and improper heat treatment. Any Alloy 600 mechanical plugs remaining in service must be considered susceptible (BL 89-01; 89-01, S 1; 89-01, S 2; IN 89-33, 94-87).	
Same as for the effect of PWSCC on Item D1.2.3 SG tube mechanical plugs made by Westinghouse.	*Same as for the effect of PWSCC on Item D1.2.3 SG tube mechanical plugs made by Westinghouse.*	No, provided Plant Technical Specifi-cations conform to EPRI inspection guidelines and NEI 97-06.

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