CHAPTER IV

(12/06/99)

REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

Major Plant Sections

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A1. Reactor Vessel (Boiling Water Reactor)

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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).

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

A1. REACTOR VESSEL (Pressurized Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A1.1.1 thru A1.1.3	Top Head Enclosure (with cladding)	Top Head, Nozzles (Vent, Top Head Spray or RCIC, and Spare), Head Flange	SA302-Gr B, SA533-Gr B, SA336, with stainless steel (SS) cladding	288°C 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, Sup. 1. ASME Section XI, 1989 Edition. Reg. Guide 1.43 BWRVIP-29. EPRI TR-103515.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination categories B-A for head welds and B-D for full penetration nozzle-to-head welds. Prevention is by material selection in accordance with guidelines of NUREG-0313, Rev. 2, and of Regulatory Guide 1.43 for control of stainless steel weld cladding of low-alloy steels. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in BWRVIP-29 and TR-103515 to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program is focused on managing the effects of stress corrosion cracking (SCC) of SS cladding on the intended function of top head enclosure. NUREG-0313, Rev. 2 and Generic Letter (GL) 88-01, respectively, describe the technical basis and staff guidance regarding the problem of IGSCC in BWRs. However, SCC is not anticipated to be an issue for the top head enclosure because analytical evaluations indicate that cracks in the SS cladding will stop growing in the ferritic base metal. ***(2) Preventive Actions:*** Selection of material considered resistant to IGSCC, e.g., grades of weld metal with a maximum carbon of 0.035% and minimum 7.5% ferrite, prevent or mitigate IGSCC, and Regulatory Guide (RG) 1.43 provides assurance that production cladding complies with ASME Section III and XI guidelines to prevent underclad cracking. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in BWRVIP-29 and TR-103515 to minimize the potential of crack initiation and growth. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of IGSCC on the intended function of top head enclosure by detection and sizing of cracks by inservice inspection (ISI). Inspection requirements of Table IWB 2500-1, examination category B-A specifies volumetric inspection of all circumferential and meridional welds and B-D specifies for all nozzles volumetric inspection of nozzle-to-vessel welds and nozzle inside radius section. ***(4) Detection of Aging Effects:*** Aging effects degradation of the top head enclosure can not occur without crack initiation; extent and schedule of inspection assure detection of cracks before the loss of intended function of the top head enclosure. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI should provide for timely detection of cracks. Top head interior is inspected at 1st refueling outage and subsequent outages at approximately 3 y intervals. ***(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-3520 for visual examination, IWB-3510 for head welds, and IWB-3512 for full penetration nozzle welds. Visual examinations that reveal relevant conditions may be supplemented by surface and volumetric examinations (IWB-3200) for flaw characterization, analytical evaluation, corrective measures, and repairs. Continued service without repair requires analytical evaluation to demonstrate acceptability. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWA-4000 and IWB-4000, and reexamination in accordance with requirements of IWA-2200. Also, some plants have removed cladding in top head because of cracking. ***(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 is effective in managing the effects of IGSCC on the intended function of top head enclosure.	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A1.1.3	Top Head Enclosure	Head Flange	SA302-Gr B, SA533-Gr B, SA336, with or without SS cladding	288°C Steam	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. GSI-190.
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	Air, Leaking Oxygenated Water and/or Steam at 288°C	Crack Initiation and Growth	SCC, IGSCC	ASME Section XI, 1989 Edition. RICSIL 055R1. Reg. Guide 1.65. ASME Section III, 1989 edition.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or Section I (Power Boilers) and Section VIII, Division 1 (Unfired Pressure Vessel).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination category B-G-1, and testing category B-P for system leakage, and additional recommendations of GE Rapid Information Communication Service Information Letter (RICSIL) 055 Revision 1, Supplement 1. Prevention and replacement in accordance with Regulatory Guide 1.65.	***(1) Scope of Program:*** The program is focused on managing the effects of IGSCC on the intended function of reactor vessel closure stud bolting. ***(2) Preventive Actions:*** Design requirements of ASME Section III, Subsection NB, and additional guidance of Regulatory Guide (RG) 1.65 on material selection, preservice inspection, 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 prevent cracking 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. Preservice inspection in conformance with NB-2580 of Section III of the Code requires ultrasonic examination of stud bolting over the entire surface prior to threading. During refueling and while the head is removed, the stud bolts and holes are protected from corrosion and contamination in accordance with RICSIL 055 R1 S1. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of IGSCC on the intended function of closure stud bolting by detection and sizing of cracks by inservice inspection (ISI). Inspection 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. Requirements for training and qualification of personnel and performance demonstration for procedures and equipment is in conformance with Appendices VII and VIII of ASME Section XI. Inspection requirements of testing category B-P conducted according to IWA–5000 specify visual VT-2 (IWA-5240) examination of all pressure retaining components extending to and including the second closed valve at the boundary extremity, 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 inspection assure detection of cracks before the loss of intended function of closure stud bolting. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI	No

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	Leaking Oxygenated Water and/or Steam up to 288°C	Crack Initiation and Growth	SCC, IGSCC	Plant Technical Specifications.
A1.2.1, A1.2.2	Vessel Shell	Vessel Flange, Upper Shell	SA302-Gr B, SA533-Gr B, SA336 with SS cladding	288°C Steam	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. GSI-190.
A1.2.3 thru 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, Oxygenated Water, max $5x10^9$ $n/cm^2 \cdot s$	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. GSI-190.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) and, based on operating experience, additional requirements of RICSIL 055 Rev. 1, are effective and adequate for timely detection of cracks. All BWRs are inspected in accordance with Program B IWB-2412 which requires 100% inspection every 10 y, at least 16% in 3 y and 50% in 7 y. 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. 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 bolting are evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3515/17. ***(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 based on ASME Section XI and industry guidelines of RICSIL 055 Revision 1 and its Supplement 1, 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, no AMP
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or Section I (Power Boilers) and Section VIII, Division 1 (Unfired Pressure Vessel).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or Section I (Power Boilers) and Section VIII, Division 1 (Unfired Pressure Vessel).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A1.2.4	Vessel Shell	Intermediate (Beltline) Shell	SA302-Gr B, SA533-Gr B with 308, 309, 308L, 309L Cladding	288°C, Oxygenated 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 92-11, Rev. 1. BWR Integrated Surveillance Program (to be submitted).
A1.2.3 thru 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, Oxygenated Water, $5x10^8$ - $5x10^9$ $n/cm^2 \cdot s$	Crack Initiation and Growth	SCC, IGSCC	NUREG-0313, Rev. 2. NRC GL 88-01. NRC GL 88-01, Sup. 1. ASME Section XI, 1989 Edition. Reg. Guide 1.43. BWRVIP-29. EPRI TR-103515. BWRVIP-05. *Supporting BWRVIP:* BWRVIP-03, BWRVIP-06, BWRVIP-14, BWRVIP-44, BWRVIP-45, BWRVIP-59, BWRVIP-60, BWRVIP-62.

Existing 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, implemented through Generic Letters (GLs) 88-11 and 92-01 Rev. 1 and Supplement 1, to predict effects of neutron irradiation on reactor vessel materials. In addition, inservice inspection of ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, examination category B-A of all pressure retaining welds in the vessel and repair welds in beltline region, defined as the region extending for the length of the thermal shield or effective length of reactor fuel elements. A NRC Generic Letter is being prepared regarding applications for exemptions from inspection requirements for circumferential welds. BWR integrated surveillance program to be submitted.	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 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, and (c) the equivalent margins analyses performed in accordance with 10 CFR 50, Appendix G. Reactor surveillance program requires that the existing reactor vessel material surveillance program be evaluated to determine whether there is sufficient material data and dosimetry to monitor irradiation embrittlement at the end of the license renewal term and whether operating restrictions (i.e., inlet temperature, neutron spectrum and flux) are necessary. If surveillance capsules are not removed during the license renewal term it will be necessary to establish operating restrictions to ensure the plant is operated within the environment of the surveillance capsules.	Yes TLAA
Inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a, Codes and Standards), Subsection IWB, Table IWB 2500-1, examination categories B-N-1 for vessel interior and B-A for shell welds. Prevention is by material selection in accordance with guidelines of NUREG-0313, Rev. 2, and of Regulatory Guide 1.43 for control of stainless steel weld cladding of low-alloy steels. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in BWRVIP-29 and TR-103515 to minimize the potential of crack initiation and growth. BWRVIP guideline is under staff review.	***(1) Scope of Program:*** The program is focused on managing the effects of stress corrosion cracking (SCC) of SS cladding on the intended function of reactor vessel shell. NUREG-0313 and GL 88-01, respectively, describe the technical basis and staff guidance regarding the problem of IGSCC in BWRs. However, SCC is not anticipated to be an issue for the vessel shell because analytical evaluations indicate that growth of the cracks in SS cladding into the ferritic base metal will be very slow. ***(2) Preventive Actions:*** Selection of material considered resistant to IGSCC, e.g., grades of weld metal with a maximum carbon of 0.035% and minimum 7.5% ferrite, prevent or mitigate IGSCC, and Regulatory Guide (RG) 1.43 provides assurance that production cladding complies with ASME Section III and XI guidelines to prevent underclad cracking. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in BWRVIP-29 and 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 effects degradation of the reactor vessel shell can not occur without crack initiation. However, because of inaccessibility, the extent and size of inspection may not be adequate to assure detection of cracks in the SS cladding before the loss of intended function of the reactor vessel. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with applicable approved BWRVIP guideline. ***(6) Acceptance Criteria:*** Any IGSCC degradation is evaluated in accordance with applicable approved BWRVIP guideline.	Yes BWRVIP Guideline

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A1.2.6	Vessel Shell	Beltline Welds	Low-alloy steel (LAS) weldments with 308, 309, 308L, 309L cladding	288°C, Oxygenated Water, $5x10^8$ - $5x10^9$ $n/cm^2 \cdot s$	Loss of Fracture Toughness	Neutron Irradia-tion Embrittle-ment	Appendices G & H to 10 CFR Part 50, Reg. Guide 1.99, NRC GL 88-11, NRC GL 92-11, Rev. 1. BWR Integrated Surveillance Program (to be submitted).
A1.2.7	Vessel Shell	Attachment Welds	SS, Inconel 182	288°C, Oxygenated Water	Crack Initiation and Growth	SCC, IGSCC	NUREG-0313, Rev. 2. NRC GL 88-01. NRC GL 88-01, Sup. 1. ASME Section XI, 1989 Edition. BWRVIP-29. EPRI TR-103515.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) ***(7) Corrective Actions:*** The corrective action proposed by the BWRVIP is under staff review. ***(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 is effective in managing crack initiation and growth due to SCC, however, because of inaccessibility, the extent and size of inspection may not be adequate to assure detection of cracks.	
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, implemented through Generic Letters (GLs) 88-11 and 92-01 Rev. 1 and Supplement 1, to predict effects of neutron irradiation on reactor vessel materials. In addition, inservice inspection of ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, examination category B-A of all pressure retaining welds in the vessel and repair welds in beltline region, defined as the region extending for the length of the thermal shield or effective length of reactor fuel elements. A NRC Generic Letter is being prepared regarding applications for exemptions from inspection requirements for circumferential welds. BWR integrated surveillance program to be submitted.	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 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, and (c) the equivalent margins analyses performed in accordance with 10 CFR 50, Appendix G. Reactor surveillance program requires that the existing reactor vessel material surveillance program be evaluated to determine whether there is sufficient material data and dosimetry to monitor irradiation embrittlement at the end of the license renewal term and whether operating restrictions (i.e., inlet temperature, neutron spectrum and flux) are necessary. If surveillance capsules are not removed during the license renewal term it will be necessary to establish operating restrictions to ensure the plant is operated within the environment of the surveillance capsules.	Yes TLAA
Inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a, Codes and Standards), Subsection IWB, Table IWB 2500-1, examination categories B-N-2 for integrally welded core support structure. Prevention is by material selection in accordance with guidelines of NUREG-0313, Rev. 2. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in BWRVIP-29 and 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 the intended function of the component. NUREG-0313 and GL 88-01, respectively, describe the technical basis and staff guidance regarding the problem of IGSCC in BWRs. ***(2) Preventive Actions:*** Mitigation is by selection of materials resistant to IGSCC and control of coolant water chemistry in accordance with EPRI guidelines in BWRVIP-29 and TR-103515 including stringent control of conductivity (many BWRs now operate at <0.15 μS/cm^2). Hydrogen additions are effective in reducing electrochemical potentials in the recirculating piping system, but are less effective in the core region. Also, the susceptibility of Ni-alloys to SCC should be evaluated. ***(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). Inspection requirements of Table IWB 2500-1, examination category B-N-2 specifies visual VT-3 examination of all accessible surfaces of	Yes Element 4 should be further evaluated

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 Steam	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. GSI-190.
A1.3.2 thru A1.3.9	Nozzles	Feedwater, HPCI, HPCS, LPCS, CRDRL, Recirculating Water, LPCI, Isolation Condenser Supply	SA508-Cl2 with SS Cladding	Up to 288°C, Oxygenated Water	Crack Initiation and Growth	SCC, IGSCC	NUREG-0313, Rev. 2. NRC GL 88-01. NRC GL 88-01, Sup. 1. ASME Section XI, 1989 Edition. Reg. Guide 1.43. BWRVIP-29. EPRI TR-103515.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) integral welds. ***(4) Detection of Aging Effects:*** Degradation due to SCC can not occur without crack initiation and growth. However, VT-3 may not be adequate to inspect tight cracks and creviced condition. Applicant should determine whether an augmented inspection program is required. Plant specific plans may include ultrasonic or other volumetric inspection. ***(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:*** 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.	
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or Section I (Power Boilers) and Section VIII, Division 1 (Unfired Pressure Vessel).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Program delineated in NUREG-0313, Rev. 2 and implemented through NRC Generic letter (GL) 88-01 and its Supplement 1, and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination categories B-D for nozzle-to-vessel welds, and testing category B-P for system leakage; and additional guidelines of Regulatory Guide 1.43 for control of stainless steel weld cladding of low-alloy steels. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in BWRVIP-29 and TR-103515 to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program is focused on managing the effects of stress corrosion cracking (SCC) of SS cladding on the intended function of reactor vessel nozzles. NUREG-0313 and GL 88-01, respectively, describe the technical basis and staff guidance regarding the problem of IGSCC in BWRs. ***(2) Preventive Actions:*** Selection of material considered resistant to IGSCC, e.g., grades of weld metal with a maximum carbon of 0.035% and minimum 7.5% ferrite, prevent or mitigate IGSCC, and Regulatory Guide (RG) 1.43 provides assurance that production cladding complies with ASME Section III and XI guidelines to prevent underclad cracking. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in BWRVIP-29 and TR-103515. Furthermore, hydrogen 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 nozzles by detection and sizing of cracks by inservice inspection (ISI). Inspection requirements of Table IWB 2500-1, examination category B-D specifies for all nozzles volumetric inspection of nozzle-to-vessel welds and nozzle inside radius section. Requirements for training and qualification of personnel and performance demonstration for procedures and equipment is in conformance with Appendices VII and VIII of ASME	No

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) Section XI, or any other formal program approved by the NRC. 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, Supplement 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 the boundary. ***(4) Detection of Aging Effects:*** Aging effects degradation of the reactor vessel nozzles can not occur without crack initiation; extent and schedule of inspection assure detection of cracks before the loss of intended function of the reactor vessel nozzles. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI should provide for timely detection of cracks. All BWRs are inspected in accordance with Program B IWB-2412 which requires 100% inspection every 10 y; for reactor vessel nozzles at least 25% but not more than 50% shall be examined by the end of 1st inspection interval. ***(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-3512. Planar and liner flaws are sized according to IWA-3300 and IWA-3400. Continued operation without repair requires that crack growth calculation be performed according to the guidance of GL 88-01 or other approved procedures. ***(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:*** IGSCC has occurred in small- and large-diameter BWR piping made of austenitic SSs or nickel alloys. The three elements that cause IGSCC are, a susceptible (sensitized) material, significant tensile stress, and an aggressive environment. The comprehensive AMP outlined in NUREG-0313 and GL 88-01 is based on substantial effort in research and development, and industry recommendations developed in response to NRC communications. The program addresses improvements in all three of the 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
A1.3.2 A1.3.6	Nozzles	Feedwater, CRDRL	SA508-Cl2 with or without SS cladding	Up to 288°C, Oxygenated Water	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition; NUREG-0619; GSI-190.
A1.3.8	Nozzles	LPCI (or RHR Injection Mode)	SA508-Cl2	Up to 288°C, Oxygenated 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, GL 88-11, GL 92-11, Rev. 1.
A1.4.1 thru 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, Oxygenated Water	Crack Initiation and Growth	SCC, IGSCC	NUREG-0313, Rev. 2. NRC GL 88-01. NRC GL 88-01, Sup. 1. ASME Section XI, 1989 Edition. Reg. Guide 1.45. BWRVIP-29. EPRI TR-103515. *Operating Experience* NRC IN 82-39. NRC IN 84-41.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or Section I (Power Boilers) and Section VIII, Division 1 (Unfired Pressure Vessel). The technical basis and staff guidance regarding the problem of feedwater nozzle cracking due to thermal cycling is described in NUREG-0619.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	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, implemented through Generic Letters (GLs) 88-11 and 92-01 Rev. 1 and Supplement 1, to predict effects of neutron irradiation on reactor vessel materials. In addition, inservice inspection of ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, examination category B-A of all pressure retaining welds in the vessel and repair welds in beltline region, defined as the region extending for the length of the thermal shield or effective length of reactor fuel elements. A NRC Generic Letter is being prepared regarding applications for exemptions from inspection requirements for circumferential welds.	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 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, and (c) the equivalent margins analyses performed in accordance with 10 CFR 50, Appendix G. Reactor surveillance program requires that the existing reactor vessel material surveillance program be evaluated to determine whether there is sufficient material data and dosimetry to monitor irradiation embrittlement at the end of the license renewal term and whether operating restrictions (i.e., inlet temperature, neutron spectrum and flux) are necessary. If surveillance capsules are not removed during the license renewal term it will be necessary to establish operating restrictions to ensure the plant is operated within the environment of the surveillance capsules.	Yes TLAA
Program delineated in NUREG-0313, Rev. 2 and implemented through NRC Generic letter (GL) 88-01 and its Supplement 1, and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination category B-F for pressure retaining dissimilar metal welds in vessel nozzles and testing category B-P for system leakage, and additional recommendations of Nuclear Services Information Letter (SIL) No. 455, Rev. 1 and Supplement 1. BWRVIP guideline is under staff review. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in BWRVIP-29 and TR-103515 to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program is focused on managing 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. NUREG-0313 and GL 88-01, respectively, describe the technical basis and staff guidance regarding the problem of IGSCC in BWRs. ***(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 (SI). 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 EPRI guidelines in BWRVIP-29 and TR-103515. Also, hydrogen 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	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

A1. REACTOR VESSEL (Pressurized Water Reactor)

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) function of reactor vessel nozzle safe ends by detection and sizing of cracks by inservice inspection (ISI). Inspection requirements of ASME Section XI, Subsection IWB, Table IWB 2500-1, examination category B-F 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. Only surface examination is conducted for all butt welds less than NPS 4. For all nozzle-to-safe end socket welds, surface examination is specified of OD surface extending 1 in. on the buttered side and 1/2 in. on the other. Requirements for training and qualification of personnel and performance demonstration for procedures and equipment is in conformance with Appendices VII and VIII of ASME Section XI, or any other formal program approved by the NRC. SIL No. 455 and Supplement 1 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. Visual VT-2 (IWA-5240) examination is performed for all pressure retaining components during system leakage test (IWB-5221), conducted prior to plant startup following each refueling outage, and during system hydrostatic test (IWB-5222) conducted at or near the end of each inspection interval. Leakage detection is in conformance with Position C of Regulatory Guide 1.45 and additional guidelines of GL 88-01, Supplement 1. ***(4) Detection of Aging Effects:*** Aging effects degradation of the nozzle safe ends can not occur without crack initiation; extent and schedule of inspection assure detection of cracks before the loss of intended function of the reactor vessel nozzle safe ends. ***(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, e.g., 25% are examined every 10 y, at least 12% in 6 y. Inspection extent and schedule are enhanced for welds of non-resistant materials, or welds that have been treated by stress improvement (SI) or reinforced by weld overlay. ***(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-3514. Planar and liner flaws are sized according to IWA-3300 and -3400. ***(7) Corrective Actions:*** Repair 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 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	

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, Oxygenated Water	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition ANSI B31.1 GSI-190
A1.5.1 thru 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, Oxygenated Water	Crack Initiation and Growth	SCC, IGSCC	NUREG-0313, Rev. 2. NRC GL 88-01. NRC GL 88-01, Sup. 1. ASME Section XI, 1989 Edition. BWRVIP-29. EPRI TR-103515. *Operating Experience* NRC IN 82-39. NRC IN 84-41.

Existing 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:*** 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, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or ANSI B31.1, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Program delineated in NUREG-0313, Rev. 2 and implemented through NRC Generic letter 88-01 and its Supplement 1, and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination category B-E for pressure retaining partial penetration welds and testing category B-P for system leakage. BWRVIP guideline is under staff review. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in BWRVIP-29 and TR-103515 to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** NUREG-0313 and GL 88-01, respectively, describe the technical basis and staff guidance regarding the problem of IGSCC in BWRs. The program is focused on managing 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:*** 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 EPRI guidelines in BWRVIP-29 and TR-103515. Also, hydrogen 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, Supplement 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 the 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 the 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	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A1.5.1 thru 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, Oxygenated Water	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. GSI-190.
A1.6	Bottom Head	-	SA302-Gr B, SA533-Gr B with 308, 309, 308L, 309L cladding	Up to 288°C, Oxygenated Water	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. GSI-190.
A1.7.1	Control Rod Drive (CRD) Mechanism	Housing	SS	Up to 288°C, Oxygenated Water	Crack Initiation and Growth	SCC, IGSCC	ASME Section XI, 1989 Edition. BWRVIP-29. EPRI TR-103515. BWRVIP-58.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) 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.	
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or Section I (Power Boilers) and Section VIII, Division 1 (Unfired Pressure Vessel).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or Section I (Power Boilers) and Section VIII, Division 1 (Unfired Pressure Vessel).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination categories B-O for pressure retaining welds in control rod housings and testing category B-P for system leakage. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in BWRVIP-29 and TR-103515 to minimize the potential of crack initiation and growth. BWRVIP guideline is under staff review.	***(1) Scope of Program:*** The program is focused on managing the effects of stress corrosion cracking (SCC) on the intended function of CRD mechanism housing. ***(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 EPRI guidelines in BWRVIP-29 and TR-103515. Also, hydrogen water chemistry and stringent control of	Yes, BWRVIP Guideline (Element 7)

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

A1. REACTOR VESSEL (Pressurized Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A1.7.1	CRD Mechanism	Housing	SS	Up to 288°C, Oxygenated Water	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. ANSI B31.1. GSI-190.
A1.8	Support Skirt & Attachment Welds	-	SA533-Gr B (Welds SS or Inconel 182)	Ambient Temperature Air	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) conductivity is used to inhibit IGSCC. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of IGSCC on the intended function of 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. ***(4) Detection of Aging Effects:*** Aging effects degradation of the CRD mechanism housing can not occur without crack initiation; 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 Program B IWB-2412 should provides timely detection of cracks. 10% peripheral CRD housings are examined each inspection interval. ***(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-3523. Planar and liner flaws are sized according to IWA-3300 and IWA-3400. ***(7) Corrective Actions:*** The corrective action proposed by the BWRVIP is under staff review. ***(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.	
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or other evaluations.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or other evaluations.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal.	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 Mechanism

- A2.2.1 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 Bottom Head

- A2.7.1 Dome

A2.8 Penetrations

A2.8.1 CRD Mechanism

A2.8.2 Instrumentation

A2.8.3 Leakage Monitoring Tubes

A2.9 Pressure Vessel Support

A2.9.1 Skirt Support

A2.9.2 Cantilever/Column Support

A2.9.3 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 closure head lifting lugs and core support pads, as well as pressure vessel support and attachment welds are also included in the table. 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	Closure Head	Dome (Outer Surface)	SA302-Gr B, SA533-Gr B with Stainless Steel (SS) Cladding	Air, Leaking Chemically Treated Borated Water or Steam up to 340°C	Loss of Material	Corrosion/ Boric Acid Wastage of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 Edition. Plant Technical Specifications. *Operating Experience* NRC IN 86-108 S 3.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Implementation of NRC Generic Letter 88-05, and based on plant technical specifications, inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, testing category B-P for pressure retaining Class 1 components.	***(1) Scope of Program:*** The staff guidance of Generic Letter (GL) 88-05 provides assurance that a program has been implemented consisting of systematic measures to ensure that the effects of corrosion caused by leaking coolant containing boric acid does not lead to degradation of the assurance that the reactor coolant pressure boundary will have a extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture. The program includes (a) determination of principal location, (b) examinations requirements and procedures for locating small leaks, and (c) engineering evaluations and corrective actions. ***(2) Preventive Actions:*** Minimizing reactor coolant leakage by frequent monitoring of the locations where potential leakage could occur and timely repair, prevent or mitigate boric acid corrosion. Modifications in the design or operating procedures to reduce the probability of leaks at locations where they may cause corrosion damage and use of suitable corrosion resistant materials or the application of protective coatings. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of boric acid corrosion on the intended function of the component by detection of coolant leakage by implementing the program delineated in GL 88-05 including guidelines for locating small leaks, conducting examinations, and performing engineering evaluations. Inspection requirements of ASME Section XI specify visual VT-2 (IWA-5240) examination during system leakage test (IWB-5221) and system hydrostatic test (IWB-5222) of all pressure retaining Class 1 components, according to Table IWB 2500-1 category B-P. ***(4) Detection of Aging Effects:*** Degradation of the component due to boric acid wastage can not occur without leakage of coolant containing boric acid; extent and schedule of the program delineated in GL 88-05 will assure detection of leakage before the loss of intended function of the component. ***(5) Monitoring and Trending:*** The program delineated in GL 88-05 provides for timely detection of leakage. ***(6) Acceptance Criteria:*** Any leakage is evaluated in accordance with the program proposed by GL 88-05. ***(7) Corrective Actions:*** The leakage source and areas of general corrosion are located and corrective actions are in conformance with the program proposed by GL 88-05, and the requirements of ASME Section XI, IWB-3142 and IWA-5250. Components with local areas of corrosion that reduces the wall thickness by more than 10% require analytical evaluation to demonstrate acceptability. ***(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:*** Boric acid wastage observed in nuclear power plants may be classified into two types: (a) corrosion that increases the rate of leakage, e.g., corrosion of closure bolting or fasteners, and (b) corrosion that occurs some distance from the source of leakage. Some recent incidents of boric acid wastage (IN 86-108 S3) indicate that, although	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A2.1.1	Closure Head	Dome	SA302-Gr B, SA533-Gr B with SS Cladding	Chemically Treated Borated Water up to 340°C	Crack Initiation and Growth	Stress Corrosion Cracking (SCC), Inter-granular Stress Corrosion Cracking (IGSCC)	ASME Section XI, 1989 Edition. Reg. Guide 1.43. Reg. Guide 1.44. EPRI TR-105714. *Operating Experience* NRC IN 84-18. NRC IN 84-89.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) implementation of GL 88-05 ensures timely detection of leakage, there may still be a lack of awareness of the conditions that can lead to boric acid wastage.	
Guidelines of Regulatory Guide (RG) 1.44 to avoid sensitization of stainless steels (SSs) and guidelines of RG 1.43 to prevent underclad cracking; and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination categories B-N-1 for vessel interior, B-A for head welds, and B-D for full penetration nozzle-to-head welds. Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 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 and growth of cracks into welds and base metal, on the intended function of component. However, SCC is not anticipated to be an issue for the closure head dome or nozzles because analytical evaluations indicate that cracks in the SS cladding will stop growing in the ferritic base metal. ***(2) Preventive Actions:*** Selection of material in compliance with the guidelines of Regulatory Guide (RG) 1.44 prevent or mitigate IGSCC, and RG 1.43 provides assurance that production cladding complies with ASME Section III and XI requirements to prevent underclad cracking. 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), and SCC could occur at creviced/cold worked locations (IN 84-89). 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 closure head dome by detection and sizing of cracks by ISI. Inspection requirements of Table IWB 2500-1, examination category B-N-1 specifies visual VT-3 of closure head interior surface made accessible for examination during normal refueling outages. Examination category B-A specifies volumetric inspection of all circumferential and meridional welds, and B-D specifies volumetric inspection of all nozzle-to-vessel welds and nozzle inside radius section. ***(4) Detection of Aging Effects:*** Aging degradation of the closure head dome 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 closure head. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI should provide for timely detection of cracks. Closure head dome interior is inspected at 1st refueling outage and subsequent outages at approximately 3 y intervals. Accessible length of all head welds and nozzle-to-head welds are examined each inspection interval. ***(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 3520 for visual examination, IWB-3510 for head welds, and IWB-3512 for full penetration nozzle welds. Visual examinations that reveal relevant conditions may be supplemented by surface and volumetric examinations (IWB-3200) for flaw characterization, analytical evaluation, corrective measures, and repairs. Continued service without repair requires analytical evaluation to demonstrate acceptability. ***(7) Corrective Actions:*** Repair	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A2.1.2	Closure Head	Flange (Outer Surface)	SA336, SA508 with SS Cladding	Air, Leaking Chemically Treated Borated Water or Steam up to 340°C	Loss of Material	Corrosion/ Boric Acid Wastage of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 Edition. Plant Technical Specifications.
A2.1.3	Closure Head	Stud Assembly	SA540-B23 & -B24, SA320-L43 Alloy 4340)	Air, Leaking Chemically Treated Borated Water and/or Steam up to 340°C	Loss of Material	Corrosion/ Boric Acid Wastage of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 Edition. Plant Technical Specifications.
A2.1.3	Closure Head	Stud Assembly	SA540-B23 & -B24, SA320-L43 (Alloy 4340)	Air, Leaking Chemically Treated Borated Water or Steam up to 340°C	Crack Initiation and Growth	SCC, IGSCC	ASME Section XI, 1989 Edition. Reg. Guide 1.65 ASME Section III, 1989 Edition.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) is 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:*** SCC has not been an issue for PWR closure head dome.	
Implementation of NRC Generic Letter 88-05, and based on plant technical specifications, inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, testing category B-P for pressure retaining Class 1 components.	*Same as effect of Corrosion/Boric Acid Wastage of external surfaces of Item A2.1.1 Closure Head Dome.*	No
Implementation of NRC Generic Letter 88-05, and based on plant technical specifications, inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, testing category B-P for pressure retaining Class 1 components.	*Same as effect of Corrosion/Boric Acid Wastage of external surfaces of Item A2.1.1 Closure Head Dome.*	No
Inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination category B-G-1, and testing category B-P for system leakage. Prevention and replacement in accordance with Regulatory Guide 1.65.	***(1) Scope of Program:*** The program is focused on managing the effects of IGSCC on the intended function of closure head stud assemblies. ***(2) Preventive Actions:*** Design requirements of ASME Section III, Subsection NB, and additional guidance of Regulatory Guide (RG) 1.65 on material selection, preservice inspection, 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 prevent cracking 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. Preservice inspection in conformance with NB-2580 of Section III of the Code requires ultrasonic examination of stud bolting over the entire surface prior to threading. 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 monitors the effects of IGSCC on the intended function of closure stud assembly by detection and sizing of cracks by inservice inspection (ISI). Inspection 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	No

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	Air, Leaking Chemically Treated Borated Water or Steam up to 340°C	Attrition	Wear	ASME Section XI, 1989 Edition.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) visual VT-1 examination of surfaces of nuts, washers, and bushings. Requirements for training and qualification of personnel and performance demonstration for procedures and equipment are in conformance with Appendices VII and VIII of ASME Section XI, or any other formal program approved by the NRC. Testing category B-P specify visual VT-2 (IWA-5240) examination of all pressure retaining components extending to and including the second closed valve at the boundary extremity, 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 ASME Section XI is effective and adequate for timely detection of cracks. Inspection schedule in accordance with IWB-2400 requires 100% inspection every 10 y. 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-3515 and 3517. ***(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 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 (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination category B-G-1, and testing category B-P for system leakage. Prevention and replacement in accordance with Regulatory Guide 1.65.	***(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, 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. Requirements for training and qualification of personnel and performance demonstration for procedures	No

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	Air, Leaking Chemically Treated Borated Water or Steam up to 340°C	Cumulative Fatigue Damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition.
A2.1.5	Closure Head	Vessel Flange Leak Detection Line	Stainless Steel	Leaking Chemically Treated Borated Water or Steam up to 340°C	Crack Initiation and Growth	SCC, IGSCC	Plant Technical Specifications.
A2.2.1	Control Rod Drive (CRD) Mechanism	Pressure Housing	SA182, Types 304 or 316, CF-8	Chemically Treated Borated Water up to 340°C	Crack Initiation and Growth	SCC, IGSCC	ASME Section XI, 1989 Edition. Reg. Guide 1.44. EPRI TR-105714. *Operating Experience* NRC IN 84-18. NRC IN 84-89.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) and equipment are in conformance with Appendices VII and VIII of ASME Section XI, or any other formal program approved by the NRC. Testing category B-P specify visual VT-2 (IWA-5240) examination of all pressure retaining components extending to and including the second closed valve at the boundary extremity, 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 attrition, 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. Inspection schedule in accordance with IWB-2400 requires 100% inspection every 10 y. 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-3515 and 3517. ***(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 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, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	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, no AMP
Inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination categories B-O for pressure retaining welds in control rod housing and testing category B-P for system leakage. Primary water chemistry is monitored and maintained in accordance with	***(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:*** Selection of material in compliance with the guidelines of Regulatory Guide (RG) 1.44 prevent or mitigate IGSCC. Control of halogens, sulfates, and oxygen in the primary water to less than 0.05, 0.05, and 0.005 ppm, respectively, during operation.	No

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
(continued from previous page) EPRI guidelines in TR-105714 to minimize the potential of crack initiation and growth.	*(continued from previous page)* 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 IGSCC 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. Requirements for training and qualification of personnel and performance demonstration for procedures and equipment are in conformance with Appendices VII and VIII of ASME Section XI, or any other formal program approved by the NRC. 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 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-2412 should provide timely detection of cracks. 10% peripheral CRD housings are examined each inspection interval. 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 IGSCC degradation is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3523. 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 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, the potential of SCC exists from inadvertent introduction of contaminants into the primary coolant system (IN 84–18). Also, SCC could occur at creviced/cold worked locations (IN 84-89).	

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
A2.2.1	Control Rod Drive (CRD) Mechanism	Pressure Housing	SB-166	Chemically Treated Borated Water up to 340°C	Crack Initiation and Growth	SCC, IGSCC	ASME Section XI, 1989 Edition. NRC GL 97-01. EPRI TR-105714. *Operating Experience* NRC IN 90-10. NRC IN 96-11.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination categories B-O for pressure retaining welds in control rod housing, B-E for pressure retaining partial penetration welds, B-F for pressure retaining dissimilar metal welds, B-N-2 for integrally-welded core support structure, and testing category B-P for system leakage. Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 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 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 review previous evaluations performed in response to Generic Letter (GL) 97-01, as updated with the most current industry susceptibility model and inspection results. ***(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 SCC has not been addressed adequately, particularly when demineralizer resins contaminate the reactor coolant system. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of IGSCC on the intended function of components by detection (and sizing) of cracks and coolant leakage by ISI. 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. B-E specifies visual VT-2 examination of partial penetration welds during the hydrostatic test. B-F specifies for all nozzle-to-safe end butt welds NPS 4 or larger, volumetric and 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. Category B-N-2 specifies visual VT-3 of accessible welds of core support pads. Training and qualification of personnel and performance demonstration for procedures and equipment are in conformance with Appendices VII and VIII of ASME Section XI, or any NRC approved program. 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). 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 degradation of the CRD mechanism housing can not occur without crack initiation and growth. Based on GL 97-01, the applicant should review whether 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 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. The applicant should either provide the technical basis that justifies the	Yes, Elements 2 to 6 should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A2.2.1	Control Rod Drive (CRD) Mechanism	Pressure Housing	SA182, Types 304 or 316, CF-8, SB-166	Chemically Treated Borated Water up to 340°C	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. GSI-190.
A2.2.1	Control Rod Drive (CRD) Mechanism	Pressure Housing	CF-8	Chemically Treated Borated Water up to 340°C	Loss of Fracture Toughness	Thermal Aging Embrittle-ment	EPRI TR-106092. ASME Section XI, 1989 Edition.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) 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 are based on the finding of the initial inspections and crack growth rate models for Ni alloys. ***(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-3523 or IWB-3522. 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 (GL 97-01). ***(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:*** 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.	
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or Section I (Power Boilers) and Section VIII, Division 1 (Unfired Pressure Vessel).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
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.	***(1) Scope of Program:*** The program includes determination of the susceptibility of CASS components to thermal aging based on casting method, Mo content, and percent ferrite, and for "potentially susceptible" components, to account for the synergistic loss of fracture toughness due to neutron embrittlement and thermal aging embrittlement, implement either a supplemental examination of the affected components as part of a 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. ***(2) Preventive Actions:*** The program provides no guidance on methods to mitigate thermal aging. ***(3) Parameters Monitored/ Inspected:*** The program specifics depend on the neutron fluence and ferrite content of the component. The criteria in EPRI TR-106092, with some modifications approved by the NRC staff, can be used to determine whether CASS components are potentially susceptibility to thermal aging embrittlement based on the casting method, Mo content, and ferrite content. For low-Mo content (0.5 wt.% max.) steels, only static-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement, all centrifugal-cast	No

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) and static-cast steels with ≤20% ferrite are not susceptible. For high-Mo content (2.0 to 3.0 wt.%) steels, static-cast steels with >14% ferrite and centrifugal-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement, static-cast steels with ≤14% ferrite and centrifugal-cast steels with ≤20% ferrite are not susceptible. Ferrite content will be calculated by the Hull's equivalent factors or a method producing an equivalent level of accuracy (±6% deviation between measured and calculated values). ***(4) Detection of Aging Effects:*** For all CASS components that have a neutron fluence greater than 10^{17} n/cm^2 (E>1 MeV), implement a 10-year ISI program during the renewal period that includes supplemental inspection that covers portions of the susceptible components determined to be limiting from the standpoint of thermal aging susceptibility (Ferrite and Mo contents, casting process, and operating temperature), neutron fluence, and cracking susceptibility (applied stress, operating temperature, and environmental conditions). The inspection technique, including the reliability in detecting the features of interest (crack appearance and size) in assuring the integrity of the component, should be specified. For example, enhancement of the visual VT-1 examination to achieve a 1/2-mil (0.0005 in.) resolution, with the conditions (lighting and surface cleanness) for the ISI bounded by those used to demonstrate the resolution of the inspection technique. Alternatively, 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. If the loading is compressive or low enough (<5 ksi) to preclude fracture, then supplemental inspection of the component is not required. Failure to meet this criteria requires continued use of supplemental inspection program. For all CASS components that have a neutron fluence of less than 10^{17} n/cm^2 (E>1 MeV), implement the supplement inspection program if they are "potentially susceptible" to thermal aging; the existing ASME Section XI inspection requirements are considered adequate if the components are "not susceptible" to thermal aging. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 should provide timely detection of cracks. ***(6) Acceptance Criteria:*** Flaws detected in CASS components are evaluated in accordance with the applicable procedures of IWB-3500. ***(7) Corrective Actions:*** Repair is in conformance with IWA-4000 and IWB-4000, and replacement according to IWA-7000 and IWB-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:*** The AMP based on susceptibility determination, neutron fluence level, and supplemental inspection is effective in managing the effects of synergistic loss of fracture	

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A2.3.1 thru A2.3.3	Nozzles	Inlet, Outlet, Safety Injection	SA336, SA508 with SS Cladding	Chemically Treated Borated Water up to 340°C	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 92-11, Rev. 1. ASME Section XI, 1989 Edition.
A2.3.1 thru A2.3.3	Nozzles	Inlet, Outlet, Safety Injection	SA336, SA508 with SS Cladding	Chemically Treated Borated Water up to 340°C	Crack Initiation and Growth	SCC, IGSCC	ASME Section XI, 1989 Edition. Reg. Guide 1.43. Reg. Guide 1.44. EPRI TR-105714.
A2.3.1 thru A2.3.3	Nozzles	Inlet, Outlet, Safety Injection	SA336, SA508 with SS Cladding	Chemically Treated Borated Water up to 340°C	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. GSI-190.
A2.4.1 thru 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	Crack Initiation and Growth	SCC, IGSCC	ASME Section XI, 1989 Edition. NRC GL 97-01. EPRI TR-105714.
A2.4.1 thru A2.4.3	Nozzle Safe Ends	Inlet, Outlet, Safety Injection	SS, SB-166 (Alloy 182 butter, and Alloy 82 or 182 weld)	Chemically Treated Borated Water up to 340°C	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition ANSI B31.1 GSI-190

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) toughness due to neutron and thermal aging embrittlement on the intended function of CASS components:	
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, implemented through NRC Generic Letters (GLs) 88-11 and 92-01 Rev. 1 and Supplement 1, to predict effects of neutron irradiation on reactor vessel materials. In addition, inservice inspection of ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, examination categories B-D for nozzle-to-vessel welds, and testing category B-P for system leakage.	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 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, and (c) the equivalent margins analyses performed in accordance with 10 CFR 50, Appendix G. Reactor surveillance program requires that the existing reactor vessel material surveillance program be evaluated to determine whether there is sufficient material data and dosimetry to monitor irradiation embrittlement at the end of the license renewal term and whether operating restrictions (i.e., inlet temperature, neutron spectrum and flux) are necessary. If surveillance capsules are not removed during the license renewal term, it will be necessary to establish operating restrictions to ensure the plant is operated within the environment of the surveillance capsules.	Yes TLAA
Same as for the effect of Stress Corrosion Cracking of Item A2.1.1 closure head dome.	*Same as for the effect of Stress Corrosion Cracking of Item A2.1.1 closure head dome.*	No
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or Section I (Power Boilers) and Section VIII, Division 1 (Unfired Pressure Vessel).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Same as for the effect of Stress Corrosion Cracking of Item A2.2.1 CRD mechanism pressure housing constructed of Ni-alloy.	*Same as for the effect of Stress Corrosion Cracking of Item A2.2.1 CRD mechanism pressure housing constructed of Ni-alloy.*	Yes Elements 2 to 6 should be further evaluated
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or ANSI B31.1, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA

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	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 92-11, Rev. 1. ASME Section XI, 1989 Edition. NRC IN 98-42.
A2.5.1 thru 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	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. GSI-190.
A2.5.3	Vessel Shell	Vessel Flange	SA336, SA508 with SS Cladding	Air, Leaking Chemically Treated Borated Water and/or Steam up to 340°C	Loss of Material	Corrosion/ Boric Acid Wastage of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 Edition.
A2.6	Core Support Pads (Lugs)	-	SB-166, SB-168	Chemically Treated Borated Water up to 340°C	Crack Initiation and Growth	SCC, IGSCC	ASME Section XI, 1989 Edition. NRC GL 97-01. EPRI TR-105714.
A2.6	Core Support Pads (Lugs)	-	SB-166, SB-168	Chemically Treated Borated Water up to 340°C	Attrition	Wear	ASME Section XI, 1989 Edition.

Existing 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, implemented through NRC Generic Letters (GLs) 88-11 and 92-01 Rev. 1 and Supplement 1, to predict effects of neutron irradiation on reactor vessel materials. In addition, inservice inspection of ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, examination category B-A of all pressure retaining welds in the vessel and repair welds in beltline region, defined as the region extending for the length of the thermal shield or effective length of reactor fuel elements. Based on NRC Information Notice 98-42, relief from inspection is in accordance with 10 CFR 50.55a (g) (5) (iii).	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 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, and (c) the equivalent margins analyses performed in accordance with 10 CFR 50, Appendix G. Reactor surveillance program requires that the existing reactor vessel material surveillance program be evaluated to determine whether there is sufficient material data and dosimetry to monitor irradiation embrittlement at the end of the license renewal term and whether operating restrictions (i.e., inlet temperature, neutron spectrum and flux) are necessary. If surveillance capsules are not removed during the license renewal term, it will be necessary to establish operating restrictions to ensure the plant is operated within the environment of the surveillance capsules.	Yes TLAA
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or Section I (Power Boilers) and Section VIII, Division 1 (Unfired Pressure Vessel).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Implementation of NRC Generic Letter 88-05, and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, testing category B-P for system leakage.	*Same as Corrosion/Boric Acid Wastage of external surfaces of Item A2.1.1 Closure Head Dome.*	No
Same as for the effect of Stress Corrosion Cracking of Item A2.2.1 CRD mechanism pressure housing constructed of Ni-alloy	*Same as for the effect of Stress Corrosion Cracking of Item A2.2.1 CRD mechanism pressure housing constructed of Ni-alloy.*	Yes Elements 2 to 6 should be further evaluated
Inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination category B-N-2 for integrally-welded core support structures.	***(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 attrition by inservice inspection (ISI). Inspection requirements of Table IWB 2500-1, examination category B-N-2 specifies 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	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A2.7.1	Bottom Head	Dome	SA302-Gr B, SA533-Gr B with SS Cladding	Chemically Treated Borated Water up to 340°C	Crack Initiation and Growth	SCC, IGSCC	ASME Section XI, 1989 Edition. Reg. Guide 1.43. Reg. Guide 1.44. EPRI TR-105714.
A2.8.1	Penetrations	CRD Mechanism	SB-166 Lower Housing Tube with SA182, Types 304 or 316 Upper Housing Flanges	Chemically Treated Borated Water up to 340°C	Crack Initiation and Growth	SCC, IGSCC	ASME Section XI, 1989 Edition. NRC GL 97-01. EPRI TR-105714.
A2.8.2	Penetrations	Instrumentation Tubes	SB-166, SB-167	Chemically Treated Borated Water up to 340°C	Crack Initiation and Growth	SCC, IGSCC	ASME Section XI, 1989 Edition. NRC GL 97-01. EPRI TR-105714.
A2.8.3	Penetrations	Leakage Monitoring Tubes	SB-166, SB-167 with SA312, Type 316 Extensions	Chemically Treated Borated Water up to 340°C	Crack Initiation and Growth	SCC, IGSCC	ASME Section XI, 1989 Edition. NRC GL 97-01. EPRI TR-105714.
A2.8.1, A2.8.2	Penetrations	CRD Mechanism, Instrumentation Tubes	SB-166, SB-167	Chemically Treated Borated Water up to 340°C	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. GSI-190.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	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 3520.2 for visual examination. 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 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 wear to maintain the intended function of core support pads.	
Same as for the effect of Stress Corrosion Cracking of Item A2.1.1 closure head dome.	*Same as for the effect of Stress Corrosion Cracking of Item A2.1.1 closure head dome.*	No
Same as for the effect of Stress Corrosion Cracking of Item A2.2.1 CRD mechanism pressure housing constructed of Ni-alloy.	*Same as for the effect of Stress Corrosion Cracking of Item A2.2.1 CRD mechanism pressure housing constructed of Ni-alloy.*	Yes Elements 2 to 6 should be further evaluated
Same as for the effect of Stress Corrosion Cracking of Item A2.2.1 CRD mechanism pressure housing constructed of Ni-alloy.	*Same as for the effect of Stress Corrosion Cracking of Item A2.2.1 CRD mechanism pressure housing constructed of Ni-alloy.*	Yes Elements 2 to 6 should be further evaluated
Same as for the effect of Stress Corrosion Cracking of Item A2.2.1 CRD mechanism pressure housing constructed of Ni-alloy.	*Same as for the effect of Stress Corrosion Cracking of Item A2.2.1 CRD mechanism pressure housing constructed of Ni-alloy.*	Yes Elements 2 to 6 should be further evaluated
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or Section I (Power Boilers) and Section VIII, Division 1 (Unfired Pressure Vessel).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	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
A2.9.1	Pressure Vessel Support	Skirt Support	SA302-Gr B, SA533-Gr B, SA516-Gr70, SA-36	Air	Cumulative Fatigue Damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition.
A2.9.2, A2.9.3	Pressure Vessel Support	Cantilever/ Column Support, Neutron Shield Tank,	SA302-Gr B, SA533-Gr B, SA516-Gr70, SA-36	Air	Loss of Fracture Toughness	Neutron Irradiation Embrittle-ment	Reg. Guide 1.99, Revision 2. NUREG-0933.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal.	Yes TLAA
Components have been designed or evaluated for structural integrity for a 40 y design life, according to the methodology of Regulatory Guide 1.99, Revision 2, or an equivalent method, to predict effects of neutron irradiation on pressure vessel support.	Based on NUREG-0933 neutron irradiation embrittlement of pressure vessel support is not a concern for the extended period of license renewal.	No

B1. Reactor Vessel Internals (Boiling Water Reactor)

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

- B1.1.1 Core Shroud Head Bolts
- B1.1.2 Core Shroud (Upper, Central, Lower)
- B1.1.3 Core Plate
- B1.1.4 Core Plate Bolts
- B1.1.5 Access Hole Cover
- B1.1.6 Shroud Support Structure
- B1.1.7 Standby Liquid Control Line
- B1.1.8 LPCI Coupling

B1.2 Top Guide

B1.3 Feedwater Spargers

- B1.3.1 Thermal Sleeve
- B1.3.2 Distribution Header
- B1.3.3 Discharge Nozzles

B1.4 Core Spray Lines and Spargers

- B1.4.1 Core Spray Lines (Headers)
- B1.4.2 Spray Ring
- B1.4.3 Spray Nozzles
- B1.4.4 Thermal Sleeve

B1.5 Jet Pump Assemblies

- B1.5.1 Thermal Sleeve
- B1.5.2 Inlet Header
- B1.5.3 Riser Brace Arm

B1.5.4 Holddown Beams

B1.5.5 Inlet Elbow

B1.5.6 Mixing Assembly

B1.5.7 Diffuser

B1.5.8 Castings

B1.6 Fuel Supports & CRD Assemblies

B1.6.1 Orificed Fuel Support

B1.7 Instrument Housings

B1.7.1 Intermediate Range Monitor (IRM) Dry Tubes

B1.7.2 Low Power Range Monitor (LPRM) Dry Tubes

B1.7.3 Source Range Monitor (SRM) Dry Tubes

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. 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).

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 Head Bolts	Alloy 600, Stainless Steel (SS)	288°C, High-Purity Water	Crack Initiation and Growth	Stress Corrosion Cracking (SCC)	ASME Section XI, 1989 Edition. GE SIL 433. BWRVIP-29. EPRI TR-103515. BWRVIP-07. BWRVIP-63. *Supporting BWRVIP:* BWRVIP-03. BWRVIP-06. BWRVIP-14. BWRVIP-44. BWRVIP-45. BWRVIP-59. BWRVIP-60. BWRVIP-62.
B1.1.1	Core Shroud, Shroud Head and Core Plate	Core Shroud Head Bolts	Alloy 600, SS	288°C, High-Purity Water	Cumulative Fatigue Damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B1.1.2	Core Shroud, Shroud Head and Core Plate	Core Shroud (Upper, Central, Lower)	SS	288°C, High-Purity Water	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. BWRVIP-29. EPRI TR-103515. BWRVIP-07. BWRVIP-63. *Operating Experience* NRC IN 94-42. NRC IN 97-17. NRC GL 94-03. NUREG-1544.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Visual inspection is performed according to ASME Section XI, IWB-2500, category B-N-2, and GE Services Information Letter (SIL) 433 recommends ultrasonic (UT) inspection during outages, verification of required torque on bolt during shroud head removal and attachment, and replacement of bolts with crevice design by a design which is crevice-free. BWRVIP guideline is under staff review. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-103515 and BWRVIP-29 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 $\mu S/cm^2$) 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 appear to increase the effectiveness of hydrogen additions in the core region, but only limited data are available to demonstrate their effectiveness. GE Services Information Letter (SIL) 433 recommends replacement of bolts with crevice-free design. ***(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, inspection schedule assures detection of cracks before the loss of intended function of the component. ***(5) Monitoring and Trending:*** 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 under staff review. ***(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 SCC on the intended function of core shroud head bolts.	Yes, BWRVIP Guideline
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes, TLAA
Visual inspection (VT-3) is performed according to ASME Section XI, IWB-2500, category B-N-2. Guidance for enhanced VT-1 and UT inspections in plant specific programs. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-103515 and BWRVIP-29 to minimize the potential of crack initiation and growth. Plant programs also may include water chemistry measures such as strict controls on conductivity, hydrogen addition, and use of noble metal additions such as palladium or	***(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 appear to increase the effectiveness of hydrogen additions in the core region, but only limited data are available	Yes, BWRVIP Guideline

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
							(continued from previous page) *Supporting BWRVIP:* BWRVIP-03. BWRVIP-06. BWRVIP-14. BWRVIP-44. BWRVIP-45. BWRVIP-59. BWRVIP-60. BWRVIP-62.
B1.1.2	Core Shroud, Shroud Head and Core Plate	Core Shroud (Upper, Central, Lower)	SS	288°C, High-Purity Water	Crack Initiation and Growth	Irradiation Assisted Stress corrosion cracking (IASCC)	*Same as for the effect of Stress Corrosion Cracking on Item B1.1.2 core shroud.*
B1.1.3, B1.1.4	Core Shroud, Shroud Head and Core Plate	Core Plate, Core Plate Bolts (used in early BWRs)	SS	288°C, High-Purity Water	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. BWRVIP-29. EPRI TR-103515. BWRVIP-25. BWRVIP-50. *Supporting BWRVIP:* BWRVIP-03. BWRVIP-06. BWRVIP-14. BWRVIP-44. BWRVIP-45. BWRVIP-59. BWRVIP-60. BWRVIP-62.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
(continued from previous page) platinum to reduce electrochemical potential. Either preventive or restorative mechanical repairs may be made to the shroud. Possible inspection relief based on hydrogen injection is currently under staff review. BWRVIP guideline is under staff review.	*(continued from previous page)* at present to demonstrate their effectiveness. ***(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. Extensive cracking has been observed at both horizontal [NRC Generic Letter (GL) 94–03] and vertical [NRC Information Notice (IN) 97–17] welds. ***(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 under staff review. ***(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 BWRs. It has affected shrouds fabricated from Type 304 SS 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.	
Same as for the effect of Stress Corrosion Cracking on Item B1.1.2 core shroud.	*Same as for the effect of Stress Corrosion Cracking on Item B1.1.2 core shroud.*	Yes, BWRVIP Guideline
Visual inspection (VT–3) is performed according to ASME Section XI, IWB-2500, category B–N–2. Guidance for enhanced VT–1 and UT inspections in plant specific programs. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-103515 and BWRVIP-29 to minimize the potential of crack initiation and growth. Plant programs also may include water chemistry measures such as strict controls on conductivity, hydrogen addition, and use of noble metal additions such as palladium or platinum to reduce electrochemical potential. Possible inspection relief based on hydrogen injection is currently under staff review. BWRVIP guideline is under staff review.	***(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 appear to increase the effectiveness of hydrogen additions in the core region, but only limited data are available at present to demonstrate their effectiveness. ***(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. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with	Yes, BWRVIP Guideline

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
							(continued from previous page) *Operating Experience* NRC GL 94–03. NRC IN 95–17. NUREG–1544.
B1.1.3, B1.1.4	Core Shroud, Shroud Head and Core Plate	Core Plate, Core Plate Bolts (used in early BWRs)	SS	288°C, High-Purity Water	Crack Initiation and Growth	IASCC	*Same as for the effect of Stress Corrosion Cracking on Item B1.1.3 core plate.*
B1.1.3	Core Shroud, Shroud Head and Core Plate	Core Plate	SS	288°C, High-Purity Water	Cumulative Fatigue Damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B1.1.5	Core Shroud, Shroud Head and Core Plate	Access Hole Cover	Alloy 600, Alloy 82 & 182 welds	288°C, High-Purity Water	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. GE SIL 462 Sup. 3. BWRVIP-29. EPRI TR-103515. *Operating Experience* NRC IN 88-03. NRC IN 92-57.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) 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 under staff review. ***(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.	
Same as for the effect of Stress Corrosion Cracking on Item B1.1.3 core plate.	*Same as for the effect of Stress Corrosion Cracking on Item B1.1.3 core plate.*	Yes, BWRVIP Guideline
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	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 ultrasonic inspection techniques. Implementation of inspection program is plant specific. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-103515 and BWRVIP-29 to minimize the potential of crack initiation and growth. Plant programs also may include water chemistry measures such as strict controls on conductivity, hydrogen addition, and use of noble metal additions such as palladium or platinum to reduce electrochemical potential.	***(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 appear to increase the effectiveness of hydrogen additions in the core region, but only limited data are available at present to demonstrate their effectiveness. 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 ultrasonic techniques for such inspections. ***(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 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	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	Access Hole Cover	Alloy 600, Alloy 82 & 182 welds	288°C, High-Purity Water	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. GE SIL 462 Sup. 3. BWRVIP-29. EPRI TR-103515.
B1.1.6	Core Shroud, Shroud Head and Core Plate	Shroud Support Structure (Shroud Support Cylinder, Shroud Support Plate, Shroud Support Legs)	Alloy 600, Alloy 82 & 182 welds	288°C, High-Purity Water	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. GE SIL 462 Sup. 3. BWRVIP-29. EPRI TR-103515. BWRVIP-38. BWRVIP-52. *Supporting BWRVIP:* BWRVIP-03. BWRVIP-06. BWRVIP-14. BWRVIP-44. BWRVIP-45. BWRVIP-59. BWRVIP-60. BWRVIP-62. *Operating Experience* NRC IN 88-03. NRC IN 92-57.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) 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 boiling water reactors (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.	
Same as for the effect of Stress Corrosion Cracking on Item B1.1.5 access hole cover.	*Same as for the effect of Stress Corrosion Cracking on Item B1.1.5 access hole cover.*	No
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 ultrasonic inspection techniques. Implementation of inspection program is plant specific. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-103515 and BWRVIP-29 to minimize the potential of crack initiation and growth. Plant programs also may include water chemistry measures such as strict controls on conductivity, hydrogen addition, and use of noble metal additions such as palladium or platinum to reduce electrochemical potential. BWRVIP guideline is under staff review.	***(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 appear to increase the effectiveness of hydrogen additions in the core region, but only limited data are available at present to demonstrate their effectiveness. 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. ***(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 under staff review. ***(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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B1.1.6	Core Shroud, Shroud Head and Core Plate	Shroud Support Structure	Alloy 600, Alloy 82 & 182 welds	288°C, High-Purity Water	Crack Initiation and Growth	IASCC	*Same as for the effect of Stress Corrosion Cracking on Item B1.1.6 shroud support structure.*
B1.1.7	Core Shroud, Shroud Head and Core Plate	Standby Liquid Control Line	SS	288°C, High-Purity Water	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. BWRVIP-29. EPRI TR-103515. BWRVIP-27. BWRVIP-53. *Supporting BWRVIP:* BWRVIP-03. BWRVIP-06. BWRVIP-14. BWRVIP-44. BWRVIP-45. BWRVIP-59. BWRVIP-60. BWRVIP-62.
B1.1.7	Core Shroud, Shroud Head and Core Plate	Standby Liquid Control Line	SS	288°C, High-Purity Water	Crack Initiation and Growth	IASCC	*Same as for the effect of Stress Corrosion Cracking on Item B1.1.7 standby liquid control line.*
B1.1.8	Core Shroud, Shroud Head and Core Plate	LPCI Coupling	SS	288°C, High-Purity Water	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. BWRVIP-29. EPRI TR-103515. BWRVIP-42. BWRVIP-56. *Supporting BWRVIP:* BWRVIP-03.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Same as for the effect of Stress Corrosion Cracking on Item B1.1.6 shroud support structure.	*Same as for the effect of Stress Corrosion Cracking on Item B1.1.6 shroud support structure.*	Yes, BWRVIP Guideline
Visual inspection (VT-3) is performed according to ASME Section XI, IWB-2500, category B-N-2. Guidance for enhanced VT-1 inspections and UT inspections in plant specific programs. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-103515 and BWRVIP-29 to minimize the potential of crack initiation and growth. Plant programs also may include water chemistry measures such as strict controls on conductivity, hydrogen addition, and use of noble metal additions to reduce electrochemical potential. BWRVIP guideline is under staff review.	***(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 appear to increase the effectiveness of hydrogen additions in the core region, but only limited data are available at present to demonstrate their effectiveness. ***(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. ***(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 under staff review. ***(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.	Yes, BWRVIP Guideline
Same as for the effect of Stress Corrosion Cracking on Item B1.1.7 standby liquid control line.	*Same as for the effect of Stress Corrosion Cracking on Item B1.1.7 standby liquid control line.*	Yes, BWRVIP Guideline
Visual inspection (VT-3) is performed according to ASME Section XI, IWB-2500, category B-N-2. Guidance for enhanced VT-1 inspections and UT inspections in plant specific programs. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-103515 and BWRVIP-29 to minimize the potential of crack initiation and growth. Plant	***(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.	Yes, BWRVIP Guideline

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
							(continued from previous page) BWRVIP-06. BWRVIP-14. BWRVIP-44. BWRVIP-45. BWRVIP-59. BWRVIP-60. BWRVIP-62.
B1.1.8	Core Shroud, Shroud Head and Core Plate	LPCI Coupling	SS	288°C, High-Purity Water	Crack Initiation and Growth	IASCC	*Same as for the effect of Stress Corrosion Cracking on Item B1.1.8 LPCI coupling.*
B1.2	Top Guide	Top Guide	SS	288°C, High-Purity Water	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. BWRVIP-29. EPRI TR-103515. BWRVIP-26. BWRVIP-50. *Supporting BWRVIP:* BWRVIP-03. BWRVIP-06. BWRVIP-14. BWRVIP-44. BWRVIP-45. BWRVIP-59. BWRVIP-60. BWRVIP-62. *Operating Experience* NRC GL 94-03. NRC IN 95-17. NUREG-1544

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
(continued from previous page) programs also may include water chemistry measures such as strict controls on conductivity, hydrogen addition, and use of noble metal additions to reduce electrochemical potential. BWRVIP guideline is under staff review.	*(continued from previous page)* Noble metal additions through a catalytic action appear to increase the effectiveness of hydrogen additions in the core region, but only limited data are available at present to demonstrate their effectiveness. ***(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. ***(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 under staff review. ***(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.	
Same as for the effect of Stress Corrosion Cracking on Item B1.1.8 LPCI coupling.	*Same as for the effect of Stress Corrosion Cracking on Item B1.1.8 LPCI coupling.*	Yes, BWRVIP Guideline
Visual inspection (VT-3) is performed according to ASME Section XI, IWB-2500, category B-N-2. Guidance for enhanced VT-1 inspections and UT inspections in plant specific programs. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-103515 and BWRVIP-29 to minimize the potential of crack initiation and growth. Plant programs also may include water chemistry measures such as strict controls on conductivity, hydrogen addition, and use of noble metal additions such as palladium or platinum to reduce electrochemical potential. BWRVIP guideline is under staff review.	***(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 appear to increase the effectiveness of hydrogen additions in the core region, but only limited data are available at present to demonstrate their effectiveness. ***(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. ***(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 under staff	Yes, BWRVIP Guideline

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B1.2	Top Guide	Top Guide	SS	288°C, High-Purity Water	Cumulative Fatigue Damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B1.2	Top Guide	Top Guide	SS	288°C, High-Purity Water	Crack Initiation and Growth	IASCC	*Same as for the effect of Stress Corrosion Cracking on Item B1.2 top guide.*
B1.3.1 thru B1.3.3	Feedwater Spargers	Thermal Sleeve, Distribution Header, Discharge Nozzles	SS	288°C, High-Purity Water	Crack Initiation and Growth	SCC	NUREG-0619. GE NEDE-21821-A. BWRVIP-29. EPRI TR-103515.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) review. ***(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.	
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Same as for the effect of Stress Corrosion Cracking on Item B1.2 top guide.	*Same as for the effect of Stress Corrosion Cracking on Item B1.2 top guide.*	Yes BWRVIP Guideline
Implementation of the program delineated in NUREG-0619 including inservice inspection (ISI) requirements (ultrasonic, visual and dye penetrant inspections) which depend upon specific plant design and other plant actions (monitoring, etc.). An update to NUREG-0619 with qualified UT inspection methods has been approved by the NRC staff. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-103515 and BWRVIP-29 to minimize the potential of crack initiation and growth. Plant programs also may include water chemistry measures such as strict controls on conductivity, hydrogen addition, and use of noble metal additions such as palladium or platinum to reduce electrochemical potential.	***(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. Design features aimed at mitigating thermal fatigue cracking, which has been the primary source of degradation for these components, have been implemented as per NUREG-0619. ***(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). ***(4) Detection of Aging Effects:*** Degradation due to SCC can not occur without crack initiation and growth. An update to NUREG-0619 with qualified UT inspection methods has been approved by the NRC staff. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with NUREG-0619 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:*** NUREG-0619 summarizes work performed by the NRC to resolve Generic Technical Activity A-10, "BWR Nozzle Cracking" and the industry	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B1.3.1 thru B1.3.3	Feedwater Spargers	Thermal Sleeve, Distribution Header, Discharge Nozzles	SS	288°C, High-Purity Water	Crack Initiation and Growth	IASCC	NUREG-0619. GE NEDE-21821-A. BWRVIP-29. EPRI TR-103515.
B1.3.1 thru B1.3.3	Feedwater Spargers	Thermal Sleeve, Distribution Header, Discharge Nozzles	SS	288°C, High-Purity Water	Cumulative Fatigue Damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B1.4.1 thru B1.4.4	Core Spray Lines and Spargers	Core Spray Lines (Headers), Spray Rings, Spray Nozzles, Thermal Sleeves	SS	288°C, High-Purity Water	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. NRC IEB 80-13. GE SIL 289. BWRVIP-29. EPRI TR-103515. BWRVIP-18. BWRVIP-16. BWRVIP-19. *Supporting BWRVIP:* BWRVIP-03. BWRVIP-06. BWRVIP-14. BWRVIP-44. BWRVIP-45. BWRVIP-59. BWRVIP-60. BWRVIP-62.
B1.4.1 thru B1.4.4	Core Spray Lines and Spargers	Core Spray Lines (Headers), Spray Rings, Spray Nozzles, Thermal Sleeves	SS	288°C, High-Purity Water	Crack Initiation and Growth	IASCC	*Same as for the effect of Stress Corrosion Cracking on Items B1.4.1-B1.4.4 core spray line components.*

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) experience with cracking in the feedwater sparger system. The industry testing and analysis program is described in GE NEDE-21821-A. The primary source of degradation in this system has been thermal fatigue. However, the inspections intended to address thermal fatigue issues are also effective in ensuring that degradation by SCC is also effectively managed.	
Same as for the effect of Stress Corrosion Cracking on Items B1.3.1-B1.3.3 feedwater sparger thermal sleeve, distribution header, and discharge nozzles.	*Same as for the effect of Stress Corrosion Cracking on Items B1.3.1-B1.3.3 feedwater sparger thermal sleeve, distribution header, and discharge nozzles.*	No
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Implementation of the program delineated in the NRC Inspection and Enforcement Bulletin (IEB) 80-13 including enhanced visual inspection techniques to supplement or replace visual inspection (VT-3) requirement of GE Services Information Letter (SIL) 289. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-103515 and BWRVIP-29 to minimize the potential of crack initiation and growth. Plant programs also may include water chemistry measures such as strict controls on conductivity, hydrogen addition, and use of noble metal additions such as palladium or platinum to reduce electrochemical potential. BWRVIP guideline is under staff review.	***(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. ***(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. ***(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:*** The corrective action proposed by the BWRVIP is under staff review. ***(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.	Yes BWRVIP Guideline
Same as for the effect of Stress Corrosion Cracking on Items B1.4.1-B1.4.4 core spray lines, spray rings, spray nozzles, and thermal sleeves.	*Same as for the effect of Stress Corrosion Cracking on Items B1.4.1-B1.4.4 core spray lines, spray rings, spray nozzles, and thermal sleeves.*	Yes BWRVIP Guideline

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B1.4.1 thru B1.4.4	Core Spray Lines and Spargers	Core Spray Lines (Headers), Spray Rings, Spray Nozzles, Thermal Sleeves	SS	288°C, High-Purity Water	Cumulative Fatigue Damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B1.5.1, thru B1.5.8	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, High-Purity Water	Crack Initiation and Growth	SCC	NRC IEB 80-07. GE SIL 330. GE SIL 605 Rev. 1. BWRVIP-29. EPRI TR-103515. BWRVIP-41. BWRVIP-28. BWRVIP-51. *Supporting BWRVIP:* BWRVIP-03. BWRVIP-06. BWRVIP-14. BWRVIP-44. BWRVIP-45. BWRVIP-59. BWRVIP-60. BWRVIP-62. *Operating Experience* NRC IN 93-101. NRC IN 97-02.
B1.5.1, thru B1.5.8	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, High-Purity Water	Crack Initiation and Growth	IASCC	*Same as for the effect of Stress Corrosion Cracking on Items B1.5.1-B1.5.8 jet pump assembly components.*

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Implementation of inspection and surveillance programs delineated in NRC Inspection and Enforcement Bulletin (IEB) 80-07 and GE Services Information Letter (SIL) 330 to ensure overall functionality and integrity of jet pump assemblies, and additional recommendations of GE Services Information Letter (SIL) 605 Rev. 1 for jet pump riser pipe. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-103515 and BWRVIP-29 to minimize the potential of crack initiation and growth. Plant programs also may include water chemistry measures such as strict controls on conductivity, and hydrogen addition. BWRVIP guideline is under staff review.	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC, performance assessment and periodic 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. ***(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, or degradation of jet pump operation. ***(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 under staff review. ***(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:*** 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).	Yes, BWRVIP Guideline
Same as for the effect of Stress Corrosion Cracking on Items B1.5.1-B1.5.8 jet pump assembly components.	*Same as for the effect of Stress Corrosion Cracking on Items B1.5.1-B1.5.8 jet pump assembly components.*	Yes BWRVIP Guideline

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B1.5.1 thru B1.5.8	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, High-Purity Water	Cumulative Fatigue Damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B1.5.4	Jet Pump Assemblies	Castings	CASS	288°C, High-Purity Water	Loss of Fracture Toughness	Thermal Aging and Neutron Irradiation Embrittle-ment	EPRI TR-106092. ASME Section XI, 1989 Edition.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
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.	***(1) Scope of Program:*** The program includes determination of the susceptibility of CASS components to thermal aging based on casting method, Mo content, and percent ferrite, and for "potentially susceptible" components, to account for the synergistic loss of fracture toughness due to neutron embrittlement and thermal aging embrittlement, implement either a supplemental examination of the affected components as part of a 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. ***(2) Preventive Actions:*** The program provides no guidance on methods to mitigate thermal aging or neutron embrittlement. ***(3) Parameters Monitored/ Inspected:*** The program specifics depend on the neutron fluence and ferrite content of the component. Based on the criteria in EPRI TR-106092, with some modifications, the susceptibility to thermal aging embrittlement of CASS components is determined in terms of casting method, Mo content, and ferrite content. For low-Mo content (0.5 wt.% max.) steels, only static-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement, all centrifugal-cast and static-cast steels with ≤20% ferrite are not susceptible. For high-Mo content (2.0 to 3.0 wt.%) steels, static-cast steels with >14% ferrite and centrifugal-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement, static-cast steels with ≤14% ferrite and centrifugal-cast steels with ≤20% ferrite are not susceptible. Ferrite content will be calculated by the Hull's equivalent factors or a method producing an equivalent level of accuracy (±6% deviation between measured and calculated values. ***(4) Detection of Aging Effects:*** For all CASS components that have a neutron fluence of greater than 10^{17} n/cm^2 (E>1 MeV), implement a 10-year ISI program during the renewal period that includes supplemental inspection that covers portions of the susceptible components determined to be limiting from the standpoint of thermal aging susceptibility (Ferrite and Mo contents, casting process, and operating temperature), neutron fluence, and cracking susceptibility (applied stress, operating temperature, and environmental conditions). The inspection technique, including the reliability in detecting the features of interest (crack appearance and size) in assuring the integrity of the component, should be specified. For example, enhancement of the visual VT-1 examination to achieve a 1/2-mil (0.0005 in.) resolution, with the conditions	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B1.6.1	Fuel Supports & CRD Assemblies	Orificed Fuel Support	CASS	288°C, High-Purity Water	Crack Initiation and Growth	Thermal Aging and Neutron Irradiation Embrittle-ment	EPRI TR-106092. ASME Section XI, 1989 Edition.
B1.6.1	Fuel Supports & CRD Assemblies	Orificed Fuel Support	SS, CASS	288°C, High-Purity Water	Cumulative Fatigue Damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) r(lighting and surface cleanness) for the ISI bounded by those used to demonstrate the resolution of the inspection technique. Alternatively, 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. If the loading is compressive or low enough (<5 ksi) to preclude fracture, then supplemental inspection of the component is not required. Failure to meet this criteria requires continued use of supplemental inspection program. For all CASS components that have a neutron fluence of less than 10^{17} n/cm^2 (E>1 MeV), implement the supplement inspection program if they are "potentially susceptible" to thermal aging; the existing ASME Section XI inspection requirements are considered adequate if the components are "not susceptible" to thermal aging. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 should provide timely detection of cracks. ***(6) Acceptance Criteria:*** Flaws detected in CASS components are evaluated in accordance with the applicable procedures of IWB-3600. ***(7) Corrective Actions:*** Repair is in conformance with IWA-4000 and IWB-4000, and replacement according to IWA-7000 and IWB-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:*** The AMP based on susceptibility determination, neutron fluence level, and supplemental inspection is effective in managing the effects of synergistic loss of fracture toughness due to neutron and thermal aging embrittlement on the intended function of CASS components.	
Same as for the effect of Thermal Aging and Neutron Irradiation Embrittlement on Item B1.5.8 jet pump castings.	*Same as for the effect of Thermal Aging and Neutron Irradiation Embrittlement on Item B1.5.8 jet pump castings.*	No
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B1.7.1, B1.7.3	Instrument Housings	Intermediate Range Monitor (IRM) Dry Tubes, Source Range Monitor (SRM) Dry Tubes	SS	288°C, High-Purity Water	Crack Initiation and Growth	SCC	GE SIL 409 Rev. 1. BWRVIP-29. EPRI TR-103515. BWRVIP-49. BWRVIP-57. *Supporting BWRVIP:* BWRVIP-03. BWRVIP-06. BWRVIP-14. BWRVIP-44. BWRVIP-45. BWRVIP-59. BWRVIP-60. BWRVIP-62.
B1.7.1, B1.7.3	Instrument Housings	IRM Dry Tubes, SRM Dry Tubes	SS	288°C, High-Purity Water	Cumulative Fatigue Damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B1.7.1 thru B1.7.3	Instrument Housings	IRM Dry Tubes, Low Power Range Monitor (LPRM) Dry Tubes, SRM Dry Tubes	SS	288°C, High-Purity Water	Crack Initiation and Growth	IASCC	*Same as for the effect of Stress Corrosion Cracking on Items B1.7.1 IRM dry tubes and B1.7.3 SRM dry tubes.*
B1.7.2	Instrument Housings	LPRM Dry Tubes	SS	288°C, High-Purity Water	Cumulative Fatigue Damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Implementation of aging management program recommended in GE Services Information Letter (SIL) 409 Rev. 1. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-103515 and BWRVIP-29 to minimize the potential of crack initiation and growth. Plant programs also may include water chemistry measures such as strict controls on conductivity, and hydrogen addition. BWRVIP guideline is under staff review.	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC and periodic 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:*** 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. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with applicable, approved BWRVIP guideline is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Crack indications are evaluated in accordance with applicable, approved BWRVIP guideline. ***(7) Corrective Actions:*** The corrective action proposed by the BWRVIP is under staff review. ***(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.	Yes, BWRVIP Guideline
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Same as for the effect of Stress Corrosion Cracking on Items B1.7.1 intermediate range monitor (IRM) dry tubes and B1.7.3 source range monitor (SRM) dry tubes.	*Same as the effect of Stress Corrosion Cracking on Items B1.7.1 intermediate range monitor (IRM) dry tubes and B1.7.3 source range monitor (SRM) dry tubes.*	Yes, BWRVIP Guideline
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA

B2. Reactor Vessel Internals (PWR) - Westinghouse

B2.1 Upper Internals Assembly

- B2.1.1 Upper Support Plate
- B2.1.2 Upper Support Column
- B2.1.3 Upper Support Column Bolts
- B2.1.4 Upper Core Plate
- B2.1.5 Upper Core Plate Alignment Pins
- B2.1.6 Fuel 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 Upper Core Barrel Flange
- B2.3.3 Core Barrel Nozzles
- B2.3.4 Thermal Shield

B2.4 Baffle/Former Assembly

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

B2.5 Lower Internal Assembly

- B2.5.1 Lower Core Plate
- B2.5.2 Fuel Pins

B2.5.3 Lower Support Plate

B2.5.4 Lower Support Plate Column

B2.5.5 Lower Support Plate Column Bolts

B2.5.6 Radial 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, and Lower Internal Assembly. 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	Crack Initiation and Growth	Stress Corrosion Cracking (SCC)	ASME Section XI, 1989 Edition. EPRI TR-105714. *Operating Experience* NRC IN 84-18. NRC IN 98-11.
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	Crack Initiation and Growth	Irradiation Assisted Stress Corrosion Cracking (IASCC)	ASME Section XI, 1989 Edition. EPRI TR-105714. *Operating Experience* NRC IN 98-11.

Existing 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 (edition specified in 10 CFR 50.55a). Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC of SS components, 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:*** 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. The AMP must rely upon inservice inspection (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 by detection and sizing of cracks by 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. VT-3 may not be adequate to detect tight cracks. Creviced and other inaccessible 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:*** 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. 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 Element 4 should be further evaluated
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI (edition specified in 10 CFR 50.55a). Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC of SS components, 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:*** 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	Yes Element 4 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	Changes in Dimensions	Void Swelling	EPRI TR-107521.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) greatly reduces susceptibility to 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 on the intended function 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 IASCC can not occur without crack initiation and growth. VT-3 may not be adequate to detect tight cracks. The inspection technique, including the reliability in detecting the features of interest (crack appearance and size) in assuring the integrity of the component, should be specified. For example, enhancement of the visual VT-1 examination to achieve a 1/2-mil (0.0005 in.) resolution, with the conditions (lighting and surface cleanness) for the ISI bounded by those used to demonstrate the resolution of the inspection technique. Creviced regions are difficult to inspect visually and supplementary UT or other nondestructive examinations may be needed. 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 the only cracking presently directly attributable to IASCC in PWRs occurred in SS fuel cladding used in early reactors, susceptibility to this degradation will increase as plant operating time and hence accumulated fluence levels increase. The role that IASCC has played in the more recent cracking that has occurred in SS baffle/former bolts in U.S. plants as well as foreign plants (IN 98-11) in not yet universally agreed upon.	
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.	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.	Yes TLAA

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	Cumula-tive fatigue damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B2.1.7	Upper Internals Assembly	Hold-Down Spring	SS	Chemically Treated Borated Water up to 340°C	Loss of Preload	Stress Relaxation	ASME Section XI, 1989 Edition.
B2.1.2	Upper Internals Assembly	Upper Support Column	SS, Cast Austenitic SS (CASS)	Chemically Treated Borated Water up to 340°C	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B2.1.2	Upper Internals Assembly	Upper Support Column	SS, CASS	Chemically Treated Borated Water up to 340°C	Crack Initiation and Growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
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 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 effects of stress relaxation on the intended function of the component by detection and sizing of cracks that could be formed by excessive vibration etc. that may occur if the preload is lost. 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, it may be adequate if there is sufficient redundancy that loss of some bolting between inspections is acceptable. In some cases additional inspection may be required. ***(4) Detection of Aging Effects:*** As part of the AMP it may be possible to identify acceptable levels of preload and demonstrate whether under the fluence of interest whether loss of acceptable preload is likely. VT-3 may not be adequate to detect tight cracks. Also, creviced regions are difficult to inspect visually. Supplementary inspections by techniques such as ultrasonic testing (UT) or other nondestructive methods may be needed to detect cracking in inaccessible regions. ***(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:*** There are no reports of stress relaxation producing damage in reactor vessel internals.	Yes Elements 3 and 4 should be further evaluated
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 Element 4 should be further evaluated
Same as for the effect of Irradiation Assisted 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 Irradiation Assisted 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 Element 4 should be further evaluated

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, CASS	Chemically Treated Borated Water up to 340°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.
B2.1.2	Upper Internals Assembly	Upper Support Column	CASS	Chemically Treated Borated Water up to 340°C	Loss of fracture toughness	Thermal Aging and Neutron Irradiation Embrittle-ment	EPRI TR-106092. ASME Section XI, 1989 Edition.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.	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.	Yes TLAA
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.	***(1) Scope of Program:*** The program includes determination of the susceptibility of CASS components to thermal aging based on casting method, Mo content, and percent ferrite, and for "potentially susceptible" components, to account for the synergistic loss of fracture toughness due to neutron embrittlement and thermal aging embrittlement, implement either a supplemental examination of the affected components as part of a 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. ***(2) Preventive Actions:*** The program provides no guidance on methods to mitigate thermal aging or neutron embrittlement. ***(3) Parameters Monitored/ Inspected:*** The program specifics depend on the neutron fluence and ferrite content of the component. The criteria in EPRI TR-106092, with some modifications approved by the NRC staff, can be used to determine whether CASS components are potentially susceptibility to thermal aging embrittlement based on the casting method, Mo content, and ferrite content. For low-Mo content (0.5 wt.% max.) steels, only static-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement, all centrifugal-cast and static-cast steels with ≤20% ferrite are not susceptible. For high-Mo content (2.0 to 3.0 wt.%) steels, static-cast steels with >14% ferrite and centrifugal-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement, static-cast steels with ≤14% ferrite and centrifugal-cast steels with ≤20% ferrite are not susceptible. Ferrite content will be calculated by the Hull's equivalent factors or a method producing an equivalent level of accuracy (±6% deviation between measured and calculated values. ***(4) Detection of Aging Effects:*** For all CASS components that have a neutron fluence of greater than 10^{17} n/cm^2 (E>1 MeV), implement a 10-year ISI program during the renewal period that includes supplemental inspection that covers portions of the susceptible components determined to be limiting from the standpoint of thermal aging susceptibility (Ferrite and Mo contents, casting process, and operating temperature), neutron fluence, and cracking susceptibility (applied stress, operating temperature, and environmental conditions). The inspection technique, including the reliability in detecting the features of interest (crack appearance and size) in assuring the integrity of the component, should be specified. For example, enhancement of the visual VT-1 examination to achieve a 1/2-mil (0.0005 in.) resolution, with the conditions (lighting and surface cleanness) for the ISI bounded by those used to demonstrate the resolution of the inspection technique. As an alternate to enhanced inspection,	No

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, CASS	Chemically Treated Borated Water up to 340°C	Cumula-tive fatigue damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B2.1.3, B2.1.5, B2.1.6	Upper Internals Assembly	Upper Support Column Bolts, Upper Core Plate Alignment Pins, Fuel Pins	SS, Ni Alloy	Chemically Treated Borated Water at temperatures up to 340°C	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714. *Operating Experience* NRC IN 82-29. NRC IN 84-18. NRC IN 98-11.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) 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. If the loading is compressive or low enough (<5 ksi) to preclude fracture, then supplemental inspection of the component is not required. Failure to meet this criteria requires continued use of supplemental inspection program. For all CASS components that have a neutron fluence of less than 10^{17} n/cm^2 (E>1 MeV), implement the supplement inspection program if they are "potentially susceptible" to thermal aging; the existing ASME Section XI inspection requirements are considered adequate if the components are "not susceptible" to thermal aging. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 should provide timely detection of cracks. ***(6) Acceptance Criteria:*** Flaws detected in CASS components are evaluated in accordance with the applicable procedures of IWB-3600. ***(7) Corrective Actions:*** Repair is in conformance with IWA-4000 and IWB-4000, and replacement according to IWA-7000 and IWB-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:*** The AMP based on susceptibility determination, neutron fluence level, and supplemental inspection is effective in managing the effects of synergistic loss of fracture toughness due to neutron and thermal aging embrittlement on the intended function of CASS components.	
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI (edition specified in 10 CFR 50.55a). Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC of components constructed from SS and Ni alloys, 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:*** 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, preventive actions are considered inadequate because of possible effects of introduction of oxygen during shutdown and the potential formation of more aggressive chemistry conditions by radiolysis in creviced regions or in low-flow stagnant regions. ***(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-3 specifies visual VT-3 examination of all accessible surfaces of reactor	Yes Element 4 should be further evaluated

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 Pins	SS, Ni Alloy	Chemically Treated Borated Water up to 340°C	Crack Initiation and Growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B2.1.3, B2.1.5, B2.1.6	Upper Internals Assembly	Upper Support Column Bolts, Upper Core Plate Alignment Pins, Fuel Pins	SS, Ni Alloy	Chemically Treated Borated Water up to 340°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.
B2.1.3	Upper Internals Assembly	Upper Support Column Bolts	SS, Ni Alloy	Chemically Treated Borated Water up to 340°C	Loss of Preload	Stress Relaxation	ASME Section XI, 1989 Edition.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) 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 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:*** 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. 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. SCC has also been observed in Ni alloy (Alloy X–750) control rod drive guide tube support pins (IN 82-29). Replacement components with a different heat treatment appear to be less susceptible to cracking.	
Same as for the effect of Stress Corrosion Cracking on Items B2.1.3 upper support column bolts, B2.1.5 upper core plate alignment pins, and B2.1.6 fuel pins.	*Same as for the effect of Stress Corrosion Cracking on Items B2.1.3 upper support column bolts, B2.1.5 upper core plate alignment pins, and B2.1.6 fuel pins.*	Yes Element 4 should be further evaluated
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.	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.	Yes TLAA
Visual inspection (VT–3) is performed according to Category B–N–3 of Subsection IWB, ASME Section XI.	*Same as the effect of Stress Relaxation on Item B2.1.7, hold-down spring.*	Yes Elements 3 and 4 should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B2.1.4	Upper Internals Assembly	Upper Core Plate	SS	Chemically Treated Borated Water up to 340°C	Cumula-tive fatigue damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B2.1.5, B2.1.6	Upper Internals Assembly	Upper Core Plate Alignment Pins, Fuel Pins	SS, Ni Alloy	Chemically Treated Borated Water up to 340°C	Attrition	Wear	ASME Section XI, 1989 Edition.
B2.1.6	Upper Internals Assembly	Fuel Pins	SS, Ni Alloy	Chemically Treated Borated Water up to 340°C	Cumula-tive fatigue damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B2.2.1	RCCA Guide Tube Assemblies	RCCA Guide Tubes	SS	Chemically Treated Borated Water up to 340°C	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B2.2.1	RCCA Guide Tube Assemblies	RCCA Guide Tubes	SS	Chemically Treated Borated Water up to 340°C	Crack Initiation and Growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B2.2.1	RCCA Guide Tube Assemblies	RCCA Guide Tubes	SS	Chemically Treated Borated Water up to 340°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
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 of 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:*** Acceptance standard for the inspection is 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
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	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 Element 4 should be further evaluated
Same as for the effect of Irradiation Assisted 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 Irradiation Assisted 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 Element 4 should be further evaluated
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.	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.	Yes TLAA

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	Attrition	Wear	ASME Section XI, 1989 Edition.
B2.2.1	RCCA Guide Tube Assemblies	RCCA Guide Tubes	SS	Chemically Treated Borated Water up to 340°C	Cumula-tive fatigue damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
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	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
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	Crack Initiation and Growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
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	Changes in Dimensions	Void Swelling	EPRI TR-107521.
B2.2.2	RCCA Guide Tube Assemblies	RCCA Guide Tube Bolts	SS, Ni Alloy	Chemically Treated Borated Water up to 340°C	Loss of Preload	Stress Relaxation	ASME Section XI, 1989 Edition.
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	Cumula-tive fatigue damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B2.3.1 thru B2.3.4	Core Barrel	Core Barrel, Upper Core Barrel Flange, Core Barrel Nozzles, Thermal Shield	SS	Chemically Treated Borated Water up to 340°C	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B2.3.1 thru B2.3.4	Core Barrel	Core Barrel, Upper Core Barrel Flange, Core Barrel Nozzles, Thermal Shield	SS	Chemically Treated Borated Water up to 340°C	Crack Initiation and Growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.

Existing 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.	*Same as the effect of Wear on Items B2.1.5 and B2.1.6, upper core plate alignment pins, fuel pins.*	No
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Same as for the effect of Stress Corrosion Cracking on Items B2.1.3 upper support column bolts, B2.1.5 upper core plate alignment pins, and B2.1.6 fuel pins.	*Same as for the effect of Stress Corrosion Cracking on Items B2.1.3 upper support column bolts, B2.1.5 upper core plate alignment pins, and B2.1.6 fuel pins.*	Yes Element 4 should be further evaluated
Same as for the effect of Stress Corrosion Cracking on Items B2.1.3 upper support column bolts, B2.1.5 upper core plate alignment pins, and B2.1.6 fuel pins.	*Same as for the effect of Stress Corrosion Cracking on Items B2.1.3 upper support column bolts, B2.1.5 upper core plate alignment pins, and B2.1.6 fuel pins.*	Yes Element 4 should be further evaluated
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.	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.	Yes TLAA
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*Same as the effect of Stress Relaxation on Item B2.1.7, hold-down spring*	Yes Elements 3 and 4 should be further evaluated
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	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 Element 4 should be further evaluated
Same as for the effect of Irradiation Assisted 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 Irradiation Assisted 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 Element 4 should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B2.3.1 thru B2.3.4	Core Barrel	Core Barrel, Upper Core Barrel Flange, Core Barrel Nozzles, Thermal Shield	SS	Chemically Treated Borated Water up to 340°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.
B2.3.1 thru B2.3.4	Core Barrel	Core Barrel, Upper Core Barrel Flange, Core Barrel Nozzles, Thermal Shield	SS	Chemically Treated Borated Water up to 340°C	Loss of Fracture Toughness	Neutron Irradiation Embrittle-ment	ASME Section XI, 1989 Edition.
B2.3.1 thru B2.3.4	Core Barrel	Core Barrel, Upper Core Barrel Flange, Core Barrel Nozzles, Thermal Shield	SS	Chemically Treated Borated Water up to 340°C	Cumula-tive fatigue damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B2.3.2	Core Barrel	Upper Core Barrel Flange	SS	Chemically Treated Borated Water up to 340°C	Attrition	Wear	ASME Section XI, 1989 Edition.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.	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.	Yes TLAA
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 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, category B-N-3 specifies visual VT-3 examination of all accessible surfaces of reactor internals. ***(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 may not be 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 Elements 3 and 4 should be further evaluated
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*Same as for the effect of Wear on Items B2.1.5 upper core plate alignment pins and B2.1.6 fuel pins.*	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B2.4.1	Baffle/Former Assembly	Baffle and Former Plates	SS	Chemically Treated Borated Water up to 340°C	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B2.4.1	Baffle/Former Assembly	Baffle and Former Plates	SS	Chemically Treated Borated Water up to 340°C	Crack Initiation and Growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B2.4.1	Baffle/Former Assembly	Baffle and Former Plates	SS	Chemically Treated Borated Water up to 340°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.
B2.4.2	Baffle/Former Assembly	Baffle/Former Bolts	SS (Type 347 & cold-worked Type 316)	Chemically Treated Borated Water up to 340°C and high fluence (>10 dpa or 7×10^{21} n/cm^2 E>1 MeV)	Crack Initiation and Growth	SCC	Plant Technical Specification
B2.4.2	Baffle/Former Assembly	Baffle/Former Bolts	SS (Type 347 & cold-worked Type 316)	Chemically Treated Borated Water up to 340°C and high fluence (>10 dpa or 7×10^{21} n/cm^2 E>1 MeV)	Crack Initiation and Growth	IASCC	Plant Technical Specification
B2.4.2	Baffle/Former Assembly	Baffle/Former Bolts	SS (Type 347 & cold-worked Type 316)	Chemically Treated Borated Water up to 340°C and high fluence	Changes in Dimensions	Void Swelling	EPRI TR-107521.

Existing 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 Element 4 should be further evaluated
Same as for the effect of Irradiation Assisted 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 Irradiation Assisted 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 Element 4 should be further evaluated
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.	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.	Yes TLAA
Plant-specific aging management program. 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 bolt 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 No AMP
Same as for the effect of Stress Corrosion Cracking on Item B2.4.2 baffle/former bolts.	*Same as for the effect of Stress Corrosion Cracking on Item B2.4.2 baffle/former bolts.*	Yes No AMP
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.	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.	Yes TLAA

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
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	Loss of Fracture Toughness	Neutron Irradiation Embrittle-ment	ASME Section XI, 1989 Edition.
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	Cumula-tive fatigue damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B2.4.2	Baffle/Former Assembly	Baffle/Former Bolts	SS, Ni Alloy	Chemically Treated Borated Water up to 340°C	Loss of Preload	Stress Relaxation	ASME Section XI, 1989 Edition.
B2.5.1, B2.5.6, B2.5.7	Lower Internal Assembly	Lower Core Plate, Radial Keys and Clevis Inserts, Clevis Insert Bolts	SS	Chemically Treated Borated Water up to 340°C	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B2.5.1, B2.5.6, B2.5.7	Lower Internal Assembly	Lower Core Plate, Radial Keys and Clevis Inserts, Clevis Insert Bolts	SS	Chemically Treated Borated Water up to 340°C	Crack Initiation and Growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B2.5.1, B2.5.6, B2.5.7	Lower Internal Assembly	Lower Core Plate, Radial Keys and Clevis Inserts, Clevis Insert Bolts	SS	Chemically Treated Borated Water up to 340°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.
B2.5.1	Lower Internal Assembly	Lower Core Plate	SS	Chemically Treated Borated Water up to 340°C	Loss of Fracture Toughness	Neutron Irradiation Embrittle-ment	ASME Section XI, 1989 Edition.
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	Cumula-tive fatigue damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B2.5.2, B2.5.5	Lower Internal Assembly	Fuel Pins, Lower Support Plate Column Bolts	SS, Ni Alloy	Chemically Treated Borated Water up to 340°C	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.

Existing 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.	*Same as for the effect of Neutron Irradiation Embrittlement on Items B2.3.1 thru B2.3.3 core barrel, upper core barrel flange, core barrel nozzles.*	Yes Elements 3 and 4 should be further evaluated
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*Same as for the effect of Stress Relaxation on Item B2.1.7 hold-down spring.*	Yes Elements 3 and 4 should be further evaluated
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 Element 4 should be further evaluated
Same as for the effect of Irradiation Assisted 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 Irradiation Assisted 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 Element 4 should be further evaluated
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.	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.	Yes TLAA
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*Same as for the effect of Neutron Irradiation Embrittlement on Items B2.3.1 thru B2.3.3 core barrel, upper core barrel flange, core barrel nozzles.*	Yes Elements 3 and 4 should be further evaluated
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Same as for the effect of Stress Corrosion Cracking on Items B2.1.3 upper support column bolts, B2.1.5 upper core plate alignment pins, and B2.1.6 fuel pins.	*Same as for the effect of Stress Corrosion Cracking on Items B2.1.3 upper support column bolts, B2.1.5 upper core plate alignment pins, and B2.1.6 fuel pins.*	Yes Element 4 should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B2.5.2, B2.5.5	Lower Internal Assembly	Fuel Pins, Lower Support Plate Column Bolts	SS, Ni Alloy	Chemically Treated Borated Water up to 340°C	Crack Initiation and Growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B2.5.2, B2.5.5	Lower Internal Assembly	Fuel Pins, Lower Support Plate Column Bolts	SS, Ni Alloy	Chemically Treated Borated Water up to 340°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.
B2.5.2, B2.5.5	Lower Internal Assembly	Fuel Pins, Lower Support Plate Column Bolts	SS, Ni Alloy	Chemically Treated Borated Water up to 340°C	Loss of Fracture Toughness	Neutron Irradiation Embrittle-ment	ASME Section XI, 1989 Edition.
B2.5.2, B2.5.5, B2.5.7	Lower Internal Assembly	Fuel Pins, Lower Support Plate Column Bolts, Clevis Insert Bolts	SS, Ni Alloy	Chemically Treated Borated Water up to 340°C	Loss of Preload	Stress Relaxation	ASME Section XI, 1989 Edition.
B2.5.2	Lower Internal Assembly	Fuel Pins	SS, Ni Alloy	Chemically Treated Borated Water up to 340°C	Attrition	Wear	ASME Section XI, 1989 Edition.
B2.5.2, B2.5.5	Lower Internal Assembly	Fuel Pins, Lower Support Plate Column Bolts	SS, Ni Alloy	Chemically Treated Borated Water up to 340°C	Cumula-tive fatigue damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B2.5.3, B2.5.4	Lower Internal Assembly	Lower Support Plate, Lower Support Plate Columns	SS, CASS	Chemically Treated Borated Water up to 340°C	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B2.5.3, B2.5.4	Lower Internal Assembly	Lower Support Plate, Lower Support Plate Columns	SS, CASS	Chemically Treated Borated Water up to 340°C	Crack Initiation and Growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B2.5.3, B2.5.4	Lower Internal Assembly	Lower Support Plate, Lower Support Plate Columns	SS, CASS	Chemically Treated Borated Water up to 340°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Same as for the effect of Stress Corrosion Cracking on Items B2.1.3 upper support column bolts, B2.1.5 upper core plate alignment pins, and B2.1.6 fuel pins.	*Same as for the effect of Stress Corrosion Cracking on Items B2.1.3 upper support column bolts, B2.1.5 upper core plate alignment pins, and B2.1.6 fuel pins.*	Yes Element 4 should be further evaluated
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.	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.	Yes TLAA
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*Same as for the effect of Neutron Irradiation Embrittlement on Items B2.3.1 thru B2.3.3 core barrel, upper core barrel flange, core barrel nozzles.*	Yes Elements 3 and 4 should be further evaluated
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*Same as for the effect of Stress Relaxation on Item B2.1.7 hold-down spring.*	Yes Elements 3 and 4 should be further evaluated
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*Same as for the effect of Wear on Items B2.1.5 upper core plate alignment pins and B2.1.6 fuel pins.*	No
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	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 Element 4 should be further evaluated
Same as for the effect of Irradiation Assisted 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 Irradiation Assisted 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 Element 4 should be further evaluated
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.	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.	Yes TLAA

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 Plate, Lower Support Plate Columns	SS, CASS	Chemically Treated Borated Water up to 340°C	Loss of fracture toughness	Thermal Aging and Neutron Irradiation Embrittle-ment	EPRI TR-106092. ASME Section XI, 1989 Edition.
B2.5.6	Lower Internal Assembly	Radial Keys and Clevis Inserts	SS	Chemically Treated Borated Water up to 340°C	Attrition	Wear	ASME Section XI, 1989 Edition.
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	Cumula-tive fatigue damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B2.6.1	Instru-mentation Support Structure	Flux Thimble Guide Tubes	SS	Chemically Treated Borated Water up to 340°C	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B2.6.1	Instru-mentation Support Structure	Flux Thimble Guide Tubes	SS	Chemically Treated Borated Water up to 340°C	Crack Initiation and Growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B2.6.1	Instru-mentation Support Structure	Flux Thimble Guide Tubes	SS	Chemically Treated Borated Water up to 340°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.
B2.6.2	Instru-mentation Support Structure	Flux Thimbles	SS	Chemically Treated Borated Water up to 340°C	Attrition	Wear	ASME Section XI, 1989 Edition.

Existing 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
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*Same as for the effect of Wear on Items B2.1.5 upper core plate alignment pins and B2.1.6 fuel pins.*	No
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	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 Element 4 should be further evaluated
Same as for the effect of Irradiation Assisted 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 Irradiation Assisted 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 Element 4 should be further evaluated
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.	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.	Yes TLAA
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*Same as for the effect of Wear on Items B2.1.5 upper core plate alignment pins and B2.1.6 fuel 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
- B3.1.4 Hold-Down Rings

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. 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 thru 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	Crack Initiation & Growth	Stress Corrosion Cracking (SCC)	ASME Section XI, 1989 Edition. EPRI TR-105714. *Operating Experience* NRC IN 84-18. NRC IN 98-11.
B3.1.1 thru 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	Crack Initiation & Growth	Irradiation Assisted Stress Corrosion Cracking (IASCC)	ASME Section XI, 1989 Edition. EPRI TR-105714. *Operating Experience* NRC IN 98-11.

Existing 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 (edition specified in 10 CFR 50.55a). Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC of SS components, 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:*** 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. The AMP must rely upon inservice inspection (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 by detection and sizing of cracks by 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. VT-3 may not be adequate to detect tight cracks. Creviced and other inaccessible 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:*** 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. 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 Element 4 should be further evaluated
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI (edition specified in 10 CFR 50.55a). Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC of SS components, 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:*** 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 IASCC. However, introduction of oxygen can occur during shutdown and	Yes Element 4 should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B3.1.1 thru 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	Changes in Dimensions	Void Swelling	EPRI TR-107521.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) 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 on the intended function 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 IASCC can not occur without crack initiation and growth. VT-3 may not be adequate to detect tight cracks. The inspection technique, including the reliability in detecting the features of interest (crack appearance and size) in assuring the integrity of the component, should be specified. For example, enhancement of the visual VT-1 examination to achieve a 1/2-mil (0.0005 in.) resolution, with the conditions (lighting and surface cleanness) for the ISI bounded by those used to demonstrate the resolution of the inspection technique. Creviced regions are difficult to inspect visually and supplementary UT or other nondestructive examinations may be needed. 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 the only cracking presently directly attributable to IASCC in PWRs occurred in SS fuel cladding used in early reactors, susceptibility to this degradation will increase as plant operating time and hence accumulated fluence levels increase. The role of IASCC in the more recent cracking that has occurred in SS baffle/former bolts in U.S. plants as well as foreign plants (IN 98-11) has not yet been universally agreed upon.	
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.	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.	Yes TLAA

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B3.1.1, B3.1.3, B3.1.4	Upper Internals Assembly	Upper Guide Structure Support Plate, Fuel Alignment Plate Guide Lugs, Hold-Down Ring	SS	Chemically Treated Borated Water up to 340°C	Attrition	Wear	ASME Section XI, 1989 Edition.
B3.2.1	CEA Shroud Assemblies	CEA Shroud	SS, Cast Austenitic SS (CASS)	Chemically Treated Borated Water up to 340°C	Crack Initiation & Growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B3.2.2	CEA Shroud Assemblies	CEA Shroud Bolts	SS, Ni alloy	Chemically Treated Borated Water up to 340°C	Crack Initiation & Growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714. *Operating Experience* NRC IN 82-29. NRC IN 84-18. NRC IN 98-11.

Existing 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:*** Acceptance standard for the inspection is 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 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 Element 4 should be further evaluated
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI (edition specified in 10 CFR 50.55a). Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC of components constructed from SS and Ni alloys, 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:*** 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, preventive actions are considered inadequate because of possible effects of introduction of oxygen during shutdown and the potential formation of more aggressive chemistry conditions by radiolysis in creviced regions or in low-flow stagnant regions. ***(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-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	Yes Element 4 should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B3.2.1	CEA Shroud Assemblies	CEA Shroud	SS, CASS	Chemically Treated Borated Water up to 340°C	Crack Initiation & Growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B3.2.2	CEA Shroud Assemblies	CEA Shroud Bolts	SS, Ni alloy	Chemically Treated Borated Water up to 340°C	Crack Initiation & Growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B3.2.1, B3.2.2	CEA Shroud Assemblies	CEA Shroud, CEA Shroud Bolts	SS, CASS, Ni alloy	Chemically Treated Borated Water up to 340°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.
B3.2.3	CEA Shroud Assemblies	CEA Shroud Extension Shaft Guides	SS	Chemically Treated Borated Water up to 340°C	Attrition	Wear	ASME Section XI, 1989 Edition.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) inaccessible regions are difficult to inspect visually. Supplementary UT 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:*** 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. 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. SCC has also been observed in Ni alloy (Alloy X–750) control rod drive guide tube support pins (IN 82-29). Replacement components with a different heat treatment appear to be less susceptible to cracking.	
Same as for the effect of Irradiation Assisted 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 Irradiation Assisted 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 Element 4 should be further evaluated
Same as for the effect of Stress Corrosion Cracking on Item B3.2.2 CEA shroud bolts.	*Same as for the effect of Stress Corrosion Cracking on Item B3.2.2 CEA shroud bolts.*	Yes Element 4 should be further evaluated
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.	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.	Yes TLAA
Visual inspection (VT–3) is performed according to Category B–N–3 of Subsection IWB, ASME Section XI.	*Same as for the effect of Wear on Items B3.1.1 upper guide structure support 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	Loss of Fracture Toughness	Thermal Aging and Neutron Irradiation Embrittle-ment	EPRI TR-106092. ASME Section XI, 1989 Edition.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
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.	***(1) Scope of Program:*** The program includes determination of the susceptibility of CASS components to thermal aging based on casting method, Mo content, and percent ferrite, and for "potentially susceptible" components, to account for the synergistic loss of fracture toughness due to neutron embrittlement and thermal aging embrittlement, implement either a supplemental examination of the affected components as part of a 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. ***(2) Preventive Actions:*** The program provides no guidance on methods to mitigate thermal aging or neutron embrittlement. ***(3) Parameters Monitored/ Inspected:*** The program specifics depend on the neutron fluence and ferrite content of the component. The criteria in EPRI TR-106092, with some modifications approved by the NRC staff, can be used to determine whether CASS components are potentially susceptibility to thermal aging embrittlement based on the casting method, Mo content, and ferrite content. For low-Mo content (0.5 wt.% max.) steels, only static-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement, all centrifugal-cast and static-cast steels with ≤20% ferrite are not susceptible. For high-Mo content (2.0 to 3.0 wt.%) steels, static-cast steels with >14% ferrite and centrifugal-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement, static-cast steels with ≤14% ferrite and centrifugal-cast steels with ≤20% ferrite are not susceptible. Ferrite content will be calculated by the Hull's equivalent factors or a method producing an equivalent level of accuracy (±6% deviation between measured and calculated values). ***(4) Detection of Aging Effects:*** For all CASS components that have a neutron fluence of greater than 10^{17} n/cm^2 (E>1 MeV), implement a 10-year ISI program during the renewal period that includes supplemental inspection that covers portions of the susceptible components determined to be limiting from the standpoint of thermal aging susceptibility (Ferrite and Mo contents, casting process, and operating temperature), neutron fluence, and cracking susceptibility (applied stress, operating temperature, and environmental conditions). The inspection technique, including the reliability in detecting the features of interest (crack appearance and size) in assuring the integrity of the component, should be specified. For example, enhancement of the visual VT-1 examination to achieve a 1/2-mil (0.0005 in.) resolution, with the conditions (lighting and surface cleanness) for the ISI bounded by those used to demonstrate the resolution of the inspection technique. perform a component-specific , perform a component-specific evaluation including a mechanical loading assessment to determine the maximum tensile loading on the component under ASME Code Level A, B, C, and D conditions. If the loading is compressive or low enough (<5 ksi) to preclude fracture, then supplemental inspection of the component is not required. Failure to meet this criteria requires continued use of supplemental inspection program. For all CASS	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B3.2.1, B3.2.2	CEA Shroud Assemblies	CEA Shroud, CEA Shroud Bolts	SS, CASS, Ni Alloy	Chemically Treated Borated Water up to 340°C	Cumulative Fatigue Damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B3.2.2	CEA Shroud Assemblies	CEA Shroud Bolts	SS, Ni Alloy	Chemically Treated Borated Water up to 340°C	Loss of Preload	Stress Relaxation	ASME Section XI, 1989 Edition.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) components that have a neutron fluence of less than 10^{17} n/cm^2 (E>1 MeV), implement the supplement inspection program if they are "potentially susceptible" to thermal aging; the existing ASME Section XI inspection requirements are considered adequate if the components are "not susceptible" to thermal aging. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 should provide timely detection of cracks. ***(6) Acceptance Criteria:*** Flaws detected in CASS components are evaluated in accordance with the applicable procedures of IWB-3600. ***(7) Corrective Actions:*** Repair is in conformance with IWA-4000 and IWB-4000, and replacement according to IWA-7000 and IWB-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:*** The AMP based on susceptibility determination, neutron fluence level, and supplemental inspection is effective in managing the effects of synergistic loss of fracture toughness due to neutron and thermal aging embrittlement on the intended function of CASS components.	
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
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 inservice inspection (ISI) to monitor the effects of stress relaxation on the intended function of the component by detection and sizing of cracks that could be formed by excessive vibration etc. if the preload is lost. 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, it may be adequate if there is sufficient redundancy that loss of some bolting between inspections is acceptable. In some cases additional inspection may be required. ***(4) Detection of Aging Effects:*** As part of the AMP it may be possible to identify acceptable levels of preload and demonstrate under the fluence of interest whether less of acceptable preload is likely. VT-3 may not be adequate to detect tight cracks. Creviced regions are difficult to inspect visually. Supplementary inspections by techniques such as ultrasonic testing (UT) or other nondestructive methods may be needed to detect cracking in inaccessible regions. ***(5) Monitoring and Trending:***	Yes Elements 3 and 4 should be further evaluated

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	Crack Initiation & Growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
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	Crack Initiation & Growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
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	Changes in Dimensions	Void Swelling	EPRI TR-107521.
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	Loss of Fracture Toughness	Neutron Irradiation Embrittle-ment	ASME Section XI, 1989 Edition.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) 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:*** There are no reports of stress relaxation producing damage in reactor vessel internals.	
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 Element 4 should be further evaluated
Same as for the effect of Irradiation Assisted 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 Irradiation Assisted 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 Element 4 should be further evaluated
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.	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.	Yes TLAA
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 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, category B-N-3 specifies visual VT-3 examination of all accessible surfaces of reactor internals. ***(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 may not be 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	Yes Elements 3 and 4 should be further evaluated

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	Cumulative Fatigue Damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
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	Attrition	Wear	ASME Section XI, 1989 Edition.
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	Crack Initiation & Growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
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	Crack Initiation & Growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
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	Changes in Dimensions	Void Swelling	EPRI TR-107521.
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	Loss of Fracture Toughness	Neutron Irradiation Embrittle-ment	ASME Section XI, 1989 Edition.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) 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 recorded have been definitely attributed to irradiation embrittlement.	
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*Same as for the effect of Wear on Items B3.1.1 upper guide structure support plate, B3.1.3 fuel alignment plate guide lugs., and B3.1.4 hold-down ring.*	No
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 Element 4 should be further evaluated
Same as for the effect of Irradiation Assisted 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 Irradiation Assisted 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 Element 4 should be further evaluated
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.	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.	Yes TLAA
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*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 Elements 3 and 4 should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B3.4.1 thru 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	Cumulative Fatigue Damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
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	Crack Initiation & Growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
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	Crack Initiation & Growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
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	Changes in Dimensions	Void Swelling	EPRI TR-107521.
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	Loss of Fracture Toughness	Neutron Irradiation Embrittle-ment	ASME Section XI, 1989 Edition.
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	Loss of Preload	Stress Relaxation	ASME Section XI, 1989 Edition.
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	Crack Initiation & Growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
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	Crack Initiation & Growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Same as for the effect of Stress Corrosion Cracking on Item B3.2.2 CEA shroud bolts.	*Same as for the effect of Stress Corrosion Cracking on Item B3.2.2 CEA shroud bolts.*	Yes Element 4 should be further evaluated
Same as for the effect of Stress Corrosion Cracking on Item B3.2.2 CEA shroud bolts.	*Same as for the effect of Stress Corrosion Cracking on Item B3.2.2 CEA shroud bolts.*	Yes Element 4 should be further evaluated
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.	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.	Yes TLAA
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*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 Elements 3 and 4 should be further evaluated
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*Same as for the effect of Stress Relaxation on Item B3.2.2 CEA shroud bolts.*	Yes Elements 3 and 4 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 Element 4 should be further evaluated
Same as for the effect of Stress Corrosion Cracking on Item B3.2.2 CEA shroud bolts.	*Same as for the effect of Stress Corrosion Cracking on Item B3.2.2 CEA shroud bolts.*	Yes Element 4 should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
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	Crack Initiation & Growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
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	Crack Initiation & Growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B3.5.1 thru 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	Changes in Dimensions	Void Swelling	EPRI TR-107521.
B3.5.1 thru 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	Loss of Fracture Toughness	Neutron Irradiation Embrittle-ment	ASME Section XI, 1989 Edition.
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	Loss of Preload	Stress Relaxation	ASME Section XI, 1989 Edition.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Same as for the effect of Irradiation Assisted 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 Irradiation Assisted 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 Element 4 should be further evaluated
Same as for the effect of Stress Corrosion Cracking on Item B3.2.2 CEA shroud bolts.	*Same as for the effect of Stress Corrosion Cracking on Item B3.2.2 CEA shroud bolts.*	Yes Element 4 should be further evaluated
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.	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.	Yes TLAA
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*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 Elements 3 and 4 should be further evaluated
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*Same as for the effect of Stress Relaxation on Item B3.2.2 CEA shroud bolts.*	Yes Elements 3 and 4 should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
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	Attrition	Wear	ASME Section XI, 1989 Edition.
B3.5.1 thru 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	Cumulative Fatigue Damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B3.5.4	Lower Internal Assembly	Core Support Column	CASS	Chemically Treated Borated Water up to 340°C	Loss of Fracture Toughness	Thermal Aging and Neutron Irradiation Embrittle-ment	EPRI TR-106092. ASME Section XI, 1989 Edition.

Existing 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.	*Same as for the effect of Wear on Items B3.1.1 upper guide structure support plate, B3.1.3 fuel alignment plate guide lugs., and B3.1.4 hold-down ring.*	No
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Same as for the effect of Thermal Aging and Neutron Irradiation Embrittlement on Item B3.2.1 CEA shroud.	*Same as for the effect of Thermal Aging and Neutron Irradiation Embrittlement on Item B3.2.1 CEA shroud.*	No

B4 Reactor Vessel Internals (PWR) - Babcock & Wilcox

B4.1 Plenum Cover and Plenum Cylinder

B4.2 CRA Guide Tube Assemblies

B4.2.1 Tubes

B4.2.2 Spacer Casting

B4.2.3 Spider Casting

B4.2.4 Bolts

B4.3 Upper Grid Assembly

B4.3.1 Upper Grid Rib Section

B4.3.2 Upper Grid Assembly Bolts

B4.3.3 Plenum Rib Pads

B4.4 Core Support Assembly

B4.4.1 Core Support Shield

B4.4.2 Core Support Shield Flange

B4.4.3 Core Support Shield to Core Barrel Bolts

B4.5 Vent Valve Assembly

B4.6 Core Barrel Assembly

B4.6.1 Core Barrel

B4.6.2 Core Barrel Bolts

B4.6.3 Core Barrel to Thermal Shield Bolts

B4.7 Lower Grid Assembly (LGA)

B4.7.1 Upper Grid

B4.7.2 Fuel Guide Pads

B4.7.3 Lower Grid

B4.7.4 Support Columns

B4.7.5 Cylinder Guide Blocks

B4.7.6 LGA Bolts

B4.7.7 Core Barrel Bolts

B4.7.8 Flow Distributor Bolts

B4.7.9 Thermal Shield Bolts

B4.8 Flow Distributor

B4.9 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, Control Rod Assemblies (CRA) Guide Tube Assemblies, Upper Grid Assembly, Core Support Assembly, Vent Valve Assembly, Core Barrel Assembly, and Lower Internal Assembly. 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	Plenum Cover and Plenum Cylinder	Plenum Cover and Plenum Cylinder	Stainless Steel (SS)	Chemically Treated Borated Water up to 340°C	Crack Initiation & Growth	Stress Corrosion Cracking (SCC)	ASME Section XI, 1989 Edition. EPRI TR-105714. *Operating Experience* NRC IN 84-18. NRC IN 98-11.
B4.1	Plenum Cover and Plenum Cylinder	Plenum Cover and Plenum Cylinder	SS	Chemically Treated Borated Water up to 340°C	Crack Initiation and Growth	Irradiation Assisted Stress Corrosion Cracking (IASCC)	ASME Section XI, 1989 Edition. EPRI TR-105714. *Operating Experience* NRC IN 98-11.

Existing 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 (edition specified in 10 CFR 50.55a). Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC of SS components, 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:*** 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. The AMP must rely upon inservice inspection (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 by detection and sizing of cracks by 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. VT-3 may not be adequate to detect tight cracks. Creviced and other inaccessible 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:*** 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. 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 Element 4 should be further evaluated
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI (edition specified in 10 CFR 50.55a). Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC of SS components, 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:*** 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	Yes Element 4 should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B4.1	Plenum Cover and Plenum Cylinder	Plenum Cover and Plenum Cylinder	SS	Chemically Treated Borated Water up to 340°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) greatly reduces susceptibility to 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 on the intended function 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 IASCC can not occur without crack initiation and growth. VT-3 may not be adequate to detect tight cracks. The inspection technique, including the reliability in detecting the features of interest (crack appearance and size) in assuring the integrity of the component, should be specified. For example, enhancement of the visual VT-1 examination to achieve a 1/2-mil (0.0005 in.) resolution, with the conditions (lighting and surface cleanness) for the ISI bounded by those used to demonstrate the resolution of the inspection technique. Creviced regions are difficult to inspect visually and supplementary UT or other nondestructive examinations may be needed. 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 the only cracking presently directly attributable to IASCC in PWRs occurred in SS fuel cladding used in early reactors, susceptibility to this degradation will increase as plant operating time and hence accumulated fluence levels increase. The role of IASCC in the more recent cracking that has occurred in SS baffle/former bolts in U.S. plants as well as foreign plants (IN 98-11) has not yet been universally agreed upon.	
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.	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.	Yes TLAA

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B4.1	Plenum Cover and Plenum Cylinder	Plenum Cover and Plenum Cylinder	SS	Chemically Treated Borated Water up to 340°C	Cumulative fatigue damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B4.2.1 thru B4.2.4	CRA Guide Tube Assemblies	Tubes, Spacer Casting, Spider Casting, Bolts	SS, Cast Austenitic Stainless Steel (CASS), Nicrobraze, Ni alloy	Chemically Treated Borated Water up to 340°C	Crack initiation & growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B4.2.1 thru B4.2.4	CRA Guide Tube Assemblies	Tubes, Spacer Casting, Spider Casting, Bolts	SS, CASS, Nicrobraze, Ni alloy	Chemically Treated Borated Water up to 340°C	Crack initiation & growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B4.2.1 thru B4.2.4	CRA Guide Tube Assemblies	Tubes, Spacer Casting, Spider Casting, Bolts	SS, CASS, Nicrobraze, Ni alloy	Chemically Treated Borated Water up to 340°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.
B4.2.1 thru B4.2.3	CRA Guide Tube Assemblies	Tubes, Spacer Casting, Spider Casting	CASS	Chemically Treated Borated Water up to 340°C	Loss of Fracture Toughness	Thermal Aging and Neutron Irradiation Embrittle-ment	EPRI TR-106092. ASME Section XI, 1989 Edition.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Same as for the effect of Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.	*Same as for the effect of Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.*	Yes Element 4 should be further evaluated
Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.	*Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.*	Yes Element 4 should be further evaluated
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.	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.	Yes TLAA
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.	***(1) Scope of Program:*** The program includes determination of the susceptibility of CASS components to thermal aging based on casting method, Mo content, and percent ferrite, and for "potentially susceptible" components, to account for the synergistic loss of fracture toughness due to neutron embrittlement and thermal aging embrittlement, implement either a supplemental examination of the affected components as part of a 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. ***(2) Preventive Actions:*** The program provides no guidance on methods to mitigate thermal aging or neutron embrittlement. ***(3) Parameters Monitored/ Inspected:*** The program specifics depend on the neutron fluence and ferrite content of the component. The criteria in EPRI TR-106092, with some modifications approved by the NRC staff, can be used to determine whether CASS components are potentially susceptibility to thermal aging embrittlement based on the casting method, Mo content, and ferrite content. For low-Mo content (0.5 wt.% max.) steels, only static-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement, all centrifugal-cast and static-cast steels with ≤20% ferrite are not susceptible. For high-Mo content (2.0 to 3.0 wt.%) steels, static-cast steels with >14% ferrite and centrifugal-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement, static-cast steels with ≤14% ferrite and centrifugal-cast steels with ≤20% ferrite are not susceptible. Ferrite content will be calculated by the Hull's equivalent factors or a method producing an equivalent level of accuracy (±6% deviation between measured and calculated values). ***(4) Detection of Aging Effects:*** For all CASS components that have a neutron	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B4.2.1	CRA Guide Tube Assemblies	Tubes	SS, CASS, Nicrobraze	Chemically Treated Borated Water up to 340°C	Attrition	Wear	ASME Section XI, 1989 Edition.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) fluence of greater than 10^{17} n/cm^2 (E>1 MeV), implement a 10-year ISI program during the renewal period that includes supplemental inspection that covers portions of the susceptible components determined to be limiting from the standpoint of thermal aging susceptibility (Ferrite and Mo contents, casting process, and operating temperature), neutron fluence, and cracking susceptibility (applied stress, operating temperature, and environmental conditions). The inspection technique, including the reliability in detecting the features of interest (crack appearance and size) in assuring the integrity of the component, should be specified. For example, enhancement of the visual VT-1 examination to achieve a 1/2-mil (0.0005 in.) resolution, with the conditions (lighting and surface cleanness) for the ISI bounded by those used to demonstrate the resolution of the inspection technique. perform a component-specific , perform a component-specific evaluation including a mechanical loading assessment to determine the maximum tensile loading on the component under ASME Code Level A, B, C, and D conditions. If the loading is compressive or low enough (<5 ksi) to preclude fracture, then supplemental inspection of the component is not required. Failure to meet this criteria requires continued use of supplemental inspection program. For all CASS components that have a neutron fluence of less than 10^{17} n/cm^2 (E>1 MeV), implement the supplement inspection program if they are "potentially susceptible" to thermal aging; the existing ASME Section XI inspection requirements are considered adequate if the components are "not susceptible" to thermal aging. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 should provide timely detection of cracks. ***(6) Acceptance Criteria:*** Flaws detected in CASS components are evaluated in accordance with the applicable procedures of IWB-3600. ***(7) Corrective Actions:*** Repair is in conformance with IWA-4000 and IWB-4000, and replacement according to IWA-7000 and IWB-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:*** The AMP based on susceptibility determination, neutron fluence level, and supplemental inspection is effective in managing the effects of synergistic loss of fracture toughness due to neutron and thermal aging embrittlement on the intended function of CASS components.	
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	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B4.2.4	CRA Guide Tube Assemblies	Bolts	SS, Ni Alloy	Chemically Treated Borated Water up to 340°C	Loss of Preload	Stress Relaxation	ASME Section XI, 1989 Edition.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) 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 of 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:*** Acceptance standard for the inspection is 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 the AMP has been effective in managing the effects of wear on reactor internals.	
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 inservice inspection (ISI) to monitor the effects of stress relaxation on the intended function of the component by detection and sizing of cracks that could be formed by excessive vibration etc. if the preload is lost. 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, it may be adequate if there is sufficient redundancy that loss of some bolting between inspections is acceptable. In some cases additional inspection may be required. ***(4) Detection of Aging Effects:*** As part of the AMP it may be possible to identify acceptable levels of preload and demonstrate under the fluence of interest whether less of acceptable preload is likely. VT-3 may not be adequate to detect tight cracks. Creviced regions are difficult to inspect visually. Supplementary inspections by techniques such as ultrasonic testing (UT) or other nondestructive methods may be needed to detect cracking in inaccessible regions.***(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:*** There are no reports of stress relaxation producing damage in reactor vessel internals.	Yes Elements 3 and 4 should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B4.2.1 thru B4.2.4	CRA Guide Tube Assemblies	Tubes, Spacer Casting, Spider Casting, Bolts	SS, CASS, Nicrobraze, Ni alloy	Chemically Treated Borated Water up to 340°C	Cumulative fatigue damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B4.3.1 thru B4.3.3	Upper Grid Assembly	Upper Grid Rib Section, Upper Grid Assembly Bolts, Fuel Guide Pads	SS, Ni alloy	Chemically Treated Borated Water up to 340°C	Crack initiation & growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B4.3.1 thru B4.3.3	Upper Grid Assembly	Upper Grid Rib Section, Upper Grid Assembly Bolts, Fuel Guide Pads	SS, Ni alloy	Chemically Treated Borated Water up to 340°C	Crack initiation & growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B4.3.1 thru B4.3.3	Upper Grid Assembly	Upper Grid Rib Section, Upper Grid Assembly Bolts, Fuel Guide Pads	SS, Ni alloy	Chemically Treated Borated Water up to 340°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.
B4.3.1 thru B4.3.3	Upper Grid Assembly	Upper Grid Rib Section, Upper Grid Assembly Bolts, Fuel Guide Pads	SS, Ni alloy	Chemically Treated Borated Water up to 340°C	Cumulative fatigue damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B4.3.1 thru B4.3.3	Upper Grid Assembly	Upper Grid Rib Section, Upper Grid Assembly Bolts, Fuel Guide Pads	SS, Ni alloy	Chemically Treated Borated Water up to 340°C	Loss of Fracture Toughness	Neutron Irradiation Embrittle-ment	ASME Section XI, 1989 Edition.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Same as for the effect of Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.	*Same as for the effect of Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.*	Yes Element 4 should be further evaluated
Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.	*Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.*	Yes Element 4 should be further evaluated
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.	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.	Yes TLAA
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
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 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, category B-N-3 specifies visual VT-3 examination of all accessible surfaces of reactor internals. ***(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 may not be 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.	Yes Elements 3 and 4 should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B4.3.3	Upper Grid Assembly	Plenum Rib Pads	SS	Chemically Treated Borated Water up to 340°C	Attrition	Wear	ASME Section XI, 1989 Edition.
B4.3.2	Upper Grid Assembly	Upper Grid Assembly Bolts	SS, Ni alloy	Chemically Treated Borated Water up to 340°C	Loss of Preload	Stress Relaxation	ASME Section XI, 1989 Edition.
B4.4.1, B4.4.2	Core Support Assembly	Core Support Shield, Core Support Shield Flange	SS	Chemically Treated Borated Water up to 340°C	Crack initiation & growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B4.4.1, B4.4.2	Core Support Assembly	Core Support Shield, Core Support Shield Flange	SS	Chemically Treated Borated Water up to 340°C	Crack initiation & growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B4.4.1, B4.4.2	Core Support Assembly	Core Support Shield, Core Support Shield Flange	SS	Chemically Treated Borated Water up to 340°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.
B4.4.1 thru B4.4.3	Core Support Assembly	Core Support Shield, Core Support Shield Flange, Core Support Shield to Core Barrel Bolts	SS	Chemically Treated Borated Water up to 340°C	Loss of Fracture Toughness	Neutron Irradiation Embrittle-ment	ASME Section XI, 1989 Edition.
B4.4.1, B4.4.2	Core Support Assembly	Core Support Shield, Core Support Shield Flange	SS	Chemically Treated Borated Water up to 340°C	Cumulative fatigue damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B4.4.2	Core Support Assembly	Core Support Shield Flange	SS	Chemically Treated Borated Water up to 340°C	Attrition	Wear	ASME Section XI, 1989 Edition.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) ***(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 recorded have been definitely attributed to irradiation embrittlement.	
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*Same as for the effect of Wear on Item B4.2.1 CRA guide tubes.*	No
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*Same as for the effect of Stress Relaxation on Item B4.2.2 CRA guide tubes bolts.*	Yes Elements 3 and 4 should be further evaluated
Same as for the effect of Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.	*Same as for the effect of Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.*	Yes Element 4 should be further evaluated
Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.	*Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.*	Yes Element 4 should be further evaluated
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.	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.	Yes TLAA
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*Same as for the effect of Neutron Irradiation Embrittlement on Items B4.2.1-B4.2.3 upper grid assembly rib section, bolts, and plenum rib pads.*	Yes Elements 3 and 4 should be further evaluated
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*Same as the effect of Wear on Item B4.2.1, CRA guide tubes*	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B4.5	Vent Valve Assembly	Vent Valve Assembly	Ni alloy, SS, CASS, Stellite, Martensitic SS	Chemically Treated Borated Water up to 340°C	Crack initiation & growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B4.5	Vent Valve Assembly	Vent Valve Assembly	Ni alloy, SS, CASS, Stellite, Martensitic SS	Chemically Treated Borated Water up to 340°C	Crack initiation & growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B4.5	Vent Valve Assembly	Vent Valve Assembly	Ni alloy, SS, CASS, Stellite, Martensitic SS	Chemically Treated Borated Water up to 340°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.
B4.5	Vent Valve Assembly	Vent Valve Assembly	Ni alloy, SS, Stellite, Martensitic SS	Chemically Treated Borated Water up to 340°C	Crack initiation & growth	Neutron Irradiation Embrittle-ment	ASME Section XI, 1989 Edition.
B4.5	Vent Valve Assembly	Vent Valve Assembly	CASS	Chemically Treated Borated Water up to 340°C	Loss of Fracture Toughness	Thermal Aging and Neutron Irradiation Embrittle-ment	EPRI TR-106092. ASME Section XI, 1989 Edition.
B4.6.1 thru B4.6.3	Core Barrel Assembly	Core Barrel (CB) (Baffle & Former Plates), CB Bolts, CB to Thermal Shield Bolts	SS, Ni alloy (bolts)	Chemically Treated Borated Water up to 340°C	Crack initiation & growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B4.6.1 thru B4.6.3	Core Barrel Assembly	Core Barrel (CB) (Baffle & Former Plates), CB Bolts, CB to Thermal Shield Bolts	SS, Ni alloy (bolts)	Chemically Treated Borated Water up to 340°C	Crack initiation & growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B4.6.1 thru B4.6.3	Core Barrel Assembly	Core Barrel (CB) (Baffle & Former Plates), CB Bolts, CB to Thermal Shield Bolts	SS, Ni alloy (bolts)	Chemically Treated Borated Water up to 340°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.
B4.6.1 thru B4.6.3	Core Barrel Assembly	Core Barrel (CB) (Baffle & Former Plates), CB Bolts, CB to Thermal Shield Bolts	SS, Ni alloy (bolts)	Chemically Treated Borated Water up to 340°C	Loss of Fracture Toughness	Neutron Irradiation Embrittle-ment	ASME Section XI, 1989 Edition.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Same as for the effect of Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.	*Same as for the effect of Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.*	Yes Element 4 should be further evaluated
Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.	*Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.*	Yes Element 4 should be further evaluated
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.	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.	Yes TLAA
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*Same as for the effect of Neutron Irradiation Embrittlement on Items B4.2.1-B4.2.3 upper grid assembly rib section, bolts, and plenum rib pads.*	Yes Elements 3 and 4 should be further evaluated
Same as for the effect of Thermal Aging and Neutron Embrittlement on Items B4.2.1- B1.2.3 CRA guide tubes, spacer casting, and spider casting.	*Same as for the effect of Thermal Aging and Neutron Embrittlement on Items B4.2.1- B1.2.3 CRA guide tubes, spacer casting, and spider casting.*	No
Same as for the effect of Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.	*Same as for the effect of Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.*	Yes Element 4 should be further evaluated
Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.	*Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.*	Yes Element 4 should be further evaluated
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.	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.	Yes TLAA
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*Same as for the effect of Neutron Irradiation Embrittlement on Items B4.2.1-B4.2.3 upper grid assembly rib section, bolts, and plenum rib pads.*	Yes Elements 3 and 4 should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B4.6.1 thru B4.6.3	Core Barrel Assembly	Core Barrel (CB) (Baffle & Former Plates), CB Bolts, CB to Thermal Shield Bolts	SS, Ni alloy (bolts)	Chemically Treated Borated Water up to 340°C	Cumulative fatigue damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B4.6.2, B4.6.3	Core Barrel Assembly	CB Bolts, CB to Thermal Shield Bolts	SS, Ni alloy	Chemically Treated Borated Water up to 340°C	Loss of Preload	Stress Relaxation	ASME Section XI, 1989 Edition.
B4.7.1 thru B4.7.9	Lower Grid Assembly (LGA)	Upper Grid, Fuel Guide Pads, Lower Grid, Support Columns, Cylinder Guide Blocks, LGA Bolts, Core Barrel Bolts, Flow Distributor Bolts, Thermal Shield Bolts	SS , Ni Alloy (bolts)	Chemically Treated Borated Water up to 340°C	Crack initiation & growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B4.7.1 thru B4.7.9	Lower Grid Assembly (LGA)	Upper Grid, Fuel Guide Pads, Lower Grid, Support Columns, Cylinder Guide Blocks, LGA Bolts, Core Barrel Bolts, Flow Distributor Bolts, Thermal Shield Bolts	SS , Ni Alloy (bolts)	Chemically Treated Borated Water up to 340°C	Crack initiation & growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B4.7.1 thru B4.7.9	Lower Grid Assembly (LGA)	Upper Grid, Fuel Guide Pads, Lower Grid, Support Columns, Cylinder Guide Blocks, LGA Bolts, Core Barrel Bolts, Flow Distributor Bolts, Thermal Shield Bolts	SS , Ni Alloy (bolts)	Chemically Treated Borated Water up to 340°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*Same as for the effect of Stress Relaxation on Item B4.2.2 CRA guide tubes bolts.*	Yes Elements 3 and 4 should be further evaluated
Same as for the effect of Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.	*Same as for the effect of Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.*	Yes Element 4 should be further evaluated
Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.	*Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.*	Yes Element 4 should be further evaluated
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.	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.	Yes TLAA

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B4.7.1 thru B4.7.9	Lower Grid Assembly (LGA)	Upper Grid, Fuel Guide Pads, Lower Grid, Support Columns, Cylinder Guide Blocks, LGA Bolts, Core Barrel Bolts, Flow Distributor Bolts, Thermal Shield Bolts	SS, Ni Alloy (bolts)	Chemically Treated Borated Water up to 340°C	Loss of Fracture Toughness	Neutron Irradiation Embrittle-ment	ASME Section XI, 1989 Edition.
B4.7.6 thru B4.7.9	Lower Grid Assembly (LGA)	LGA Bolts, Core Barrel Bolts, Flow Distributor Bolts, Thermal Shield Bolts	SS, Ni Alloy	Chemically Treated Borated Water up to 340°C	Loss of Preload	Stress Relaxation	ASME Section XI, 1989 Edition.
B4.7.2, B4.7.5	Lower Grid Assembly (LGA)	Fuel Guide Pads, Cylinder Guide Blocks	SS	Chemically Treated Borated Water up to 340°C	Attrition	Wear	ASME Section XI, 1989 Edition.
B4.7.1 thru B4.7.9	Lower Grid Assembly (LGA)	Upper Grid, Fuel Guide Pads, Lower Grid, Support Columns, Cylinder Guide Blocks, LGA Bolts, Core Barrel Bolts, Flow Distributor Bolts, Thermal Shield Bolts	SS, Ni Alloy (bolts)	Chemically Treated Borated Water up to 340°C	Cumulative fatigue damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B4.8	Flow Distributor	-	SS	Chemically Treated Borated Water up to 340°C	Crack initiation & growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B4.8	Flow Distributor	-	SS	Chemically Treated Borated Water up to 340°C	Crack initiation & growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.

Existing 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.	*Same as for the effect of Neutron Irradiation Embrittlement on Items B4.2.1-B4.2.3 upper grid assembly rib section, bolts, and plenum rib pads.*	Yes Elements 3 and 4 should be further evaluated
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*Same as for the effect of Stress Relaxation on Item B4.2.2 CRA guide tubes bolts.*	Yes Elements 3 and 4 should be further evaluated
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*Same as the effect of Wear on Item B4.2.1, CRA guide tubes*	No
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Same as for the effect of Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.	*Same as for the effect of Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.*	Yes Element 4 should be further evaluated
Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.	*Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.*	Yes Element 4 should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B4.8	Flow Distributor	-	SS	Chemically Treated Borated Water up to 340°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.
B4.8	Flow Distributor	-	SS	Chemically Treated Borated Water up to 340°C	Loss of Fracture Toughness	Neutron Irradiation Embrittle-ment	ASME Section XI, 1989 Edition.
B4.9	Thermal Shield	-	SS	Chemically Treated Borated Water up to 340°C	Crack initiation & growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B4.9	Thermal Shield	-	SS	Chemically Treated Borated Water up to 340°C	Crack initiation & growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B4.9	Thermal Shield	-	SS	Chemically Treated Borated Water up to 340°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.
B4.9	Thermal Shield	-	SS	Chemically Treated Borated Water up to 340°C	Loss of Fracture Toughness	Neutron Irradiation Embrittle-ment	ASME Section XI, 1989 Edition.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.	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.	Yes TLAA
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*Same as for the effect of Neutron Irradiation Embrittlement on Items B4.2.1-B4.2.3 upper grid assembly rib section, bolts, and plenum rib pads.*	Yes Elements 3 and 4 should be further evaluated
Same as for the effect of Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.	*Same as for the effect of Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.*	Yes Element 4 should be further evaluated
Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.	*Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.*	Yes Element 4 should be further evaluated
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.	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.	Yes TLAA
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*Same as for the effect of Neutron Irradiation Embrittlement on Items B4.2.1-B4.2.3 upper grid assembly rib section, bolts, and plenum rib pads.*	Yes Elements 3 and 4 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 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

C1.2 Recirculation Pump

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

C1.3 Safety & Relief Valves

- C1.3.1 Valve 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 first isolation valve outside of containment or to the first anchor point. 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. All systems, structures, and components in the reactor coolant pressure boundary are classified as Group A Quality Standards. The aging management program for containment isolation valves is reviewed in Table V C.

The pump and valve internals are considered to be active components. They perform their intended functions with moving parts or with a change in configuration and are not subject to aging management review pursuant to 10 CFR 54.21(a)(1)(i).

System Interfaces

The systems that interface with the reactor coolant pressure boundary include the reactor pressure vessel (Table IV A1), containment isolation components (Table V C), emergency core cooling system (Table V D2), main steam system (Table VIII B2), and feedwater system (Table VIII D2).

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 Steam	Wall Thinning	Erosion/ Corrosion (E/C)	NUREG-1344. EPRI NSAC-202L-R2. NRC IN 93-21. EPRI TR-103515. BWRVIP-29. *Operating Experience* NRC BI 87-01. NRC GL 89-08. NRC IN 89-53. NRC IN 91-18. NRC IN 91-18 S1. NRC IN 91-28. NRC IN 92-35. NRC IN 95-11. NRC IN 97-84.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Program delineated in NUREG-1344 for single phase lines and implemented through NRC Generic Letter 89-08; CHECWORKS Code; EPRI guidelines of NSAC-202L-R2 for effective erosion/corrosion program; and water chemistry program based on EPRI guidelines in TR-103515 and BWRVIP-29 for water chemistry in BWRs.	***(1) Scope of Program:*** The AMP includes NUMARC program delineated in Appendix A of NUREG-1344 and implemented through NRC Generic Letter (GL) 89-08; CHECWORKS computer Code; and EPRI guidelines of NSAC-202L-R2. The program includes the following recommendations: (a) conduct appropriate analysis and limited baseline inspection, (b) determine the extent of thinning and repair/replace components, and (c) perform follow-up inspections to confirm or quantify and take longer term corrective actions. Technical aspects of the CHECWORKS Code, including the parameters and inputs, are acceptable. However, the EPRI guidance document NSAC-202L-R2 (April 1999) is too general to ensure applicant's flow-accelerated corrosion program will be effective in managing aging in safety-related systems. ***(2) Preventive Actions:*** The rate of E/C is affected by piping material, geometry and hydrodynamic conditions, and operating conditions such as temperature, pH, and dissolved oxygen content. Mitigation is by selecting material considered resistant to E/C, adjusting water chemistry and operating conditions, and improving hydrodynamic conditions through design modifications. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of E/C on the intended function of piping by measuring wall thickness by nondestructive examination and performing analytical evaluations. The inspection program delineated in NUREG-1344 requires ultrasonic or radiographic testing of 10 most susceptible locations and 5 additional locations based on unique operating conditions or special considerations. For each location outside the acceptance guidelines, the inspection sample is expanded based on engineering judgment. Analytical models such as those incorporated into the CHECWORKS code are used to predict E/C in piping systems based on specific plant data including material and hydrodynamic and operating conditions. The inspection data are used to calibrate and benchmark the models and code. ***(4) Detection of Aging Effects:*** Aging degradation of piping and fittings occurs by wall thinning; extent and schedule of inspection assure detection of wall thinning before the loss of intended function of the piping. ***(5) Monitoring and Trending:*** Inspection schedule of NUREG-1344 and EPRI guidelines should provide for timely detection of leakage. Inspections and analytical evaluations are performed during plant outage. If analysis shows unacceptable conditions, inspection of initial sample is performed within 6 months. ***(6) Acceptance Criteria:*** Inspection results are used to calculate number of refueling or operating cycles remaining before the component reaches Code minimum allowable wall thickness. If calculations indicate that an area will reach Code minimum (plus 10% margin), the component must be repaired or replaced. However, NRC staff has identified the problems in implementing E/C program that pertain to weakness or errors in (a) using predictive models, (b) calculating minimum wall thickness acceptance criteria, (c) analyzing the results of UT examinations, and (d) assessment of E/C program activities (NRC	Yes, Element 1 should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C1.1.1	Piping & Fittings	Main Steam	CS SA106-Gr B, SA333-Gr 6, SA155-Gr KCF70	288°C Steam	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. ANSI B31.1. GSI-190.
C1.1.2	Piping & Fittings	Feedwater	CS SA106-Gr B, SA333-Gr 6, SA155-Gr KCF70	Up to 225°C, Oxygenated Water	Wall Thinning	Erosion/ Corrosion	*Same as the effect of E/C on Item C1.1.1 Main Steam Line Piping and Fittings*
C1.1.2	Piping & Fittings	Feedwater	CS SA106-Gr B, SA333-Gr 6, SA155-Gr KCF70	Up to 225°C, Oxygenated Water	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. ANSI B31.1. GSI-190.
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 Oxygenated Water or Steam	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. ANSI B31.1. GSI-190.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) Information Notice IN 93-21). ***(7) Corrective Actions:*** Prior to service, repair or replace to meet the requirements of NUREG-1344. Follow-up inspections are performed to confirm or quantify thinning and take longer term corrective actions such as adjustment of chemistry and operating parameters, or selection of materials resistant to E/C. However, NRC staff has identified weakness or errors in (a) dispositioning components after reviewing the results of the inspection analysis, and (b) repairing or replacing components that failed to meet the acceptance criteria (IN 93-21). ***(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:*** Wall-thinning problems in single-phase systems have occurred in feedwater and condensate systems (NRC Bulletin No. 87-01, INs 81-28, 92-35, 95-11), and in two-phase piping in extraction steam lines (INs 89-53, 97-84) and moisture separation reheater and feedwater heater drains (INs 89-53, 91-18, 93-21, 97-84). The AMP outlined in NUREG-1344 and EPRI report and implemented through GL 89-08 has provided effective means of ensuring the structural integrity of all high-energy carbon steel systems.	
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or ANSI B31.1, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Same as the effect of Erosion/Corrosion on Item C1.1.1 Main Steam Line Piping and Fittings.	*Same as the effect of Erosion/Corrosion on Item C1.1.1 Main Steam Line Piping and Fittings.*	No
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or ANSI B31.1, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or ANSI B31.1, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C1.1.5 thru 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 Oxygenated Water 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 Edition. BWRVIP-29. EPRI TR-103515. BWRVIP-42. BWRVIP-56. BWRVIP-61. BWRVIP-75. *Supporting BWRVIP:* BWRVIP-03.
C1.1.5, C1. 1.11	Piping & Fittings	RHR, LPCI, LPCS, HPCS, Lines to IC, Lines to RWC & SLC Systems	CASS	288°C Oxygenated Water or Steam	Loss of Fracture Toughness	Thermal Aging Embrittle-ment	EPRI TR-106092. ASME Section XI, 1989 Edition.

Existing 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 (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination categories B-J for pressure retaining welds in piping and B-F for pressure retaining dissimilar metal welds, and testing category B-P for system leakage. BWRVIP guideline is under staff review. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-103515 and BWRVIP-29 to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program is focused on managing and implementing the counter measures 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:*** 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 (SI). Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-103515 and BWRVIP-29 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 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. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with applicable approved BWRVIP guideline. ***(6) Acceptance Criteria:*** Any IGSCC degradation is evaluated in accordance with applicable approved BWRVIP guideline. ***(7) Corrective Actions:*** The corrective action proposed by the BWRVIP is under staff review. ***(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. Significant cracking has occurred in RHR system and reactor water cleanup system piping welds.	Yes, BWRVIP Guideline
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,	***(1) Scope of Program:*** The program includes determination of the susceptibility of CASS components to thermal aging based on casting method, Mo content, and percent ferrite, and for potentially susceptible components aging management is accomplished either through volumetric examination or plant/component-specific flaw tolerance evaluation. ***(2) Preventive Actions:*** The program provides no guidance on methods to mitigate thermal aging. ***(3) Parameters Monitored/ Inspected:*** Based on the criteria in EPRI TR-106092, with some modifications, the susceptibility to thermal aging embrittlement of CASS piping is determined in terms of casting method, Mo content, and ferrite content. For low-Mo content (0.5 wt.% max.) steels, only static-cast steels	No

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
(continued from previous page) screening for susceptibility to thermal aging is not required. Also, the existing ASME Section XI inspection requirements, including the alternative requirements of ASME Code Case N-481 for pump casings, are considered adequate for all pump casings and valve bodies.	*(continued from previous page)* with >20% ferrite are potentially susceptible to thermal embrittlement, all centrifugal-cast and static-cast steels with ≤20% ferrite are not susceptible. For high-Mo content (2.0 to 3.0 wt.%) steels, static-cast steels with >14% ferrite and centrifugal-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement, static-cast steels with ≤14% ferrite and centrifugal-cast steels with ≤20% ferrite are not susceptible. Ferrite content will be calculated by the Hull's equivalent factors or a method producing an equivalent level of accuracy (±6% deviation between measured and calculated values). For pump casings and valve bodies, screening for susceptibility to thermal aging is not required. ***(4) Detection of Aging Effects:*** For "not susceptible" piping, no additional inspection or evaluations are required to demonstrate that the material has adequate fracture toughness. For "potentially susceptible" piping, because the base metal does not receive periodic inspection per ASME Section XI, volumetric examination should be performed on the base metal, with the scope of the inspection covering the portions determined to be limiting from the standpoint of applied stress, operating time, and environmental considerations. Alternatively, a plant/component- specific flaw tolerance evaluation, using specific geometry and stress information, can be used to demonstrate that the thermally-embrittled material has adequate toughness. Current volumetric examination methods are inadequate for reliable detection of cracks in CASS components; the performance of the equipment and techniques when developed, should be demonstrated through the program consistent with the ASME Section XI, Appendix VIII. For all pump casings and valve bodies, the existing ASME Section XI inspection requirements, including the alternative requirements of ASME Code Case N-481 for pump casings, are considered adequate. For valve bodies less than NPS 4, the adequacy of inservice inspection according to ASME Section XI has been demonstrated by a NRC performed bounding fracture analysis. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 should provide timely detection of cracks. ***(6) Acceptance Criteria:*** Flaws detected in CASS components are evaluated in accordance with the applicable procedures of IWB-3500. If aging management is accomplished through plant/component-specific flaw tolerance evaluation, e.g., for potentially susceptible piping, flaw evaluation for piping with <25% ferrite is performed according to the principles associated with IWB-3640 procedures for submerged arc welds (SAW), disregarding the Code restriction of 20% ferrite in IWB-3641(b)(1). Flaw evaluation for piping with >25% ferrite is performed on a case-by-case basis using fracture toughness data provided by the applicant. ***(7) Corrective Actions:*** Repair is in conformance with IWA-4000 and IWB-4000, and replacement according to IWA-7000 and IWB-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	

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C1.1.5, C1. 1.11	Piping & Fittings	Recirculation, Lines to RWC and SLC Systems	SS	288°C, Oxygenated Water	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. ANSI B31.1. GSI-190.
C1.1.6 thru C1. 1.10	Piping & Fittings	RHR, LPCI, LPCS, HPCS, IC	CS, SS	288°C Oxygenated Water or Steam	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. ANSI B31.1. GSI-190.
C1.2.1 thru C1.2.3	Recirculation Pump	Bowl/Casing, Cover, Seal Flange	CASS, SS	288°C, Oxygenated Water	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. ANSI B31.1. GSI-190.
C1.2.1, C1.2.2	Recirculation Pump	Bowl/Casing, Cover	CASS (SA351 CF-8 or CF-8M)	288°C, Oxygenated Water	Loss of Fracture Toughness	Thermal Aging Embrittle-ment	EPRI TR-106092. ASME Section XI, 1989 Edition.
C1.2.1	Recirculation Pump	Bowl/Casing	CASS, SS	288°C, Oxygenated Water	Crack Initiation and Growth	SCC, IGSCC	ASME Section XI, 1989 Edition. NUREG-0313, Rev. 2. NRC GL 88-01. NRC GL 88-01, S 1. BWRVIP-29. EPRI TR-103515.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** The AMP based on susceptibility determination and examination requirements or flaw tolerance evaluations of potentially susceptible components is effective in managing the effects of thermal aging on the intended function of CASS components.	
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or ANSI B31.1, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or ANSI B31.1, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or ANSI B31.1, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Same as for the effect of Thermal Aging Embrittlement on piping and fittings in various reactor coolant pressure boundary systems Items C1.1.5 - C1.1.11.	*Same as for the effect of Thermal Aging Embrittlement on piping and fittings in various reactor coolant pressure boundary systems Items C1.1.5 - C1.1.11.*	No
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 (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination categories B-L-1 for pump casing welds and B-L-2 for pump casing, and testing category B-P for system leakage. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-103515 and BWRVIP-29 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 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:*** 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. However, high-carbon grades of cast SS, e.g., CF-8 and CF-8M may be 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 the intended function of the pump by detection and sizing of cracks by ISI. The inspection requirements of pump casing welds are delineated in GL 88-01. Inspection requirements of Table IWB 2500-1, examination category B-L-2 specifies visual VT-3 examination of internal surfaces of the pump. Inspection requirements of testing category B-P conducted according to IWA-5000 specify visual VT-2 (IWA-5240) examination of all pressure	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 Oxygenated Water and/or Steam at 288°C	Attrition	Wear	NUREG-1339. EPRI NP-5769. NRC GL 91-17. IEB 82-02. ASME Section XI, 1989 Edition.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) 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 EPRI guidelines in TR-103515 and BWRVIP-29 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; IWB-3518 for volumetric examination of welds and IWB-3519 for visual examination of pump internal surfaces. 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, e.g., susceptible material, significant tensile stress, and an aggressive environment, and has provided effective means of ensuring structural integrity of the primary coolant pressure boundary.	
Recommendations for a comprehensive bolting integrity program delineated in NUREG-1339 on resolution of Generic Safety Issue 29 and implemented through NRC Generic Letter 91-17; additional details on bolting integrity outlined in EPRI NP-5769; and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination categories B-G-1 or B-G-2 for pressure retaining bolting, and category B-P for system leakage.	***(1) Scope of Program:*** The staff guidance of NRC Generic Letter (GL) 91-17 provides assurance that plant specific comprehensive bolting integrity programs have been implemented to ensure bolting reliability. The NRC staff recommendations and guidelines for a comprehensive bolting integrity program is delineated in NUREG-1339, and the industry's technical basis for the program is outlined in EPRI NP-5769. ***(2) Preventive Actions:*** Selection of bolting material and the use of lubricants and sealants in accordance with guidelines of EPRI NP-5769 and additional requirements of NUREG 1339, prevent or mitigate degradation and failure of all safety-related closure bolting. ***(3) Parameter Monitored/ Inspected:***	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C1.2.4	Recirculation Pump	Closure Bolting	HSLAS SA193 GrB7	Air, Leaking Oxygenated Water and/or Steam at 288°C	Loss of Preload	Stress Relaxation	NUREG-1339. EPRI NP-5769. NRC GL 91-17. IEB 82-02. ASME Section XI, 1989 Edition.
C1.2.4	Recirculation Pump	Closure Bolting	HSLAS SA193 GrB7	Air, Leaking Oxygenated Water and/or Steam at 288°C	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. ANSI B31.1. GSI-190.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) The AMP monitors the effects of aging degradation on the intended function of closure bolting by detection of coolant leakage, and by detection and sizing of cracks by inservice inspection (ISI). Inspection requirements of ASME Section XI, Table IWB 2500-1, examination category B-G-1 for bolting greater than 2 in. in diameter specify volumetric examination of studs and bolts, and visual VT-1 examination of surfaces of nuts, washers, bushings, and flanges. Examination category B-G-2 for bolting 2 in. or smaller specifies only visual VT-1 examination of surfaces of bolts, studs, and nuts. However, because most failures have occurred in fasteners 2 in. or smaller, based on IE Bulletin 82-02, enhanced inspection and improved techniques are recommended. Inspection requirements of ASME Section XI 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 of the closure bolting due to crack initiation, loss of prestress, or attrition of the closure bolting would result in leakage. The extent and schedule of inspection assure detection of aging degradation before the loss of intended function of closure bolting. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI are effective and adequate for timely detection of cracks and leakage. ***(6) Acceptance Criteria:*** Any cracks in closure bolting are evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3515 and 3517. ***(7) Corrective Actions:*** Repair and replacement is in conformance with IWB-4000 and guidelines and recommendations of EPRI NP-5769. ***(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 bolting integrity programs developed and implemented in accordance with commitments made in response to NRC communications on bolting events have provided effective means of ensuring bolting reliability.	
Same as the effect of wear on Item C1.2.4 Closure Bolting for Recirculation Pump.	*Same as 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, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or ANSI B31.1, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C1.3.1	Valves (Check, Control, Hand, Motor-Operated, and Relief Valves)	Body	CS	288°C, Oxygenated Water	Wall Thinning	Erosion/ Corrosion	NUREG-1344. EPRI NSAC-202L-R2. NRC IN 93-21. EPRI TR-103515. BWRVIP-29.
C1.3.1, C1.3.2	Valves (Check, Control, Hand, MO, and Relief Valves)	Body, Bonnet	CASS	288°C, Oxygenated Water	Loss of Fracture Toughness	Thermal Aging Embrittle-ment	EPRI TR-106092. ASME Section XI, 1989 Edition.
C1.3.1, C1.3.2	Valves (Check, Control, Hand, Motor-Operated, and Relief Valves)	Valve Body, Bonnet	CASS, SS	288°C, Oxygenated Water	Crack Initiation and Growth	SCC, IGSCC	NUREG-0313, Rev. 2. ASME Section XI, 1989 Edition. NRC GL 88-01. NRC GL 88-01, S 1. BWRVIP-29. EPRI TR-103515.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Same as for the effect of Erosion/Corrosion on Item C1.1.1 main steam piping and fittings.	*Same as for the effect of Erosion/Corrosion on Item C1.1.1 main steam piping and fittings.*	Yes, Element 1 should be further evaluated
Same as for the effect of Thermal Aging Embrittlement on piping and fittings in various reactor coolant pressure boundary systems Items C1.1.5 - C1.1.11.	*Same as for the effect of Thermal Aging Embrittlement on piping and fittings in various reactor coolant pressure boundary systems Items C1.1.5 - C1.1.11.*	No
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 (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination categories B-M-1 for valve body welds and B-M-2 for valve body, and testing category B-P for system leakage. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-103515 and BWRVIP-29 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:*** 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. However, high-carbon grades of cast SS, e.g., CF-8 and CF-8M may be 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 GL 88-01. Inspection requirements of Table IWB 2500-1, examination category B-M-2 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 (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 EPRI guidelines in TR-103515 and BWRVIP-29 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 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	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C1.3.3, C1.3.4	Valves	Seal Flange, Closure Bolting	Flange: CS, SS Bolting: HSLAS	Air, Leaking Oxygenated Water and/or Steam at 288°C	Attrition	Wear	NUREG-1339. EPRI NP-5769. NRC GL 91-17. IEB 82-02. ASME Section XI, 1989 Edition.
C1.3.1 thru C1.3.3	Valves (Check, Control, Hand, Motor-Operated, and Relief Valves)	Valve Body, Bonnet, Seal Flange	CS, CASS, SS	288°C, Oxygenated Water	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. ANSI B31.1. GSI-190.
C1.3.4	Valves	Closure Bolting	HSLAS SA193 GrB7	Air, Leaking Oxygenated Water and/or Steam at 288°C	Loss of Preload	Stress Relaxation	NUREG-1339. EPRI NP-5769. NRC GL 91-17. IEB 82-02. ASME Section XI, 1989 Edition.
C1.3.4	Valves	Closure Bolting	HSLAS SA193 GrB7	Air, Leaking Oxygenated Water and/or Steam at 288°C	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. ANSI B31.1. GSI-190.
C1.4.1 thru 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: demineralized water	Loss of Material	Crevice and Pitting Corrosion	ASME Section XI, 1989 Edition. ASME OM S/G, Pt 2. NRC GL 89-13. Plant Technical Specifications.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) 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.	
Same as the effect of wear on Item C1.2.4 Closure Bolting for Recirculation Pump.	*Same as 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, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or ANSI B31.1, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Same as the effect of wear on Item C1.2.4 Closure Bolting for Recirculation Pump.	*Same as 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, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or ANSI B31.1, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Detection of reactor coolant leakage by radiation and temperature monitors in component cooling system; inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWC 2500-1, examination category C-H for pressure retaining Class 2 components; performance testing in accordance with ASME OM Standards and Guides, Part 2 to ensure	***(1) Scope of Program:*** The program includes monitoring and control of ECCS water chemistry to minimize exposure to aggressive environments, and performance testing in accordance with ASME OM-Standards and Guides, Part 2 provides assurance that the heat exchanger serviced by the closed-cycle cooling water system is performing its function acceptably. ***(2) Preventive Actions:*** Monitor and control of suppression pool and component cooling system water chemistries based on the plant technical specifications to minimize impurities,	No

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
(continued from previous page) that the heat exchanger serviced by the closed-cycle cooling water system is performing its function acceptably; and control of system water chemistries in suppression pool and component cooling water based on the plant technical specifications to minimize exposure to aggressive environments.	*(continued from previous page)* and timely corrective action prevent or mitigate corrosion. The parameters monitored include halogens, sulfates, oxygen, and pH in the suppression pool water, and in addition to these, dissolved copper and iron, suspended solids, and gamma activity in the component cooling water. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of corrosion by surveillance program to detect coolant leakage and inservice testing to evaluate component performance. The parameters monitored are directly related to corrosion, e.g., dissolved iron and copper, and by detection of leakage by radiation and temperature monitors, in the component cooling system. Also, inspection requirements of ASME Section XI specify visual VT-2 (IWA-5240) examination during system leakage test and hydrostatic test of all pressure retaining Class 2 components according to Table IWC 2500-1 category C-H. Performance testing is conducted in accordance with ASME OM S/G, Part 2. ***(4) Detection of Aging Effects:*** Degradation of component due to corrosion would result in leakage of coolant or degradation of component performance. Monitoring of radiation and suspended solids would detect leakage; extent and schedule of inspection/ testing assure detection of corrosion before the loss of intended function of the component. ***(5) Monitoring and Trending:*** Results from performance tests to verify the heat transfer capabilities are trended. Also, based on the recommendations of NRC GL 89-13 or its equivalent, if adequacy of cooling water chemistry control can not be confirmed, implement Action III of GL 89-13 to include inspection and maintenance program for closed-cycle cooling water system to ensure that corrosion, erosion, and protective coating failure can not degrade the performance of safety-related systems serviced by closed-cycle cooling water. ***(6) Acceptance Criteria:*** Heat exchanger performance test results are evaluated in accordance with the guidelines of ASME OM S/G Part 2. Any relevant conditions related to corrosion causing leakage of ECCS water are compared with established acceptable limits. Results of Section XI leakage tests are evaluated in accordance with IWC-3100 and acceptance standards of IWC-3400 and IWB-3516. ***(7) Corrective Actions:*** Root cause evaluation and appropriate corrective action taken when acceptable limits are exceeded or leakage is detected. Repair is in conformance with IWA-4000 and replacement is in 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:*** Operating plant experience with this AMP indicates timely detection of corrosion in ECCS heat exchangers.	

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

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 Chemical and Volume Control System or High-Pressure Injection System (Makeup & Letdown Functions)
- C2.2.4 Sampling System
- C2.2.5 Drains and Instrument Lines
- C2.2.6 Nozzles and Safe Ends
- C2.2.7 Small-Bore Piping in Connected Systems

C2.3 Reactor Coolant Pump

- C2.3.1 Bowl / Casing
- C2.3.2 Cover
- C2.3.3 Closure Bolting

C2.4 Safety & Relief Valves

- C2.4.1 Valve 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 Nozzle

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, pressurizer, pressurizer relief tank, and portions of other connected systems extending up to the first isolation valve outside of containment or to the first anchor point. The connected systems include residual heat removal (RHR) or low-pressure injection system, core flood spray (CFS) or safety injection tank (SIT), chemical and volume control system (CVCS) or high-pressure injection system, and sampling system. 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 "Emergency Core Cooling System." The aging management program for the variety of lines that penetrate the containment and associated isolation valves is reviewed in Table V C "Containment Isolation Components."

The pump and valve internals are considered to be active components. They perform their intended functions with moving parts or with a change in configuration and are not subject to aging management review pursuant to 10 CFR 54.21(a)(1)(i).

System Interfaces

The systems that interface with the reactor coolant pressure boundary include the reactor pressure vessel (Table IV A2), containment isolation components (Table V C), emergency core cooling system (Table V D1), and steam generators (Tables IV D1 and IV D2).

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	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. ANSI B31.1. GSI-190.
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	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. ANSI B31.1. GSI-190.
C2.1.1 thru 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	Crack Initiation and Growth	Stress Corrosion Cracking (SCC), Inter-granular Stress Corrosion Cracking (IGSCC)	ASME Section XI, 1989 Edition. Reg. Guide 1.43. Reg. Guide 1.44. Reg. Guide 1.45. EPRI TR-105714. Plant Technical Specifications. *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.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or ANSI B31.1, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or ANSI B31.1, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Guidelines of Regulatory Guide (RG) 1.44 to avoid sensitization of stainless steels (SSs) and guidelines of RG 1.43 to prevent underclad cracking and inservice inspection in conformance with ASME Section XI (edition specified 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. Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 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:*** Selection of material in compliance with the guidelines of Regulatory Guide (RG) 1.44 prevent or mitigate SCC, and RG 1.43 provides assurance that production cladding complies with ASME Section III and XI requirements to prevent underclad cracking. Also, 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 ISI in accordance with ASME Section XI to detect possible degradation. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of IGSCC on the intended function of piping and fittings by detection and sizing of cracks by ISI. Inspection 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. Requirements for training and qualification of personnel and performance demonstration for procedures and equipment are in conformance with Appendices VII and VIII of ASME Section XI, or any other formal program approved by the NRC. Inspection requirements of testing category B-P	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 with SS Cladding	Air, Leaking Chemically Treated Borated Water	Loss of Material	Corrosion/ Boric Acid Wastage of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 Edition. Plant Technical Specifications. *Operating Experience* NRC IN 86-108 S 3.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) conducted according to IWA–5000 specify visual VT-2 (IWA-5240) examination of all pressure retaining components extending to and including the second closed valve at the boundary extremity, during system leakage test (IWB-5221) and system hydrostatic test (IWB-5222). Leakage detection is in conformance with Position C of Regulatory Guide 1.45. ***(4) Detection of Aging Effects:*** Degradation of the piping 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 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 IGSCC 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 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 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 based on plant technical specifications, inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, testing category B-P for pressure retaining Class 1 components, and Table IWC 2500-1, examination category C-H for pressure retaining Class 2 components.	***(1) Scope of Program:*** The staff guidance of Generic Letter (GL) 88-05 provides assurances that a program has been implemented consisting of systematic measures to ensure that the effects of corrosion caused by leaking coolant containing boric acid does not lead to degradation and provides assurance that the reactor coolant pressure boundary will have a extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture. The program includes (a) determination of principal location,	No

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) (b) examinations requirements and procedures for locating small leaks, and (c) engineering evaluations and corrective actions. ***(2) Preventive Actions:*** Minimizing reactor coolant leakage by frequent monitoring of the locations where potential leakage could occur and timely repair, prevent or mitigate boric acid corrosion. Modifications in the design or operating procedures to reduce the probability of leaks at locations where they may cause corrosion damage and use of suitable corrosion resistant materials or the application of protective coatings. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of boric acid corrosion on the intended function of the component by detection of coolant leakage by implementing the program delineated in GL 88-05 including guidelines for locating small leaks, conducting examinations, and performing engineering evaluations. Inspection requirements of ASME Section XI specify visual VT-2 (IWA-5240) examination during system leakage test (IWB-5221) and system hydrostatic test (IWB-5222) of all pressure retaining Class 1 components, according to Table IWB 2500-1 category B-P and pressure retaining Class 2 components according to Table IWC 2500-1 category C-H. ***(4) Detection of Aging Effects:*** Degradation of the component due to boric acid wastage can not occur without leakage of coolant containing boric acid; extent and schedule of the program delineated in GL 88-05 will assure detection of leakage before the loss of intended function of the component. ***(5) Monitoring and Trending:*** The program delineated in GL 88-05 provides for timely detection of leakage. ***(6) Acceptance Criteria:*** Any leakage is evaluated in accordance with the program proposed by GL 88-05. ***(7) Corrective Actions:*** The leakage source and areas of general corrosion are located and corrective actions are in conformance with the program proposed by GL 88-05, and the requirements of ASME Section XI, IWB-3142 and IWA-5250. Components with local areas of corrosion that reduces the wall thickness by more than 10% require analytical evaluation to demonstrate acceptability. ***(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:*** Boric acid wastage observed in nuclear power plants may be classified into two types: (a) corrosion that increases the rate of leakage, e.g., corrosion of closure bolting or fasteners, and (b) corrosion that occurs some distance from the source of leakage. Some recent incidents of boric acid wastage (IN 86-108 S3) indicate that, although implementation of GL 88-05 ensures timely detection of leakage, there may still be a lack of awareness of the conditions that can lead to boric acid wastage.	

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C2.1.1 thru C2.1.3	Reactor Coolant System (RCS) Piping & Fittings	Cold-Leg, Hot-Leg, Surge Line	CASS	Chemically Treated Borated Water up to 340°C	Crack Initiation and Growth	SCC, IGSCC	EPRI TR-105714. NUREG-0313, Rev. 2.
C2.1.1 thru C2.1.3	Reactor Coolant System (RCS) Piping & Fittings	Cold-Leg, Hot-Leg, Surge Line	CASS	Chemically Treated Borated Water up to 340°C	Loss of Fracture Toughness	Thermal Aging Embrittle-ment	EPRI TR-106092. ASME Section XI, 1989 Edition.

Existing 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 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, no AMP
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 to demonstrate that the material has adequate fracture toughness are not required for "not susceptible" piping. Because of low applied stresses for pump casings and valve bodies, screening for susceptibility to thermal aging. Also, the existing ASME Section XI inspection requirements, including the alternative requirements of ASME Code Case N-481 for pump casings, are considered adequate for all pump casings and valve bodies.	***(1) Scope of Program:*** The program includes determination of the susceptibility of CASS components to thermal aging based on casting method, Mo content, and percent ferrite, and for potentially susceptible components aging management is accomplished either through volumetric examination or plant/component-specific flaw tolerance evaluation. ***(2) Preventive Actions:*** The program provides no guidance on methods to mitigate thermal aging. ***(3) Parameters Monitored/ Inspected:*** Based on the criteria in EPRI TR-106092, with some modifications, the susceptibility to thermal aging embrittlement of CASS piping is determined in terms of casting method, Mo content, and ferrite content. For low-Mo content (0.5 wt.% max.) steels, only static-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement, all centrifugal-cast and static-cast steels with ≤20% ferrite are not susceptible. For high-Mo content (2.0 to 3.0 wt.%) steels, static-cast steels with >14% ferrite and centrifugal-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement, static-cast steels with ≤14% ferrite and centrifugal-cast steels with ≤20% ferrite are not susceptible. Ferrite content will be calculated by the Hull's equivalent factors or a method producing an equivalent level of accuracy (±6% deviation between measured and calculated values.) For pump casings and valve bodies, screening for susceptibility to thermal aging is not required. ***(4) Detection of Aging Effects:*** For "not susceptible" piping, no additional inspection or evaluations to demonstrate that the material has adequate fracture toughness are required. For "potentially susceptible" piping, because the base metal does not receive periodic inspection per ASME Section XI, volumetric examination should be performed on the base metal, with the scope of the inspection covering the portions determined to be limiting from the standpoint of applied stress, operating time, and environmental considerations. Alternatively, a plant/component- specific flaw tolerance evaluation, using specific geometry and stress information, can be used to demonstrate that the thermally-embrittled material has adequate toughness. Current volumetric examination methods are inadequate for reliable detection of cracks in CASS components; the performance of the equipment and techniques when developed, should be demonstrated through the program consistent with the ASME Section XI, Appendix VIII. For all pump casings and valve bodies, the existing ASME Section XI inspection	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C2.1.5	Reactor Coolant System (RCS) Piping & Fittings	Small-Bore RCS Piping	SS, CS	Chemically Treated Borated Water up to 340°C	Crack Initiation and Growth	Unanticipated Thermal and Mechanical Loading	ASME Section XI, 1989 Edition. EPRI TR-105714. Plant Technical Specifications. NRC IN 97-46.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) requirements, including the alternative requirements of ASME Code Case N-481 for pump casings, are considered adequate. For valve bodies less than NPS 4, the adequacy of inservice inspection according to ASME Section XI has been demonstrated by a NRC performed bounding fracture analysis. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 should provide timely detection of cracks. ***(6) Acceptance Criteria:*** Flaws detected in CASS components are evaluated in accordance with the applicable procedures of IWB-3500. If aging management is accomplished through plant/component-specific flaw tolerance evaluation, e.g., for potentially susceptible piping, flaw evaluation for piping with <25% ferrite is performed according to the principles associated with IWB-3640 procedures for submerged arc welds (SAW), disregarding the Code restriction of 20% ferrite in IWB-3641(b)(1). Flaw evaluation for piping with >25% ferrite is performed on a case-by-case basis using fracture toughness data provided by the applicant. ***(7) Corrective Actions:*** Repair is in conformance with IWA-4000 and IWB-4000, and replacement according to IWA-7000 and IWB-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:*** The AMP based on susceptibility determination and examination requirements or flaw tolerance evaluations of potentially susceptible components is effective in managing the effects of thermal aging on the intended function of CASS components.	
Inservice inspection in conformance with ASME Section XI (edition specified 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 EPRI guidelines in TR-105714 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 cracking. 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 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 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	Yes Elements 3 and 4 should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C2.2.1 thru C2.2.3	Connected Systems Piping & Fittings	Residual Heat Removal (RHR), Core Flood System (CFS), Chemical & Volume Control System(CVCS)	SS	Chemically Treated Borated Water up to 340°C	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. ANSI B31.1. GSI-190.
C2.2.4, C2.2.5	Connected Systems Piping & Fittings	Sampling, Drains & In-strumentation	CS, SS	Chemically Treated Borated Water up to 340°C	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. ANSI B31.1. GSI-190.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) 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 connections. Actual inspection locations should be based on risk-informed approaches and 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 unisolable section of combined makeup and high-pressure injection lines.	
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or ANSI B31.1, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or ANSI B31.1, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C2.2.6	Connected Systems Piping & Fittings	Nozzles & Safe Ends	SS, CASS	Chemically Treated Borated Water up to 340°C	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. ANSI B31.1. GSI-190.
C2.2.4, C2.2.5	Connected Systems Piping & Fittings	Sampling, Drains & In-strumentation (External Surfaces)	CS	Air, Leaking Chemically Treated Borated Water	Loss of Material	Corrosion /Boric Acid Wastage of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 Edition. Plant Technical Specifications.
C2.2.6	Connected Systems Piping & Fittings	Nozzles & Safe Ends	CASS	Chemically Treated Borated Water up to 340°C	Loss of Fracture Toughness	Thermal Aging Embrittle-ment	EPRI TR-106092. ASME Section XI, 1989 Edition.
C2.2.1 thru C2.2.6	Connected Systems Piping & Fittings	RHR, CFS, CVCS, Sampling, Drains & In-strumentation , Nozzles & Safe Ends	SS	Chemically Treated Borated Water up to 340°C	Crack Initiation and Growth	SCC	*Same as for the effect of Stress Corrosion Cracking on Items C2.1.1 thru C2.1.4 cold-leg, hot-leg, surge line, and spray line piping and fittings constructed of SS*
C2.2.6	Connected Systems Piping & Fittings	Nozzles & Safe Ends	CASS	Chemically Treated Borated Water up to 340°C	Crack Initiation and Growth	SCC	EPRI TR-105714. NUREG-0313, Rev. 2.
C2.2.7	Connected Systems Piping & Fittings	Small-Bore Piping in Connected Systems	SS, CS	Chemically Treated Borated Water up to 340°C	Crack Initiation and Growth	SCC, IGSCC	ASME Section XI, 1989 Edition. EPRI TR-105714. Plant Technical Specifications. NRC IN 97-46.
C2.3.1, C2.3.2	Reactor Coolant Pump	Bowl/Casings, Cover	Bowl: CASS SA351 CF-8 or CF-8M; Cover: SS	Chemically Treated Borated Water up to 340°C	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. GSI-190.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or ANSI B31.1, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Same as for the effect of corrosion/boric acid wastage 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 corrosion/boric acid wastage 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 Thermal Aging Embrittlement on Items C2.1.1 thru 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 thru 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 thru 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 thru C2.1.4 cold-leg, hot-leg, surge line, and spray line piping and fittings constructed of SS.*	No
Monitoring and control of primary water chemistry in accordance with the EPRI guidelines in TR-105714 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, no AMP
Same as for the effect of Stress Corrosion Cracking on Item C2.1.7 small-bore RCS piping.	*Same as for the effect of Stress Corrosion Cracking on Item C2.1.7 small-bore RCS piping.*	Yes, Elements 3 and 4 should be further evaluated
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C2.3.1	Reactor Coolant Pump	Bowl/Casings	CASS SA351 CF-8 or CF-8M	Chemically Treated Borated Water up to 340°C	Crack Initiation and Growth	SCC	EPRI TR-105714. NUREG-0313, Rev. 2. ASME Section XI, 1989 Edition. *Operating Experience* NRC IN 84-18.

Existing 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 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 above guidelines, inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination categories B-L-1 for pump casing welds and B-L-2 for pump casings, and testing category B-P for system leakage.	***(1) Scope of Program:*** Control of primary water chemistry in accordance with EPRI TR-105714 guidelines and material selection according to the guidance in NUREG-0313, Rev. 2 mitigate the potential of SCC in CASS components. For components that do not meet either one of the above guidelines, inservice inspection (ISI) to monitor the effects of SCC on the intended function of the pump. ***(2) Preventive Actions:*** Control of primary water chemistry and selection of appropriate materials prevents or mitigates 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 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 pump by detection and sizing of cracks by ISI. Inspection requirements of Table IWB 2500-1, examination category B-L-1 specify 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 for training and qualification of personnel and performance demonstration for procedures and equipment are in conformance with Appendices VII and VIII of ASME Section XI, or any other formal program approved by the NRC. Inspection requirements of category B-P 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:*** Degradation of the pump due to SCC can not occur without crack initiation and growth; extent and schedule of inspection assures detection of cracks before the loss of the intended function of the component. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 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 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 pump internal surfaces. Planar and liner flaws are sized according to IWA-3300 and IWA-3400. 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, replacement according to IWA-7000 and IWB-7000, and reexamination in accordance with requirements of IWA-2200. ***(8 & 9) Confirmation Process and***	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C2.3.1	Reactor Coolant Pump	Bowl/Casings	CASS SA351 CF-8 or CF-8M	Chemically Treated Borated Water up to 340°C	Loss of Fracture Toughness	Thermal Aging Embrittle-ment	EPRI TR-106092. ASME Section XI, 1989 Edition.
C2.3.3	Reactor Coolant Pump	Closure Bolting	High Strength Low-Alloy Steel (HSLAS) SA540 GrB23, SA193 GrB7	Air, Leaking Chemically Treated Borated Water and/or Steam up to 340°C	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. GSI-190.
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	Crack Initiation and Growth	SCC	NUREG-1339 EPRI NP-5769 NRC GL 91-17 ASME Section XI, 1989 Edition. *Operating Experience* IEB 82-02

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) ***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, potential of SCC exists from inadvertent introduction of contaminants into the primary coolant system (IN 84–18).	
Same as for the effect of Thermal Aging Embrittlement on Items C2.1.1 thru 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 thru C2.1.3 Cold-Leg, Hot-Leg, and Surge Line Piping and Fittings.*	No
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Recommendations for a comprehensive bolting integrity program delineated in NUREG–1339 on resolution of Generic Safety Issue 29 and implemented through NRC Generic Letter 91-17; additional details on bolting integrity outlined in EPRI NP-5769; and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination categories B-G–1 or B-G-2 for pressure retaining bolting, and testing category B-P for system leakage.	***(1) Scope of Program:*** The staff guidance of NRC Generic Letter (GL) 91-17 provides assurance that plant specific comprehensive bolting integrity programs have been implemented to ensure bolting reliability. The NRC staff recommendations and guidelines for a comprehensive bolting integrity program are delineated in NUREG-1339, and the industry's technical basis for the program is outlined in EPRI NP-5769. ***(2) Preventive Actions:*** Selection of bolting material and the use of lubricants and sealants in accordance with guidelines of EPRI NP-5769 and additional requirements of NUREG 1339, prevent or mitigate degradation and failure of all safety-related closure bolting. ***(3) Parameter Monitored/ Inspected:*** The AMP monitors the effects of aging degradation on the intended function of closure bolting by detection of coolant leakage, and by detection and sizing of cracks by inservice inspection (ISI). Inspection requirements of ASME Section XI, Table IWB 2500-1, examination category B-G-1 for bolting greater than 2 in. in diameter specify volumetric examination of studs and bolts from top of the nut to bottom of the flange hole, and visual VT-1 examination of surfaces of nuts, washers, bushings, and flanges including an 1 in. annular surface of flange surrounding each stud. Examination category B-G-2 for bolting 2 in. or smaller specifies only visual VT-1 examination of surfaces of bolts, studs, and nuts. However, because most failures have occurred in fasteners 2 in. or smaller, based on IE Bulletin 82-02 enhanced inspection and improved techniques are recommended. These examinations are limited to components selected for inspection under examination category B-M-2, B-L-2, or B-J, and may be performed in place under tension, when	No

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) connection is disassembled, or when bolting is removed. Requirements for training and qualification of personnel and performance demonstration for procedures and equipment are in conformance with Appendices VII and VIII of ASME Section XI, and additional requirements of EPRI NP-5769. Inspection requirements of ASME Section XI testing category B-P specify visual VT-2 (IWA-5240) examination of all pressure retaining components extending to and including the second closed valve at the boundary extremity, during system leakage test (IWB-5221) and system hydrostatic test (IWB-5222). ***(4) Detection of Aging Effects:*** Degradation of the closure bolting due to crack initiation, loss of prestress, or attrition of the closure bolting would result in leakage. The extent and schedule of inspection assure detection of aging degradation before the loss of intended function of closure bolting. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI is effective and adequate for timely detection of cracks and leakage. Inspection schedule in accordance with IWB-2400 requires inspection of all bolts, studs, nuts, bushing, and flange surfaces every 10 y. 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 bolting are evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3515 and 3517. Any relevant conditions that may be detected during the leakage and hydrostatic tests are compared with the acceptance standards of IWB-3522. ***(7) Corrective Actions:*** Repair and replacement is according to IWB-4000 and IWB-7000, respectively, and recommendations and guidelines of EPRI NP-5769. ***(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:*** Significant number of incidents have been reported on bolts and nuts that have failed or become degraded because of boric acid wastage or SCC. Examples of affected fasteners include (a) steam generator and pressurizer manway closures, (b) valve bonnets and pump flange connections on lines 6 in. or greater, and (c) control rod drive and pressurizer heater connections. The bolting integrity programs developed and implemented in accordance with commitments made in response to NRC communications on bolting events have provided effective means of ensuring bolting reliability.	

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
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	Loss of Material	Corrosion /Boric Acid Wastage of External Surfaces	NUREG-1339. EPRI NP-5769. NRC GL 91-17. NRC GL 88-05. ASME Section XI, 1989 Edition.
C2.3.3	Reactor Coolant Pump	Closure Bolting	HSLAS SA540 GrB23, SA193 GrB7	Air, Leaking Chemically Treated Borated Water and/or Steam up to 340°C	Loss of Prestress	Stress Relaxation	NUREG-1339. EPRI NP-5769. NRC GL 91-17. ASME Section XI, 1989 Edition.
C2.3.3	Reactor Coolant Pump	Closure Bolting	HSLAS SA540 GrB23, SA193 GrB7	Air, Leaking Chemically Treated Borated Water and/or Steam up to 340°C	Attrition	Wear	NUREG-1339. EPRI NP-5769. NRC GL 91-17. ASME Section XI, 1989 Edition.
C2.4.1, C2.4.2	Valves (Check, Control, Hand, Motor-Operated, and Relief Valves)	Bodies, Bonnet	CASS SA351 CF-8M, SA182 F316, SA582 Type 416	Chemically Treated Borated Water up to 340°C	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. GSI-190.
C2.4.1	Valves (Check, Control, Hand, Motor-Operated, and Relief Valves)	Bodies	CASS SA351 CF-8M	Chemically Treated Borated Water up to 340°C	Crack Initiation and Growth	SCC	EPRI TR-105714. NUREG-0313, Rev. 2. ASME Section XI, 1989 Edition. *Operating Experience* NRC IN 84-18.

Existing 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 implemented through NRC Generic Letter 91-17; additional details on bolting integrity outlined in EPRI NP-5769; implementation of NRC Generic Letter 88-05; and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination categories B-G-1 or B-G-2 for pressure retaining bolting, and testing category B-P for system leakage.	*Combination of (a) the effect of stress corrosion cracking on Item C2.3.3 closure bolting for reactor coolant pump, and (b) the effect of corrosion/boric acid wastage on 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 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
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Monitoring and control of primary water chemistry in accordance with the EPRI guidelines in TR-105714 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, inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination categories B-M-1 for valve body welds and B-M-2 for valve body, and testing category B-P for system leakage.	***(1) Scope of Program:*** Control of primary water chemistry in accordance with EPRI TR-105714 guidelines and material selection according to the guidance in NUREG-0313, Rev. 2 mitigate the potential of SCC in CASS components. For components that do not meet either one of the above guidelines, inservice inspection (ISI) should be performed to monitor the effects of SCC on the intended function of the pump. ***(2) Preventive Actions:*** Control of primary water chemistry and selection of appropriate materials prevents or mitigates 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 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 valve by detection and sizing of cracks by ISI. Inspection requirements of Table IWB 2500-1, examination category B-M-1 specify for all welds NPS 4 or larger, volumetric	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C2.4.1	Valves (Check, Control, Hand, Motor-Operated, and Relief Valves)	Bodies	CASS SA351 CF-8M	Chemically Treated Borated Water up to 340°C	Loss of Fracture Toughness	Thermal Aging Embrittle-ment	EPRI TR-106092. ASME Section XI, 1989 Edition.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) 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 specifies visual VT-3 examination of internal surfaces of the valve. Requirements for training and qualification of personnel and performance demonstration for procedures and equipment are in conformance with Appendices VII and VIII of ASME Section XI, or any other formal program approved by the NRC. Inspection requirements of 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 of the valve bodies 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 valves. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 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 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 and surface examination of welds and 3519 for visual examination of valve internal surfaces. Planar and liner flaws are sized according to IWA-3300 and IWA-3400. 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, replacement according to IWA-7000 and IWB-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:*** Although the primary pressure boundary piping of PWRs have 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).	
Same as for the effect of Thermal Aging Embrittlement on Items C2.1.1 thru 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 thru 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
C2.4.3	Valves	Closure Bolting	HSLAS	Air, Leaking Chemically Treated Borated Water or Steam up to 340°C	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. GSI-190.
C2.4.3	Valves	Closure Bolting	HSLAS	Air, Leaking Chemically Treated Borated Water or Steam up to 340°C	Crack Initiation and Growth	SCC	NUREG-1339. EPRI NP-5769. NRC GL 91-17. ASME Section XI, 1989 Edition.
C2.4.3	Valves	Closure Bolting	HSLAS	Air, Leaking Chemically Treated Borated Water or Steam up to 340°C	Loss of Material	Corrosion /Boric Acid Wastage of External Surfaces	NUREG-1339. EPRI NP-5769. NRC GL 91-17. NRC GL 88-05. ASME Section XI, 1989 Edition.
C2.4.3	Valves	Closure Bolting	HSLAS	Air, Leaking Chemically Treated Borated Water or Steam	Loss of Prestress	Stress Relaxation	NUREG-1339. EPRI NP-5769. NRC GL 91-17. ASME Section XI, 1989 Edition.
C2.4.3	Valves	Closure Bolting	HSLAS	Air, Leaking Chemically Treated Borated Water or Steam	Attrition	Wear	NUREG-1339. EPRI NP-5769. NRC GL 91-17. ASME Section XI, 1989 Edition.
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	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. GSI-190.
C2.5.1	Pressurizer	Shell/Heads (Outer surfaces)	LAS	Air, Leaking Chemically Treated Borated Water or Steam at up to 340°C	Loss of Material	Corrosion /Boric Acid Wastage of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 Edition. Plant Technical Specifications.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	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
Recommendations for a comprehensive bolting integrity program delineated in NUREG–1339 on resolution of Generic Safety Issue 29 and implemented through NRC Generic Letter 91-17; additional details on bolting integrity outlined in EPRI NP-5769; implementation of NRC Generic Letter 88-05; and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination categories B-G–1 or B-G-2 for pressure retaining bolting, and testing category B-P for system leakage.	*Combination of (a) the effect of stress corrosion cracking on Item C2.3.3 closure bolting for reactor coolant pump, and (b) the effect of corrosion/boric acid wastage on 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 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
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Same as for the effect of corrosion/boric acid wastage 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 corrosion/boric acid wastage 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
C2.5.1	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	Crack Initiation and Growth	SCC/ IGSCC	ASME Section XI, 1989 Edition. Reg. Guide 1.43. Reg. Guide 1.44. Reg. Guide 1.45. EPRI TR-105714. *Operating Experience* NRC IN 84-18. NRC IN 84-89.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Guidelines of Regulatory Guide (RG) 1.44 to avoid sensitization of stainless steels (SSs) and requirements of RG 1.43 to prevent underclad cracking and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination category B-P for the pressurizer pressure boundary, category B-B for pressure-retaining welds, category B-D for full-penetration welded nozzles, category B-E for pressure retaining partial retaining welds, and category B-F for pressure-retaining dissimilar metal welds in nozzles. Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 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 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:*** Selection of material in compliance with the requirements of Regulatory Guide (RG) 1.44 prevent or mitigate SCC, and RG 1.43 provides assurance that production cladding complies with ASME Section III and XI requirements to prevent underclad cracking. Also, 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. 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 ISI in accordance with ASME Section XI to detect possible degradation. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of SCC/IGSCC on the intended function of pressurizer components by detection and sizing of cracks by ISI. Inspection requirements of Table IWB 2500-1, specify the extent and schedule for inspection of the components making up the pressurizer pressure boundary. Requirements for training and qualification of personnel and performance demonstration for procedures and equipment are in conformance with Appendices VII and VIII of ASME Section XI, or any other formal program approved by the NRC. Inspection requirements of testing category B-P conducted according to IWA-5000 specify visual VT-2 (IWA-5240) examination of all pressure retaining components extending to and including the second closed valve at the boundary extremity, during system leakage test (IWB-5221) and system hydrostatic test (IWB-5222). Leakage detection is in conformance with Position C of Regulatory Guide 1.45. ***(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 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/IGSCC 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	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C2.5.2	Pressurizer	Spray Line Nozzle	CS or LAS with SS Cladding	Chemically Treated Borated Water or Saturated Steam 290-343°C	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. GSI-190.
C2.5.3	Pressurizer	Surge Line Nozzle	CS or LAS with SS Cladding, CASS	Chemically Treated Borated Water up to 340°C	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. GSI-190.
C2.5.4	Pressurizer	Spray Head	Alloy 600, SS, CASS	Chemically Treated Borated Water or Saturated Steam 290-343°C	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. GSI-190.
C2.5.5 thru C2.5.7	Pressurizer	Thermal Sleeves, Instru-mentation Penetrations, Safe Ends	Thermal Sleeves: Alloy 600; Penetrations: A 600, SS; Safe Ends: SS	Chemically Treated Borated Water or Saturated Steam 290-343°C	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. GSI-190.
C2.5.2, C2.5.3, C2.5.6	Pressurizer	Spray Line Nozzle, Surge Line Nozzle, Instru-mentation Penetrations	CS or LAS with SS Cladding; or SS	Chemically Treated Borated Water or Saturated Steam 290-343°C	Crack Initiation and Growth	SCC	*Same as for the effect of SCC on C2.5.1 pressurizer shell/heads*

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) 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).	
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes. TLAA
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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C2.5.7	Pressurizer	Safe Ends	SS	Chemically Treated Borated Water or Saturated Steam 290-343°C	Crack Initiation and Growth	SCC	*Same as for the effect of Stress Corrosion Cracking on Items C2.1.1 thru C2.1.4 cold-leg, hot-leg, surge line, and spray line piping and fittings constructed of SS.*
C2.5.3	Pressurizer	Surge Line Nozzle	CASS	Chemically Treated Borated Water up to 340°C	Crack Initiation and Growth	SCC	EPRI TR-105714. NUREG-0313, Rev. 2.
C2.5.4	Pressurizer	Spray Head	Alloy 600, SS, CASS	Chemically Treated Borated Water or Saturated Steam 290-343°C	Crack Initiation and Growth	SCC	Plant Technical Specifications
C2.5.6	Pressurizer	Instru-mentation Penetrations	Alloy 600	Chemically Treated Borated Water or Saturated Steam 290-343°C	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. Reg. Guide 1.45. NRC GL 97-01. EPRI TR-105714. *Operating Experience* NRC IN 90-10. NRC IN 96-11.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Same as for the effect of Stress Corrosion Cracking on Items C2.1.1 thru 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 thru C2.1.4 cold-leg, hot-leg, surge line, and spray line piping and fittings constructed of SS.*	No
Monitoring and control of primary water chemistry in accordance with the EPRI guidelines in TR-105714 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, no 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 (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination categories B-E for pressure retaining partial penetration welds and testing category B-P for system leakage. Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 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 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 review previous evaluations performed in response to Generic Letter (GL) 97-01, as updated with the most current industry susceptibility model and inspection results. ***(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 SCC has not been addressed adequately, particularly when demineralizer resins contaminate the reactor coolant system. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of SCC on the intended function of instrument nozzles and penetrations by performance monitoring and detection of cracks by inservice inspection (ISI). Testing category B-P specify visual VT-2 (IWA-5240) examination of all pressure retaining components extending to and including the second closed valve at the boundary extremity, during system leakage test (IWB-5221) and system hydrostatic test (IWB-5222). Leakage detection is in conformance with Position C of Regulatory Guide 1.45. Examination category B-E specifies visual VT-2 examination of partial	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	Loss of Fracture Toughness	Thermal Aging Embrittle-ment	EPRI TR-106092. ASME Section XI, 1989 Edition.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) 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 SCC. 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 3522 for visual examination. Acceptance of the component for continued service is based on the results of visual examination, supplemental volumetric and surface examination (IWB-3200), corrective measure or repair, analytical evaluation , or replacement (IWB-3142). 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 (GL 97-01). ***(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:*** 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 thru 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 thru 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
C2.5.4	Pressurizer	Spray head	Alloy 600, SS, CASS	Chemically Treated Borated Water or Saturated Steam 290-343°C	Wall Thinning	Erosion	Plant Technical Specifications.
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	Crack Initiation and Growth	SCC	*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	Crack Initiation and Growth	SCC	NUREG-1339. EPRI NP-5769. NRC GL 91-17. ASME Section XI, 1989 Edition.
C2.5.9	Pressurizer	Manway and Flange Bolting	HSLAS	Air, Leaking Chemically Treated Borated Water or Steam up to 340°C	Loss of Material	Corrosion /Boric Acid Wastage of External Surfaces	NUREG-1339. EPRI NP-5769. NRC GL 91-17. NRC GL 88-05. ASME Section XI, 1989 Edition.
C2.5.9	Pressurizer	Manway and Flange Bolting	HSLAS	Air, Leaking Chemically Treated Borated Water or Steam up to 340°C	Loss of Prestress	Stress Relaxation	NUREG-1339. EPRI NP-5769. NRC GL 91-17. ASME Section XI, 1989 Edition.
C2. 5.10	Pressurizer	Heater Sheaths and Sleeves	Alloy 600 or Austenitic SS	Chemically Treated Borated Water up to 340°C	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. GSI-190.
C2. 5.10	Pressurizer	Heater Sheaths and Sleeves	Austenitic SS	Chemically Treated Borated Water up to 340°C	Crack Initiation and Growth	SCC	*Same as for the effect of SCC on Item C2.5.1 pressurizer shell/heads*

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Plant-specific aging management program.	Plant-specific aging management program is to be evaluated.	Yes No AMP
Same as for the effect of SCC on Item C2.5.1 pressurizer shell/heads.	*Same as for the effect of SCC 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
Recommendations for a comprehensive bolting integrity program delineated in NUREG–1339 on resolution of Generic Safety Issue 29 and implemented through NRC Generic Letter 91-17; additional details on bolting integrity outlined in EPRI NP-5769; implementation of NRC Generic Letter 88-05; and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination categories B-G–1 or B-G-2 for pressure retaining bolting, and testing category B-P for system leakage.	*Combination of (a) the effect of stress corrosion cracking on Item C2.3.3 closure bolting for reactor coolant pump, and (b) the effect of corrosion/boric acid wastage on 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
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes, TLAA
Same as for the effect of SCC on Item C2.5.1 pressurizer shell/heads.	*Same as for the effect of SCC on Item C2.5.1 pressurizer shell/heads.*	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C2. 5.10	Pressurizer	Heater Sheaths and Sleeves	Alloy 600	Chemically Treated Borated Water up to 340°C	Crack Initiation and Growth	SCC	*Same as for the effect of SCC on Item C2.5.6 pressurizer instrumentation penetrations constructed of Alloy 600*
C2. 5.10	Pressurizer	Heater Sheaths and Sleeves	Alloy 600 or Austenitic SS	Chemically Treated Borated Water up to 340°C	Attrition	Wear	Plant Technical Specifications.
C2. 5.11	Pressurizer	Support Keys, Skirts, and Shear Lugs	CS, LAS	Air, with metal temperature s up to 340°C	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition.
C2. 5.12	Pressurizer	Integral Support	CS	Air, Leaking Chemically Treated Borated Water	Loss of Material	Corrosion /Boric Acid Wastage of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 Edition. NRC IN 86-108 S 3.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Same as for the effect of SCC on Item C2.5.6 pressurizer instrumentation penetrations constructed of Alloy 600.	*Same as for the effect of SCC on Item C2.5.6 pressurizer instrumentation penetrations constructed of Alloy 600.*	Yes, Elements 2 to 6 should be further evaluated
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, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal.	Yes, TLAA
Implementation of NRC Generic Letter 88-05, and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination category B-K-1 for integral attachments for piping, pumps, and valves.	***(1) Scope of Program:*** The staff guidance of Generic Letter (GL) 88-05 provides assurance that a program has been implemented consisting of systematic measures to ensure that the effects of corrosion caused by leaking coolant containing boric acid does not lead to degradation and provides assurance that the reactor coolant pressure boundary will have a extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture. The program includes (a) determination of principal location, (b) examinations requirements and procedures for locating small leaks, and (c) engineering evaluations and corrective actions. ***(2) Preventive Actions:*** Minimizing reactor coolant leakage by frequent monitoring of the locations where potential leakage could occur and repairing the leaky components as soon as possible, prevent or mitigate boric acid corrosion. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of boric acid corrosion on the intended function of the component by detection of flaws and coolant leakage by inservice inspection (ISI). Inspection requirements of ASME Section XI, Table IWB 2500-1, examination category B-K-1 specify volumetric or surface examination extending 1/2 in. on either side of the weld. Inspection includes the welded attachments of piping required to be examined by examination category B-J and the welded attachments to associated pumps and valves integral to the piping. ***(4) Detection of Aging Effects:*** Degradation of the component due to boric acid wastage can not occur without leakage of coolant containing boric acid; extent and schedule of inspection assure detection of flaws and leakage before the loss of intended function of the component. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI should provide for timely detection of flaws and coolant leakage. ***(6) Acceptance Criteria:*** Any relevant conditions that may be detected during the inspection are evaluated in accordance with IWB-3100 and acceptance standards of IWB-3400 and IWB-3516. ***(7) Corrective Actions:*** Repair	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C2. 5.12	Pressurizer	Integral Support	CS, SS	Air	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition.
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 94°C	Cumulative Fatigue Damage	Fatigue	Original Licensing Basis or ASME Section III, 1989 Edition.
C2.6.1	Pressurizer Relief Tank	Tank Shells and Heads (External Surfaces)	CS	94°C, air, Chemically Treated Borated Water	Loss of Material	Corrosion /Boric Acid Wastage of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 Edition. NRC IN 86-108 S 3. Plant Technical Specifications.
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 94°C	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. Regulatory Guide 1.43. Regulatory Guide 1.44. Regulatory Guide 1.45.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) 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:*** Boric acid wastage observed in nuclear power plants may be classified into two distinct types: (a) corrosion that increases the rate of leakage, e.g., corrosion of closure bolting or fasteners in reactor coolant pressure boundary, and (b) corrosion that occurs some distance from the source of leakage. Some recent incidents of boric acid wastage (IN 86-108 S3) indicate that, although implementation of GL 88-05 ensures timely detection of leakage, there may still be a lack of awareness of the conditions that can lead to boric acid wastage.	
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal.	Yes, TLAA
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NC, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal.	Yes, TLAA
Same as for the effect of corrosion/boric acid wastage 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 corrosion/boric acid wastage of external surfaces of Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.*	No
Guidelines of Regulatory Guide (RG) 1.44 to avoid sensitization of stainless steels (SSs) and guidelines of RG 1.43 to prevent underclad cracking and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWC, Table IWC 2500-1, examination category C-A for pressure-retaining welds, category C-B for pressure-retaining nozzle welds, and category C-H for all pressure-retaining components.	**(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 clad components. **(2) Preventive Actions:** Selection of material in compliance with the guidelines of Regulatory Guide (RG) 1.44 prevent or mitigate SCC, and RG 1.43 provides assurance that production cladding complies with ASME Section III and XI requirements to prevent underclad cracking. Also, 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 can occur, and cladding regions with low delta ferrite may be sensitized during post-weld heat treatment and thus become susceptible to SCC. Nonetheless, oxygen-induced SCC is not anticipated for this component because of the low operating	No

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) temperatures. **(3) Parameters Monitored/Inspected:** The AMP monitors the effects of IGSCC on the intended function of clad components by detection and sizing of cracks by inservice inspection (ISI). Inspection requirements of Table IWC 2500-1, specify various extents of inspection and inspection intervals for the pressurizer relief tank. Requirements for training and qualification of personnel and performance demonstration for procedures and equipment are in conformance with Appendices VII and VIII of ASME Section XI, or any other formal program approved by the NRC. ***(4) Detection of Aging Effects:*** Degradation of the pressurizer relief tank 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 pressurizer relief tank. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWC-2500 should provide timely detection of cracks. ***(6) Acceptance Criteria:*** Any IGSCC 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 Manways and Bolting
- D1.1.8 Secondary Handholes and Bolting
- D1.1.9 Instrument Nozzles
- D1.1.10 Primary Manways and Bolting

D1.2 Tube Bundle

- D1.2.1 Tubes
- D1.2.2 Tube Support Plates
- D1.2.3 Tube Support Lattice Bars (Combustion Engineering)
- D1.2.4 Tube Plugs
- D1.2.5 Tube Repair Sleeves

D1.3 Upper Assembly and Separators

- D1.3.1 Feed Water Inlet Ring

D1.4 Piping and Fittings

- D1.4.1 Main Steam
- D1.4.2 Feedwater
- D1.4.3 Auxiliary Feedwater

D1.5 Safety and Relief Valves

D1.5.1 Body

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, external primary and secondary water/steam nozzles and safe ends, and portions of main steam, feedwater, and auxiliary feedwater systems extending from the steam generator up to the first isolation valve outside of containment or to the first anchor point. 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, including portions of main steam, feedwater, and auxiliary feedwater systems extending up to the first isolation valve outside of containment, is classified as Group B Quality Standard. The aging management program for the lines that penetrate the containment, and associated isolation valves, is reviewed in Table V C.

The valve internals are considered to be active components. They perform their intended functions with moving parts or with a change in configuration and are not subject to aging management review pursuant to 10 CFR 54.21(a)(1)(i).

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).

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 Steam	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. GSI-190.
D1.1.2	Pressure Boundary and Structural	Steam Nozzle & Safe End	LAS	Up to 300°C Steam	Wall Thinning	Erosion/ Corrosion (E/C)	NUREG-1344. EPRI NSAC-202L-R2. NRC IN 93-21. EPRI TR-102134. *Operating Experience* NRC BL 87-01. NRC GL 89-08. NRC IN 89-53. NRC IN 91-18. NRC IN 91-18 S1. NRC IN 91-28. NRC IN 92-35. NRC IN 95-11. NRC IN 97-84.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a) or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Program delineated in NUREG-1344 for single phase lines and implemented through NRC Generic Letter 89-08; CHECWORKS Code; EPRI guidelines of NSAC-202L-R2 for effective erosion/corrosion program; and water chemistry program based on EPRI guidelines for secondary water chemistry (EPRI TR-102134).	***(1) Scope of Program:*** The AMP includes NUMARC program delineated in Appendix A of NUREG-1344 and implemented through NRC Generic Letter (GL) 89-08; CHECWORKS computer Code; and EPRI guidelines of NSAC-202L-R2. The program includes the following recommendations: (a) conduct appropriate analysis and limited baseline inspection, (b) determine the extent of thinning and repair/replace components, and (c) perform follow-up inspections to confirm or quantify and take longer term corrective actions. Technical aspects of the CHECWORKS Code, including the parameters and inputs, are acceptable. However, the EPRI guidance document NSAC-202L-R2 (April 1999) is too general to ensure applicant's flow-accelerated corrosion program will be effective in managing aging in safety-related systems. ***(2) Preventive Actions:*** The rate of E/C is affected by piping material, geometry and hydrodynamic conditions, and operating conditions such as temperature, pH, and dissolved oxygen content. Mitigation is by selecting material considered resistant to E/C, adjusting water chemistry and operating conditions, and improving hydrodynamic conditions through design modifications. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of E/C on the intended function of piping by measuring wall thickness by nondestructive examination and performing analytical evaluations. The inspection program delineated in NUREG-1344 requires ultrasonic or radiographic testing of 10 most susceptible locations and 5 additional locations based on unique operating conditions or special considerations. For each location outside the acceptance guidelines, the inspection sample is expanded based on engineering judgment. Analytical models such as those incorporated into the CHECWORKS code are used to predict E/C in piping systems based on specific plant data including material and hydrodynamic and operating conditions. The inspection data is used to calibrate and benchmark the models and code. ***(4) Detection of Aging Effects:*** Aging degradation of piping and fittings occurs by wall thinning; extent and schedule of inspection assure detection of wall thinning before the loss of intended function of the piping. ***(5) Monitoring and Trending:*** Inspection schedule of NUREG-1344 and EPRI guidelines should provide for timely detection of leakage. Inspections and analytical evaluations are performed during plant outage. If analysis shows unacceptable conditions, inspection of initial sample is performed within 6 months. ***(6) Acceptance Criteria:*** Inspection results are used to calculate number of refueling or operating cycles remaining before the component reaches Code minimum allowable wall thickness. If calculations indicate that an area will	Yes, Element 1 should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.1.3 thru 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 Secondary-side Water Chemistry at 5.3-7.2 MPa	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. GSI-190.
D1.1.3, D1.1.4	Pressure Boundary and Structural	Upper & Lower Shell, Transition Cone	CS, LAS	Up to 300°C Secondary-side Water Chemistry at 5.3-7.2 MPa	Loss of Material	Crevice and Pitting Corrosion	ASME Section XI, 1989 Edition. EPRI TR-102134. *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) reach Code minimum (plus 10% margin), the component must be repaired or replaced. However, NRC staff has identified the problems in implementing E/C program that pertain to weakness or errors in (a) using predictive models, (b) calculating minimum wall thickness acceptance criteria, (c) analyzing the results of UT examinations, and (d) assessment of E/C program activities (NRC Information Notice IN 93-21). ***(7) Corrective Actions:*** Prior to service, repair or replace to meet the requirements of NUREG-1344. Follow-up inspections are performed to confirm or quantify thinning and take longer term corrective actions such as adjustment of chemistry and operating parameters, or selection of materials resistant to E/C. However, NRC staff has identified weakness or errors in (a) dispositioning components after reviewing the results of the inspection analysis, and (b) repairing or replacing components that failed to meet the acceptance criteria (IN 93-21). ***(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:*** Wall-thinning problems in single-phase systems have occurred in feedwater and condensate systems (NRC Bulletin No. 87-01, INs 81-28, 92-35, 95-11), and in two-phase piping in extraction steam lines (INs 89-53, 97-84) and moisture separation reheater and feedwater heater drains (INs 89-53, 91-18, 93-21, 97-84). The AMP outlined in NUREG-1344 and EPRI report and implemented through GL 89-08 has provided effective means of ensuring the structural integrity of all high-energy carbon steel systems.	
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a) or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
The program includes preventive measures to mitigate crevice corrosion by recommendations on secondary water chemistry of EPRI TR-102134 and, based on plant specifications, inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a) Table IWC 2500-1, examination category C-A for pressure retaining welds in pressure vessels, and examination category C-H for pressure retaining Class 2 components.	***(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 and by detection of coolant leakage by inservice inspection (ISI). Inspection requirements of Table IWC 2500-1, examination	Yes Element 4 should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.1.5	Upper Assembly and Separators	FW Nozzle & Safe End	LAS	Up to 225°C Secondary-side Water Chemistry	Wall Thinning	Erosion/ Corrosion (E/C)	*Same as the effect of Erosion/ Corrosion on Item D1.1.2 Steam Nozzle and Safe End*
D1.1.6	Upper Assembly and Separators	Feedwater Impingement Plate Support	CS	Up to 300°C Secondary-side Water Chemistry	Loss of section thickness	Erosion	Plant Technical Specifications

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) category C-A specify volumetric examination, extending 1/2 in. each side, of all circumferential and tubesheet-to-shell welds, and visual VT-2 (IWA-5240) examination during system leakage and hydrostatic tests of all pressure retaining Class 2 components. Requirements for training and qualification of personnel and performance demonstration for procedures and equipment is in conformance with Appendix VII and VIII of ASME Section XI, or any other formal program approved by the NRC. ***(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 requirements may not be sufficient to differentiate isolated cracks from inherent geometric conditions, 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 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 defect detected is compared with the requirements of Section XI, IWC 3510. Any relevant conditions that may be detected during the leakage and hydrostatic tests are evaluated in accordance with IWC-3410 and IWC-3516. ***(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. These requirements, in conjunction with enhanced UT techniques, should continue to ensure the timely detection and correction of virtually all forms of this type of degradation but cannot be depended upon to prevent further degradation. ***(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).	
Same as the effect of Erosion/Corrosion on Item D1.1.2 Steam Nozzle and Safe End	*Same as the effect of Erosion/Corrosion on Item D1.1.2 Steam Nozzle and Safe End*	No
Plant-specific aging management program	Plant-specific aging management program will be evaluated	Yes no AMP

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.1.7, D1.1.8	Pressure Boundary and Structural	Secondary Manway & Bolting, Secondary Handhold & Bolting	CS, LAS	Up to 300°C Secondary-side Water Chemistry at 5.3-7.2 MPa	Attrition	Wear	NUREG-1339. EPRI NP-5769. NRC GL 91-17. ASME Section XI, 1989 Edition. *Operating Experience* IEB 82-02.

Existing 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 implemented through NRC Generic Letter 91-17; additional details on bolting integrity outlined in EPRI NP-5769; and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWC 2500-1, examination categories C-D for pressure retaining bolting in Class 2 components, and testing category C-H for system leakage.	***(1) Scope of Program:*** The staff guidance of NRC Generic Letter (GL) 91-17 provide assurance that plant specific comprehensive bolting integrity programs have been implemented to ensure bolting reliability. The NRC staff recommendations and guidelines for a comprehensive bolting integrity program are delineated in NUREG-1339, and the industry's technical basis for the program is outlined in EPRI NP-5769. ***(2) Preventive Actions:*** Selection of bolting material and the use of lubricants and sealants in accordance with guidelines of EPRI NP-5769 and additional requirements of NUREG 1339, prevent or mitigate degradation and failure of all safety-related closure bolting. ***(3) Parameter Monitored/ Inspected:*** The AMP monitors the effects of aging degradation on the intended function of closure bolting by detection of coolant leakage, and by detection and sizing of cracks by inservice inspection (ISI). Inspection requirements of ASME Section XI, Table IWC 2500-1, examination category C-D for pressure retaining bolting greater than 2 in. in diameter specify volumetric examination of the entire length of bolts. However, because most failures have occurred in fasteners 2 in. or smaller, based on IE Bulletin 82-02, enhanced inspection and improved techniques are recommended. These examinations may be conducted on one component among a group of components with similar design and performing similar functions in the system. Requirements for training and qualification of personnel and performance demonstration for procedures and equipment are in conformance with Appendices VII and VIII of ASME Section XI, and additional requirements of EPRI NP-5769. Requirements of Table IWC 2500-1 category C-H specify visual VT-2 (IWA-5240) examination during system leakage test and hydrostatic test of all pressure retaining Class 2 components. ***(4) Detection of Aging Effects:*** Degradation of the closure bolting can not occur without crack initiation. Also, loss of prestress or attrition of the closure bolting would result in leakage. The extent and schedule of inspection assure detection of aging degradation before the loss of intended function of closure bolting. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI is effective and adequate for timely detection of cracks and leakage. 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 cracks in closure bolting are evaluated in accordance with IWC-3100 by comparing ISI results with the acceptance standards of IWC-3400 and IWC-3513 and 3517. Any relevant conditions that may be detected during the leakage and hydrostatic tests are compared with the acceptance standards of IWC-3516. ***(7) Corrective Actions:*** Repair and replacement is in conformance with IWA-4000 and guidelines and recommendations of EPRI NP-5769. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.1.7, D1.1.8	Pressure Boundary and Structural	Primary Manways Bolting, Secondary Manways Handholes Bolting	LAS	Air, with metal temperatures up to 340°C	Loss of Preload	Stress Relaxation	NUREG-1339. EPRI NP-5769. NRC GL 91-17. ASME Section XI, 1989 Edition.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) 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:*** Significant number of incidents have been reported on bolts and nuts that have failed or become degraded because of SCC. Examples of affected fasteners include (a) steam generator and pressurizer manway closures, (b) valve bonnets and pump flange connections on lines 6 in. or greater, and (c) control rod drive and pressurizer heater connections. The bolting integrity programs developed and implemented in accordance with commitments made in response to NRC communications on bolting events have provided effective means of ensuring bolting reliability.	
Recommendations for a comprehensive bolting integrity program delineated in NUREG–1339 on resolution of Generic Safety Issue 29 and implemented through NRC Generic Letter 91-17; additional details on bolting integrity outlined in EPRI NP-5769. Inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, testing categories B-G-1 and B-G-2 for pressure-retaining bolting and B-P for system leakage (Class 1 components); Subsection IWC, Table IWC 2500-1, testing category C-D for all bolting >2 inches in diameter (Class 2 components).	***(1) Scope of Program:*** The staff guidance of NRC Generic Letter (GL) 91-17 provides assurances that plant specific comprehensive bolting integrity programs have been implemented to ensure bolting reliability. The NRC staff recommendations and guidelines for a comprehensive bolting integrity program are delineated in NUREG-1339, and the industry's technical basis for the program is outlined in EPRI NP-5769. ***(2) Preventive Actions:*** Selection of bolting material and the use of lubricants and sealants in accordance with guidelines of EPRI NP-5769 and additional requirements of NUREG 1339, prevent or mitigate degradation and failure of all safety-related closure bolting. ***(3) Parameter Monitored/ Inspected:*** The AMP monitors the effects of aging degradation on the intended function of closure bolting by detection of coolant leakage. Inspection requirements of ASME Section XI testing category B-P specifies visual VT-2 (IWA-5240) examination of Class 1 pressure retaining components during system leakage and hydrostatic testing. Inspection requirements of ASME Section XI testing categories C-D and C-H specify volumetric inspection of bolting >2 inches in diameter and visual VT-2 (IWA-5240) examination of all pressure retaining components (Class 2 components). ***(4) Detection of Aging Effects:*** Degradation of the closure bolting due to loss of prestress or attrition would result in leakage. The extent and schedule of inspection assure detection of aging degradation before the loss of intended function of closure bolting. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI are effective and adequate for timely detection of cracks and leakage. System leakage test is typically conducted at 40-month intervals. ***(6) Acceptance Criteria:*** Inspection results are evaluated in accordance with IWB-3100 (Class 1 components) and IWC-3100 (Class 2 components). Any relevant conditions that may be detected during the leakage and hydrostatic tests are compared with the acceptance standards of IWB-3522 or IWC-3522. ***(7) Corrective Actions:*** Repair and replacement is in conformance with IWA-4000 and recommendations and guidelines of EPRI NP-5769. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.1.9	Pressure Boundary and Structural	Instrument Nozzles	Alloy 600	Chemically Treated Borated Water at temperatures up to 340°C and 15.5 MPa	Crack Initiation and Growth	SCC, IGSCC, Primary Water Stress Corrosion Cracking (PWSCC)	ASME Section XI, 1989 Edition. Reg. Guide 1.45. NRC GL 97-01, EPRI TR-105714, Plant Technical Specifications. *Operating Experience* NRC IN 90-10. NRC IN 96-11.

Existing 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:*** Significant number of incidents have been reported on bolts and nuts that have failed or become degraded. Examples of affected fasteners include (a) steam generator and pressurizer manway closures, (b) valve bonnets and pump flange connections on lines 6 in. or greater, and (c) control rod drive and pressurizer heater connections. The bolting integrity programs developed and implemented in accordance with commitments made in response to NRC communications on bolting events have provided effective means of ensuring bolting reliability.	
Operational limits on leakage from Plant Technical Specifications. Inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination category B-E for pressure-retaining partial penetration welds and testing category B-P for system leakage. PWR primary water chemistry guidelines from EPRI TR-105714.	***(1) Scope of Program:*** The program is focused on managing the effects of stress corrosion cracking (SCC) on the intended function of vessel penetrations. ***(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, based upon guidelines given in EPRI TR-105714. 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 SCC has not been addressed adequately, particularly when demineralizer resins contaminate the reactor coolant system. As described in IN 96-11 and implemented through GL 97-01, applicant should review the primary coolant system water chemistry to identify any primary system resin bed intrusions that exceed the PWR primary water chemistry guidelines. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of IGSCC on the intended function of instrument nozzles by performance monitoring and detection of cracks by inservice inspection (ISI). Testing category B-P specifies visual VT-2 (IWA-5240) examination of all pressure retaining components extending to and including the second closed valve at the boundary extremity, during system leakage test (IWB-5221) and system hydrostatic test (IWB-5230). Leakage detection is in conformance with Position C of Regulatory Guide 1.45. However, in compliance with GL 97-01, applicant should determine whether an augmented inspection program is necessary. ***(4) Detection of Aging Effects:*** Aging effects degradation of the vessel penetrations can not occur without crack initiation. In accordance with GL 97-01, applicant should review whether the scope and schedule of inspection, including leakage detection system, 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	Yes Elements 2 thru 6 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	Primary Manway & Bolting	CS, LAS	Chemically Treated Borated Water at temperatures up to 340°C and 15.5 MPa	Attrition	Wear	ASME Section XI, 1989 Edition. Reg. Guide 1.45.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) program or develop an integrated long-term program which includes periodic inspection and performance monitoring. ***(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 3522 for visual examination. Acceptance of the component for continued service is based on the results of visual examination, supplemental volumetric and surface examination (IWB-3200), corrective measure or repair, analytical evaluation, or replacement (IWB-3132). However, in accordance with GL 97-01, 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 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:*** SCC of Alloy 600 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.	
Inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, testing category B-P for pressure retaining components. Supplemental inspection requirements may be contained in plant-specific Technical Specifications.	***(1) Scope of Program:*** The program is focused on managing the effects of wear due to deterioration of the sealing surfaces of manes and handholds, and closure bolting. ***(2) Preventive Actions:*** Verification of the sealing function by frequent monitoring and timely corrective action prevent or mitigate wear damage of manway and handhold. ***(3) Parameter Monitored/ Inspected:*** The AMP monitors the effects of attrition on the sealing function of manway and handhold by detection of leakage or attrition by inservice inspection (ISI). Inspection requirements of ASME Section XI, Table IWB 2500-1, testing category B-P specify visual VT-2 (IWA-5240) examination of all Class 1 pressure retaining components during system leakage and hydrostatic tests. Leakage detection is in conformance with Position C of Regulatory Guide 1.45. ***(4) Detection of Aging Effects:*** Aging degradation of manway and handhold can not occur without attrition and leakage; extent and schedule of inspection assure detection of leakage or attrition before the loss of the sealing function of manway and handhold. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI should provide for timely detection of leakage. System leakage test is conducted at each refueling outage. ***(6) Acceptance Criteria:*** Any relevant conditions that may be detected during the leakage and hydrostatic tests are evaluated in accordance with IWB-3100 and acceptance standards of IWB-3400 and IWB-3522. ***(7) Corrective Actions:*** Prior to service, corrective measures are	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1. 1.10	Pressure Boundary and Structural	Primary Manway & Bolting	CS, LAS	Air, Leaking Chemically Treated Borated Water and/or Steam at temperatures up to 340°C	Loss of Material	Corrosion/ Boric Acid Wastage of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 Edition.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) needed to meet the requirements of IWB-3142 and IWA-5250. The leakage source and areas of wear 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:*** To date, the present AMP has been effective in ensuring the timely detection and correction of wear degradation in steam generator manway and handhold.	
Implementation of NRC Generic Letter 88-05, and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWC, Table IWB 2500-1, testing category B-P.	***(1) Scope of Program:*** The staff guidance of Generic Letter (GL) 88-05 provides assurances that a program has been implemented consisting of systematic measures to ensure that the effects of corrosion caused by leaking coolant containing boric acid does not lead to degradation of the assurance that the reactor coolant pressure boundary will have a extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture. The program includes (a) determination of principal location, (b) examinations requirements and procedures for locating small leaks, and (c) engineering evaluations and corrective actions. ***(2) Preventive Actions:*** Minimizing reactor coolant leakage by frequent monitoring of the locations where potential leakage could occur and repairing the leaky components as soon as possible, prevent or mitigate boric acid corrosion. ***(3) Parameters Monitored/Inspected:*** The AMP detects coolant leakage by inservice inspection (ISI). Inspection requirements of ASME Section XI, Table IWB 2500-1, testing category B-P specifies visual inspection of all Class 1 pressure-retaining components during system leakage and hydrostatic tests. ***(4) Detection of Aging Effects:*** Aging degradation of the component can not occur without leakage of coolant containing boric acid; extent and schedule of inspection assure detection of leakage before the loss of intended function of the component. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI should provide for timely detection of leakage. System leakage test is 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 IWB-3100 and acceptance standards of NRC Generic Letter 88-05. ***(7) Corrective Actions:*** Prior to service, corrective measures are needed to meet the requirements of NRC Generic Letter 88-05. The leakage source and areas of general corrosion are located. Components with local areas of corrosion that reduces the wall thickness by more than 10% require analytical evaluation to demonstrate acceptability. ***(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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1. 1.10	Pressure Boundary and Structural	Primary Manway Bolting	LAS	Air, Leaking Chemically Treated Borated Water and/or Steam at temperatures up to 340°C	Crack Initiation and Growth	Stress Corrosion Cracking (SCC)	NUREG-1339. EPRI NP-5769. NRC GL 91-17. IEB 82-02. ASME Section XI, 1989 Edition.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) Boric acid wastage observed in nuclear power plants may be classified into two distinct types: (a) corrosion that increases the rate of leakage, e.g., corrosion of closure bolting or fasteners in reactor coolant pressure boundary, and (b) corrosion that occurs some distance from the source of leakage. Some recent incidents of boric acid wastage (IN 86-108 S3) at Calvert Cliffs Unit 1 (Feb. 1994) and Three Mile Island Unit 1 (March 1994) indicate that, although implementation of GL 88-05 ensures timely detection of leakage, there may still be a lack of awareness of the conditions that can lead to boric acid wastage.	
Recommendations for a comprehensive bolting integrity program delineated in NUREG–1339 on resolution of Generic Safety Issue 29 as described in NRC Generic Letter 91-17; additional details on bolting integrity outlined in EPRI NP-5769; and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination categories B-G-1 and B-G-2 for pressure retaining bolting and category B-P for system leakage. Limits on operational leakage specified by Plant Technical Specifications.	***(1) Scope of Program:*** NRC Generic Letter (GL) 91-17 suggests an approach to implement a plant-specific comprehensive bolting integrity programs to ensure bolting reliability. The NRC staff recommendations and guidelines for a comprehensive bolting integrity program are delineated in NUREG-1339, and the industry's technical basis for the program is outlined in EPRI NP-5769. ***(2) Preventive Actions:*** Selection of bolting material and the use of lubricants and sealants in accordance with guidelines of EPRI NP-5769 and additional requirements of NUREG 1339, prevent or mitigate degradation and failure of all safety-related closure bolting. ***(3) Parameter Monitored/ Inspected:*** The AMP monitors the effects of aging degradation on the intended function of closure bolting by detection of coolant leakage, and by detection and sizing of cracks by inservice inspection (ISI). Inspection requirements of ASME Section XI, Table IWB 2500-1, examination category B-G-1 for bolting >2 inches in diameter specify volumetric examination of studs and bolts from top of nut to bottom of the flange hole, and visual VT-I examination of surfaces of nuts, washers, bushings, and flanges, including a 1-inch annular surface of flange surrounding each stud. Examination category B-G-2 for bolting <2 inches in diameter specifics only visual VT-1 examination of surfaces of bolts, studs, and nuts.. However, most failures have occurred in fasteners 2 in. or smaller, and, based on IE Bulletin 82-02, enhanced inspection and improved techniques are recommended. Requirements for training and qualification of personnel and performance demonstration for procedures and equipment is in conformance with Appendices VII and VIII of ASME Section XI, and additional requirements of EPRI NP-5769. Inspection requirements of ASME Section XI testing category B-P specify visual VT-2 (IWA-5240) examination of all pressure retaining components during system leakage and hydrostatic testing. ***(4) Detection of Aging Effects:*** Degradation of the closure bolting can not occur without crack initiation. Also, loss of prestress or attrition of the closure bolting would result in leakage. The extent and schedule of inspection assure detection of aging degradation before the loss of intended function of closure bolting. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI is effective and adequate for timely detection of cracks and leakage. Inspection schedule in	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.2.1	Tube Bundle	Steam Generator Tubes and Sleeves	Alloy 600	Up to 300°C Secondary-side Water Chemistry at 5.3-7.2 MPa	Crack Initiation and Growth	PWSCC	Plant Technical Specifications, Reg. Guide 1.83. Reg. Guide 1.121. NRC GL 95-05. NRC IN 97-26. NRC IN 97-88. EPRI TR-105714. EPRI document "PWR Steam Generator Examination Guidelines: Rev. 5" NEI 97-06

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) accordance with IWB-2400 requires inspection of all bolts, studs, nuts, bushing, and flange surfaces every 10 y. System leakage test is conducted at each refueling outage. ***(6) Acceptance Criteria:*** Any cracks in closure bolting are evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3515 and IWB-3517. Any relevant conditions that may be detected are compared with the acceptance standards of IWB-3522. ***(7) Corrective Actions:*** Repair and replacement is in conformance with IWA-4000 and recommendations and guidelines of EPRI NP-5769. ***(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:*** Significant number of incidents have been reported on bolts and nuts that have failed or become degraded because of boric acid wastage or SCC. Examples of affected fasteners include (a) steam generator and pressurizer manway closures, (b) valve bonnets and pump flange connections on lines 6 in. or greater, and (c) control rod drive and pressurizer heater connections. The present AMP has provided effective means of ensuring bolting reliability.	
Inservice inspection in conformance with Plant Technical Specifications and NRC Regulatory Guide (RG) 1.83 ("Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes." Plugging or repair of defective tubes is governed by NRC Regulatory Guide 1.121 ("Bases for Plugging Degraded PWR Steam Generator Tubes"), but defects at the tube sheet may be shown to satisfy the 1.121 requirements using the P*, F*, or L* criterion. Alternative criteria applicable to cracking at the tube support plates in Westinghouse-designed steam generators under certain circumstances are provided in NRC Generic Letter (GL) 95-05 ("Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking"). Specific guidance on the inservice inspection of tubes is also provided in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5" and NEI 97-06. Supplemental inspection requirements and plugging criteria may be contained in plant-specific Technical Specifications.	***(1) Scope of Program:*** The program deals with the effects of SCC of the steam generator tubes on the integrity of the pressure boundary. Inservice inspection in accordance with Plant Technical Specifications and RG 1.83 deal with the detection of flaws, and RG 1.121 and NRC GL 95-05 describe the basis for determining when flawed tubes must be plugged or repaired. ***(2) Preventive Actions:*** Primary water chemistry guidelines given in EPRI TR-105714 provide guidance on the prevention of PWSCC. ***(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 threaten the integrity 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 SCC. ***(6) Acceptance Criteria:*** Any cracking detected is compared with the requirements of the Plant Technical Specifications, RG 1.121, and GL 95-05. ***(7) Corrective Actions:*** Tubes containing flaws that exceed the maximum permissible size must be plugged or repaired. ***(8 & 9)***	No, provided Plant Technical Specifications conform to EPRI inspection guidelines and NEI 97-06 and are adhered to.

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) ***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.	
Inservice inspection in conformance with Plant Technical Specifications and NRC Regulatory Guide 1.83 ("Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes". Plugging or repair of defective tubes is governed by NRC Regulatory Guide 1.121 ("Bases for Plugging Degraded PWR Steam Generator Tubes"), but defects at the tube sheet may be shown to satisfy the 1.121 requirements using the P*, F*, or L* criterion. Alternative criteria applicable to cracking at the tube support plates in Westinghouse-designed steam generators under certain circumstances are provided in NRC Generic Letter 95-05 ("Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking"). Recommendations on secondary water chemistry given in EPRI TR-102134 ("PWR Secondary Water Chemistry Guidelines—Revision 3"). Specific guidance on the inservice inspection of tubes is also provided in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5" and NEI 97-06. Supplemental inspection requirements and plugging criteria may be contained in plant-specific Technical Specifications.	***(1) Scope of Program:*** The program deals with the effects of SCC of the steam generator tubes on the integrity of the pressure boundary. Plant Technical Specifications and Reg. Guide 1.83 deal with the detection of flaws, and Reg. Guide 1.121 and GL 95-05 describe the basis for determining when flawed tubes must be plugged or repaired. ***(2) Preventive Actions:*** Secondary water chemistry guidelines given in EPRI TR-102134 provide guidance on the prevention of ODSCC. ***(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 threaten the integrity of the tubes. Problems with tube inspection, e.g., failures to detect some flaws, uncertainties in flaw sizing, and inaccuracies in flaw locations, 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 SCC. ***(6) Acceptance Criteria:*** Any cracking detected is compared with the requirements of the Plant Technical Specifications, Reg. Guide 1.121, and GL 95-05. ***(7) Corrective Actions:*** Tubes containing flaws 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. 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 SCC degradation in steam generator tubes.	No, provided Plant Technical Specifications conform to EPRI inspection guidelines and NEI 97-06 and are adhered to.

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.2.1	Tube Bundle	Steam Generator Tubes	Alloy 600	Up to 300°C Secondary-side Water Chemistry at 5.3-7.2 MPa	Crack Initiation and Growth	Intergranu-lar Attack (IGA)	Plant Technical Specifications. Reg. Guide 1.83. Reg. Guide 1.121. NRC GL 95-05. EPRI document "PWR Steam Generator Examination Guidelines: Rev. 5". NEI 97-06
D1.2.1	Tube Bundle	Steam Generator Tubes	Alloy 600	ID Chemically Treated Borated Water up to 340°C; OD up to 300°C Secondary-side Water Chemistry	Cumulative Fatigue Damage	Fatigue	Plant Technical Specifications. GSI-190.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Inservice inspection in conformance with Plant Technical Specifications and NRC Regulatory Guide 1.83 ("Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes". Plugging or repair of defective tubes is governed by NRC Regulatory Guide 1.121 ("Bases for Plugging Degraded PWR Steam Generator Tubes"), but defects at the tube sheet may be shown to satisfy the 1.121 requirements using the P*, F*, or L* criterion. Alternative criteria applicable to cracking at the tube support plates in Westinghouse-designed steam generators under certain circumstances are provided in NRC Generic Letter 95-05 ("Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking"). Recommendations on secondary water chemistry given in EPRI TR-102134 ("PWR Secondary Water Chemistry Guidelines—Revision 3"). Specific guidance on the inservice inspection of tubes is also provided in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5" and NEI 97-06. Supplemental inspection requirements and plugging criteria may be contained in plant-specific Technical Specifications.	***(1) Scope of Program:*** The program deals with the effects of IGA of the steam generator tubes on the integrity of the pressure boundary. Plant Technical Specifications and Reg. Guide 1.83 deal with the detection of flaws, and Reg. Guide 1.121 and GL 95-05 describe the basis for determining when flawed tubes must be plugged or repaired. ***(2) Preventive Actions:*** Secondary water chemistry guidelines given in EPRI TR-102134 provide guidance on the prevention of 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 IGA. ***(6) Acceptance Criteria:*** Any degradation detected is compared with the requirements of the Plant Technical Specifications, Reg. Guide 1.121, and GL 95-05. ***(7) Corrective Actions:*** Tubes containing flaws 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. 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 steam generator tubes.	No, provided Plant Technical Specifications conform to EPRI inspection guidelines and NEI 97-06 and are adhered to.
Components have been designed or evaluated for fatigue for a 40 y design life, according to Plant Technical Specifications or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.2.1	Tube Bundle	Steam Generator Tubes	Alloy 600	Up to 300°C Secondary-side Water Chemistry at 5.3-7.2 MPa	Loss of section thickness	Fretting and wear	Plant Technical Specifications. Reg. Guide 1.83. Reg. Guide 1.121. EPRI document "PWR Steam Generator Examination Guidelines: Rev. 5", NEI 97-06.
D1.2.1	Tube Bundle	Steam Generator Tubes	Alloy 600	Up to 300°C Secondary-side Water Chemistry at 5.3-7.2 MPa	Loss of material	General and Pitting corrosion (OD)	Plant Technical Specifications. Reg. Guide 1.83. Reg. Guide 1.121. EPRI document "PWR Steam Generator Examination Guidelines: Rev. 5", NEI 97-06.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Inservice inspection in conformance with Plant Technical Specifications and NRC Regulatory Guide 1.83 ("Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes". Plugging or repair of defective tubes is governed by NRC Regulatory Guide 1.121 ("Bases for Plugging Degraded PWR Steam Generator Tubes"). Specific guidance on the inservice inspection of tubes is also provided in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5." Specific guidance on the inservice inspection of tubes is also provided in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5" and NEI 97-06. Supplemental inspection requirements and plugging criteria may be contained in plant-specific Technical Specifications.	***(1) Scope of Program:*** The program deals with the effects of the fretting and wear of the steam generator tubes on the integrity of the pressure boundary. Plant Technical Specifications and Reg. Guide 1.83 deal with the detection of flaws, and Reg. Guide 1.121 describes the basis for deciding when flawed tubes must be plugged or repaired. ***(2) Preventive Actions:*** The program provides no guidance or recommendations on measures to prevent 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 prescribed by the program are designed to ensure that flaws cannot attain a depth sufficient to threaten the integrity of the tubes. Guidance on tube inspection is 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) should provide for timely detection of fretting and wear. ***(6) Acceptance Criteria:*** Any cracks detected are compared with the requirements 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. ***(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 continue to ensure the timely detection and correction of fretting and wear but cannot be depended upon to prevent further degradation. ***(10) Operating Experience:*** To date, the present program has been effective in ensuring the timely detection and correction of fretting and wear degradation in steam generator tubes.	No, provided Plant Technical Specifications conform to EPRI inspection guidelines and NEI 97-06 and are adhered to.
Inservice inspection in conformance with Plant Technical Specifications and NRC Regulatory Guide 1.83 ("Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes". Plugging or repair of defective tubes is governed by NRC Regulatory Guide 1.121 ("Bases for Plugging Degraded PWR Steam Generator Tubes"). Recommendations on secondary water chemistry given in EPRI TR-102134 ("PWR Secondary Water Chemistry Guidelines—Revision 3"). Specific guidance on the inservice inspection of tubes is also provided in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5" and NEI 97-06. Supplemental inspection requirements and plugging criteria may be contained in plant-specific Technical Specifications.	***(1) Scope of Program:*** The program deals with the effects of pitting of the steam generator tubes on the integrity of the pressure boundary. Plant Technical Specifications and Reg. Guide 1.83 deal with the detection of flaws, and Reg. Guide 1.121 describes the basis for deciding when flawed tubes must be plugged or repaired. ***(2) Preventive Actions:*** Guidance provided 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 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	No, provided Plant Technical Specifications conform to EPRI inspection guidelines and NEI 97-06 and are adhered to.

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.2.1	Tube Bundle	Steam Generator Tubes	Alloy 600	Up to 300°C Secondary-side Water Chemistry at 5.3-7.2 MPa	Deforma-tion (denting)	Corrosion of tube support plate at intersec-tions with tubes	Plant Technical Specifications. Reg. Guide 1.83. Reg. Guide 1.121. NRC GL 95-05.
D1.2.3	Tube Bundle	Tube Support Lattice Bars	CS	Up to 300°C Secondary-side Water Chemistry at 5.3-7.2 MPa	Loss of section thickness	Erosion/ Corrosion	NRC GL 97-06

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) outage) should provide for timely detection of virtually all forms of pitting. ***(6) Acceptance Criteria:*** Any cracks detected are compared with the requirements 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. This should continue to ensure the timely detection and correction of virtually all forms of pitting but cannot be depended upon to prevent further degradation. ***(10) Operating Experience:*** To date, the present program has been effective in ensuring the timely detection and correction of pitting in steam generator tubes.	
Inservice inspection in conformance with Plant Technical Specifications and NRC Regulatory Guide 1.83 ("Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes". Plugging or repair of defective tubes is governed by NRC Regulatory Guide 1.121 ("Bases for Plugging Degraded PWR Steam Generator Tubes"). Recommendations on secondary water chemistry given in EPRI TR-102134 ("PWR Secondary Water Chemistry Guidelines—Revision 3"). Specific guidance on the inservice inspection of tubes is also provided in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5" and NEI 97-06. Supplemental inspection requirements and plugging criteria may be contained in plant-specific Technical Specifications.	***(1) Scope of Program:*** The program deals with the effects of denting of the steam generator tubes on the integrity of the pressure boundary. Plant Technical Specifications and Reg. Guide 1.83 deal with the detection of flaws, and Reg. Guide 1.121 describes the basis for deciding when degraded tubes must be plugged or repaired. ***(2) Preventive Actions:*** Guidance provided 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 requirements 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. This should continue to ensure the timely detection and correction of virtually all forms of denting degradation but cannot be depended upon to prevent further degradation. ***(10) Operating Experience:*** The present AMP has been effective in ensuring timely detection and correction of denting in steam generator tubes.	No
Plant-specific aging management program will be evaluated and redactions 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 redactions of NRC Generic Letter 97-06 for creation of an inspection program for steam generator internals is to be considered.	Yes no generic AMP

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.2.4	Tube Bundle	Tube Mechanical Plugs (Westinghouse)	Alloy 600. Alloy 690	Chemically Treated Borated Water at temperatures up to 340°C and 15.5 MPa	Crack Initiation and Growth	PWSCC	Plant Technical Specifications; NRC IN 89-33; NRC BL 89-01 & 89-01, S 1 and S 2; NRC IN 94-87; WCAP-12244; EPRI TR-102134
D1.2.4	Tube Bundle	Tube Mechanical Plugs (Babcock & Wilcox)	Alloy 600, Alloy 690	Chemically Treated Borated Water at temperatures up to 340°C and 15.5 MPa	Crack Initiation and Growth	PWSCC	Plant Technical Specifications; NRC Reg. Guide 1.83; NRC IN 89-65

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Inservice inspection in conformance with Plant Technical Specifications and NRC Regulatory Guide 1.83 ("Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes". Correlations for estimating plug life contained in Westinghouse reports WCAP-12244 and WCAP 12245. Specific guidance on the inservice inspection of plugs is provided in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5" and NEI 97-06. Supplemental inspection requirements may be contained in plant-specific Technical Specifications.	***(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-102134 can significantly reduce PWSCC. The program also recommends that certain susceptible heats 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 tubes. 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. However, note item (4) immediately above ***(6) Acceptance Criteria:*** Any cracking detected requires replacement. Plug lives 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. However, 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 Specifications conform to EPRI inspection guidelines and NEI 97-06 and are adhered to.
Inservice inspection in conformance with Plant Technical Specifications and NRC Regulatory Guide 1.83 ("Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes". Specific guidance on the inservice inspection of plugs is provided in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5" and NEI 97-06. Supplemental inspection requirements may be contained in plant-specific Technical Specifications.	***(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-102134 can significantly reduce PWSCC. The program also recommends that certain susceptible heats 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 tubes. ***(5) Monitoring and Trending:*** Required inspection intervals (typically ≈18 months or each refueling or maintenance/repair outage) appears to provide for timely detection of SCC, and no leakage due to failed plugs has been reported. ***(6) Acceptance Criteria:*** Any cracking detected is	No, provided Plant Technical Specifications conform to EPRI inspection guidelines and NEI 97-06 and are adhered to.

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.2..5	Tube Bundle	Tubes in the Region of Repair Sleeves	Alloy 600	Chemically Treated Borated Water at temperatures up to 340°C and 15.5 MPa	Crack Initiation and Growth	PWSCC	Plant Technical Specifications; NRC Reg. Guide 1.83; NRC IN 94-05; EPRI document "PWR Steam Generator Examination Guidelines: Rev. 5"; NEI 97-06
D1.3.1	Upper Assembly and Separators	Feedwater Inlet Ring and Supports	CS	Up to 300°C Secondary-side Water Chemistry at 5.3-7.2 MPa	Loss of Material	Erosion/ corrosion	ASME Section XI, 1989 Edition.; NRC IN 91-19; NRC LER 50-362/90-05-01; Combustion Engineering Info-bulletin 90-04

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) compared with the requirements of the Plant Technical Specifications. ***(7) Corrective Actions:*** Tube plugs containing flaws 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 continue to 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. However, any Alloy 600 mechanical plugs remaining in service must be considered susceptible.	
Inservice inspection in conformance with Plant Technical Specifications and NRC Regulatory Guide 1.83 ("Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes". Specific guidance on the inservice inspection of repair sleeves is provided in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5" and NEI 97-06. Supplemental inspection requirements may be contained in plant-specific Technical Specifications.	***(1) Scope of Program:*** The program deals with the periodic inspection of steam generator tube repair sleeves and the associated parent tubing material. ***(2) Preventive Actions:*** Guidance on primary water chemistry provided in EPRI TR-102134 can significantly reduce PWSCC. ***(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 tubes. ***(5) Monitoring and Trending:*** Required inspection intervals (typically ≈18 months or each refueling or maintenance/repair outage) provide for timely detection of SCC. ***(6) Acceptance Criteria:*** Any cracking detected is compared with the requirements of the Plant Technical Specifications. ***(7) Corrective Actions:*** Tubes containing flaws must be repaired or plugged. ***(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 continue to 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 (NRC Information Notice 94-05).	No
Periodic inservice inspections in conformance with Plant Technical Specifications. Additional recommendations on inservice inspection are contained in Combustion Engineering Info-bulletin 90-04: ("Feedwater Distribution System Degradation"), and supplemental inspection requirements may be included 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	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.4.1	Piping and Fittings	Main Steam	CS	Up to 300°C Steam	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. ANSI B31.1, GSI-190.
D1.4.1	Piping and Fittings	Main Steam	CS	Up to 300°C Steam	Wall Thinning	E/C	*Same as the effect of Erosion/Corrosion on Item D1.1.2 Steam Nozzle and Safe End*
D1.4.2	Piping and Fittings	Feedwater	CS	Up to 225°C Secondary-side Water Chemistry	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. ANSI B31.1. GSI-190.
D1.4.2	Piping and Fittings	Feedwater	CS	Up to 225°C Secondary-side Water Chemistry	Wall Thinning	E/C	*Same as the effect of Erosion/Corrosion on Item D1.1.2 Steam Nozzle and Safe End*
D1.4.3	Piping and Fittings	Auxiliary Feedwater	CS	<90°C Secondary-side Water Chemistry	Wall Thinning	E/C	*Same as the effect of Erosion/Corrosion on Item D1.1.2 Steam Nozzle and Safe End*

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) forms of degradation. ***(6) Acceptance Criteria:*** Any defect detected is compared with the requirements of the Plant Technical Specifications and with the recommendations of Combustion Engineering Info-bulletin 90-04. These requirements 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 requirements 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).	
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), ANSI B31.1, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Same as the effect of Erosion/Corrosion on Item D1.1.2 Steam Nozzle and Safe End	*Same as the effect of Erosion/Corrosion on Item D1.1.2 Steam Nozzle and Safe End*	No
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or ANSI B31.1, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Same as the effect of Erosion/Corrosion on Item D1.1.2 Steam Nozzle and Safe End	*Same as the effect of Erosion/Corrosion on Item D1.1.2 Steam Nozzle and Safe End*	No
Same as the effect of Erosion/Corrosion on Item D1.1.2 Steam Nozzle and Safe End	*Same as the effect of Erosion/Corrosion on Item D1.1.2 Steam Nozzle and Safe End*	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.4.1	Piping and Fittings	Main Steam	CS	Up to 300°C Steam	Loss of Material	Crevice and Pitting Corrosion	ASME Section XI, 1989 Edition
D1.4.2, D1.4.3	Piping and Fittings	Feedwater, Auxiliary Feedwater	CS	Up to 300°C Secondary-side Water Chemistry	Loss of Material	Crevice and Pitting Corrosion	ASME Section XI, 1989 Edition
D1.5.1	Safety and Relief Valves	Body	CS	Up to 300°C Steam or Secondary-side Water Chemistry	Wall Thinning	E/C	*Same as the effect of Erosion/Corrosion on Item D1.1.2 Steam Nozzle and Safe End*

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
The program includes preventive measures to mitigate crevice corrosion by control of secondary water chemistry and, based on plant specifications, inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWC 2500-1, examination category C-H for pressure retaining Class 2 components.	***(1) Scope of Program:*** The program relies on preventive measures to mitigate general, crevice, or pitting corrosion, and inservice inspection (ISI) to monitor the effects of corrosion on the intended function of the main steam system components. ***(2) Preventive Actions:*** Mitigation is by monitoring and control of secondary water chemistry and demineralized water chemistry to minimize concentration of corrosive impurities such as chlorides, sulfates, and oxygen. Chemical species and impurities are monitored either continuous, daily, weekly, or as required based on plant operating conditions. Corrective actions are taken when specified values are exceeded. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of corrosion by detection of coolant leakage by inservice inspection (ISI). Inspection requirements of ASME Section XI specify visual VT-2 (IWA-5240) examination during system leakage test and hydrostatic test of all pressure retaining Class 2 components up to the first normally closed valve, according to Table IWC 2500-1 category C-H. ***(4) Detection of Aging Effects:*** Degradation of the component due to corrosion can not occur without leakage of coolant; extent and schedule of inspection assure detection of leakage before the loss of intended function of the component. ***(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-3516. ***(7) Corrective Actions:*** Prior to service, corrective measures are needed to meet the requirements of IWC-3132 and IWA-5250. Repair and replacement are in conformance with IWA-4000 and 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:*** No significant corrosion related problem has been reported for piping and fittings in secondary side systems of steam generators.	No
Same as the effect of Crevice and Pitting Corrosion on Item D1.4.1 Piping and Fittings in Main Steam System	*Same as the effect of Crevice and Pitting Corrosion on Item D1.4.1 Piping and Fittings in Main Steam System*	No
Same as the effect of Erosion/Corrosion on Item D1.1.2 Steam Nozzle and Safe End	*Same as the effect of Erosion/Corrosion on Item D1.1.2 Steam Nozzle and Safe End*	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
- D2.1.5 Feed Water and Auxiliary Feed Water Nozzles & Safe Ends
- D2.1.6 Steam Nozzles & Safe Ends
- D2.1.7 Instrument & Drain Nozzles
- D2.1.8 Primary Manways & Bolting
- D2.1.9 Secondary Manways Handholes & Bolting

D2.2 Tube Bundle

- D2.2.1 Tubes

D2.3 Piping and Fittings

- D1.4.1 Main Steam
- D1.4.2 Feedwater
- D1.4.3 Auxiliary Feedwater

D2.4 Safety and Relief Valves

- D1.5.1 Body

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, external primary and secondary water/steam nozzles and safe ends, and portions of main steam, feedwater, and auxiliary feedwater systems extending from the steam generator up to the first isolation valve outside of containment or to the first anchor point. 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, including portions of main steam, feedwater, and auxiliary feedwater systems extending up to the first isolation valve outside of containment, is classified as Group B Quality Standard. The aging management program for the lines that penetrate the containment, and associated isolation valves, is reviewed in Table V C.

The valves internals are considered to be active components. They perform their intended functions with moving parts or with a change in configuration and are not subject to aging management review pursuant to 10 CFR 54.21(a)(1)(i).

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).

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) Cladding	Up to 300°C Secondary -side Water Chemistry at 5.3-7.2 MPa	Crack Initiation and Growth	Stress Corrosion Cracking (SCC), Intergranu -lar Stress Corrosion Cracking (IGSCC)	ASME Section XI, 1989 Edition. Reg. Guide 1.43. Reg. Guide 1.44, EPRI TR-102134

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Guidelines of Regulatory Guide (RG) 1.44 to avoid sensitization of stainless steels (SSs) and requirements of RG 1.43 to prevent underclad cracking; and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWC, Table IWC 2500-1, examination category C-A for pressure retaining welds and category C-H for all pressure retaining components. Secondary water chemistry guidelines of EPRI TR-102134. Supplemental inspection requirements may be contained in Plant Technical Specifications.	***(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 upper and lower heads and tubesheets. ***(2) Preventive Actions:*** Selection of material in compliance with the requirements of Regulatory Guide (RG) 1.44 prevent or mitigate IGSCC, and requirements of RG 1.43 prevent underclad cracking. In addition, secondary water chemistry guidelines given in EPRI TR-105714 provide guidance on the prevention of SCC. Also, control of halogens and oxygen in the primary water to less than 5 and 0.01 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, 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 ISI in accordance with ASME Section XI to detect possible degradation. ***(3) Parameters Monitored/ Inspected:*** The program monitors flaw size and depth, or, alternatively, remaining wall thickness in pressure retaining welds. Inspection requirements of Table IWC 2500-1, examination category C-H specifies visual VT-2 inspection of all pressure retaining components. Requirements for training and qualification of personnel and performance demonstration for procedures and equipment is in conformance with Appendices VII and VIII of ASME Section XI, or any other formal program approved by the NRC. ***(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, 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:*** Required inspection intervals should provide for timely detection of virtually all forms of degradation. ***(6) Acceptance Criteria:*** Any defect detected is compared with the requirements of Section XI, IWC 3516. These requirements should ensure structural integrity. ***(7) Corrective Actions:*** Welds containing flaws that exceed the maximum permissible size must be repaired. Repair and replacement are in conformance with IWA-4000 and 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. These requirements, in conjunction with enhanced UT techniques, should continue to ensure the timely detection and correction of virtually all forms of this type of degradation but cannot be depended upon to prevent further degradation. ***(10) Operating Experience:*** The present program has been effective in managing the effects of SCC degradation in once-through steam generator upper and lower heads.	Yes Element 4 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 Chemicall y Treated Borated Water and/or Steam at temperatu res up to 340°C	Loss of Material	Corrosion /Boric Acid Wastage of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 Edition.
D2.1.3	Pressure Boundary and Structural	Primary Nozzles and Safe Ends	Carbon Steel (CS) with SS cladding, Safe Ends: SS	Chemicall y Treated Borated Water up to 340°C and 15.2 MPa	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. GSI-190.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Implementation of NRC Generic Letter 88-05, and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWC, Table IWC 2500-1, testing category C-H for all pressure retaining components.	***(1) Scope of Program:*** The staff guidance of Generic Letter (GL) 88-05 provides assurances that a program has been implemented consisting of systematic measures to ensure that the effects of corrosion caused by leaking coolant containing boric acid does not lead to degradation of the assurance that the reactor coolant pressure boundary will have a extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture. The program includes (a) determination of principal location, (b) examinations requirements and procedures for locating small leaks, and (c) engineering evaluations and corrective actions. ***(2) Preventive Actions:*** Minimizing reactor coolant leakage by frequent monitoring of the locations where potential leakage could occur and repairing the leaky components as soon as possible, prevent or mitigate boric acid corrosion. ***(3) Parameters Monitored/Inspected:*** The AMP detects coolant leakage 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 effects degradation of the component can not occur without leakage of coolant containing boric acid; extent and schedule of inspection assure detection of leakage before the loss of intended function of the component. ***(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 NRC GL 88-05. ***(7) Corrective Actions:*** Prior to service, corrective measures are needed to meet the requirements of GL 88-05. The leakage source and areas of general corrosion are located. Components with local areas of corrosion that reduces the wall thickness by more than 10% require analytical evaluation to demonstrate acceptability. ***(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:*** Boric acid wastage in nuclear power plants may either be classified as (a) corrosion that increases the rate of leakage, e.g., corrosion of closure bolting, or (b) corrosion that occurs some distance from the source of leakage. Some recent incidents of boric acid wastage (IN 86-108 S3) indicate that, although implementation of GL 88-05 ensures timely detection of leakage, there may still be a lack of awareness of the conditions that can lead to boric acid wastage.	No
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a) or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
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)	Chemicall y Treated Borated Water at temperatu res up to 340°C and 15.2 MPa	Crack Initiation and Growth	SCC, IGSCC	ASME Section XI, 1989 Edition. Reg. Guide 1.43. Reg. Guide 1.44. NRC IN 84-18.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Guidelines of Regulatory Guide (RG) 1.44 to avoid sensitization and RG 1.43 to prevent underclad cracking. Primary water chemistry guidelines of EPRI TR-105714. Inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination category B-D for nozzle-to-vessel welds, category B-F for pressure-retaining dissimilar metal welds in vessel nozzles, and testing category B-P for system leakage	***(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 reactor vessel nozzles. ***(2) Preventive Actions:*** Selection of material in compliance with the requirements of Regulatory Guide (RG) 1.44 prevent or mitigate IGSCC, and requirements of RG 1.43 prevent underclad cracking. In addition, primary water chemistry guidelines given in EPRI TR-102134 provide guidance on the prevention of SCC. Also, control of halogens and oxygen in the primary water to less than 5 and 0.01 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, 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 ISI in accordance with ASME Section XI to detect possible degradation. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of IGSCC on the intended function of reactor vessel nozzles by detection and sizing of cracks by ISI. Inspection requirements of Table IWC 2500-1, examination category C-A specifies volumetric inspection for all pressure retaining welds. Requirements for training and qualification of personnel and performance demonstration for procedures and equipment is in conformance with Appendices VII and VIII of ASME Section XI, or any other formal program approved by the NRC. ***(4) Detection of Aging Effects:*** Aging degradation of the reactor vessel nozzles can not occur without crack initiation. 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 IWC-2400 should provide timely detection of cracks. System leakage test is typically conducted at 40-month intervals. ***(6) Acceptance Criteria:*** Any IGSCC degradation is evaluated in accordance with IWC-3100 by comparing ISI results with the acceptance standards of IWC-3410. Planar and liner flaws are sized according to IWA-3300 and IWA-3400. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWA-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	Yes Element 4 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	ASME Section III, 1989 Edition. GSI-190.
D2.1.4	Pressure Boundary and Structural	Shell Assembly	CS	Up to 300°C Secondary -side Water Chemistry at 5.3-7.2 MPa	Loss of Material	Crevice and Pitting Corrosion	ASME Section XI, 1989 Edition. EPRI TR-102134. 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) will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** The primary pressure boundary components of PWRs are generally not affected by SCC because of low dissolved oxygen levels and control of primary water chemistry. The potential of SCC exists from inadvertent introduction of contaminants into the primary coolant system; sources are (a) contaminants in boric acid and (b) spent fuel pool exposed to airborne contaminants (IN 84–18).	
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a) or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
The program includes preventive measures to mitigate crevice corrosion by recommendations on secondary water chemistry of EPRI TR-102134 and, based on Plant Technical Specifications, inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a) Table IWC 2500-1, examination category C-A for pressure retaining welds in pressure vessels, and examination category C-H for pressure retaining Class 2 components.	***(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 and by detection of coolant leakage by inservice inspection (ISI). 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 visual VT-2 (IWA-5240) examination during system leakage and hydrostatic tests of all pressure retaining Class 2 components. Requirements for training and qualification of personnel and performance demonstration for procedures and equipment is in conformance with Appendix VII and VIII of ASME Section XI, or any other formal program approved by the NRC. ***(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 requirements may not be sufficient to differentiate isolated cracks from inherent geometric conditions, 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. ***(6) Acceptance Criteria:*** Any defect detected is compared with the requirements of Section XI, IWC 3510. Any relevant conditions that may be detected during the leakage and hydrostatic tests are evaluated in accordance with IWC-3400 and IWC-3516. ***(7) Corrective Actions:*** Welds containing flaws that exceed the maximum	Yes Element 4 should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D2.1.5	Pressure Boundary and Structural	FW and AFW Nozzles & Safe Ends	CS	Up to 300°C Secondary -side Water Chemistry at 5.3-7.2 MPa	Wall Thinning	Erosion/ Corrosion (E/C)	NUREG-1344. EPRI NSAC-202L-R2. NRC IN 93-21. EPRI TR-102134. *Operating Experience* NRC BL 87-01. NRC GL 89-08. NRC IN 89-53. NRC IN 91-18. NRC IN 91-18 S1. NRC IN 91-28. NRC IN 92-35. NRC IN 95-11. NRC IN 97-84.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) 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. These requirements, in conjunction with enhanced UT techniques, should continue to ensure the timely detection and correction of virtually all forms of this type of degradation but cannot be depended upon to prevent further degradation. ***(10) Operating Experience:*** This AMP has resulted in the timely detection and correction of corrosion in several steam generators in the U.S. (NRC Information Notices 82-37 & 85-65).	
Program delineated in NUREG-1344 for single phase lines and implemented through NRC Generic Letter 89-08; CHECWORKS Code; EPRI guidelines of NSAC-202L-R2 for effective erosion/corrosion program; and water chemistry program based on EPRI guidelines for secondary water chemistry (EPRI TR-102134).	***(1) Scope of Program:*** The AMP includes NUMARC program delineated in Appendix A of NUREG-1344 and implemented through NRC Generic Letter (GL) 89-08; CHECWORKS computer Code; and EPRI guidelines of NSAC-202L-R2. The program includes the following recommendations: (a) conduct appropriate analysis and limited baseline inspection, (b) determine the extent of thinning and repair/replace components, and (c) perform follow-up inspections to confirm or quantify and take longer term corrective actions. Technical aspects of the CHECWORKS Code, including the parameters and inputs, are acceptable. However, the EPRI guidance document NSAC-202L-R2 (April 1999) is too general to ensure applicant's flow-accelerated corrosion program will be effective in managing aging in safety-related systems. ***(2) Preventive Actions:*** The rate of E/C is affected by piping material, geometry and hydrodynamic conditions, and operating conditions such as temperature, pH, and dissolved oxygen content. Mitigation is by selecting material considered resistant to E/C, adjusting water chemistry and operating conditions, and improving hydrodynamic conditions through design modifications. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of E/C on the intended function of piping by measuring wall thickness by nondestructive examination and performing analytical evaluations. The inspection program delineated in NUREG-1344 requires ultrasonic or radiographic testing of 10 most susceptible locations and 5 additional locations based on unique operating conditions or special considerations. For each location outside the acceptance guidelines, the inspection sample is expanded based on engineering judgment. Analytical models such as those incorporated into the CHECWORKS code are used to predict E/C in piping systems based on specific plant data including material and hydrodynamic and operating conditions. The inspection data are used to calibrate and benchmark the models and code. ***(4) Detection of Aging Effects:*** Aging degradation of piping and fittings occurs by wall thinning; extent and schedule of inspection assure detection of wall thinning before the loss of intended function of the piping. ***(5) Monitoring and Trending:*** Inspection schedule of NUREG-1344 and EPRI	Yes, Element 1 should be further evaluated

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 Steam	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. GSI-190.
D2.1.6	Pressure Boundary and Structural	Steam Nozzles & Safe Ends	CS	Up to 300°C Steam	Wall Thinning	Erosion/ Corrosion (E/C)	*Same as the effect of Erosion/ Corrosion on Item D2.1.5 FW and AFW Nozzles and Safe Ends*

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) guidelines should provide for timely detection of leakage. Inspections and analytical evaluations are performed during plant outage. If analysis shows unacceptable conditions, inspection of initial sample is performed within 6 months. ***(6) Acceptance Criteria:*** Inspection results are used to calculate number of refueling or operating cycles remaining before the component reaches Code minimum allowable wall thickness. If calculations indicate that an area will reach Code minimum (plus 10% margin), the component must be repaired or replaced. However, NRC staff has identified the problems in implementing E/C program that pertain to weakness or errors in (a) using predictive models, (b) calculating minimum wall thickness acceptance criteria, (c) analyzing the results of UT examinations, and (d) assessment of E/C program activities (NRC Information Notice IN 93-21). ***(7) Corrective Actions:*** Prior to service, repair or replace to meet the requirements of NUREG-1344. Follow-up inspections are performed to confirm or quantify thinning and take longer term corrective actions such as adjustment of chemistry and operating parameters, or selection of materials resistant to E/C. However, NRC staff has identified weakness or errors in (a) dispositioning components after reviewing the results of the inspection analysis, and (b) repairing or replacing components that failed to meet the acceptance criteria (IN 93-21). ***(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:*** Wall-thinning problems in single-phase systems have occurred in feedwater and condensate systems (NRC Bulletin No. 87-01, INs 81-28, 92-35, 95-11), and in two-phase piping in extraction steam lines (INs 89-53, 97-84) and moisture separation reheater and feedwater heater drains (INs 89-53, 91-18, 93-21, 97-84). The AMP outlined in NUREG-1344 and EPRI report and implemented through GL 89-08 has provided effective means of ensuring the structural integrity of all high-energy carbon steel systems.	
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a) or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Same as the effect of Erosion/Corrosion on Item D2.1.5 FW and AFW Nozzles and Safe Ends	*Same as the effect of Erosion/Corrosion on Item D2.1.5 FW and AFW Nozzles and Safe Ends*	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D2.1.7	Pressure Boundary and Structural	Instruments & Drain Nozzles	Alloy 600	Chemicall y Treated Borated Water at temperatu res up to 340°C and 15.2 MPa	Crack Initiation and Growth	SCC, IGSCC, Primary Water Stress Corrosion Cracking (PWSCC)	ASME Section XI, 1989 Edition. Reg. Guide 1.45. NRC IN 90-10. NRC IN 96-11. NRC GL 97-01. Plant Technical Specifications

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Operational limits from Plant Technical Specifications. Inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination category B-E for pressure-retaining partial penetration welds and testing category B-P for system leakage. PWR primary water chemistry guidelines from EPRI TR-105714.	***(1) Scope of Program:*** The program is focused on managing the effects of stress corrosion cracking (SCC) on the intended function of vessel penetrations. ***(2) Preventive Actions:*** Control of halogens and oxygen in the primary water to less than 5 and 0.01 ppm, respectively, during operation, and monitor and control of water chemistry during shut down, mitigate potential of SCC, based upon guidelines given in EPRI TR-105714. 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 SCC has not been addressed adequately, particularly when demineralizer resins contaminate the reactor coolant system. As described in IN 96-11 and implemented through GL 97-01, applicant should review the primary coolant system water chemistry to identify any primary system resin bed intrusions that exceed the PWR primary water chemistry guidelines. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of IGSCC on the intended function of instrument nozzles by performance monitoring and detection of cracks by inservice inspection (ISI). Testing category B-P specifies visual VT-2 (IWA-5240) examination of all pressure retaining components extending to and including the second closed valve at the boundary extremity, during system leakage test (IWB-5221) and system hydrostatic test (IWB-5230). Leakage detection is in conformance with Position C of Regulatory Guide 1.45. However, in compliance with GL 97-01, applicant should determine whether an augmented inspection program is necessary. ***(4) Detection of Aging Effects:*** Aging effects degradation of the vessel penetrations can not occur without crack initiation. In accordance with GL 97-01, applicant should review whether the scope and schedule of inspection, including leakage detection system, 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 and performance monitoring. ***(6) Acceptance Criteria:*** Any IGSCC degradation is evaluated in accordance with WB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and 3522 for visual examination. Acceptance of the component for continued service is based on the results of visual examination, supplemental volumetric and surface examination (IWB-3200), corrective measure or repair, analytical evaluation, or replacement (IWB-3132). However, in accordance with GL 97-01, 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 are in conformance with IWA-4000 and IWB-4000, and	Yes Elements 2 thru 6 should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D2.1.4 thru D2.1.8	Pressure Boundary and Structural	External Surfaces of Shell Assembly, FW and AFW Nozzles, and Steam Nozzles; and Bolting of Primary Manways, and Secondary Manways & Handholes	CS, LAS	Air, Leaking Chemicall y Treated Borated Water and/or Steam at temperatu res up to 340°C	Loss of Material	Corrosion /Boric Acid Wastage of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 Edition.
D2.1.8, D2.1.9	Pressure Boundary and Structural	Primary Manways Bolting, Secondary Manways Handholes Bolting	LAS	Air, with metal temperatu res up to 340°C	Loss of Preload	Stress Relaxation	NUREG-1339. EPRI NP-5769. NRC GL 91-17. ASME Section XI, 1989 Edition.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) 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:*** SCC of Alloy 600 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.	
Implementation of NRC Generic Letter 88-05, and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWC, Table IWC 2500-1, testing category C-H for all pressure retaining components.	*Same as the effect of Corrosion/Boric Acid Wastage on Items D2.1.1 Upper and Lower Heads*	No
Recommendations for a comprehensive bolting integrity program delineated in NUREG–1339 on resolution of Generic Safety Issue 29 and implemented through NRC Generic Letter 91-17; additional details on bolting integrity outlined in EPRI NP-5769; and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWC, Table IWC 2500-1, testing category C-D for all bolting >2 inches in diameter.	***(1) Scope of Program:*** The staff guidance of NRC Generic Letter (GL) 91-17 provides assurances that plant specific comprehensive bolting integrity programs have been implemented to ensure bolting reliability. The NRC staff recommendations and guidelines for a comprehensive bolting integrity program are delineated in NUREG-1339, and the industry's technical basis for the program is outlined in EPRI NP-5769. ***(2) Preventive Actions:*** Selection of bolting material and the use of lubricants and sealants in accordance with guidelines of EPRI NP-5769 and additional requirements of NUREG 1339, prevent or mitigate degradation and failure of all safety-related closure bolting. ***(3) Parameter Monitored/ Inspected:*** The AMP monitors the effects of aging degradation on the intended function of closure bolting by detection of coolant leakage. Inspection requirements of ASME Section XI testing categories C-D and C-H specify volumetric inspection of bolting >2 inches in diameter and visual VT-2 (IWA-5240) examination of all pressure retaining components.***(4) Detection of Aging Effects:*** Degradation of the closure bolting due to loss of prestress or attrition would result in leakage. The extent and schedule of inspection assure detection of aging degradation before the loss of intended function of closure bolting. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI are effective and adequate for timely detection of cracks and leakage. System leakage test is typically conducted at 40-month intervals. ***(6) Acceptance Criteria:*** Inspection results are evaluated in accordance with IWC-3100. Any relevant conditions that may be detected during the leakage and hydrostatic tests are compared with the acceptance standards of IWC-3522.	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D2.1.9	Pressure Boundary and Structural	Secondary Manways and Handholes	CS	Air, Leaking Secondary -side Water and/or Steam at temperatu res up to 300°C	Wall Thinning	Erosion/ Corrosion (E/C)	ASME Section XI, 1989 Edition. Reg. Guide 1.45.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) ***(7) Corrective Actions:*** Repair and replacement is in conformance with IWA-4000 and recommendations and guidelines of EPRI NP-5769. ***(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:*** Significant number of incidents have been reported on bolts and nuts that have failed or become degraded. Examples of affected fasteners include (a) steam generator and pressurizer manway closures, (b) valve bonnets and pump flange connections on lines 6 in. or greater, and (c) control rod drive and pressurizer heater connections. The bolting integrity programs developed and implemented in accordance with commitments made in response to NRC communications on bolting events have provided effective means of ensuring bolting reliability.	
Inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWC, Table IWC 2500-1, testing category C-H for pressure retaining components.	***(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. Leakage detection is in conformance with Position C of Regulatory Guide 1.45. ***(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-3410 and IWC-3516. ***(7) Corrective Actions:*** Prior to service, corrective measures are needed to meet the requirements of IWC-3132. 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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D2.2.1	Tube Bundle (Babcock & Wilcox)	Steam Generator Tubes and Sleeves	Alloy 600	Chemi-cally Treated Borated Water up to 340°C and 15.2 MPa	Crack Initiation and Growth	PWSCC	Plant Technical Specifications, Reg. Guide 1.83. Reg. Guide 1.121. NRC GL 95-05. NRC IN 97-26. NRC IN 97-88. EPRI TR-105714, EPRI document "PWR Steam Generator Examination Guidelines: Rev. 5" NEI 97-06
D2.2.1	Tube Bundle (Babcock & Wilcox)	Steam Generator Tubes	Alloy 600	Up to 300°C Secondary -side Water Chemistry at 5.3-7.2 MPa	Crack Initiation and Growth	Outer Diameter Stress Corrosion Cracking (ODSCC)	Plant Technical Specifications. Reg. Guide 1.83. Reg. Guide 1.121. NRC GL 95-05, EPRI document "PWR Steam Generator Examination Guidelines: Rev. 5", NEI 97-06

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Inservice inspection in conformance with Plant Technical Specifications and NRC Regulatory Guide (RG) 1.83 ("Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes." Plugging or repair of defective tubes is governed by NRC Regulatory Guide 1.121 ("Bases for Plugging Degraded PWR Steam Generator Tubes"). Specific guidance on the inservice inspection of tubes is also provided in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5" and NEI 97-06. Supplemental inspection requirements and plugging criteria may be contained in Plant Technical Specifications.	***(1) Scope of Program:*** The program deals with the effects of SCC of the steam generator tubes on the integrity of the pressure boundary. Inservice inspection in accordance with Plant Technical Specifications and RG 1.83 deal with the detection of flaws, and RG 1.121 and NRC GL 95-05 describe the basis for determining when flawed tubes must be plugged or repaired. ***(2) Preventive Actions:*** Primary water chemistry guidelines given in EPRI TR-105714 provide guidance on the prevention of PWSCC. ***(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 threaten the integrity 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 SCC. ***(6) Acceptance Criteria:*** Any cracking detected is compared with the requirements of the Plant Technical Specifications, RG 1.121, and GL 95-05. ***(7) Corrective Actions:*** Tubes containing flaws 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. 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 Specifications conform to EPRI inspection guidelines and NEI 97-06 and are adhered to.
Inservice inspection in conformance with Plant Technical Specifications and NRC Regulatory Guide 1.83 ("Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes". Plugging or repair of defective tubes is governed by NRC Regulatory Guide 1.121 ("Bases for Plugging Degraded PWR Steam Generator Tubes"). Recommendations on secondary water chemistry given in EPRI TR-102134 ("PWR Secondary Water Chemistry Guidelines—Revision 3"). Specific guidance on the inservice inspection of	***(1) Scope of Program:*** The program deals with the effects of SCC of the steam generator tubes on the integrity of the pressure boundary. Plant Technical Specifications and Reg. Guide 1.83 deal with the detection of flaws, and Reg. Guide 1.121 and GL 95-05 describe the basis for determining when flawed tubes must be plugged or repaired. ***(2) Preventive Actions:*** Secondary water chemistry guidelines given in EPRI TR-102134 provide guidance on the prevention of ODSCC. ***(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	No, provided Plant Technical Specifications conform to EPRI inspection guidelines and NEI 97-06 and 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)	Steam Generator Tubes	Alloy 600	Up to 300°C Secondary -side Water Chemistry at 5.3-7.2 MPa	Crack Initiation and Growth	Intergranu -lar Attack (IGA)	Plant Technical Specifications. Reg. Guide 1.83. Reg. Guide 1.121. NRC GL 95-05, EPRI document "PWR Steam Generator Examination Guidelines: Rev. 5", NEI 97-06

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
(continued from previous page) tubes is also provided in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5" and NEI 97-06. Supplemental inspection requirements and plugging criteria may be contained in plant-specific Technical Specifications.	*(continued from previous page)* cannot attain a depth sufficient to threaten the integrity of the tubes. Problems with tube inspection, e.g., failures to detect some flaws, uncertainties in flaw sizing, and inaccuracies in flaw locations, 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 SCC. ***(6) Acceptance Criteria:*** Any cracking detected is compared with the requirements of the Plant Technical Specifications, Reg. Guide 1.121, and GL 95-05. ***(7) Corrective Actions:*** Tubes containing flaws 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. 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 SCC degradation in steam generator tubes.	
Inservice inspection in conformance with Plant Technical Specifications and NRC Regulatory Guide 1.83 ("Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes". Plugging or repair of defective tubes is governed by NRC Regulatory Guide 1.121 ("Bases for Plugging Degraded PWR Steam Generator Tubes"). Recommendations on secondary water chemistry given in EPRI TR-102134 ("PWR Secondary Water Chemistry Guidelines—Revision 3"). Specific guidance on the inservice inspection of tubes is also provided in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5" and NEI 97-06. Supplemental inspection requirements and plugging criteria may be contained in plant-specific Technical Specifications.	***(1) Scope of Program:*** The program deals with the effects of IGA of the steam generator tubes on the integrity of the pressure boundary. Plant Technical Specifications and Reg. Guide 1.83 deal with the detection of flaws, and Reg. Guide 1.121 and GL 95-05 describe the basis for determining when flawed tubes must be plugged or repaired. ***(2) Preventive Actions:*** Secondary water chemistry guidelines given in EPRI TR-102134 provide guidance on the prevention of 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 IGA. ***(6) Acceptance Criteria:*** Any degradation detected is compared with the requirements of the Plant Technical Specifications, Reg. Guide 1.121, and GL 95-05. ***(7) Corrective Actions:*** Tubes containing flaws 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	No, provided Plant Technical Specifications conform to EPRI inspection guidelines and NEI 97-06 and 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)	Steam Generator Tubes	Alloy 600	Up to 300°C Secondary -side Water Chemistry at 5.3-7.2 MPa	Crack Initiation and Growth	Fatigue	Plant Technical Specifications, GSI-190
D2.3.1	Piping and Fittings	Main Steam	CS	Up to 300°C Steam	Cumulative Fatigue Damage	Fatigue	ANSI B31.1
D2.3.1	Piping and Fittings	Main Steam	CS	Up to 300°C Steam	Wall Thinning	E/C	*Same as the effect of Erosion/ Corrosion on Item D2.1.5 FW and AFW Nozzles and Safe Ends*
D2.3.2	Piping and Fittings	Feedwater	CS	Up to 225°C Secondary -side Water Chemistry	Cumulative Fatigue Damage	Fatigue	ANSI B31.1
D2.3.2	Piping and Fittings	Feedwater	CS	Up to 225°C Secondary -side Water Chemistry	Wall Thinning	E/C	*Same as the effect of Erosion/ Corrosion on Item D2.1.5 FW and AFW Nozzles and Safe Ends*
D2.3.3	Piping and Fittings	Auxiliary Feedwater	CS	<90°C Secondary -side Water Chemistry	Wall Thinning	E/C	*Same as the effect of Erosion/ Corrosion on Item D2.1.5 FW and AFW Nozzles and Safe Ends*
D2.3.1	Piping and Fittings	Main Steam	CS	Up to 300°C Steam	Loss of Material	Crevice and Pitting Corrosion	ASME Section XI, 1989 Edition; Plant Technical Specifications

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) 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 steam generator tubes.	
Components have been designed or evaluated for fatigue for a 40 y design life, according to Plant Technical Specifications or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Components have been designed for fatigue for a 40 y design life according to the requirements of ANSI B31.1.	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.	Yes TLAA
Same as the effect of Erosion/Corrosion on Item D2.1.5 FW and AFW Nozzles and Safe Ends	*Same as the effect of Erosion/Corrosion on Item D2.1.5 FW and AFW Nozzles and Safe Ends*	No
Components have been designed for fatigue for a 40 y design life according to the requirements of ANSI B31.1.	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.	Yes TLAA
Same as the effect of Erosion/Corrosion on Item D2.1.5 FW and AFW Nozzles and Safe Ends	*Same as the effect of Erosion/Corrosion on Item D2.1.5 FW and AFW Nozzles and Safe Ends*	No
Same as the effect of Erosion/Corrosion on Item D2.1.5 FW and AFW Nozzles and Safe Ends	*Same as the effect of Erosion/Corrosion on Item D2.1.5 FW and AFW Nozzles and Safe Ends*	No
The program includes preventive measures to mitigate crevice corrosion by control of secondary water chemistry and, based on Plant Technical Specifications, inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWC 2500-1, examination category C-H for pressure retaining Class 2 components.	***(1) Scope of Program:*** The program relies on preventive measures to mitigate general, crevice, or pitting corrosion, and inservice inspection (ISI) to monitor the effects of corrosion on the intended function of the auxiliary feedwater system components. ***(2) Preventive Actions:*** Mitigation is by monitoring and control of secondary water chemistry and demineralized water chemistry to minimize concentration of corrosive impurities such as chlorides, sulfates, and oxygen. Chemical species and impurities are monitored either continuous, daily, weekly, or as required based on plant operating conditions. Corrective actions are taken when specified values are exceeded. ***(3) Parameters Monitored/ Inspected:***	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D2.3.2, D2.3.3	Piping and Fittings	Feedwater, Auxiliary Feedwater	CS	Up to 300°C Secondary -side Water Chemistry	Loss of Material	Crevice and Pitting Corrosion	ASME Section XI, 1989 Edition
D2.4.1	Safety and Relief Valves	Body	CS	Up to 300°C Steam or Secondary -side Water Chemistry	Wall Thinning	E/C	*Same as the effect of Erosion/ Corrosion on Item D2.1.5 FW and AFW Nozzles and Safe Ends*

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) The AMP monitors the effects of corrosion by detection of coolant leakage by inservice inspection (ISI). Inspection requirements of ASME Section XI specify visual VT-2 (IWA-5240) examination during system leakage test and hydrostatic test of all pressure retaining Class 2 components up to the first normally closed valve, according to Table IWC 2500-1 category C-H. ***(4) Detection of Aging Effects:*** Degradation of the component due to corrosion can not occur without leakage of coolant; extent and schedule of inspection assure detection of leakage before the loss of intended function of the component. ***(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-3516. ***(7) Corrective Actions:*** Prior to service, corrective measures are needed to meet the requirements of IWC-3132 and IWA-5250. 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:*** No significant corrosion related problem has been reported for piping and fittings in secondary side systems of steam generators.	
Same as the effect of Crevice and Pitting Corrosion on Item D2.3.1 Piping and Fittings in Main Steam System	*Same as the effect of Crevice and Pitting Corrosion on Item D2.3.1 Piping and Fittings in Main Steam System*	No
Same as the effect of Erosion/Corrosion on Item D2.1.5 FW and AFW Nozzles and Safe Ends	*Same as the effect of Erosion/Corrosion on Item D2.1.5 FW and AFW Nozzles and Safe Ends*	No

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NRC Information Notice 85-30, *Microbiologically Induced Corrosion of Containment Service Water System*, April 19, 1985.

NRC Information Notice 86-96, *Heat Exchanger Fouling can Cause Inadequate Operability of Service Water Systems*, November 20, 1986.

NRC Information Notice 86-106, *Feedwater Line Break*, December 16, 1986.

NRC Information Notice 86-106, Supplement 1, *Feedwater Line Break*, February 13, 1987.

NRC Information Notice 86-106, Supplement 2, *Feedwater Line Break*, March 18, 1987.

NRC Information Notice 86-106, Supplement 3, *Feedwater Line Break*, November 10, 1988.

NRC Information Notice 88-70, *Check Valve Inservice Testing Program Deficiencies*, August 29, 1988.

NRC Information Notice 89-53, *Rupture of Extraction Steam Line on High Pressure Turbine*, June 13, 1989.

NRC Information Notice 91-18, *High-Energy Piping Failures Caused by Wall Thinning*, March 12, 1991.

NRC Information Notice 91-18, Supplement 1, *High-Energy Piping Failures Caused by Wall Thinning*, December 18, 1991.

NRC Information Notice 91-19, *Steam Generator Feedwater Distribution Piping Damage*, March 12, 1991.

NRC Information Notice 91-28, *Cracking in Feedwater System Piping*, April 15, 1991.

NRC Information Notice 92-35, *Higher than Predicted Erosion/Corrosion in Unisolable Reactor Coolant Pressure Boundary Piping inside Containment at a Boiling Water Reactor*, May 6, 1992.

NRC Information Notice 93-21, *Summary of NRC Staff Observations Compiled During Engineering Audits or Inspections of Licensee Erosion/Corrosion Programs*, March 25, 1993.

NRC Information Notice 94-03, *Deficiencies Identified During Service Water System Operational Performance Inspections*, January 11, 1994.

NRC Information Notice 95-11, *Failure of Condensate Piping Because of Erosion/Corrosion at a Flow Straightening Device*, February 24, 1995.

NRC Information Notice 97-84, *Rupture in Extraction Steam Piping as a Result of Flow-Accelerated Corrosion*, December 11, 1997.

NUREG-1344, *Erosion/Corrosion-Induced Pipe Wall Thinning in U.S. Nuclear Power Plants*, P. C. Wu, U.S. Nuclear Regulatory Commission, April 1989.

NUREG-1339, *Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants*, Richard E. Johnson, U.S. Nuclear Regulatory Commission, June 1990.

CHAPTER V

(12/06/99)

ENGINEERED SAFETY FEATURES

Major Plant Sections

A. Containment Spray System

B. Standby Gas Treatment System (Boiling Water Reactor)

C. Containment Isolation Components

D1. Emergency Core Cooling System (Pressurized Water Reactor)

D2. Emergency Core Cooling System (Boiling Water Reactor)

E. Fan Cooler System

A Containment Spray System

A.1 Containment Spray System

A.1.1 Piping and Fittings up to Isolation Valve

A.1.2 Flow Orifice/Elements

A.1.3 Temperature Elements/Indicators

A.1.4 Bolting

A.1.5 Eductors

A.2 Header and Spray Nozzles System

A.2.1 Piping and Fittings

A.2.2 Flow Orifice

A.2.3 Headers

A.2.4 Spray Nozzles

A.3 Chemical Addition System

A.3.1 Piping and Fittings

A.3.2 Storage Tank

A.4 Pumps

A.4.1 Bowl/Casing

A.4.2 Bolting

A.5 Valves (Hand, Control, Check, Motor-Operated) (in Containment Spray System)

A.5.1 Body and Bonnet

A.5.2 Bolting

A.6 Valves (Hand, Control) (in Header and Spray Nozzle System)

A.6.1 Body and Bonnet

A.6.2 Bolting

A.7 Containment Spray Heat Exchanger

A.7.1 Bonnet/Cover

A.7.2 Tubing

A.7.3 Shell

A.7.4 Case/Cover

A.7.5 Bolting

A Containment Spray System (Pressurized Water Reactors)

System, Structures, and Components

The system, structures, and components included in this table comprise the containment spray system for pressurized water reactors (PWRs) designed to limit the pressure and temperature of the containment atmosphere following a design basis-event, and to remove heat from the reactor coolant system (RCS) during plant cooldown and to maintain the RCS temperature during plant shutdown. Spray systems using borated water and NaOH solutions are reviewed. The system consists of piping, valves, flow elements and orifices, pumps, spray nozzles, eductors, containment spray system heat exchanger (some plants), and chemical addition system. 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," all components in the containment spray system outside the containment are classified as Group B Quality Standards and those inside the containment and up to the isolation valve are classified Group A Quality Standard. The aging management programs for the residual heat removel (RHR) heat exchanger, refueling water tank, and shutdown coolers are reviewed in Table V D1, and for the containment isolation valves in Table V C.

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 and are not subject to aging management review pursuant to 10 CFR 54.21(a)(1)(i).

System Interfaces

The systems that interface with the containment spray system are the emergency core cooling or safety injection system (Table V D1), coolant storage/refueling water system (Table VII E4), open- and closed-cycle cooling water system (Tables VII C1 and C2), containment isolation components (Table V C), and fan cooler system (Table V E).

V ENGINEERED SAFETY FEATURES

A. CONTAINMENT SPRAY SYSTEM (Pressurized Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A.1.1 thru A.1.3	Containment Spray System	Piping and Fittings up to Isolation Valve, Flow Orifice/ Elements, Temperature Elements/ Indicators	Stainless Steel (SS)	Chemically Treated Borated Water at Maximum Design Temperature of ≈205°C	Local Loss of Material	Pitting and Crevice Corrosion	ASME Section XI, 1989 Edition. NRC IN 84-18. Plant Technical Specifications. EPRI TR-105714.

V ENGINEERED SAFETY FEATURES

A. CONTAINMENT SPRAY SYSTEM (Pressurized Water Reactor)

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Inservice inspection is in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWC 2500-1, examination category C-H for pressure retaining Class 2 components. Water chemistry program for minimizing impurities by monitoring and maintaining water chemistry conditions based on guidelines of EPRI TR-105714 for primary water chemistry and plant technical specifications for refueling water storage tank water chemistry.	***(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 containment spray system components. ***(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 for pitting and crevice corrosion. However, inadvertent introduction of contaminants into the coolant system can occur, e.g., contaminants in the boric acid, or introduced through the free surface of spent fuel pool [NRC Information Notice (IN) 84–18], or from water from the sump. 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 corrosion by control of system water chemistry and by detection of coolant leakage by inservice inspection (ISI). Inspection requirements of ASME Section XI specify visual VT-2 (IWA-5240) examination during system leakage and hydrostatic tests of all pressure retaining Class 2 components required to operate or support the safety function according to Table IWC 2500-1 category C-H. ***(4) Detection of Aging Effects:*** Degradation of the component due to corrosion would result in leakage of coolant. However, a one-time inspection of representative sample of the system population and most susceptible locations in the system should be conducted to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period. Follow up actions are based on the inspection results and plant technical specification. Inspection is performed in accordance with the requirements of ASME Code, 10CFR50 Appendix B, and ASTM standards, using a variety of nondestructive techniques including visual, ultrasonic, and surface techniques. Selection of susceptible locations is based on severity of conditions, time of service, and lowest design margin. ***(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 and hydrostatic tests are evaluated in accordance with IWC-3100 and acceptance standards of IWC-3400 and IWB-3516 for Class 2 components. ***(7) Corrective Actions:*** Prior to service, corrective measures are needed to meet the requirements of IWB-3142 and IWA-5250. Repairs are in conformance with IWA-4000 and replacement according to 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 50 and will continue to be adequate	Yes, Element 4 should be further evaluated

V ENGINEERED SAFETY FEATURES

A. CONTAINMENT SPRAY SYSTEM (Pressurized Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A.1.1	Containment Spray System	Piping and Fittings up to Isolation Valve	Stainless Steel (SS)	Chemically Treated Borated Water at Maximum Design Temperature of ≈205°C	Crack Initiation and Growth	Stress Corrosion Cracking (SCC)	ASME Section XI, 1989 Edition. Reg. Guide 1.44. NRC IN 79-19. NRC IN 80-38. NRC IN 84-18. NRC IN 91-05. NRC IN 94-63. NRC IN 97-19. EPRI TR-105714. Plant Technical Specifications.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) for the period of license renewal. ***(10) Operating Experience:*** Corrosion related degradation has not been reported for containment spray system components, but cracking has occurred in safety-related SS piping systems and portions of systems which contain oxygenated, stagnant, or essentially stagnant borated water.	
Guidelines of Regulatory Guide (RG) 1.44 to avoid sensitization of stainless steels and, based on plant specifications, inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWC 2500-1, examination category C-F-1 for pressure retaining welds in Class 2 stainless steel piping. Water chemistry program for minimizing impurities by monitoring and maintaining water chemistry conditions based on guidelines of EPRI TR-105714 for primary water chemistry and plant technical specifications for refueling water storage tank water chemistry.	***(1) Scope of Program:*** The program includes preventive measures to mitigate stress corrosion cracking (SCC) of stainless steel (SS) and inservice inspection (ISI) to monitor the effects of SCC on the intended function of containment spray system components. ***(2) Preventive Actions:*** Selection of material in compliance with the guidelines of Regulatory Guide (RG) 1.44 prevents or mitigates SCC. 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, e.g., contaminants in the boric acid, or introduced through the free surface of spent fuel pool [NRC Information Notice (IN) 84–18], or from water from the sump. 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 piping by control of system water chemistry and by detection and sizing of cracks by ISI. Inspection requirements of IWC 2500-1 category C-F-1, specify for circumferential and longitudinal welds in each pipe or branch run NPS 4 or larger, volumetric and surface examination of ID region, and surface examination of OD surface. Surface examination is conducted for circumferential and longitudinal welds in each pipe or branch run less than NPS 4. Requirements for training and qualification of personnel and performance demonstration for procedures and equipment are in conformance with Appendices VII and VIII of ASME Section XI, or any other formal program approved by the NRC. ***(4) Detection of Aging Effects:*** Degradation of piping and fittings due to SCC can not occur without crack initiation; inspection schedule assures detection of cracks before the loss of intended function of the piping. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWC-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 IWC-3100 by comparing ISI results with the acceptance standards of IWC-3400 and IWC-3514. Supplementary surface examination may be performed on interior and/or exterior surfaces when flaws are detected in volumetric examination. ***(7) Corrective Actions:*** Repairs are in conformance with IWA-4000, replacement according to IWA-7000, and reexamination in accordance	No

V ENGINEERED SAFETY FEATURES

A. CONTAINMENT SPRAY SYSTEM (Pressurized Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A.1.4	Containment Spray System	Bolting	Carbon Steel (CS), Low-Alloy Steel (LAS)	Air, Leaking Chemically Treated Borated Water	Loss of Material	Corrosion/ Boric Acid Wastage of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 Edition. NRC IN 86-108 S3. Plant Technical Specifications.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) 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:*** Although the primary pressure boundary piping of PWRs has generally not been found to be affected by SCC because of control of primary water chemistry, cracking has occurred in safety injection lines (IN 97-19 and 84-18), charging pump casing cladding (INs 80-38 and 94-63), instrument nozzles in safety injection tanks (IN 91-05), and safety-related SS piping systems which contain oxygenated, stagnant, or essentially stagnant borated water (IN 97-19).	
Implementation of NRC Generic Letter 88-05 and, based on plant specifications, inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWC 2500-1, examination category C-H for pressure retaining Class 2 components.	***(1) Scope of Program:*** The staff guidance of Generic Letter (GL) 88-05 provides assurance that a program has been implemented consisting of systematic measures to ensure that the effects of corrosion caused by leaking coolant containing boric acid does not lead to degradation and provides assurance that the reactor coolant pressure boundary will have a extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture. The program includes (a) determination of principal location, (b) examinations requirements and procedures for locating small leaks, and (c) engineering evaluations and corrective actions. ***(2) Preventive Actions:*** Minimizing reactor coolant leakage by frequent monitoring of the locations where potential leakage could occur and repairing the leaky components as soon as possible, prevent or mitigate boric acid corrosion. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of boric acid corrosion on the intended function of the component by detection of coolant leakage by inservice inspection (ISI). Inspection requirements of ASME Section XI specify visual VT-2 (IWA-5240) examination during system leakage test and system hydrostatic test of all pressure retaining Class 2 components required to operate or support the safety function up to and including the first normally closed valve, according to Table IWC 2500-1 category C-H. ***(4) Detection of Aging Effects:*** Degradation of the component due to boric acid wastage can not occur without leakage of coolant containing boric acid; extent and schedule of inspection assure detection of leakage before the loss of intended function of the component. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI should provide for timely detection of leakage. 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 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 IWB-3516 for Class 2 components. ***(7) Corrective Actions:*** Prior to service, corrective measures are needed to meet the requirements of	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A.1.5	Containment Spray System	Eductors	SS	Chemically Treated Borated Water and NaOH Solution	Local Loss of Material	Pitting and Crevice Corrosion	ASME Section XI, 1989 Edition. NRC IN 84-18. Plant Technical Specifications. EPRI TR-105714.
A.1.5	Containment Spray System	Eductors	SS	Chemically Treated Borated Water and NaOH Solution	Crack Initiation and Growth	Stress Corrosion Cracking (SCC)	*Same as effects of SCC on containment spray system components (A.1.1-A.1.3).*
A.2.1 thru A.2.4	Header and Spray Nozzles System	Piping and Fittings, Flow Orifice, Headers, Spray Nozzles	CS, SS	Air	Loss of Material	General, Pitting, and Crevice Corrosion	Plant Technical Specifications.
A.3.1	Chemical Addition System	Piping and Fittings	SS	NaOH Solution at near Ambient Temperature	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. Regulatory Guide 1.44.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) IWB-3142 and IWA-5250. The leakage source and areas of general corrosion are located. Components with local areas of corrosion that reduces the wall thickness by more than 10% require analytical evaluation to demonstrate acceptability. ***(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:*** Boric acid wastage observed in nuclear power plants may be classified into two distinct types: (a) corrosion that increases the rate of leakage, e.g., corrosion of closure bolting or fasteners in reactor coolant pressure boundary, and (b) corrosion that occurs some distance from the source of leakage. Some recent incidents of boric acid wastage (IN 86-108 S3) at Calvert Cliffs Unit 1 (Feb. 1994) and Three Mile Island Unit 1 (March 1994) indicate that, although implementation of GL 88-05 ensures timely detection of leakage, there may still be a lack of awareness of the conditions that can lead to boric acid wastage.	
Same as effects of Pitting and Crevice Corrosion on containment spray system piping and fittings (A.1.1), flow orifice/elements (A.1.2), temperature elements/indicators (A.1.3).	*Same as effects of Pitting and Crevice Corrosion on containment spray system piping and fittings (A.1.1), flow orifice/elements (A.1.2), temperature elements/indicators (A.1.3).*	Yes, Element 4 should be further evaluated
Same as effects of SCC on containment spray system components (A.1.1-A.1.3).	*Same as effects of SCC on containment spray system components (A.1.1-A.1.3).*	No
Plant specific aging management program.	Plant specific aging management program is to be evaluated.	Yes, no AMP
Guidelines of Regulatory Guide (RG) 1.44 to avoid sensitization of stainless steels (SSs) and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWC, Table IWC 2500-1, examination category C-F-1 for pressure retaining welds in austenitic stainless steel and high alloy piping.	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC of SS components, and inservice inspection (ISI) to monitor the effects of SCC on the intended function of piping and fittings and connected lines. ***(2) Preventive Actions:*** Selection of material in compliance with the guidelines of Regulatory Guide (RG) 1.44 to prevent or mitigate SCC. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of SCC on the intended function of the piping by detection and sizing of cracks by ISI. Inspection requirements of IWC 2500-1 category C-F-1, specify for circumferential and longitudinal welds in each pipe or branch run NPS 4 or larger, volumetric and surface examination of ID region, and surface examination of OD surface. Requirements for training and qualification of personnel and performance demonstration for procedures and equipment are	No

V ENGINEERED SAFETY FEATURES

A. CONTAINMENT SPRAY SYSTEM (Pressurized Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A.3.2	Chemical Addition System	Storage Tank	SS	NaOH Solution at near Ambient Tempera-ture	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. Regulatory Guide 1.44.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) in conformance with Appendices VII and VIII of ASME Section XI, or any other formal program approved by the NRC. ***(4) Detection of Aging Effects:*** Degradation of the piping due to SCC can not occur without crack initiation. The extent of inspection required by ASME Section XI is considered adequate to detect cracking of susceptible SS components and cladding in the weld regions. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWC-2400 should provides 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 and replacement are in conformance with IWA-4000 and IWC-3133, 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:*** No significant cracking has been reported for chemical addition lines in PWRs.	
Guidelines of Regulatory Guide (RG) 1.44 to avoid sensitization of stainless steels and, based on plant specifications, inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWC 2500-1, examination category C-H for pressure retaining Class 2 components.	***(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 the storage tank. ***(2) Preventive Actions:*** Selection of material in compliance with the guidelines of Regulatory Guide (RG) 1.44 prevents or mitigates SCC. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of SCC by detection of leakage. Inspection requirements of ASME Section XI specify visual VT-2 (IWA-5240) examination during system leakage test and system hydrostatic test of all pressure retaining Class 2 components required to operate or support the safety function, according to Table IWC 2500-1 category C-H. ***(4) Detection of Aging Effects:*** Degradation of the component due to SCC can not occur without leakage of coolant. However, a one-time inspection of representative sample of the system population and most susceptible locations in the system should be conducted to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period. Follow up actions are based on the inspection results and plant technical specification. Inspection is performed in accordance with the requirements of ASME Code, 10CFR50 Appendix B, and ASTM standards, using a variety of nondestructive techniques including visual, ultrasonic, and surface techniques. Selection of susceptible locations is based on severity of conditions, time of service, and lowest design margin. ***(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	Yes, Element 4 should be further evaluated

V ENGINEERED SAFETY FEATURES

A. CONTAINMENT SPRAY SYSTEM (Pressurized Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A.4.1	Pump	Bowl/Casing	SS	Chemically Treated Borated Water	Local Loss of Material	Pitting and Crevice Corrosion	ASME Section XI, 1989 Edition. ASME OM Code-1990, Subsection ISTB. NRC GL 89-04. NRC IN 84-18. NRC IN 96-11. EPRI TR-105714. Plant Technical Specifications.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) 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 IWB-3516. Any evidence of aging effects or unacceptable results should be evaluated. ***(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:*** No significant cracking has been reported for chemical addition storage tanks in PWRs.	
Inservice inspection is in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWC 2500-1, examination category C-H for pressure retaining Class 2 components; and based on the testing requirements of 10 CFR 50.55a for ASME Code Class 2 pumps, and additional NRC staff guidelines of NRC Generic Letter 89-04, inservice testing performed in accordance with ASME Subsection IWP (or Operation and Maintenance Code Subsection ISTB) for pumps, or other approved program in the plant specifications. Water chemistry program based on EPRI guidelines of TR-105714 for minimizing impurities by monitoring and maintaining primary water chemistry.	***(1) Scope of Program:*** The program relies on preventive measures to mitigate crevice or pitting corrosion and combination of inservice inspection (ISI) and inservice testing (IST) to monitor the effects of corrosion on the intended function of containment spray system components. ***(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 for pitting and crevice corrosion. However, inadvertent introduction of contaminants into the coolant system can occur, e.g., contaminants in the boric acid, or introduced through the free surface of spent fuel pool [NRC Information Notice (IN) 84–18], or from water from the sump. 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 corrosion by control of water chemistry and by ISI to detect coolant leakage and IST to evaluate component performance. Inspection requirements of ASME Section XI specify visual VT-2 (IWA-5240) examination during system leakage and hydrostatic tests of all pressure retaining Class 2 components required to operate or support the safety function according to Table IWC 2500-1 category C-H. Based on the requirements of 10 CFR 50.55a for ASME Code Class 2 pumps and additional guidelines of NRC Generic Letter (GL) 89-04, IST is performed in accordance with ASME Subsection IWP (or OM Code Subsection ISTB). ***(4) Detection of Aging Effects:*** Degradation of the component due to corrosion would result in leakage of coolant or degradation of pump performance. However, a one-time inspection of representative sample of the system population and most susceptible locations in the system should be conducted to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period. Follow up actions are based on the inspection results and plant technical specification. Inspection is performed in accordance with the requirements of ASME Code, 10CFR50 Appendix B, and ASTM standards, using a variety of	Yes, Element 4 should be further evaluated

V ENGINEERED SAFETY FEATURES

A. CONTAINMENT SPRAY SYSTEM (Pressurized Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A.4.1	Pump	Bowl/Casing	SS	Chemically Treated Borated Water	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. Reg. Guide 1.44. NRC IN 80-38. NRC IN 84-18. NRC IN 91-05. NRC IN 94-63. NRC IN 97-19. EPRI TR-105714 Plant Technical Specifications.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) nondestructive techniques including visual, ultrasonic, and surface techniques. Selection of susceptible locations is based on severity of conditions, time of service, and lowest design margin. ***(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 and hydrostatic tests are evaluated in accordance with IWC-3100 and acceptance standards of IWC-3400 and IWB-3516. ***(7) Corrective Actions:*** Repairs are 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:*** Localized corrosion is likely to occur at flange connections where buildup of impurities can occur.	
Guidelines of Regulatory Guide (RG) 1.44 to avoid sensitization of stainless steels and, based on plant specifications, inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWC 2500-1, examination category C-G for pressure retaining welds in pumps. Water chemistry program based on EPRI guidelines of TR-105714 for minimizing impurities by monitoring and maintaining primary water chemistry.	***(1) Scope of Program:*** The program includes preventive measures to mitigate stress corrosion cracking (SCC) of stainless steel (SS) and inservice inspection (ISI) to monitor the effects of SCC on the intended function of containment spray system components. ***(2) Preventive Actions:*** Selection of material in compliance with the guidelines of Regulatory Guide (RG) 1.44 prevents or mitigates SCC. 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, e.g., contaminants in the boric acid, or introduced through the free surface of spent fuel pool [NRC Information Notice (IN) 84–18], or from water from the sump. 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 intended function of the pump by control of primary water chemistry and by detection and sizing of cracks by ISI. Inspection requirements of IWC 2500-1 category C-G, specifies surface examination of either the inside or outside surface of all welds extending 1/2 in. on either side of the weld. In a group of multiple pumps of similar design, size, function, and service in a system, examination of only one pump is required. ***(4) Detection of Aging Effects:*** Degradation of pumps due to SCC can not occur without crack initiation and growth; ISI schedule assures detection of cracks before the loss of intended function of the pump. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWC-2400 should provide timely detection	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A.4.1	Pump	Bolting	CS, LAS	Air, Leaking Chemically Treated Borated Water	Loss of Material	Corrosion/ Boric Acid Wastage	NRC GL 88-05. ASME Section XI, 1989 Edition. NRC IN 86-108 S3, Plant Technical Specifications.
A.5.1	Valves (Hand, Control, Check, and Motor-Operated Valves) (in Containment Spray System)	Body and Bonnet	SS	Chemically Treated Borated Water	Local Loss of Material	Pitting and Crevice Corrosion	ASME Section XI, 1989 Edition. ASME OM Code-1990, Appendix I and Subsection ISTC. NRC GL 89-04. NRC GL 96-05. NRC IN 84-18. NRC IN 88-70. NRC IN 94-63. NRC IN 98-23. EPRI TR-105714 Plant Technical Specifications.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) of cracks. Surface examination of welds is conducted during each inspection interval. ***(6) Acceptance Criteria:*** Any SCC degradation is evaluated in accordance with IWC-3100 by comparing ISI results with the acceptance standards of IWC-3400 and IWC-3515. ***(7) Corrective Actions:*** Repairs are 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:*** Although the primary pressure boundary piping of PWRs have 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). 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).	
Same as effects of Corrosion/Boric Acid Wastage on containment spray system bolting (A.1.4).	*Same as effects of Corrosion/Boric Acid Wastage on containment spray system bolting (A.1.4).*	No
Inservice inspection is in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWC 2500-1, examination category C-H for pressure retaining Class 2 components; and based on the testing requirements of 10 CFR 50.55a for ASME Code Class 2 valves, staff guidelines of NRC Generic Letter (GL) 89-04 regarding the scope of inservice testing (IST), and information in NRC IN 88-70 regarding scope and testing of safety-related check valves, and GL 96-05 regarding safety-related motor-operated valves. IST is performed in accordance with ASME Subsection IWV (Operation and Maintenance Code Appendix I and Subsection ISTC), to ensure that the changes in design-basis performance of safety-related valves resulting from degradation can be identified and managed. Water chemistry program for minimizing impurities by monitoring and maintaining water chemistry conditions based on guidelines of EPRI TR-105714 for primary water chemistry.	**(1) Scope of Program:** The program relies on preventive measures to mitigate crevice or pitting corrosion and combination of inservice inspection (ISI) and inservice testing (IST) to monitor the effects of corrosion on the intended function of containment spray components. **(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 for pitting and crevice corrosion. However, inadvertent introduction of contaminants into the coolant system can occur, e.g., contaminants in the boric acid, or introduced through the free surface of spent fuel pool [NRC Information Notice (IN) 84–18], or from water from the sump. 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 corrosion by ISI to detect coolant leakage and IST to evaluate component performance. Inspection requirements of ASME Section XI specify visual VT-2 (IWA-5240) examination during system leakage test and hydrostatic test of all pressure retaining Class 2 components required to operate or support the safety function, according to Table IWC 2500-1 category C-H. Based on the requirements of 10 CFR 50.55a for ASME Code Class 2 valves and additional guidelines of NRC GLs	Yes, Element 4 should be further evaluated

V ENGINEERED SAFETY FEATURES

A. CONTAINMENT SPRAY SYSTEM (Pressurized Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A.5.1	Valves (Hand, Control, Check, and Motor-Operated Valves) (in Containment Spray System)	Body and Bonnet	SS	Chemically Treated Borated Water	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. Reg. Guide 1.44. NRC IN 84-18. NRC IN 89-02. NRC IN 97-19. EPRI TR-105714. Plant Technical Specifications.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) 89-04 and 96-05, IST is performed in accordance with ASME Subsection IWV (OM Code Appendix I and Subsection ISTC). ***(4) Detection of Aging Effects:*** Degradation of the component due to corrosion would result in leakage of coolant. However, a one-time inspection of representative sample of the system population and most susceptible locations in the system should be conducted to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period. Follow up actions are based on the inspection results and plant technical specification. Inspection is performed in accordance with the requirements of ASME Code, 10CFR50 Appendix B, and ASTM standards, using a variety of nondestructive techniques including visual, ultrasonic, and surface techniques. Selection of susceptible locations is based on severity of conditions, time of service, and lowest design margin. ***(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 and hydrostatic tests are evaluated in accordance with IWC-3100 and acceptance standards of IWC-3400 and IWB-3516 for Class 2 components. ***(7) Corrective Actions:*** Prior to service, corrective measures are needed to meet the requirements of IWB-3142 and IWA-5250. Repairs are in conformance with IWA-4000 and repair according to 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:*** Corrosion has been observed in guide rings of relief valves (IN 98-23) and charging pump casing (IN 94-63).	
Guidelines of Regulatory Guide (RG) 1.44 to avoid sensitization of stainless steels and, based on plant specifications, inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWC 2500-1, examination category C-G for pressure retaining welds in Class 2 valves. Water chemistry program for minimizing impurities by monitoring and maintaining water chemistry conditions based on guidelines of EPRI TR-105714 for primary water chemistry.	***(1) Scope of Program:*** The program includes preventive measures to mitigate stress corrosion cracking (SCC) of stainless steel (SS) and combination of inservice inspection (ISI) and inservice testing (IST) to monitor the effects of SCC on the intended function of containment spray system components. ***(2) Preventive Actions:*** Selection of material in compliance with the guidelines of Regulatory Guide (RG) 1.44 prevents or mitigates SCC. 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 either e.g., contaminants in the boric acid, or introduced through the free surface of spent fuel pool [NRC Information Notice (IN) 84–18], or from water from the sump. The AMP must therefore rely	No

V ENGINEERED SAFETY FEATURES

A. CONTAINMENT SPRAY SYSTEM (Pressurized Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A.5.2	Valves (Hand, Control, Check, and Motor-Operated Valves) (in Containment Spray System)	Bolting	CS, LAS	Air, Leaking Chemically Treated Borated Water	Loss of Material	Corrosion/ Boric Acid Wastage	NRC GL 88-05. ASME Section XI, 1989 Edition. NRC IN 86-108 S3, Plant Technical Specifications.
A.6.1	Valves (Hand and Control) (in Header and Spray Nozzles System)	Body and Bonnet	CS	Air	Loss of Material	General, Pitting, and Crevice Corrosion	Plant Technical Specifications.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) 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 valves by detection and sizing of cracks by ISI. Inspection requirements of Table IWC 2500-1, category C-G specify for all valves in each piping run examined under category C-F, surface examination of either the inside or outside surface of all welds extending 1/2 in. on either side of the weld. In a group of multiple valves of similar design, size, function, and service in a system, examination of only one valve is required. ***(4) Detection of Aging Effects:*** Degradation of valves due to SCC can not occur without crack initiation and growth; ISI schedule assures detection of cracks before the loss of intended function of the valves. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWC-2400 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. ***(6) Acceptance Criteria:*** Any SCC degradation is evaluated in accordance with IWC-3100 by comparing ISI results with the acceptance standards of IWC-3400 and IWC-3515 for surface examination of welds in Class 2 valves. ***(7) Corrective Actions:*** Repairs are in conformance with IWA-4000 and replacement according to 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:*** 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, cracking has occurred in safety injection lines (IN 97-19 and 84-18), internal bolting in swing check valves (IN 89-02), and safety-related SS piping systems which contain oxygenated, stagnant, or essentially stagnant borated water (IN 97-19).	
Same as effects of Corrosion/Boric Acid Wastage on containment spray system bolting (A.1.4).	*Same as effects of Corrosion/Boric Acid Wastage on containment spray system bolting (A.1.4).*	No
Plant specific aging management program.	Plant specific aging management program is to be evaluated.	Yes, no AMP

V ENGINEERED SAFETY FEATURES

A. CONTAINMENT SPRAY SYSTEM (Pressurized Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A.7.1 thru A.7.4	Containment Spray Heat Exchanger (Serviced by Closed Cycle Cooling Water)	Bonnet/Cover, Tubing, Shell, Case/Cover	CS, SS	Chemically Treated Borated Water; and Closed Cycle Cooling Water (Treated Water)	Loss of Material	Crevice and Pitting Corrosion	ASME Section XI, 1989 Edition. ASME OM S/G, Pt 2. NRC GL 89-13. EPRI TR-105714. Plant Technical Specifications.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Detection of reactor coolant leakage by temperature monitors in component cooling system; inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWC 2500-1, examination category C-H for pressure retaining Class 2 components; performance testing in accordance with ASME OM Standards and Guides, Part 2 to ensure that the heat exchanger serviced by the closed-cycle cooling water system is performing its function acceptably; and water chemistry control program based on EPRI TR-105714 for primary water and plant technical specifications for cooling water.	***Scope of Program:*** The program includes monitoring and control of water chemistry and cooling water chemistry to minimize exposure to aggressive environments, and performance testing in accordance with ASME OM-Standards and Guides, Part 2 provides assurance that the heat exchanger serviced by the closed-cycle cooling water system is performing its function acceptably. If adequacy of cooling water chemistry control can not be confirmed, implement Action III of GL 89-13 to include inspection and maintenance program to ensure that corrosion, erosion, and protective coating failure can not degrade the performance of safety-related systems serviced by closed-cycle cooling water. ***(2) Preventive Actions:*** Use of appropriate materials, lining or coating to protect the underlying metal surfaces, and control of primary water chemistry and cooling water chemistry to minimize exposure to aggressive environment. The parameters monitored include halogens, sulfates, oxygen, and pH in the primary water, and in addition to these, dissolved copper and iron, and suspended solids, in the component cooling water. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of corrosion by surveillance program to detect coolant leakage and inservice testing to evaluate component performance. The parameters monitored are directly related to corrosion, e.g., dissolved iron and copper, and by detection of leakage by temperature monitors, in the component cooling system. Also, inspection requirements of ASME Section XI specify visual VT-2 (IWA-5240) examination during system leakage test and hydrostatic test of all pressure retaining Class 2 components required to operate or support the safety function, according to Table IWC 2500-1 category C-H. Performance testing is conducted in accordance with ASME OM S/G, Part 2. ***(4) Detection of Aging Effects:*** Degradation of component due to corrosion would result in leakage of coolant or degradation of component performance. Monitoring of temperature and suspended solids would detect the existence of corrosion; extent and schedule of inspection/testing assure detection of corrosion before the loss of intended function of the component. ***(5) Monitoring and Trending:*** Results from performance tests to verify the heat transfer capabilities are trended. Also, based on the recommendations of NRC GL 89-13 or its equivalent, if adequacy of cooling water chemistry control can not be confirmed, implement Action III of GL 89-13 to include inspection and maintenance program for closed-cycle cooling water system to ensure that corrosion, erosion, and protective coating failure can not degrade the performance of safety-related systems serviced by closed-cycle cooling water. ***(6) Acceptance Criteria:*** Heat exchanger performance test results are evaluated in accordance with the guidelines of ASME OM S/G Part 2. Results of Section XI leakage tests are evaluated in accordance with IWC-3100 and acceptance standards of IWC-3400 and IWB- 3516 for Class 2 components. ***(7) Corrective Actions:*** If the heat	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A.7.1 thru A.7.4	Containment Spray Heat Exchanger (Serviced by Open Cycle Cooling Water)	Bonnet/Cover, Tubing, Shell, Case/Cover	CS, SS	Chemically Treated Borated Water; and Open Cycle Cooling Water (Raw Water)	Loss of Material	General and Microbio-logically influenced Corrosion	NRC GL 89-13. EPRI TR-105714. Plant Technical Specifications. *Operating Experience* NRC IN 81-21. NRC IN 85-24. NRC IN 85-30. NRC IN 86-96. NRC IN 94-03.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) exchanger fails to perform adequately, corrective actions are taken in accordance with OM S/G Part 2. Root cause evaluation and appropriate corrective action are taken when acceptable limits are exceeded or leakage is detected. ***(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 plant experience with this AMP indicates timely detection of corrosion in the containment spray heat exchangers.	
The program relies on preventive measures to mitigate corrosion by monitoring and control of water chemistry to minimize exposure to aggressive environments, and implementation of the recommendations of Generic Letter 89-13 or an equally effective program to ensure that open-cycle cooling water system is in compliance with General Design Criteria and Quality Assurance requirements. Water chemistry control program based on EPRI TR-105714 for primary water and plant technical specifications for cooling water.	***(1) Scope of Program:*** The program includes monitoring and control of water chemistry to minimize exposure to aggressive environments, and staff recommendations of Generic Letter (GL) 89-13 or an equivalent program provide assurance that open-cycle cooling water system is in compliance with General Design Criteria and Quality Assurance requirements. Guidelines of GL 89-13 include (a) surveillance and control of biofouling, (b) test program to verify heat transfer capabilities, (c) routine inspection and maintenance program to ensure that corrosion, erosion, protective coating failure, silting, and biofouling, can not degrade the performance of safety-related systems serviced by open-cycle cooling water, (d) system walkdown inspection to ensure compliance with licensing basis, and (e) review of maintenance, operating, and training practices and procedures. ***(2) Preventive Actions:*** The component is constructed of appropriate materials, control of secondary side water chemistry, and lining or coating protect the underlying metal surfaces from being exposed to aggressive cooling water environment. Based on GL 89-13 cooling water system is continuously chlorinated or treated with biocide whenever the potential for biological fouling species exists. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of corrosion by surveillance program to detect coolant leakage and inservice testing to evaluate component performance. Based on recommendations of GL 89-13 or its equivalent, cooling water system is inspected for biofouling organisms, sediment, protective coating failure, and corrosion; and cooling water flow and temperature are monitored for component performance evaluation to ensure that flow blockage or excessive fouling accumulation does not exist. ***(4) Detection of Aging Effects:*** Degradation of component due to corrosion would result in leakage of coolant or degradation of component performance; extent and schedule of inspection/testing assure detection of corrosion before the loss of intended function of the component. ***(5) Monitoring and Trending:*** Results from performance tests to verify heat transfer capabilities are trended. ***(6) Acceptance Criteria:*** Any relevant conditions related to corrosion or leakage are compared to established acceptable limits. Maximum levels for various impurities in secondary side water and cooling water	No

V ENGINEERED SAFETY FEATURES

A. CONTAINMENT SPRAY SYSTEM (Pressurized Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A.7.1 thru A.7.4	Containment Spray Heat Exchanger (Serviced by Open Cycle Cooling Water)	Bonnet/Cover, Tubing, Shell, Case/Cover	CS, SS	Chemically Treated Borated Water; and Open Cycle Cooling Water (Raw Water)	Buildup of Deposit	Biofouling	*Same as effect of general and microbiologically influenced corrosion on containment spray heat exchanger components (A.7.1-A.7.4).*
A.7.3 thru A.7.5	Containment Spray Heat Exchanger	Shell, Case/Cover, (External Surfaces); Bolting	CS, LAS	Air, Leaking Chemically Treated Borated Water	Loss of Material	Corrosion/ Boric Acid Wastage of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 Edition. NRC IN 86-108 S3. Plant Technical Specifications.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) are specified. ***(7) Corrective Actions:*** If the heat exchanger fails to perform adequately, corrective actions are taken. Root cause evaluation is performed when acceptable limits are exceeded or leakage is detected. ***(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:*** Significant microbiologically influenced corrosion [NRC Information Notice (IN) 85-30], failure of protective coatings (IN 85-24), and fouling (IN 81-21, 86-96) has been observed in a number of heat exchangers. Although the AMP provides an effective means to manage the effects of corrosion on the intended function of heat exchanger, results of service water system operational performance inspections (IN 94-03) indicate that deficiencies still exist in implementation of GL 89-13.	
Same as effect of general and microbiologically influenced corrosion on containment spray heat exchanger components (A.7.1-A.7.4).	*Same as effect of general and microbiologically influenced corrosion on containment spray heat exchanger components (A.7.1-A.7.4).*	No
Same as effect of Corrosion/Boric Acid Wastage of Item D1.1.7 Bolting for flange connections in Items D1.1.1 thru D1.1.5	*Same as effect of Corrosion/Boric Acid Wastage of Item D1.1.7 Bolting for flange connections in Items D1.1.1 thru D1.1.5*	No

B Standby Gas Treatment System (Boiling Water Reactor)

B.1 Ductwork and Dampers

B.2 Electric Heater

B.3 Filters

B.3.1 Filter Housing and Supports

B.3.2 Charcoal Absorber Filter

B.3.3 Elastomer Seals

B.4 Fan

B. Standby Gas Treatment System (Boiling Water Reactor)

System, Structures, and Components

The system, structures, and components included in this table comprise the standby gas treatment system found in boiling water reactors (BWRs) and consist of ductwork, dampers, electrical heaters, filters, fans, and associated instrumentation and controls. 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," all components in the standby gas treatment system are classified as Group B Quality Standards.

System Interfaces

The system that interfaces with the standby gas treatment system is the containment isolation components system (Table V C).

V ENGINEERED SAFETY FEATURES

B. STANDBY GAS TREATMENT SYSTEMS (Boiling Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B.1	Ductwork and dampers	Internal and external surfaces	Carbon steel	Internal: Occasional exposure to Moist Air; External: Ambient Plant Air Environ-ment	Loss of Material	General Corrosion	NRC Reg. Guide 1.52. NRC IN 82-43.
B.1	Ductwork and dampers	Internal and external surfaces	Carbon steel	Internal: Occasional exposure to Moist Air; External: Ambient Plant Air Environ-ment	Local Loss of Material	Crevice Corrosion	NRC Reg. Guide 1.52. NRC IN 82-43.
B.1	Ductwork and dampers	Internal and external surfaces	Carbon steel	Internal: Occasional exposure to Moist Air; External: Ambient Plant Air Environ-ment	Local Loss of Material	Pitting Corrosion	NRC Reg. Guide 1.52. NRC IN 82-43.
B.2	Electric Heater	Housing	Stainless steel	Occasional exposure to Moist Air and entrained impurities at elevated temperature	Crack Initiation and Growth	Stress Corrosion Cracking	
B.3.1	Filters	Filter housing and supports	Carbon and stainless steel	Internal: Occasional exposure to Moist Air; External: Ambient Plant Air Environ-ment	Loss of Material	General Corrosion	NRC Reg. Guide 1.52. NRC IN 82-43.
B.3.1	Filters	Filter housing and supports	Carbon and stainless steel	Internal: Occasional exposure to Moist Air; External: Ambient Plant Air Environ-ment	Local Loss of Material	Crevice Corrosion	NRC Reg. Guide 1.52. NRC IN 82-43.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
NRC Reg. Guide 1.52 recommends operation of the Standby Gas Treatment System (SGTS) at 70% relative humidity or less to prevent condensation on interior surfaces (see NRC IN 82-43). Otherwise, the methodologies for the prevention and detection of degradation in the SGTS components are covered in plant-specific preventative maintenance and inspection programs.	Plant-specific preventative maintenance and inspection programs will be evaluated.	Yes. No generic AMP
Same as the effect of general corrosion on Item B.1, Ductwork and Dampers.	Plant-specific preventative maintenance and inspection programs will be evaluated.	Yes. No generic AMP
Same as the effect of general corrosion on Item B.1, Ductwork and Dampers.	Plant-specific preventative maintenance and inspection programs will be evaluated.	Yes. No generic AMP
Methodologies for the prevention and detection of degradation in the SGTS components are covered in plant-specific preventative maintenance and inspection programs.	Plant-specific preventative maintenance and inspection programs will be evaluated.	Yes. No generic AMP
Same as the effect of general corrosion on Item B.1, Ductwork and Dampers.	Plant-specific preventative maintenance and inspection programs will be evaluated.	Yes. No generic AMP
Same as the effect of general corrosion on Item B.1, Ductwork and Dampers.	Plant-specific preventative maintenance and inspection programs will be evaluated.	Yes. No generic AMP

V ENGINEERED SAFETY FEATURES

B. STANDBY GAS TREATMENT SYSTEMS (Boiling Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B.3.1	Filters	Filter housing and supports	Carbon and stainless steel	Internal: Occasional exposure to Moist Air; External: Ambient Plant Air Environ-ment	Local Loss of Material	Pitting Corrosion	NRC Reg. Guide 1.52. NRC IN 82-43.
B.3.2	Filters	Charcoal absorber filter	Activated charcoal	Occasional exposure to Moist Air	Loss of Iodine Retention Capacity	Absorption of Moisture	NRC Reg. Guide 1.52. NRC IN 82-43.
B.3.3	Filters	Elastomer Seals	Elastomers (neoprene and similar materials)	Occasional exposure to Moist Air	Changes in Hardness, Compres-sion Strength, and other Physical Properties	Elastomer Degrada-tion	
B.4	Fan	Internal components	Carbon steel	Occasional exposure to Moist Air	Loss of Material	General Corrosion	NRC Reg. Guide 1.52. NRC IN 82-43.
B.4	Fan	Internal components	Carbon steel	Occasional exposure to Moist Air	Local Loss of Material	Crevice Corrosion	NRC Reg. Guide 1.52. NRC IN 82-43.
B.4	Fan	Internal components	Carbon steel	Occasional exposure to Moist Air	Local Loss of Material	Pitting Corrosion	NRC Reg. Guide 1.52. NRC IN 82-43.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Same as the effect of general corrosion on Item B.1, Ductwork and Dampers.	Plant-specific preventative maintenance and inspection programs will be evaluated.	Yes. No generic AMP
Same as the effect of general corrosion on Item B.1, Ductwork and Dampers.	Plant-specific preventative maintenance and inspection programs will be evaluated.	Yes. No generic AMP
Methodologies for the prevention and detection of degradation in the SGTS components are covered in plant-specific preventative maintenance and inspection programs.	Plant-specific preventative maintenance and inspection programs will be evaluated.	Yes. No generic AMP
Same as the effect of general corrosion on Item B.1, Ductwork and Dampers.	Plant-specific preventative maintenance and inspection programs will be evaluated.	Yes. No generic AMP
Same as the effect of general corrosion on Item B.1, Ductwork and Dampers.	Plant-specific preventative maintenance and inspection programs will be evaluated.	Yes. No generic AMP
Same as the effect of general corrosion on Item B.1, Ductwork and Dampers.	Plant-specific preventative maintenance and inspection programs will be evaluated.	Yes. No generic AMP

C. Containment Isolation Components

C.1 Personnel Hatch

- C.1.1 Hatchway
- C.1.2 Inner Door
- C.1.3 Outer Door

C.2 Equipment Hatch

- C.2.1 Hatchway
- C.2.2 Cover Plate

C.3 Mechanical (pipe) Penetrations

- C.3.1 Sleeve
- C.3.2 Seal
- C.3.3 Closure Plate
- C.3.4 Anchors
- C.3.5 Fasteners

C.4 Electrical Penetrations

- C.4.1 Sleeve
- C.4.2 Header Plate
- C.4.3 Seal
- C.4.4 Anchors

C.5 Fuel Transfer Penetrations

- C.5.1 Sleeve
- C.5.2 Closure Plate
- C.5.3 Anchors

C.6 Purge/Vent

C.6.1 Seal

C.7 Leak Testing (Penetration, Integrated, & Isolation Valve Leak Test Systems)

C.7.1 Mechanical Penetrations

C.7.2 Sleeves

C.7.3 Seal

C.8 Isolation Barriers - Valves (BWR, in Lines for Emergency Core Cooling Systems, Feedwater, Main Steam)

C.8.1 Body

C.8.2 Bonnet

C.9 Isolation Barriers - Valves (PWR, in Lines for Emergency Core Cooling Systems, Feedwater, Auxiliary Feedwater, Main Steam, and Blowdown Piping)

C.9.1 Body

C.9.2 Bonnet

C.10 Isolation Barriers - Valves (BWR & PWR, in Lines for Fire Protection, Plant Heating, Waste Gas, Plant Drain, Liquid Waste, & Cooling Water)

C.10.1 Body

C.10.2 Bonnet

C. Containment Isolation Components

System, Structures, and Components

The system, structures, and components included in this table comprise the containment isolation components found in all designs of boiling water reactors (BWR) and pressurized water reactors (PWR) in the U.S. BWR and PWR components are treated separately in the table where differences in the operating environments may influence their degradation behavior. The system consists of personnel and equipment hatches; mechanical, electrical, fuel transfer, and leak-testing penetrations; purges and vents; and BWR and PWR isolation barriers valves. 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," all components in the containment isolation components are classified as Group A or B Quality Standards.

System Interfaces

All of the systems that penetrate the containment pressure boundary interface with the containment isolation components.

V ENGINEERED SAFETY FEATURES

C. CONTAINMENT ISOLATION COMPONENTS

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C.1.1 thru C.1.3	Personnel Hatch	Hatchway, Inner Door, Outer Door	Carbon steel (Coating)	Air; Occasional Leaking Borated Water (PWRs) or Oxygenated Water (BWRs)	Loss of Material	General Corrosion	ASME Code Section XI, 1992 Edition. 10 CFR 50 Appendix J. Regulatory Guide 1.54, ANSI 101.2. ANSI 101.4. *Operating Experience* NRC IN 89-79.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Implementation of periodic inspections of containment structures in accordance with ASME Section XI, Subsection IWE; in conjunction with a coatings program in accordance with ANSI 101.2, ANSI 101.4, and Regulatory Guide 1.54; and the containment leak rate test requirements of Appendix J of 10 CFR Part 50, provide assurance that the containment steel components will continue to perform their intended function during the extended period. Inspection requirements of Subsection IWE include examination categories E-A for containment surfaces, E-C for containment surfaces requiring augmented inspection, E-D for seals, gaskets, and moisture barriers, E-G for pressure retaining bolting, and E-P for all pressure retaining components. The coatings program includes qualification of Service Level I coatings, guidance for coating selection and procurement, specifications of surface preparation and coating application, and inspection requirements and criteria. The leak rate tests verify the leak-tight integrity of the containment pressure boundary. Type A (integrated) tests measure the overall leakage rate of the containment, and Type B (local) tests measure leakage of containment penetrations whose design incorporates resilient seals and gaskets including airlock door seals and equipment hatch gaskets.	***(1) Scope of Program:*** The program relies on periodic inspections to detect degradation of the containment steel components, periodic assessment of existing coatings, and leak-rate testing to manage leak-tight integrity of the containment pressure boundary. ***(2) Preventive Actions:*** An effective implementation of containment coating program assures that the essential leak-tight barriers of the containment steel structures will be protected from the environmental effects. ***(3) Parameters Monitored/ Inspected:*** Periodic inspection in accordance with ASME Section XI, Subsection IWE monitors integrity of the coating, loss of material from corrosion, expansion joint sealant integrity, and seal integrity of the liner behind the attachment welds. Coatings program monitors application of the coatings including surface preparation, dry film thickness, and visual inspection of the coating for damage. Periodic assessment of the coatings include the visual appearance and presence of corrosion products. ***(4) Detection of Aging Effects:*** Effects of aging for coatings include rusted areas, flaking, blistering, crazing, peeling, discoloration, and other signs of damage to accessible metal surface coatings. The visual inspections are supplemented by remote inspections to provide access to otherwise inaccessible areas. Implementation of Subsection IWE, supplemented by the additional requirements of 10 CFR 50.55a, assures detection of significant aging effects by visual inspection. Significant reduction in thickness of the leak-tight barrier requires thickness measurements by volumetric examination. ***(5) Monitoring and Trending:*** A visual examination of the containment pressure retaining boundary is suggested prior to each leakage rate test under 10 CFR 50.55a. The periodic assessments of coatings are conducted during each refueling outage. ***(6) Acceptance Criteria:*** Any significant degradation is reported and evaluated in accordance with ASME Subsection IWE-3500. Acceptance criteria for coatings are stated in ANSI 101.2 and 101.4. ***(7, 8 & 9) Corrective Actions, Confirmation Process and Administrative Controls:*** Site corrective actions program, QA procedures, site review and approval process, and administrative controls are implemented in accordance with Appendix B to 10 CFR Part 50 requirements and will continue to be adequate for license renewal. ***(10) Operating Experience:*** Severe degradation of coatings and the corrosion of steel ice condenser containment vessels have been reported for the McGuire 1 containment. The corrosion was caused by boric acid and collected condensation in the annular space between the steel shell and the surrounding concrete shield, coating damage and base. Metal corrosion was also detected during inspections of the steel shells of the containment at Catawba (NRC IN 89-79).	No

V ENGINEERED SAFETY FEATURES

C. CONTAINMENT ISOLATION COMPONENTS

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C.1.1 thru C.1.3	Personnel Hatch	Hatchway, Inner Door, Outer Door	Carbon steel (Coating)	Air; Occasional Leaking Borated Water (PWRs) or Oxygenated Water (BWRs)	Attrition, Loss of Material	Wear	ASME Code Section XI, 1992 Edition. 10 CFR 50 Appendix J. Regulatory Guide 1.54, ANSI 101.2. ANSI 101.4.
C.2.1, C.2.2	Equipment Hatch	Hatchway, Cover Plate	Carbon steel (Coating)	Air; Occasional Leaking Borated Water (PWRs) or Oxygenated Water (BWRs)	Loss of Material	General Corrosion	ASME Code Section XI, 1992 Edition. 10 CFR 50 Appendix J. Regulatory Guide 1.54, ANSI 101.2. ANSI 101.4.
C.2.1, C.2.2	Equipment Hatch	Hatchway, Cover Plate	Carbon steel (Coating)	Air; Occasional Leaking Borated Water (PWRs) or Oxygenated Water (BWRs)	Attrition, Loss of Material	Wear	ASME Code Section XI, 1992 Edition. 10 CFR 50 Appendix J. Regulatory Guide 1.54, ANSI 101.2. ANSI 101.4.
C.3.1, C.3.3, C.3.4, C.3.5	Mechanical (pipe) Penetrations	Sleeve, Closure Plate, Anchors, Fasteners,	Carbon Steel (typically A516, Gr 70)	Air; Occasional Leaking Borated Water (PWRs) or Oxygenated Water (BWRs)	Loss of Material	General Corrosion	ASME Code Section XI, 1992 Edition. 10 CFR 50 Appendix J. Regulatory Guide 1.54, ANSI 101.2. ANSI 101.4.
C.3.2	Mechanical (pipe) Penetrations	Seal	Elastomers	Internal: Occasional Moist Air; External: Ambient Air	Changes in Hardness, Compres-sion Strength, and other Physical Properties	Elastomer Degrada-tion	ASME Code Section XI, 1992 Edition. 10 CFR 50, Appendices B & J.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Same as for the effect of General Corrosion on personnel hatch hatchway (C.1.1), inner door (C.1.2), and outer door (C.1.3).	*Same as for the effect of General Corrosion on personnel hatch hatchway (C.1.1), inner door (C.1.2), and outer door (C.1.3).*	No
Same as for the effect of General Corrosion on personnel hatch hatchway (C.1.1), inner door (C.1.2), and outer door (C.1.3).	*Same as for the effect of General Corrosion on personnel hatch hatchway (C.1.1), inner door (C.1.2), and outer door (C.1.3).*	No
Same as for the effect of General Corrosion on personnel hatch hatchway (C.1.1), inner door (C.1.2), and outer door (C.1.3).	*Same as for the effect of General Corrosion on personnel hatch hatchway (C.1.1), inner door (C.1.2), and outer door (C.1.3).*	No
Same as for the effect of General Corrosion on personnel hatch hatchway (C.1.1), inner door (C.1.2), and outer door (C.1.3).	*Same as for the effect of General Corrosion on personnel hatch hatchway (C.1.1), inner door (C.1.2), and outer door (C.1.3).*	No
Elastomeric components have been designed and evaluated for specific lifetimes within the initial 40-year licensing period under 10 CFR 50, Appendix B. A similar evaluation is needed for the additional license renewal time period. During the initial 40-year licensing period, the aging management program consists of periodic visual inspections and pressure leakage rate tests in accordance with ASME Section XI, Subsection IWE. A visual examination of the containment vessel pressure retaining boundary is required prior to each 10 CFR 50, Appendix J, Type A leakage rate test.	***(1) Scope of Program:*** he program relies on periodic inspections to detect degradation of the containment isolation components and leak-rate testing to manage leak-tight integrity of the containment pressure boundary. ***(2) Preventive Actions:*** The program does not address prevention of component degradation, but instead focuses on its timely detection. Preventative actions are provided by routine plant-specific maintenance procedures. ***(3) Parameters Monitored/ Inspected:*** Periodic inspection in accordance with ASME Section XI, Subsection IWE monitors seal integrity of isolation components and leak rate tests verify the leak-tight integrity of the containment pressure boundary. ***(4) Detection of Aging Effects:*** Pressure and leakage rate tests detect the presence of leaks through the containment boundary components, including the elastomer seals. ***(5) Monitoring and Trending:*** A visual examination of the containment vessel pressure retaining boundary is required prior to each leakage rate test. ***(6) Acceptance Criteria:*** Any significant degradation is reported and required further evaluation in accordance	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C.4.1, C.4.2, C.4.4	Electrical Penetrations	Sleeve Header Plate Anchors	Carbon Steel (typically A516, Gr 70)	Air; Occasional Leaking Borated Water (PWRs) or Oxygenated Water (BWRs)	Loss of Material	General Corrosion	ASME Code Section XI, 1992 Edition. 10 CFR 50 Appendix J. Regulatory Guide 1.54, ANSI 101.2. ANSI 101.4.
C.4.3	Electrical Penetrations	Seal	Elastomers	Air; Occasional Leaking Borated Water (PWRs) or Oxygenated Water (BWRs)	Changes in Hardness, Compres-sion Strength, & Physical Properties	Elastomer Degrada-tion	ASME Code Section XI, 1992 Edition. 10 CFR 50, Appendices B and J.
C.5.1 thru C.5.3	Fuel Transfer Penetrations	Sleeve, Closure Plate, Anchors	Carbon Steel (typically A516, Gr 70)	Air; Occasional Leaking Borated Water	Loss of Material	General Corrosion	ASME Code Section XI, 1992 Edition. 10 CFR 50 Appendix J. Regulatory Guide 1.54, ANSI 101.2. ANSI 101.4.
C.6.1	Purge/Vent	Seal	Elastomers: Nitrile, Ethylene Propylene	Air; Occasional Leaking Borated Water (PWRs) or Oxygenated Water (BWRs)	Changes in Hardness, Compres-sion Strength, & Physical Properties	Elastomer Degrada-tion	ASME Code Section XI, 1992 Edition. 10 CFR 50, Appendices B and J.
C.7.1 C.7.2	Leak Testing Penetration	Mechanical Penetrations, Sleeves	Carbon Steel (typically A516, Gr 70)	Air; Occasional Leaking Borated Water (PWRs) or Oxygenated Water (BWRs)	Loss of Material	General Corrosion	ASME Code Section XI, 1992 Edition. 10 CFR 50 Appendix J. Regulatory Guide 1.54, ANSI 101.2. ANSI 101.4.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) with ASME Subsection IWE-3500. ***(7, 8 & 9) Corrective Actions, Confirmation Process and Administrative Controls:*** Site corrective actions program, QA procedures, site review and approval process, and administrative controls are implemented in accordance with Appendix B to 10 CFR Part 50 requirements and will continue to be adequate for license renewal. ***(10) Operating Experience:*** No significant failure problems reported.	
Same as for the effect of General Corrosion on personnel hatch hatchway (C.1.1), inner door (C.1.2), and outer door (C.1.3).	*Same as for the effect of General Corrosion on personnel hatch hatchway (C.1.1), inner door (C.1.2), and outer door (C.1.3).*	No
Same as for the effect of elastomer degradation on Item C.3.2 mechanical (pipe) penetration seals. In addition, elastomeric materials for electrical components are required to meet the requirements of 10 CFR 50.49.	*Same as for the effect of elastomer degradation on Item C.3.2 mechanical (pipe) penetration seals.*	No
Same as for the effect of General Corrosion on personnel hatch hatchway (C.1.1), inner door (C.1.2), and outer door (C.1.3).	*Same as for the effect of General Corrosion on personnel hatch hatchway (C.1.1), inner door (C.1.2), and outer door (C.1.3).*	No
Same as for the effect of elastomer degradation on Item C.3.2 mechanical (pipe) penetration seals.	*Same as for the effect of elastomer degradation on Item C.3.2 mechanical (pipe) penetration seals.*	No
Same as for the effect of General Corrosion on personnel hatch hatchway (C.1.1), inner door (C.1.2), and outer door (C.1.3).	*Same as for the effect of General Corrosion on personnel hatch hatchway (C.1.1), inner door (C.1.2), and outer door (C.1.3).*	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C.7.3	Leak Testing Penetration	Seal	Elastomers: Nitrile, Ethylene Propylene	Air; Occasional Leaking Borated Water (PWRs) or Oxygenated Water (BWRs)	Changes in Hardness, Compres-sion Strength, & Physical Properties	Elastomer Degrada-tion	ASME Code Section XI, 1992 Edition. 10 CFR 50, Appendices B and J.
C.8.1, C.8.2	BWR Isolation Barriers— Valves (in Lines for Emergency Core Cooling Systems, Feedwater, Main Steam).	Body, Bonnet	Carbon Steel, Low-Alloy Steel	Inside Surface: Chemically Treated Oxygenated Water or Steam up to 288°C; Outside Surface: Air	Loss of Material	General, Crevice, and Pitting Corrosion	ASME Section XI, 1989 Edition. ASME OM Code-1990, Appendix I and Subsection ISTC. 10 CFR 50 Appendix J. EPRI TR-103515. NRC GL 89-04. NRC GL 90-06. NRC GL 96-05. NRC IN 88-70. Plant Technical Specifications.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Same as for the effect of elastomer degradation on Item C.3.2 mechanical (pipe) penetration seals.	*Same as for the effect of elastomer degradation on Item C.3.2 mechanical (pipe) penetration seals.*	No
Water chemistry program based on EPRI guidelines of TR-103515 and implemented through the plant technical specifications for minimizing impurities by monitoring and maintaining water chemistry conditions; inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWB 2500-1, testing category B-P for pressure retaining Class 1 components; and Table IWC 2500-1, examination category C-H for pressure retaining Class 2 components; and based on the testing requirements of 10 CFR 50.55a for ASME Code Class 1 and 2 valves, additional staff guidelines of NRC Generic Letter (GL) 89-04 regarding the scope of inservice testing (IST), and information in NRC IN 88-70 regarding scope and testing of safety-related check valves, and in GL 96-05 regarding safety-related motor-operated valves, IST is performed in accordance with ASME Subsection IWV (Operation and Maintenance Code Appendix I and Subsection ISTC), to ensure that the changes in design-basis performance of safety-related valves resulting from degradation can be identified and managed. Also, the integrity of the containment isolations valves is verified in Type C leak rate tests in accordance with Appendix J of 10 CFR 50.	***(1) Scope of Program:*** The program relies on preventive measures to mitigate crevice or pitting corrosion and combination of inservice inspection (ISI) and inservice testing (IST) to monitor the effects of corrosion on the intended function of emergency core cooling system components. ***(2) Preventive Actions:*** Mitigation is by monitoring and control of water chemistry to minimize concentration of corrosive impurities by following EPRI guidelines of TR-103515 and implemented through the plant technical specifications. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of corrosion by ISI to detect coolant leakage and IST to evaluate component performance. Inspection requirements of ASME Section XI specify visual VT-2 (IWA-5240) examination during system leakage test and hydrostatic test of all pressure retaining Class 1 components according to Table IWB 2500-1 category B-P; and pressure retaining Class 2 components required to operate or support the safety function, according to Table IWC 2500-1 category C-H. Based on the requirements of 10 CFR 50.55a for ASME Code Class 1 and 2 valves and additional guidelines of NRC GLs 89-04 and 96-05, IST is performed in accordance with ASME Subsection IWV (OM Code Appendix I and Subsection ISTC). ***(4) Detection of Aging Effects:*** Degradation of the component due to corrosion would result in leakage of coolant or degradation of component performance. However, a one-time inspection of representative sample of the system population and most susceptible locations in the system should be conducted to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period. Inspection is performed in accordance with the requirements of ASME Code, 10CFR50 Appendix B, and ASTM standards, using a variety of nondestructive techniques including visual, ultrasonic, and surface techniques. ***(5) Monitoring and Trending:*** ISI/IST schedule of ASME Section XI should provide for timely detection of corrosion. 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 and hydrostatic tests are evaluated in accordance with IWC-3100 and acceptance standards of IWC-3400 and IWC-3516 for Class 2 components. Any evidence of aging effects or unacceptable results are evaluated. ***(7) Corrective Actions:*** Prior to service,	Yes, Element 4 should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C.8.1, C.8.2	BWR Isolation Barriers—Valves (in Lines for Emergency Core Cooling Systems, Feedwater, Main Steam).	Body, Bonnet	Carbon Steel, Low-Alloy Steel	Inside Surface: Chemically Treated Oxygenated Water or Steam up to 288°C; Outside Surface: Air	Crack Initiation and Growth	SCC	NUREG-0313, Rev. 2. NRC GL 88-01. ASME Section XI, 1989 Edition. 10 CFR 50 Appendix J. BWRVIP-29. EPRI TR-103515.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) corrective measures are needed to meet the requirements of IWB-3142 and IWA-5250. Repair is in conformance with IWA-4000 and replacement according to 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:*** Localized corrosion is likely to occur at crevice geometry where buildup of impurities can occur.	
Program delineated in NUREG-0313, Rev. 2 and implemented through NRC Generic letter (GL) 88-01, and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWB 2500-1, examination categories B-M-1 and B-M-2 for Class 1 valves; Table IWC 2500-1, examination category C-G for pressure retaining welds in Class 2 valves. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-103515 and BWRVIP-29 to minimize the potential of crack initiation and growth. Also, the integrity of the containment isolations valves is verified in Type C leak rate tests in accordance with Appendix J of 10 CFR 50.	***(1) Scope of Program:*** The program includes implementing counter measures to mitigate stress corrosion cracking (SCC) of stainless steel (SS) and combination of inservice inspection (ISI) to monitor SCC and its effects on the intended function of valves. NUREG-0313 and GL 88-01, respectively, describe the technical basis and staff guidance regarding mitigating IGSCC in BWRs. ***(2) Preventive Actions:*** 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. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors SCC of valves by detection and sizing of cracks by implementing the inspection schedule, methods, personnel, sample expansion, and leak detection requirements of GL 88-01. In a group of multiple valves of similar design, size, function, and service in a system, examination of only one valve is required. Coolant water chemistry is monitored and maintained in accordance with the EPRI guidelines in BWRVIP-29 to minimize the potential of crack initiation and growth. ***(4) Detection of Aging Effects:*** Degradation of valves due to SCC can not occur without crack initiation and growth; ISI schedule delineated in the AMP is adequate and will assure detection of cracks or degradation of valve performance before the loss of intended function of 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 performed only when the valve is disassembled for maintenance, repair, or volumetric examination, but at least once during the period. ***(6) Acceptance Criteria:*** Any SCC degradation is evaluated in accordance with IWC-3100 by comparing ISI results with the acceptance standards of IWC-3400 and IWC-3515. ***(7) Corrective Actions:*** Repair is in conformance with IWA-4000 and replacement is in 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:*** The	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C.9.1, C.9.2	PWR Isolation Barriers—Valves (in Lines for Emergency Core Cooling Systems, Feedwater, Auxiliary Feedwater, Main Steam, and Blowdown Piping),	Body, Bonnet	Carbon Steel, Low-Alloy, Steel	Inside Surface: Chemically Treated Borated Water or Steam up to 340°C; Outside Surface: Air	Loss of Material	Crevice and Pitting Corrosion	ASME Section XI, 1989 Edition. ASME OM Code-1990, Appendix I and Subsection ISTC. 10 CFR 50 Appendix J. NRC GL 89-04. NRC GL 96-05. NRC IN 88-70. EPRI TR-105714. Plant Technical Specifications. *Operating Experience* NRC IN 84-18. NRC IN 98-23. NRC IN 94-63.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) comprehensive AMP outlined in GL 88-01 addresses improvements in all three elements that cause SCC, e.g., a susceptible material, significant tensile stress, and an aggressive environment, and has provided effective means of ensuring structural integrity of BWR components.	
Inservice inspection is in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWB 2500-1, testing category B-P for pressure retaining Class 1 components; and Table IWC 2500-1, examination category C-H for pressure retaining Class 2 components; and based on the testing requirements of 10 CFR 50.55a for ASME Code Class 1 and 2 valves, additional staff guidelines of NRC Generic Letter (GL) 89-04 regarding the scope of inservice testing (IST), and information in NRC IN 88-70 regarding scope and testing of safety-related check valves, and in GL 96-05 regarding safety-related motor-operated valves. IST is performed in accordance with ASME Subsection IWV (Operation and Maintenance Code Appendix I and Subsection ISTC), to ensure that the changes in design-basis performance of safety-related valves resulting from degradation can be identified and managed. Water chemistry program for minimizing impurities by monitoring and maintaining water chemistry conditions based on guidelines of EPRI TR-105714 for primary water chemistry. Also, the integrity of the containment isolations valves is verified in Type C leak rate tests in accordance with Appendix J of 10 CFR 50.	***(1) Scope of Program:*** The program relies on preventive measures to mitigate crevice or pitting corrosion and combination of inservice inspection (ISI) and inservice testing (IST) to monitor the effects of corrosion on the intended function of emergency core cooling system components. ***(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 corrosion. However, inadvertent introduction of contaminants into the coolant system can occur, e.g., contaminants in the boric acid, or introduced through the free surface of spent fuel pool [NRC Information Notice (IN) 84–18], or from water from the sump. The aging management program must therefore rely upon inservice inspections in accordance with ASME Section XI to detect possible degradation. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of corrosion by ISI to detect coolant leakage and IST to evaluate component performance. Inspection requirements of ASME Section XI specify visual VT-2 (IWA-5240) examination during system leakage test and hydrostatic test of all pressure retaining Class 1 components according to Table IWB 2500-1 category B-P; and pressure retaining Class 2 components required to operate or support the safety function, according to Table IWC 2500-1 category C-H. Based on the requirements of 10 CFR 50.55a for ASME Code Class 1 and 2 valves and additional guidelines of NRC GLs 89-04 and 96-05, IST is performed in accordance with ASME Subsection IWV (OM Code Appendix I and Subsection ISTC). ***(4) Detection of Aging Effects:*** Degradation of the component due to corrosion would result in leakage of coolant. However, a one-time inspection of representative sample of the system population and most susceptible locations in the system should be conducted to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period. Follow up actions are based on the inspection results and plant technical specification. Inspection is performed in accordance with the requirements of ASME Code, 10CFR50 Appendix B, and ASTM standards, using a variety of nondestructive techniques including visual, ultrasonic, and surface techniques. Selection of susceptible locations is based on severity of conditions, time of service, and lowest design margin. ***(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	Yes, Element 4 should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C.9.1, C.9.2	PWR Isolation Barriers—Valves (in Lines for Emergency Core Cooling Systems, Feedwater, Auxiliary Feedwater, Main Steam, and Blowdown Piping),	Body, Bonnet	SS Forging SS Casting	Inside Surface: Chemically Treated Borated Water or Steam up to 340°C; Outside Surface: Air	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. NRC Reg. Guide 1.44. 10 CFR 50 Appendix J. EPRI TR-105714 Plant Technical Specifications. *Operating Experience* NRC IN 80-38. NRC IN 84-18. NRC IN 91-05. NRC IN 94-63. NRC IN 97-19,

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) frequency of future inspections. ***(6) Acceptance Criteria:*** Any relevant conditions that may be detected during the leakage and hydrostatic tests are evaluated in accordance with IWB-3100 and acceptance standards of IWB-3400 and IWB-3522 for Class 1 components, IWC-3100 and acceptance standards of IWC-3400 and IWB-3516 for Class 2 components, IWD-3000 for Class 3 components. ***(7) Corrective Actions:*** Prior to service, corrective measures are needed to meet the requirements of IWB-3142 and IWA-5250. 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:*** Corrosion has been observed in guide rings of relief valves (IN 98-23) and charging pump casing (IN 94-63).	
Guidelines of Regulatory Guide (RG) 1.44 to avoid sensitization of stainless steels and, based on plant specifications, inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWB 2500-1, examination categories B-M-1 and B-M-2 for Class 1 valves; Table IWC 2500-1, examination category C-G for pressure retaining welds in Class 2 valves. Water chemistry program for minimizing impurities by monitoring and maintaining water chemistry conditions based on guidelines of EPRI TR-105714 for primary water chemistry. Also, the integrity of the containment isolations valves is verified in Type C leak rate tests in accordance with Appendix J of 10 CFR 50.	***(1) Scope of Program:*** The program includes preventive measures to mitigate stress corrosion cracking (SCC) of stainless steel (SS) and inservice inspection (ISI) to monitor the effects of SCC on the intended function of emergency core cooling system components. ***(2) Preventive Actions:*** Selection of material in compliance with the requirements of Regulatory Guide (RG) 1.44 prevents or mitigates SCC. 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, e.g., contaminants in the boric acid, or introduced through the free surface of spent fuel pool [NRC Information Notice (IN) 84–18], or from water from the sump. 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 intended function of the valves by detection and sizing of cracks by ISI. Inspection requirements of Table IWB 2500-1 for Class 1 valves, examination category B-M-1 specify 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 specifies visual VT–3 examination of internal surfaces of the valve. Table IWC 2500-1 for Class 2 valves, category C-G specifies for all valves in each piping run examined under category C-F, surface examination of either the inside or outside surface of all welds extending 1/2 in. on either side of the weld. In a group of multiple valves of similar design, size, function, and service in a system, examination of only one valve is required. ***(4) Detection of Aging Effects:*** Degradation of valves due to SCC can not occur without crack initiation and growth; ISI schedule assures detection of cracks or	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C.9.1, C.9.2	PWR Isolation Barriers—Valves (in Lines for Emergency Core Cooling Systems, Feedwater, Auxiliary Feedwater, Main Steam, and Blowdown Piping),	Body, Bonnet (External Surfaces)	Carbon Steel, Low-Alloy, Steel	Air, Leaking Chemically Treated Borated Water	Loss of Material	Corrosion/ Boric Acid Wastage of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 Edition. Plant Technical Specifications. *Operating Experience* NRC IN 86-108 S3,

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) initiation and growth; ISI schedule assures detection of cracks or degradation of valve performance before the loss of intended function of the valves. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 or IWC-2400 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. ***(6) Acceptance Criteria:*** Any SCC degradation is evaluated in accordance with IWC-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3518 for volumetric and surface examination of welds or IWB-3519 for visual examination of valve internal surfaces and IWC-3400 and IWC-3515 for surface examination of welds in Class 2 valves. ***(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:*** 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, significant potential of SCC exists from inadvertent introduction of contaminants into the primary coolant system (IN 84–18). 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, based on plant specifications, inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWB 2500-1, testing category B-P for pressure retaining Class 1 components, e.g., CFS and other components within the containment; Table IWC 2500-1, examination category C-H for pressure retaining Class 2 components, e.g., most components in the safety injection system; and Table IWD 2500-1, test and examination category D-B for systems in support of emergency core cooling.	***(1) Scope of Program:*** The staff guidance of Generic Letter (GL) 88-05 provides assurances that a program has been implemented consisting of systematic measures to ensure that the effects of corrosion caused by leaking coolant containing boric acid does not lead to degradation and provides assurance that the reactor coolant pressure boundary will have a extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture. The program includes (a) determination of principal location, (b) examinations requirements and procedures for locating small leaks, and (c) engineering evaluations and corrective actions. ***(2) Preventive Actions:*** Minimizing reactor coolant leakage by frequent monitoring of the locations where potential leakage could occur and repairing the leaky components as soon as possible, prevent or mitigate boric acid corrosion. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of boric acid corrosion on the intended function of the component by detection of coolant leakage by implementing the program delineated in GL 88-05 including guidelines for locating small leaks, conducting examinations, and performing engineering evaluations. Inspection requirements of ASME Section XI specify	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C.10.1, C.10.2	BWR and PWR Isolation Barriers—Valves (in Lines for Fire Protection, Plant Heating, Waste Gas, Plant Drain, Liquid Waste, & Cooling Water).	Body, Bonnet	Carbon Steel, Low-Alloy Steel, and Stainless Steel	Inside Surface: Treated or Raw Water, Gaseous or Liquid Waste, Reactor Coolant; Outside Surface: Ambient Air	Loss of Material	General, Pitting, and Crevice Corrosion	Plant Technical Specifications.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) visual VT-2 (IWA-5240) examination during system leakage test and system hydrostatic test of all pressure retaining Class 1 components extending to and including the second closed valve at the boundary extremity, according to Table IWB 2500-1 category B-P; pressure retaining Class 2 components required to operate or support the safety function up to and including the first normally closed valve, according to Table IWC 2500-1 category C-H; and Class 3 components in support of emergency core cooling, containment heat removal, and reactor residual heat removal, according to Table IWD 2500-1 category D-B. ***(4) Detection of Aging Effects:*** Degradation of the component due to boric acid wastage can not occur without leakage of coolant containing boric acid; extent and schedule of the program delineated in GL 88-05 will assure detection of leakage before the loss of intended function of the component. ***(5) Monitoring and Trending:*** The program delineated in GL 88-05 provides for timely detection of leakage. 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 coolant leakage is evaluated in accordance with the program proposed by GL 88-05. ***(7) Corrective Actions:*** The leakage source and areas of general corrosion are located and corrective actions are in conformance with the program proposed by GL 88-05, and the requirements of ASME Section XI, IWB-3142 and IWA-5250. Components with local areas of corrosion that reduces the wall thickness by more than 10% require analytical evaluation to demonstrate acceptability. ***(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:*** Boric acid wastage observed in nuclear power plants may be classified into two distinct types: (a) corrosion that increases the rate of leakage, e.g., corrosion of closure bolting or fasteners in reactor coolant pressure boundary, and (b) corrosion that occurs some distance from the source of leakage. Some recent incidents of boric acid wastage (IN 86-108 S3) indicate that, although implementation of GL 88-05 ensures timely detection of leakage, there may still be a lack of awareness of the conditions that can lead to boric acid wastage.	
Plant-specific aging management program.	Plant-specific aging management program is to be evaluated.	Yes, no AMP

D1 Emergency Core Cooling System (Pressurized Water Reactor)

D1.1 Piping & Fittings

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

D1.2 HPSI & LPSI Pumps

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

D1.3 RWT Circulation Pump

- D1.3.1 Bowl/Casing
- D1.3.2 Bolting

D1.4 Valves

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

D1.5 Heat Exchangers (RCP, HPSI, & LPSI Pump Seals; & RHR)

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

D1.6 Heat Exchangers (RWT Heating)

D1.6.1 Bonnet/Cover

D1.6.2 Tubing

D1.6.3 Shell

D1.6.4 Bolting

D1.7 Safety Injection Tank (Accumulator)

D1.7.1 Shell

D1.7.2 Manway

D1.7.3 Penetrations/Nozzles

D1.8 Refueling Water Tank (RWT)

D1.8.1 Shell

D1.8.2 Manhole

D1.8.3 Penetrations/Nozzles

D1.8.4 Bolting

D1.8.5 Perimeter Seal

D1. Emergency Core Cooling System (Pressurized Water Reactors)

System, Structures, and Components

The system, structures, and components included in this table comprise the emergency core cooling systems for pressurized water reactors (PWRs) designed to cool the reactor core and provide safe shutdown following a design basis accident, and consist of the core flood system (CFS), residual heat removal (RHR) or shutdown cooling (SDC), high–pressure safety injection (HPSI) system, low–pressure safety injection (LPSI) system, lines to chemical and volume control system (CVCS), spent fuel pool (SFP) cooling, and emergency sump, and HPSI and LPSI pumps, pump seal coolers, RHR heat exchanger, and refueling water tank (RWT). 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," all components in the emergency core cooling system are classified as Group B Quality Standards. Portion of the RHR, HPSI, LPSI systems and CVCS extending from the reactor coolant system up to the second isolation valve outside the containment are classified as Group A and covered in Table IV C2. The aging management program for the variety of lines that penetrate the containment and associated isolation valves is reviewed in Table V C.

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 and are not subject to aging management review pursuant to 10 CFR 54.21(a)(1)(i).

System Interfaces

The systems that interface with the emergency core cooling system include the reactor coolant system (Table IV C2), containment spray system (Table V A), spent fuel pool cooling and cleanup (Table VII A3), closed cycle cooling water (Table VII C2), chemical and volume control system (Table VII E1), and coolant storage/refueling water system (Table VII E4).

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.1.1 thru D1.1.6	Piping & Fittings	Core Flood System (CFS), Residual Heat Removal (RHR), High-Pressure Safety Injection (HPSI), Low-Pressure Safety Injection (LPSI), Connecting lines to Chemical & Volume Control System (CVCS) and Spent Fuel Pool (SFP) Cooling, Lines to Emergency Sump	Stainless Steel (SS)	25–340°C, chemically treated borated water	Crack Initiation and Growth	Stress Corrosion Cracking (SCC)	ASME Section XI, 1989 Edition. Regulatory Guide 1.44. EPRI TR-105714. Plant Technical Specifications. *Operating Experience* NRC IN 80-38. NRC IN 84-18. NRC IN 91-05. NRC IN 94-63. NRC IN 97-19.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Guidelines of Regulatory Guide (RG) 1.44 to avoid sensitization of stainless steels and, based on plant specifications, inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWB 2500-1, examination category B-J for pressure retaining welds in Class 1 piping, e.g., CFS and other components within the containment; Table IWC 2500-1, examination category C-F-1 for pressure retaining welds in Class 2 SS piping, e.g., most of the safety injection piping; and Table IWD 2500-1, test and examination category D-B for systems in support of emergency core cooling, e.g., refueling water tank (RWT) heating system. Water chemistry program for minimizing impurities by monitoring and maintaining water chemistry conditions based on guidelines of EPRI TR-105714 for primary water chemistry and plant technical specifications for refueling water storage tank water chemistry.	***(1) Scope of Program:*** The program includes preventive measures to mitigate stress corrosion cracking (SCC) of stainless steel (SS) and inservice inspection (ISI) to monitor the effects of SCC on the intended function of emergency core cooling system piping and fittings. ***(2) Preventive Actions:*** Selection of material in compliance with the guidelines of Regulatory Guide (RG) 1.44 prevents or mitigates SCC. 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, e.g., contaminants in the boric acid, or introduced through the free surface of spent fuel pool [NRC Information Notice (IN) 84–18], or from water from the sump. 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 intended function of the piping by control of system water chemistry and by detection and sizing of cracks by ISI. Inspection requirements of IWB 2500-1 category B-J or IWC 2500-1 category C-F-1, specify 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. Requirements for training and qualification of personnel and performance demonstration for procedures and equipment is in conformance with Appendices VII and VIII of ASME Section XI, or any other formal program approved by the NRC. ***(4) Detection of Aging Effects:*** Degradation of piping and fittings due to SCC can not occur without crack initiation; inspection schedule assures detection of cracks before the loss of intended function of the piping. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400, IWC-2400, or IWD-2400 for Class 1, 2, or 3 piping, respectively, 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 or IWC-3400 and IWC-3514. Planar and liner flaws are sized according to IWA-3300 and IWA-3400. Supplementary surface examination may be performed on interior and/or exterior surfaces when flaws are detected in volumetric examination. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWA-4000 and IWB-4000, and reexamination in accordance with	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.1.1 thru D1.1.5	Piping & Fittings	CFS, RHR or SDC, HPSI, LPSI, Lines to CVCS & SFP cooling	SS	25–340°C, chemically treated borated water	Loss of Material	Crevice and Pitting Corrosion	ASME Section XI, 1989 Edition. EPRI TR-105714. Plant Technical Specifications. *Operating Experience* NRC IN 84-18.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) 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 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 from inadvertent introduction of contaminants into the primary coolant system (IN 84–18). 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).	
Inservice inspection is in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWB 2500-1, testing category B-P for pressure retaining Class 1 components, e.g., CFS and other components within the containment; Table IWC 2500-1, examination category C-H for pressure retaining Class 2 components, e.g., most components in the safety injection system; and Table IWD 2500-1, test and examination category D-B for systems in support of emergency core cooling, e.g., refueling water tank (RWT) heating system. Water chemistry program for minimizing impurities by monitoring and maintaining water chemistry conditions based on guidelines of EPRI TR-105714 for primary water chemistry and plant technical specifications for refueling water storage tank water chemistry.	***(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 emergency core cooling system components. ***(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. However, inadvertent introduction of contaminants into the coolant system can occur either e.g., contaminants in the boric acid, or introduced through the free surface of spent fuel pool [NRC Information Notice (IN) 84–18], or from water from the sump. 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 corrosion by control of system water chemistry and by detection of coolant leakage by inservice inspection (ISI). Inspection requirements of ASME Section XI specify visual VT-2 (IWA-5240) examination during system leakage and hydrostatic tests of all pressure retaining Class 1* components according to Table IWB 2500-1 category B-P; Class 2 components required to operate or support the safety function according to Table IWC 2500-1 category C-H; and Class 3 components in support of emergency core cooling according to Table IWD 2500-1 category D-B. ***(4) Detection of Aging Effects:*** Degradation of the component due to corrosion would result in leakage of coolant. However, a one-time inspection of representative sample of the system population and most susceptible locations in the system should be conducted to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period. Follow up actions are based on the inspection results and plant technical specification. Inspection is performed in accordance with the requirements of ASME Code, 10CFR50 Appendix B, and ASTM standards, using a variety of nondestructive techniques including visual, ultrasonic,	Yes, Element 4 should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.1.1 thru D1.1.4	Piping & Fittings	CFS, RHR or SDC, HPSI, LPSI	SS	25–340°C, chemically treated borated water	Cumulative Fatigue Damage	Fatigue	ANSI B31.1.
D1.1.6	Piping & Fittings	Lines to Emergency Sump	SS	Stagnant chemically treated borated water	Loss of Material	Microbio-logically Induced Corrosion (MIC)	–
D1.1.7	Piping & Fittings	Bolting for flange connections in Items D1.1.1 thru D1.1.5	Nuts: Carbon Steel (CS), Bolts/Studs Alloy steel	Air, leaking chemically treated borated water	Loss of Material	Corrosion/ Boric Acid Wastage of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 Edition. Plant Technical Specifications. *Operating Experience* NRC IN 86-108 S3,

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) and surface techniques. Selection of susceptible locations is based on severity of conditions, time of service, and lowest design margin. ***(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 and hydrostatic tests are evaluated in accordance with IWB-3100 and acceptance standards of IWB-3400 and IWB-3522 for Class 1 components, IWC-3100 and acceptance standards of IWC-3400 and IWB-3516 for Class 2 components, IWD-3000 for Class 3 components. ***(7) Corrective Actions:*** Prior to service, corrective measures are needed to meet the requirements of IWB-3142 and IWA-5250. 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:*** Localized corrosion is likely to occur at mechanical joints, e.g., flange connections, because they present a crevice geometry at the sealing surfaces that may allow buildup of impurities due to stagnant conditions.	
Components have been designed for fatigue for a 40 y design life according to the requirements of ANSI B31.1.	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.	Yes TLAA
Plant-specific aging management program.	Plant-specific aging management program will be evaluated.	Yes no AMP
Implementation of NRC Generic Letter 88-05 and, based on plant specifications, inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWB 2500-1, testing category B-P for pressure retaining Class 1 components, e.g., CFS and other components within the containment; Table IWC 2500-1, examination category C-H for pressure retaining Class 2 components, e.g., most components in the safety injection system; and Table IWD 2500-1, test and examination category D-B for systems in support of emergency core cooling.	***(1) Scope of Program:*** The staff guidance of Generic Letter (GL) 88-05 provides assurances that a program has been implemented consisting of systematic measures to ensure that the effects of corrosion caused by leaking coolant containing boric acid does not lead to degradation and provides assurance that the reactor coolant pressure boundary will have a extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture. The program includes (a) determination of principal location, (b) examinations requirements and procedures for locating small leaks, and (c) engineering evaluations and corrective actions. ***(2) Preventive Actions:*** Minimizing reactor coolant leakage by frequent monitoring of the locations where potential leakage could occur and repairing the leaky components as soon as possible, prevent or mitigate boric acid corrosion. ***(3) Parameters Monitored/Inspected:*** The AMP	No

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) monitors the effects of boric acid corrosion on the intended function of the component by detection of coolant leakage by implementing the program delineated in GL 88-05 including guidelines for locating small leaks, conducting examinations, and performing engineering evaluations. Inspection requirements of ASME Section XI specify visual VT-2 (IWA-5240) examination during system leakage test and system hydrostatic test of all pressure retaining Class 1 components extending to and including the second closed valve at the boundary extremity, according to Table IWB 2500-1 category B-P; pressure retaining Class 2 components required to operate or support the safety function up to and including the first normally closed valve, according to Table IWC 2500-1 category C-H; and Class 3 components in support of emergency core cooling, containment heat removal, and reactor residual heat removal, according to Table IWD 2500-1 category D-B. ***(4) Detection of Aging Effects:*** Degradation of the component due to boric acid wastage can not occur without leakage of coolant containing boric acid; extent and schedule of the program delineated in GL 88-05 will assure detection of leakage before the loss of intended function of the component. ***(5) Monitoring and Trending:*** The program delineated in GL 88-05 provides for timely detection of leakage. 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 coolant leakage is evaluated in accordance with the program proposed by GL 88-05. ***(7) Corrective Actions:*** The leakage source and areas of general corrosion are located and corrective actions are in conformance with the program proposed by GL 88-05, and the requirements of ASME Section XI, IWB-3142 and IWA-5250. Components with local areas of corrosion that reduces the wall thickness by more than 10% require analytical evaluation to demonstrate acceptability. ***(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:*** Boric acid wastage observed in nuclear power plants may be classified into two distinct types: (a) corrosion that increases the rate of leakage, e.g., corrosion of closure bolting or fasteners in reactor coolant pressure boundary, and (b) corrosion that occurs some distance from the source of leakage. Some recent incidents of boric acid wastage (IN 86-108 S3) indicate that, although implementation of GL 88-05 ensures timely detection of leakage, there may still be a lack of awareness of the conditions that can lead to boric acid wastage.	

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.2.1	HPSI & LPSI Pumps	Bowl/Casing	SS, CS with SS cladding	25–340°C, chemically treated borated water	Loss of Material	Crevice and Pitting Corrosion	ASME Section XI, 1989 Edition. ASME OM Code–1990, Subsection ISTB. NRC GL 89-04 EPRI TR-105714. Plant Technical Specifications. *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 (edition specified in 10 CFR 50.55a), Table IWC 2500-1, examination category C-H for pressure retaining Class 2 components, e.g., most components in the safety injection system; and Table IWD 2500-1, test and examination category D-B for systems in support of emergency core cooling, e.g., refueling water tank (RWT) heating system; and based on the testing requirements of 10 CFR 50.55a for ASME Code Class 1, 2, and 3 pumps and valves, and additional NRC staff guidelines of NRC Generic Letter 89-04, inservice testing performed in accordance with ASME Subsection IWP (or Operation and Maintenance Code Subsection ISTB) for pumps, or other approved program in the plant specifications. Water chemistry program based on EPRI guidelines of TR-105714 for minimizing impurities by monitoring and maintaining primary water chemistry.	***(1) Scope of Program:*** The program relies on preventive measures to mitigate crevice or pitting corrosion and combination of inservice inspection (ISI) and inservice testing (IST) to monitor the effects of corrosion on the intended function of emergency core cooling system components. ***(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 corrosion. However, inadvertent introduction of contaminants into the coolant system can occur, e.g., contaminants in the boric acid, or introduced through the free surface of spent fuel pool [NRC Information Notice (IN) 84–18], or from water from the sump. 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 corrosion by control of water chemistry and by ISI to detect coolant leakage and IST to evaluate component performance. Inspection requirements of ASME Section XI specify visual VT-2 (IWA-5240) examination during system leakage and hydrostatic tests of all pressure retaining Class 2 components required to operate or support the safety function according to Table IWC 2500-1 category C-H; and Class 3 components in support of emergency core cooling according to Table IWD 2500-1 category D-B. Based on the requirements of 10 CFR 50.55a for ASME Code Class 1, 2, and 3 pumps and valves and additional guidelines of NRC Generic Letter (GL) 89-04, IST is performed in accordance with ASME Subsection IWP (or OM Code Subsection ISTB). ***(4) Detection of Aging Effects:*** Degradation of the component due to corrosion would result in leakage of coolant. However, a one-time inspection of representative sample of the system population and most susceptible locations in the system should be conducted to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period. Follow up actions are based on the inspection results and plant technical specification. Inspection is performed in accordance with the requirements of ASME Code, 10CFR50 Appendix B, and ASTM standards, using a variety of nondestructive techniques including visual, ultrasonic, and surface techniques. Selection of susceptible locations is based on severity of conditions, time of service, and lowest design margin. ***(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 and hydrostatic tests are evaluated in accordance with IWB-3100 and acceptance standards of IWB-3400 and IWB-3522 for Class 1 components, IWC-3100 and acceptance standards of IWC-3400 and IWB-3516 for Class 2 components, IWD-3000 for Class 3	Yes, Element 4 should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.2.1	HPSI & LPSI Pumps	Bowl/Casing	SS, CS with SS cladding	25-340°C, chemically treated borated water	Crack Initiation and Growth	Stress Corrosion Cracking (SCC)	ASME Section XI, 1989 Edition. Regulatory Guide 1.44. EPRI TR-105714. Plant Technical Specifications. *Operating Experience* NRC IN 80-38. NRC IN 84-18. NRC IN 91-05. NRC IN 94-63. NRC IN 97-19.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	components. ***(7) Corrective Actions:*** Prior to service, corrective measures are needed to meet the requirements of IWB-3142 and IWA-5250. 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:*** Localized corrosion is likely to occur at flange connections where buildup of impurities can occur.	
Guidelines of Regulatory Guide (RG) 1.44 to avoid sensitization of stainless steels and, based on plant specifications, inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWC 2500-1, examination category C-G for pressure retaining welds in pumps. Water chemistry program based on EPRI guidelines of TR-105714 for minimizing impurities by monitoring and maintaining primary water chemistry.	***(1) Scope of Program:*** The program includes preventive measures to mitigate stress corrosion cracking (SCC) of stainless steel (SS) and inservice inspection (ISI) to monitor the effects of SCC on the intended function of emergency core cooling system components. ***(2) Preventive Actions:*** Selection of material in compliance with the requirements of Regulatory Guide (RG) 1.44 prevents or mitigates SCC. 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, e.g., contaminants in the boric acid, or introduced through the free surface of spent fuel pool [NRC Information Notice (IN) 84–18], or from water from the sump. 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 intended function of the pump by control of primary water chemistry and by detection and sizing of cracks by ISI. Inspection requirements of IWC 2500-1 category C-G, specifies surface examination of either the inside or outside surface of all welds extending 1/2 in. on either side of the weld. In a group of multiple pumps of similar design, size, function, and service in a system, examination of only one pump is required. ***(4) Detection of Aging Effects:*** Degradation of pumps due to SCC can not occur without crack initiation and growth; ISI schedule assures detection of cracks or degradation of pump performance before the loss of intended function of the pump. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWC-2400 should provide timely detection of cracks. Surface examination of welds is conducted during each inspection interval. ***(6) Acceptance Criteria:*** Any SCC degradation is evaluated in accordance with IWC-3100 by comparing ISI results with the acceptance standards of IWC-3400 and IWC-3515. ***(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	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.2.1, D1.2.2	HPSI & LPSI Pumps	Bowl/Casing (External Surfaces), Bolting	Casing: CS; Nuts: CS, Bolts/Studs: Alloy Steel	Air, leaking chemically treated borated water	Loss of Material	Corrosion/ Boric Acid Wastage of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 Edition. Plant Technical Specifications.
D1.3.1	RWT Circulation Pump	Bowl/Casing	SS	25°C chemically treated borated water	Loss of Material	Crevice and Pitting Corrosion	*Same as effect of Crevice and Pitting Corrosion of Item D1.2.1 bowl/casing for HPSI and LPSI pumps*
D1.3.2	RWT Circulation Pump	Bolting	Nuts: CS, Bolts/Studs: Alloy Steel	Air, leaking chemically treated borated water	Loss of Material	Corrosion/ Boric Acid Wastage of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 Edition. Plant Technical Specifications.
D1.4.1	Valves (Check, Control, Hand, Motor Operated, and Relief Valves)	Body and Bonnet	SS, CS with SS cladding	25–340°C, chemically treated borated water	Cumulative Fatigue Damage	Fatigue	ANSI B31.1.
D1.4.1	Valves (Check, Control, Hand, Motor Operated, and Relief Valves)	Body and Bonnet	SS, CS with SS cladding	25–340°C, chemically treated borated water	Loss of Material	Crevice and Pitting Corrosion	ASME Section XI, 1989 Edition. ASME OM Code–1990, Appendix I and Subsection ISTC. NRC GL 89-04. NRC GL 96-05. NRC IN 88-70. EPRI TR-105714. Plant Technical Specifications. *Operating Experience* NRC IN 84-18. NRC IN 98-23. NRC IN 94-63.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) 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, potential of SCC exists from inadvertent introduction of contaminants into the primary coolant system (IN 84–18). 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).	
Same as effect of Corrosion/Boric Acid Wastage of Item D1.1.7 Bolting for flange connections in Items D1.1.1 thru D1.1.5.	*Same as effect of Corrosion/Boric Acid Wastage of Item D1.1.7 Bolting for flange connections in Items D1.1.1 thru D1.1.5.*	No
Same as effect of Crevice and Pitting Corrosion of Item D1.2.1 bowl/casing for HPSI and LPSI pumps.	*Same as effect of Crevice and Pitting Corrosion of Item D1.2.1 bowl/casing for HPSI and LPSI pumps.*	Yes, Element 4 should be further evaluated
Same as effect of Corrosion/Boric Acid Wastage of Item D1.1.8 Bolting for flange connections in Items D1.1.1 thru D1.1.7.	*Same as effect of Corrosion/Boric Acid Wastage of Item D1.1.8 Bolting for flange connections in Items D1.1.1 thru D1.1.7.*	No
Components have been designed for fatigue for a 40 y design life according to the requirements of ANSI B31.1.	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.	Yes TLAA
Inservice inspection is in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWB 2500-1, testing category B-P for pressure retaining Class 1 components; Table IWC 2500-1, examination category C-H for pressure retaining Class 2 components; and Table IWD 2500-1, test and examination category D-B for systems in support of emergency core cooling system; and based on the testing requirements of 10 CFR 50.55a for ASME Code Class 1, 2, and 3 pumps and valves, additional staff guidelines of NRC Generic Letter (GL) 89-04 regarding the scope of inservice testing (IST), and information in NRC IN 88-70 regarding scope and testing of safety-related check	***(1) Scope of Program:*** The program relies on preventive measures to mitigate crevice or pitting corrosion and combination of inservice inspection (ISI) and inservice testing (IST) to monitor the effects of corrosion on the intended function of emergency core cooling system components. ***(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 corrosion. However, inadvertent introduction of contaminants into the coolant system can occur, e.g., contaminants in the boric acid, or introduced through the free surface of spent fuel pool [NRC Information Notice (IN) 84–18], or from water from the sump. The aging management program must therefore rely upon inservice inspections in accordance with ASME Section XI to detect possible degradation. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors	Yes, Element 4 should be further evaluated

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
(continued from previous page) valves, and in GL 96-05 regarding safety-related motor-operated valves, IST is performed in accordance with ASME Subsection IWV (Operation and Maintenance Code Appendix I and Subsection ISTC), to ensure that the changes in design-basis performance of safety-related valves resulting from degradation can be identified and managed. Water chemistry program for minimizing impurities by monitoring and maintaining water chemistry conditions based on guidelines of EPRI TR-105714 for primary water chemistry.	*(continued from previous page)* the effects of corrosion by ISI to detect coolant leakage and IST to evaluate component performance. Inspection requirements of ASME Section XI specify visual VT-2 (IWA-5240) examination during system leakage test and hydrostatic test of all pressure retaining Class 1 components according to Table IWB 2500-1 category B-P; pressure retaining Class 2 components required to operate or support the safety function, according to Table IWC 2500-1 category C-H; and Class 3 components in support of emergency core cooling, according to Table IWD 2500-1 category D-B. Based on the requirements of 10 CFR 50.55a for ASME Code Class 1, 2, and 3 pumps and valves and additional guidelines of NRC GLs 89-04 and 96-05, IST is performed in accordance with ASME Subsection IWV (OM Code Appendix I and Subsection ISTC). ***(4) Detection of Aging Effects:*** Degradation of the component due to corrosion would result in leakage of coolant. However, a one-time inspection of representative sample of the system population and most susceptible locations in the system should be conducted to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period. Follow up actions are based on the inspection results and plant technical specification. Inspection is performed in accordance with the requirements of ASME Code, 10CFR50 Appendix B, and ASTM standards, using a variety of nondestructive techniques including visual, ultrasonic, and surface techniques. Selection of susceptible locations is based on severity of conditions, time of service, and lowest design margin. ***(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 and hydrostatic tests are evaluated in accordance with IWB-3100 and acceptance standards of IWB-3400 and IWB-3522 for Class 1 components, IWC-3100 and acceptance standards of IWC-3400 and IWB-3516 for Class 2 components, IWD-3000 for Class 3 components. ***(7) Corrective Actions:*** Prior to service, corrective measures are needed to meet the requirements of IWB-3142 and IWA-5250. 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:*** Corrosion has been observed in guide rings of relief valves (IN 98-23) and charging pump casing (IN 94-63).	

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.4.1	Valves (Check, Control, Hand, Motor Operated, and Relief Valves)	Body and Bonnet	SS, CS with SS cladding	25–340°C, chemically treated borated water	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. Regulatory Guide 1.44. EPRI TR-105714 Plant Technical Specifications. *Operating Experience* NRC IN 80-38. NRC IN 84-18. NRC IN 91-05. NRC IN 94-63. NRC IN 97-19,

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Guidelines of Regulatory Guide (RG) 1.44 to avoid sensitization of stainless steels and, based on plant specifications, inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWB 2500-1, examination categories B-M-1 and B-M-2 for Class 1 valves; Table IWC 2500-1, examination category C-G for pressure retaining welds in Class 2 valves. Water chemistry program for minimizing impurities by monitoring and maintaining water chemistry conditions based on guidelines of EPRI TR-105714 for primary water chemistry.	***(1) Scope of Program:*** The program includes preventive measures to mitigate stress corrosion cracking (SCC) of stainless steel (SS) and inservice inspection (ISI) to monitor the effects of SCC on the intended function of emergency core cooling system components. ***(2) Preventive Actions:*** Selection of material in compliance with the requirements of Regulatory Guide (RG) 1.44 prevents or mitigates SCC. 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, e.g., contaminants in the boric acid, or introduced through the free surface of spent fuel pool [NRC Information Notice (IN) 84–18], or from water from the sump. 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 intended function of the valves by detection and sizing of cracks by ISI. Inspection requirements of Table IWB 2500-1 for Class 1 valves, examination category B-M–1 specify 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 specifies visual VT–3 examination of internal surfaces of the valve. Table IWC 2500-1 for Class 2 valves, category C-G specifies for all valves in each piping run examined under category C-F, surface examination of either the inside or outside surface of all welds extending 1/2 in. on either side of the weld. In a group of multiple valves of similar design, size, function, and service in a system, examination of only one valve is required. ***(4) Detection of Aging Effects:*** Degradation of valves due to SCC can not occur without crack initiation and growth; ISI schedule assures detection of cracks or degradation of valve performance before the loss of intended function of the valves. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 or IWC-2400 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. ***(6) Acceptance Criteria:*** Any SCC degradation is evaluated in accordance with IWC-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3518 for volumetric and surface examination of welds or IWB-3519 for visual examination of valve internal surfaces and IWC-3400 and IWC-3515 for surface examination of welds in Class 2 valves. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWA-4000 and IWB-4000. ***(8 & 9) Confirmation Process and Administrative***	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.4.1, D1.4.2	Valves (Check, Control, Hand, Motor Operated, and Relief Valves)	Body and Bonnet (External Surfaces), Bolting	Body and Bonnet: CS; Nuts: CS, Bolts/Studs: Alloy Steel	Air, leaking chemically treated borated water	Loss of Material	Corrosion/ Boric Acid Wastage of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 Edition. Plant Technical Specifications.
D1.5.1 thru D1.5.4	Heat Exchangers (Reactor Coolant Pump Seal, HPSI Pump Seal, LPSI Pump Seal, RHR or SDC)	Bonnet/Cover, Tubing, Shell, Case/Cover	Bonnet/ Cover & Tubing: SS, Shell: CS, Case/Cover: Cast iron	Chemically treated borated water; and treated component cooling water	Loss of Material	Crevice and Pitting Corrosion	ASME Section XI, 1989 Edition. ASME OM S/G, Pt 2. NRC GL 89-13. EPRI TR-105714. Plant Technical Specifications.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) **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, significant potential of SCC exists from inadvertent introduction of contaminants into the primary coolant system (IN 84–18). 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).	
Same as effect of Corrosion/Boric Acid Wastage of Item D1.1.7 Bolting for flange connections in Items D1.1.1 thru D1.1.5.	*Same as effect of Corrosion/Boric Acid Wastage of Item D1.1.7 Bolting for flange connections in Items D1.1.1 thru D1.1.5.*	No
Detection of reactor coolant leakage by radiation monitors in component cooling system; inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWB 2500-1, testing category B-P for pressure retaining Class 1 components, Table IWC 2500-1, examination category C-H for pressure retaining Class 2 components; performance testing in accordance with ASME OM Standards and Guides, Part 2 to ensure that the heat exchanger serviced by the closed-cycle cooling water system is performing its function acceptably; and water chemistry control program based on EPRI TR-105714 for primary water and plant technical specifications for cooling water.	***Scope of Program:*** The program includes monitoring and control of water chemistry and cooling water chemistry to minimize exposure to aggressive environments, and performance testing in accordance with ASME OM-Standards and Guides, Part 2 provides assurance that the heat exchanger serviced by the closed-cycle cooling water system is performing its function acceptably. ***(2) Preventive Actions:*** Use of appropriate materials, lining or coating to protect the underlying metal surfaces, and control of primary water chemistry and cooling water chemistry to minimize exposure to aggressive environment. The parameters monitored include halogens, sulfates, oxygen, and pH in the primary water, and in addition to these, dissolved copper and iron, suspended solids, and gamma activity, in the component cooling water. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of corrosion by surveillance program to detect coolant leakage and inservice testing to evaluate component performance. The parameters monitored are directly related to corrosion, e.g., dissolved iron and copper, and by detection of leakage by radiation monitors, in the component cooling system. Also, inspection requirements of ASME Section XI specify visual VT-2 (IWA-5240) examination during system leakage test and hydrostatic test of all pressure retaining Class 1 components extending to and including the second closed valve at the boundary extremity, according to Table IWB 2500-1 category B-P, and pressure retaining Class 2 components required to operate or support the safety function, according to Table IWC 2500-1 category C-H. Performance testing is conducted in accordance with ASME OM S/G, Part 2. ***(4) Detection of Aging Effects:*** Degradation of component due to corrosion would result in leakage of coolant or degradation of component performance. Monitoring of radiation and suspended	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.5.3 thru D1.5.5	Heat Exchangers (RCP Seal, HPSI Pump Seal, LPSI Pump Seal, RHR or SDC)	Shell, Case/Cover, (External Surfaces); Bolting	Shell: CS; Case/Cover: Cast iron; Nuts: CS, Bolts/Studs: Alloy Steel	Air, leaking chemically treated borated water	Loss of Material	Corrosion/ Boric Acid Wastage of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 Edition. Plant Technical Specifications.
D1.6.1 thru D1.6.3	Heat Exchanger (RWT Heating)	Bonnet and Cover, Tubing, Shell	Bonnet/ Cover & Tubing: SS, Shell: CS	Chemically treated borated water; and untreated cooling water	Loss of Material	Crevice and Pitting Corrosion	ASME Section XI, 1989 Edition. ASME OM S/G, Pt 2. NRC GL 89-13. EPRI TR-105714. Plant Technical Specifications.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) solids would detect the existence of corrosion; extent and schedule of inspection/testing assure detection of corrosion before the loss of intended function of the component. ***(5) Monitoring and Trending:*** Results from performance tests to verify the heat transfer capabilities are trended. Also, based on the recommendations of NRC GL 89-13 or its equivalent, if adequacy of cooling water chemistry control can not be confirmed, implement Action III of GL 89-13 to include inspection and maintenance program for closed-cycle cooling water system to ensure that corrosion, erosion, and protective coating failure can not degrade the performance of safety-related systems serviced by closed-cycle cooling water. ***(6) Acceptance Criteria:*** Heat exchanger performance test results are evaluated in accordance with the guidelines of ASME OM S/G Part 2. Maximum levels for various impurities in secondary side water and cooling water are specified. Results of Section XI leakage tests are evaluated in accordance with IWB-3100 and acceptance standards of IWB-3400 and IWB-3522 for Class 1 components, and IWC-3100 and acceptance standards of IWC-3400 and IWB- 3516 for Class 2 components. ***(7) Corrective Actions:*** If the heat exchanger fails to perform adequately, corrective actions are taken in accordance with OM S/G Part 2. Root cause evaluation and appropriate corrective action when acceptable limits are exceeded or leakage is detected. ***(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 plant experience with this AMP indicate timely detection of corrosion in the ECCS heat exchangers.	
Same as effect of Corrosion/Boric Acid Wastage of Item D1.1.7 Bolting for flange connections in Items D1.1.1 thru D1.1.5.	*Same as effect of Corrosion/Boric Acid Wastage of Item D1.1.7 Bolting for flange connections in Items D1.1.1 thru D1.1.5.*	No
Inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWD 2500-1, examination category D-B for systems in support of emergency core cooling; performance testing in accordance with ASME OM Standards and Guides, Part 2 to ensure that the heat exchanger serviced by the closed-cycle cooling water system is performing its function acceptably; and water chemistry control program based on EPRI TR-105714 for primary water and plant technical specifications for cooling water.	***(1) Scope of Program:*** The program includes monitoring and control of water chemistry and cooling water chemistry to minimize exposure to aggressive environments, and inservice inspection and performance testing provides assurance that the heat exchanger serviced by the closed-cycle cooling water system is performing its function acceptably. ***(2) Preventive Actions:*** Use of appropriate materials, lining or coating to protect the underlying metal surfaces, and control of primary water chemistry based on EPRI guidelines of TR-105714 and cooling water chemistry based on the plant technical specifications to minimize exposure to aggressive environment. The parameters monitored include halogens, sulfates, oxygen, and pH in the	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.6.3, D1.6.4	Heat Exchanger (RWT Heating)	Shell (External Surface), Bolting	Shell: CS, Nuts: CS, Bolts/Studs: Alloy Steel	Air, leaking chemically treated borated water	Loss of Material	Corrosion/ Boric Acid Wastage of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 Edition. Plant Technical Specifications.
D1.7.1	Safety Injection Tank (Accumulators)	Shell (External Surface)	CS with SS cladding	Air, leaking chemically treated borated water	Loss of Material	Corrosion/ Boric Acid Wastage of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 Edition. Plant Technical Specifications.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) primary water, and in addition to these, dissolved copper and iron, and suspended solids in the component cooling water. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of corrosion by surveillance program to detect coolant leakage and inservice testing to evaluate component performance. Inspection requirements of ASME Section XI, Table IWD 2500-1, examination category D-B specify visual VT-2 (IWA-5240) examination during system leakage and hydrostatic test of all pressure retaining Class 3 components in support of emergency core cooling. Performance testing is conducted in accordance with ASME OM S/G, Part 2. ***(4) Detection of Aging Effects:*** Degradation of component due to corrosion would result in leakage of coolant or degradation of component performance; extent and schedule of inspection/testing assure detection of corrosion before the loss of intended function of the component. ***(5) Monitoring and Trending:*** Results from performance tests to verify the heat transfer capabilities are trended. Also, based on the recommendations of NRC GL 89-13 or its equivalent, if adequacy of cooling water chemistry control can not be confirmed, implement Action III of GL 89-13 to include inspection and maintenance program for closed-cycle cooling water system to ensure that corrosion, erosion, and protective coating failure can not degrade the performance of safety-related systems serviced by closed-cycle cooling water. ***(6) Acceptance Criteria:*** Heat exchanger performance test results are evaluated in accordance with the guidelines of ASME OM S/G Part 2. Maximum levels for various impurities in secondary side water and cooling water are specified. ***(7) Corrective Actions:*** If the heat exchanger fails to perform adequately, corrective actions are taken in accordance with OM S/G Part 2. ***(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 the effects of corrosion in RWT heat exchanger.	
Same as effect of Corrosion/Boric Acid Wastage of Item D1.1.7 Bolting for flange connections in Items D1.1.1 thru D1.1.5.	*Same as effect of Corrosion/Boric Acid Wastage of Item D1.1.7 Bolting for flange connections in Items D1.1.1 thru D1.1.5.*	No
Same as effect of Corrosion/Boric Acid Wastage of Item D1.1.7 Bolting for flange connections in Items D1.1.1 thru D1.1.5.	*Same as effect of Corrosion/Boric Acid Wastage of Item D1.1.7 Bolting for flange connections in Items D1.1.1 thru D1.1.5.*	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.7.1 thru D1.7.3	Safety Injection Tank (Accumulators)	Shell, Manway, Penetrations/ Nozzles	Shell: CS with SS cladding, Other: SS	Chemically treated borated water	Loss of Material	Crevice and Pitting Corrosion	ASME Section XI, 1989 Edition. Plant Technical Specifications.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWC 2500-1, examination category C-H for pressure retaining Class 2 components, and water chemistry control program based on plant technical specifications.	***(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 emergency core cooling system components. ***(2) Preventive Actions:*** The system is filled with borated water with controlled water chemistry. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of corrosion by detection of coolant leakage by ISI. Inspection requirements of ASME Section XI specify visual VT-2 (IWA-5240) examination during system leakage test and system hydrostatic test of all pressure retaining Class 2 components required to operate or support the safety function up to the first normally closed valve, according to Table IWC 2500-1 category C-H. ***(4) Detection of Aging Effects:*** Degradation of the component due to corrosion would result in leakage of coolant. However, a one-time inspection of representative sample of the system population and most susceptible locations in the system should be conducted to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period. Follow up actions are based on the inspection results and plant technical specification. Inspection is performed in accordance with the requirements of ASME Code, 10CFR50 Appendix B, and ASTM standards, using a variety of nondestructive techniques including visual, ultrasonic, and surface techniques. Selection of susceptible locations is based on severity of conditions, time of service, and lowest design margin. ***(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 and hydrostatic tests are evaluated in accordance with IWC-3100 and acceptance standards of IWC-3400 and IWB-3516. Any evidence of aging effects or unacceptable results are evaluated. ***(7) Corrective Actions:*** Prior to service, corrective measures are needed to meet the requirements of IWA-5250. 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:*** Localized corrosion is likely to occur at mechanical joints because they present a crevice geometry at the sealing surfaces that may allow buildup of impurities due to stagnant conditions.	Yes, Element 4 should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.7.3	Safety Injection Tank (Accumulators)	Penetrations/ Nozzles	CS with SS cladding	Chemically treated borated water	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. Regulatory Guide 1.44. Plant Technical Specifications. *Operating Experience* NRC IN 80-38. NRC IN 84-18. NRC IN 91-05. NRC IN 94-63. NRC IN 97-19.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Guidelines of Regulatory Guide (RG) 1.44 to avoid sensitization of stainless steels and, based on plant specifications, inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWC 2500-1, examination category C-H for pressure retaining Class 2 components, and water chemistry control program based on plant technical specifications.	***(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 the intended function of the RWT. ***(2) Preventive Actions:*** Selection of material in compliance with the requirements of Regulatory Guide (RG) 1.44 prevents or mitigates SCC. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of SCC on intended function of the RWT by detection of leakage. Inspection requirements of ASME Section XI specify visual VT-2 (IWA-5240) examination during system leakage test and system hydrostatic test of all pressure retaining Class 2 components required to operate or support the safety function, according to Table IWC 2500-1 category C-H. ***(4) Detection of Aging Effects:*** Degradation of the component due to SCC can not occur without leakage of coolant. However, a one-time inspection of representative sample of the system population and most susceptible locations in the system should be conducted to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period. Follow up actions are based on the inspection results and plant technical specification. Inspection is performed in accordance with the requirements of ASME Code, 10CFR50 Appendix B, and ASTM standards, using a variety of nondestructive techniques including visual, ultrasonic, and surface techniques. Selection of susceptible locations is based on severity of conditions, time of service, and lowest design margin. ***(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 and hydrostatic tests are evaluated in accordance with IWC-3100 and acceptance standards of IWC-3400 and IWB-3516. Any evidence of aging effects or unacceptable results are evaluated. ***(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 has been observed in safety injection lines (IN 97-19 & 84-18), charging pump casing cladding (INs 80-38 and 94-63), and instrument nozzles in safety injection tanks (IN 91-05).	Yes, Element 4 should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.8.1 thru D1.8.3	Refueling Water Tank (RWT)	Shell, Manhole, Penetrations/ Nozzles	SS	Chemically treated borated water	Loss of Material	Crevice and Pitting Corrosion	ASME Section XI, 1989 Edition.
D1.8.3	Refueling Water Tank (RWT)	Penetrations/ Nozzles	SS	Chemically treated borated water	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. Regulatory Guide 1.44. NRC IN 80-38. NRC IN 84-18. NRC IN 91-05. NRC IN 94-63. NRC IN 97-19.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWC 2500-1, test and examination category D-B for systems in support of emergency core cooling, and water chemistry control program based on plant technical specifications.	***(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 emergency core cooling system components. ***(2) Preventive Actions:*** Control of water chemistry based on plant technical specifications. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of corrosion by detection of coolant leakage by ISI. Inspection requirements of ASME Section XI, Table IWD 2500-1, category D-B specify visual VT-2 (IWA-5240) examination during system leakage and hydrostatic tests of all pressure retaining Class 3 components in support of emergency core cooling. ***(4) Detection of Aging Effects: :*** Degradation of the component due to corrosion would result in leakage of coolant. However, a one-time inspection of representative sample of the system population and most susceptible locations in the system should be conducted to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period. Follow up actions are based on the inspection results and plant technical specification. Inspection is performed in accordance with the requirements of ASME Code, 10CFR50 Appendix B, and ASTM standards, using a variety of nondestructive techniques including visual, ultrasonic, and surface techniques. Selection of susceptible locations is based on severity of conditions, time of service, and lowest design margin. ***(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 and hydrostatic tests are evaluated in accordance with IWD-3000 for Class 3 components. Any evidence of aging effects or unacceptable results are evaluated. ***(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:*** Localized corrosion is likely to occur at crevices that may allow buildup of impurities due to stagnant conditions.	Yes, Element 4 should be further evaluated
Guidelines of Regulatory Guide (RG) 1.44 to avoid sensitization of stainless steels, inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWD 2500-1, test and examination category D-B for systems in support of emergency core cooling, and water chemistry control program based on plant technical specifications.	***(1) Scope of Program:*** The program includes preventive measures to mitigate stress corrosion cracking (SCC) of stainless steel (SS) and inservice inspection (ISI) to monitor the effects of SCC on the intended function of the RWT. ***(2) Preventive Actions:*** Selection of material in compliance with the requirements of Regulatory Guide (RG) 1.44 prevents or mitigates SCC. Control of water chemistry is based on plant technical specifications. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of SCC on intended function of the RWT by	Yes, Element 4 should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.8.4	Refueling Water Tank (RWT)	Bolting	Nuts: CS, Bolts/Studs: Alloy Steel	Air, leaking chemically treated borated water	Loss of Material	Corrosion/ Boric Acid Wastage of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 Edition. Plant Technical Specifications.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) detection of leakage. Inspection requirements of ASME Section XI, Table IWD 2500-1, category D-B specify visual VT-2 (IWA-5240) examination during system leakage and hydrostatic tests of all pressure retaining Class 3 components in support of emergency core cooling. ***(4) Detection of Aging Effects:*** Degradation of the component due to SCC can not occur without leakage of coolant. However, a one-time inspection of representative sample of the system population and most susceptible locations in the system should be conducted to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period. Follow up actions are based on the inspection results and plant technical specification. Inspection is performed in accordance with the requirements of ASME Code, 10CFR50 Appendix B, and ASTM standards, using a variety of nondestructive techniques including visual, ultrasonic, and surface techniques. Selection of susceptible locations is based on severity of conditions, time of service, and lowest design margin. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI should provide for timely detection of leakage. 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 and hydrostatic tests are evaluated in accordance with IWD-3000 for Class 3 components. Any evidence of aging effects or unacceptable results are evaluated. ***(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:*** 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, significant potential of SCC exists from inadvertent introduction of contaminants into the primary coolant system (IN 84–18). SCC has been observed in safety injection lines (IN 97-19 and 84-18), charging pump casing cladding (INs 80-38 and 94-63), internal bolting in swing check valves (IN 89-02), and instrument nozzles in safety injection tanks (IN 91-05).	
Same as effect of Corrosion/Boric Acid Wastage of Item D1.1.7 Bolting for flange connections in Items D1.1.1 thru D1.1.5.	*Same as effect of Corrosion/Boric Acid Wastage of Item D1.1.7 Bolting for flange connections in Items D1.1.1 thru D1.1.5.*	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.8.5	Refueling Water Tank (RWT)	Perimeter Seal	Cold plastic coal tar pitch flashing or other organic sealants	Outside atmosphere	Loss of elasticity (drying out)	Weathering	Plant Technical Specifications

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Plant-specific aging management program.	Plant-specific aging management program will be evaluated.	Yes no AMP

D2 Emergency Core Cooling System (BWR)

D2.1 Piping & Fittings

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

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

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

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

- D2.3.1 Body and Bonnet

D2.4 Heat Exchangers (RHR & LPCI)

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

D2.5 Header and Spray Nozzles System

D2.5.1 Piping and Fittings

D2.5.2 Flow Orifice

D2.5.3 Headers

D2.5.4 Spray Nozzles

D2.6 Isolation Condenser

D2.6.1 Tubing

D2.6.2 Tubesheet

D2.6.3 Channel Head

D2.6.4 Shell

D2. Emergency Core Cooling System (Boiling Water Reactors)

System, Structures, and Components

The system, structures, and components included in this table comprise the emergency core cooling systems for boiling water reactors (BWRs) designed to cool the reactor core and provide safe shutdown following a design basis accident, and consist of the high-pressure coolant injection (HPCI), reactor core isolation cooling (RCIC), high-pressure core spray (HPCS), automatic depressurization system (ADS), low-pressure core spray (LPCS), low-pressure coolant injection (LPCI) or residual heat removal (RHR), including various pumps and valves, RHR heat exchangers, isolation condenser, and containment spray system (CSS). 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," all components in the emergency core cooling system outside the containment are classified as Group B Quality Standards and portion of the CSS inside the containment up to the isolation valve is classified Group A Quality Standard. Portion of the HPCI, RCIC, HPCS, LPCS, and LPCI (or RHR) systems extending from the reactor vessel up to the second isolation valve outside the containment are classified as Group A and are covered in Table IV C1. The aging management program for the variety of lines that penetrate the containment and associated isolation valves is reviewed in Table V C.

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 and are not subject to aging management review pursuant to 10 CFR 54.21(a)(1)(i).

System Interfaces

The systems that interface with the steam turbine system include the reactor coolant pressure boundary (Table IV C1), feedwater system (Table VIII D2), condensate system (Table VIII E), nitrogen system (Table VIII H), spent fuel pool cooling and cleanup (Table VII A4), suppression pool cleanup system (Table VII A5), and closed cycle cooling water (Table VII C2).

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D2.1.1 thru D2.1.7	Piping & Fittings	High Pressure Coolant Injection (HPCI), Reactor Core Isolation Cooling (RCIC), High-Pressure Core Spray (HPCS), Low-Pressure Core Spray (LPCS), Low Pressure Coolant Injection (LPCI) or Residual Heat Removal (RHR), Lines to Spent Fuel Pool (SFP) & Suppression Chamber (SC), Lines to Containment Spray System (CSS)	Carbon Steel (CS)	25–288°C, Oxygenated Water	Loss of Material	General, Crevice, and Pitting Corrosion	ASME Section XI, 1989 Edition. EPRI TR-103515. Plant Technical Specifications

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Water chemistry program based on EPRI guidelines of TR-103515 and implemented through the plant technical specifications for minimizing impurities by monitoring and maintaining water chemistry conditions, and inservice inspection is in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWC 2500-1, examination category C-H for pressure retaining Class 2 components.	***(1) Scope of Program:*** The program relies on preventive measures to mitigate general, crevice, and pitting corrosion and inservice inspection (ISI) to monitor the effects of corrosion on emergency core cooling system components. ***(2) Preventive Actions:*** Mitigation is by monitoring and control of water chemistry to minimize concentration of corrosive impurities in accordance with the EPRI guidelines of TR-103515. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of corrosion by detection of coolant leakage by inservice inspection (ISI). Inspection requirements of ASME Section XI specify visual VT-2 (IWA-5240) examination during system leakage and hydrostatic tests of all pressure retaining Class 2 components according to Table IWC 2500-1 category C-H. ***(4) Detection of Aging Effects:*** Degradation of the component due to corrosion would result in leakage of coolant. However, a one-time inspection of representative sample of the system population and most susceptible locations in the system should be conducted to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period. Follow up actions are based on the inspection results and plant technical specification. Inspection is performed in accordance with the requirements of ASME Code, 10CFR50 Appendix B, and ASTM standards, using a variety of nondestructive techniques including visual, ultrasonic, and surface techniques. Selection of susceptible locations is based on severity of conditions, time of service, and lowest design margin. ***(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 and hydrostatic tests are evaluated in accordance with IWC-3100 and acceptance standards of IWC-3400 and IWB-3516 for Class 2 components. Any evidence of aging effects or unacceptable results are evaluated. ***(7) Corrective Actions:*** Prior to service, corrective measures are needed to meet the requirements of IWB-3142 and IWA-5250. Repair are in conformance with IWA-4000 and IWB-4000 and replacement according to IWA-7000 and IWB-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:*** Localized corrosion is likely to occur at mechanical joints, because of crevice geometry at the sealing surfaces that may allow buildup of impurities due to stagnant conditions. No significant corrosion related problem has been reported for piping and fittings in BWR emergency core cooling system.	Yes, Element 4 should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D2.1.1	Piping & Fittings	HPCI	Carbon Steel (CS), Stainless Steel (SS)	25–288°C, Oxygenated Water	Cumulative Fatigue Damage	Fatigue	ANSI B31.1.
D2.1.1 thru D2.1.7	Piping & Fittings	HPCI, RCIC, HPCS, LPCS, LPCI or RHR, Lines to SFP and SC, Lines to CSS	Stainless Steel (SS)	25–288°C, Oxygenated Water	Crack Initiation and Growth	Stress Corrosion Cracking (SCC), Inter-granular Stress Corrosion Cracking (IGSCC)	NUREG-0313, Rev. 2. NRC GL 88-01. ASME Section XI, 1989 Edition. BWRVIP-29. EPRI TR-103515. Regulatory Guide 1.45.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Components have been designed for fatigue for a 40 y design life according to the requirements of ANSI B31.1.	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.	Yes TLAA
Program delineated in NUREG-0313, Rev. 2 and implemented through NRC Generic letter (GL) 88-01, and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWC 2500-1, examination category C-F-1 for pressure retaining welds in Class 2 stainless steel piping, and testing category C-H for system leakage. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-103515 and BWRVIP-29 to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program is focused on managing and implementing the counter measures 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 contains water at a temperature above 93°C (above 200°F) during power operation regardless of Code classification. NUREG-0313 and GL 88-01, respectively, describe the technical basis and staff guidance regarding mitigating IGSCC in BWRs. ***(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. ***(3) Parameters Monitored/Inspected:*** The AMP monitors IGSCC of austenitic SS piping by detection and sizing of cracks by implementing the inspection requirements delineated in GL 88-01 including guidelines for inspection schedule, methods, personnel, sample expansion, and leak detection requirements. Requirements for training and qualification of personnel and performance demonstration for procedures and equipment is in conformance with Appendices VII and VIII of ASME Section XI, or any other formal program approved by the NRC. Inspection requirements of testing category C-H specify visual VT-2 (IWA-5240) examination of all pressure retaining Class 2 components during system leakage test (IWB-5221) and system hydrostatic test (IWB-5222). Leakage detection is in conformance with Position C of Regulatory Guide 1.45 and the guidelines in GL 88-01. Coolant water chemistry is monitored and maintained in accordance with the EPRI guidelines in BWRVIP-29 to minimize the potential of crack initiation and growth. ***(4) Detection of Aging Effects:*** Aging degradation of the piping can not occur without crack initiation and growth; extent and schedule of inspection as delineated in GL 88-01 is adequate and assures timely detection of cracks before the loss of intended function of austenitic SS piping and fittings. ***(5) Monitoring and Trending:*** Inspection schedule and sample size specified in Table 1 of GL 88-01 are based on the IGSCC susceptibility 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, e.g., 25% are examined every 10 y, at least 12% in 6 y. Inspection extent and schedule are enhanced for welds of non-resistant materials, or such welds that have been treated by SI or reinforced by weld overlay. ***(6) Acceptance***	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D2.1.8	Piping & Fittings	Automatic Depressuri-zation System (ADS)	CS, SS	Moist Containment Atmosphere (Air/Nitrogen), Steam, or Oxygenated Water	Loss of Material	General, Crevice, and Pitting Corrosion	Plant Technical Specifications.
D2.1.9, D2. 1.10	Piping & Fittings	Lines to HPCI & RCIC Pump Turbine, Lines from HPCI & RCIC Pump Turbine to Condenser	CS	Air and Steam up to 320°C	Wall Thinning	Erosion/ Corrosion (E/C)	NUREG-1344. EPRI NSAC-202L-R2. NRC GL 89-08 NRC IN 93-21. EPRI TR-103515. BWRVIP-29. *Operating Experience* NRC IN 89-53 NRC IN 91-18 NRC IN 91-18 S1 NRC IN 97-84

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) **Criteria:** Any IGSCC degradation is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWC-3400 and IWC-3514. Planar and liner flaws are evaluated according to IWA-3300 and IWA-3400. ***(7) Corrective Actions:*** Repair is in conformance with IWA-4000 and IWB-4000 or GL 88-01. Continued operation without repair requires that crack growth calculation be performed according to the guidance of GL 88-01 or other approved procedures. Repair methods include weld overlay reinforcement or partial replacement. Approved clamping devices may be used for temporary reinforcement of cracked weldments. SI is considered effective mitigation for short and shallow cracks, e.g., not longer than 10% of circumference and not deeper than 30% of wall thickness. ***(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. Significant cracking has occurred in RHR system and reactor water cleanup system piping welds. The AMP delineated in GL 88-01 has been effective in managing the effects of IGSCC in SS piping.	
Plant specific aging management program.	Plant specific aging management program is to be evaluated.	Yes, no AMP
Program delineated in NUREG-1344 for single phase lines and implemented through NRC Generic Letter 89-08; CHECWORKS Code; EPRI guidelines of NSAC-202L-R2 for effective erosion/corrosion program; and water chemistry program based on EPRI guidelines in TR-103515 and BWRVIP-29 for water chemistry in BWRs.	***(1) Scope of Program:*** The NUMARC program delineated in Appendix A of NUREG-1344 and implemented through NRC Generic Letter (GL) 89-08 provides assurances that procedures or administrative controls are in place to assure that the NUMARC program or other equally effective programs are implemented and the structural integrity of all high-energy (two phase as well as single phase) carbon steel systems is maintained. The program includes the following recommendations: (a) conduct appropriate analysis and limited baseline inspection, (b) determine the extent of thinning and repair/replace components, and (c) perform follow-up inspections to confirm or quantify and take longer term corrective actions. Although technical aspects of the CHECWORKS Code, including the parameters and inputs, are acceptable, the EPRI guidance document NSAC-202L-R2 (April 1999) is too general to ensure applicant's flow-accelerated corrosion program will be effective in managing aging in safety-related systems. However, E/C is not considered to	Yes, Element 1 should be further evaluated

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) be an issue for the steam lines to pump turbines. ***(2) Preventive Actions:*** The rate of E/C is affected by piping material, geometry and hydrodynamic conditions, and operating conditions such as temperature, pH, and dissolved oxygen content. Mitigation is by selecting material considered resistant to E/C, adjusting water chemistry and operating conditions, and improving hydrodynamic conditions through design modifications. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of E/C on the intended function of piping by measuring wall thickness by nondestructive examination and performing analytical evaluations. The inspection program delineated in NUREG-1344 requires ultrasonic and radiographic testing of 10 most susceptible locations and 5 additional locations based on unique operating conditions or special considerations. For each location outside the acceptance guidelines, the inspection sample is expanded based on engineering judgment. In addition analytical models are used to predict E/C in piping systems based on specific plant data including material and hydrodynamic and operating conditions. The CHECWORKS Code used for predicting wall thinning. ***(4) Detection of Aging Effects:*** Aging degradation of piping and fittings occurs by wall thinning; extent and schedule of inspection assure detection of wall thinning before the loss of intended function of the piping. ***(5) Monitoring and Trending:*** Inspection schedule of NUREG-1344 and EPRI guidelines should provide for timely detection of leakage. Inspections and analytical evaluations are performed during plant outage. If analysis shows unacceptable conditions, inspection of initial sample is performed within 6 months. ***(6) Acceptance Criteria:*** Inspection results are used to calculate number of refueling or operating cycles remaining before the component reaches Code minimum allowable wall thickness. If calculations indicate that an area will reach Code minimum (plus 10% margin), the component must be repaired or replaced. However, NRC staff has identified the problems in implementing E/C program that pertain to weakness or errors in (a) using predictive models, (b) calculating minimum wall thickness acceptance criteria, (c) analyzing the results of UT examinations, and (d) assessment of E/C program activities (NRC Information Notice IN 93-21). ***(7) Corrective Actions:*** Prior to service, repair or replace to meet the requirements of NUREG-1344. Follow-up inspections are performed to confirm or quantify thinning and take longer term corrective actions such as adjustment of chemistry and operating parameters, or selection of materials resistant to E/C. However, NRC staff has identified weakness or errors in (a) dispositioning components after reviewing the results of the inspection analysis, and (b) repairing or replacing components that failed to meet the acceptance criteria (IN 93-21). ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are	

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D2.2.1 thru D2.2.3	Pumps HPCS or HPCI Main & Booster, LPCS, LPCI or RHR, & RCIC	Bowl/Casing, Suction Head, Discharge Head	CS Casting, CS	25–288°C, Oxygenated Water	Loss of Material	General, Crevice, and Pitting Corrosion	ASME Section XI, 1989 Edition. ASME OM Code–1990, Subsection ISTB. NRC GL 89-04. EPRI TR-103515. Plant Technical Specifications.

Existing 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:*** Wall-thinning problems in two-phase piping have occurred in extraction steam lines (INs 89-53, 97-84) and moisture separation reheater and feedwater heater drains (INs 89-53, 91-18, 93-21, 97-84).	
Water chemistry program based on EPRI guidelines of TR-103515 and implemented through the plant technical specifications for minimizing impurities by monitoring and maintaining water chemistry conditions; inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWC 2500-1, examination category C-H for pressure retaining Class 2 components; and based on the testing requirements of 10 CFR 50.55a for ASME Code Class 2 pumps, and additional NRC staff guidelines of NRC Generic Letter 89-04, inservice testing performed in accordance with ASME Subsection IWP (or Operation and Maintenance Code Subsection ISTB) for pumps, or other approved program in the plant specifications.	***(1) Scope of Program:*** The program relies on preventive measures to mitigate general, crevice, and pitting corrosion and combination of inservice inspection (ISI) and inservice testing (IST) to monitor the effects of corrosion on the intended function of emergency core cooling system components. ***(2) Preventive Actions:*** Mitigation is by monitoring and control of water chemistry to minimize concentration of corrosive impurities in accordance with EPRI guidelines of TR-103515 and implemented through the plant technical specifications. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of corrosion by ISI to detect coolant leakage and IST to evaluate component performance. Inspection requirements of ASME Section XI specify visual VT-2 (IWA-5240) examination during system leakage test and hydrostatic test of all pressure retaining Class 2 components according to Table IWC 2500-1 category C-H. Based on the requirements of 10 CFR 50.55a for ASME Code Class 2 pumps and additional guidelines of NRC Generic Letter (GL) 89-04, IST is performed in accordance with ASME Subsection IWP (or OM Code Subsection ISTB). ***(4) Detection of Aging Effects:*** Degradation of the component due to corrosion would result in leakage of coolant or degradation of component performance. However, a one-time inspection of representative sample of the system population and most susceptible locations in the system should be conducted to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period. Inspection is performed in accordance with the requirements of ASME Code, 10CFR50 Appendix B, and ASTM standards, using a variety of nondestructive techniques including visual, ultrasonic, and surface techniques. ***(5) Monitoring and Trending:*** ISI/IST schedule of ASME Section XI should provide for timely detection of corrosion. 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 and hydrostatic tests are evaluated in accordance with IWC-3100 and acceptance standards of IWC-3400 and IWB-3516 for Class 2 components. Any evidence of aging effects or unacceptable results are evaluated. ***(7) Corrective Actions:*** Prior to service,	Yes, Element 4 should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D2.3.1	Valves (Check, Control, Hand, Motor Operated, & Relief Valves)	Body and Bonnet	CS forging, CS casting	25–288°C, Oxygenated Water	Wall Thinning	Erosion/ Corrosion	NUREG-1344. EPRI NSAC-202L-R2. NRC GL 89-08 NRC IN 93-21. EPRI TR-103515. BWRVIP-29.
D2.3.1	Valves (Check, Control, Hand, Motor Operated, & Relief Valves)	Body and Bonnet	CS forging, CS casting	25–288°C, Oxygenated Water	Loss of Material	General, Crevice, and Pitting Corrosion	ASME Section XI, 1989 Edition. ASME OM Code-1990, Appendix I and Subsection ISTC. EPRI TR-103515. NRC GL 89-04. NRC GL 90-06. NRC GL 96-05. NRC IN 88-70. NRC IN 98-24, Plant Technical Specifications.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) corrective measures are needed to meet the requirements of IWB-3142 and IWA-5250. 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:*** Localized corrosion is likely to occur at flange connections and other crevices where buildup of impurities can occur. No significant corrosion related problem has been reported for pumps in BWR emergency core cooling system.	
Same as for Erosion/Corrosion of Item D2.1.9 lines to HPCI & RCIC pump turbine and D2.1.10 lines from HPCI & RCIC pump turbine to condenser.	*Same as for Erosion/Corrosion of Item D2.1.9 lines to HPCI & RCIC pump turbine and D2.1.10 lines from HPCI & RCIC pump turbine to condenser.*	Yes, Element 1 should be further evaluated
Water chemistry program based on EPRI guidelines of TR-103515 and implemented through the plant technical specifications for minimizing impurities by monitoring and maintaining water chemistry conditions; inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWC 2500-1, examination category C-H for pressure retaining Class 2 components; and based on the testing requirements of 10 CFR 50.55a for ASME Code Class 2 valves, staff guidelines of NRC Generic Letter (GL) 89-04 regarding the scope of inservice testing (IST), and information in NRC IN 88-70 regarding scope and testing of safety-related check valves, and in GL 96-05 regarding safety-related motor-operated valves, IST is performed in accordance with ASME Subsection IWV (Operation and Maintenance Code Appendix I and Subsection ISTC), to ensure that the changes in design-basis performance of safety-related valves resulting from degradation can be identified and managed.	***(1) Scope of Program:*** The program relies on preventive measures to mitigate crevice or pitting corrosion and combination of inservice inspection (ISI) and inservice testing (IST) to monitor the effects of corrosion on the intended function of emergency core cooling system components. ***(2) Preventive Actions:*** Mitigation is by monitoring and control of water chemistry to minimize concentration of corrosive impurities by following EPRI guidelines of TR-103515 and implemented through the plant technical specifications. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of corrosion by ISI to detect coolant leakage and IST to evaluate component performance. Inspection requirements of ASME Section XI specify visual VT-2 (IWA-5240) examination during system leakage test and hydrostatic test of all pressure retaining Class 2 components, according to Table IWC 2500-1 category C-H. Based on the requirements of 10 CFR 50.55a for ASME Code Class 2 valves and additional guidelines of NRC GLs 89-04 and 96-05, IST is performed in accordance with ASME Subsection IWV (OM Code Appendix I and Subsection ISTC). ***(4) Detection of Aging Effects:*** Degradation of the component due to corrosion would result in leakage of coolant or degradation of component performance. However, a one-time inspection of representative sample of the system population and most susceptible locations in the system should be conducted to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period. Inspection is performed in accordance with the requirements of ASME Code, 10CFR50 Appendix B, and ASTM standards, using a variety of nondestructive techniques including visual, ultrasonic, and surface techniques. ***(5) Monitoring and Trending:*** ISI/IST	Yes, Element 4 should be further evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D2.3.1	Valves (Check, Control, Hand, Motor Operated, & Relief Valves)	Body and Bonnet	SS forging, SS casting	25–288°C, Oxygenated Water	Crack Initiation and Growth	SCC	NUREG-0313, Rev. 2. NRC GL 88-01. ASME Section XI, 1989 Edition. BWRVIP-29. EPRI TR-103515.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) schedule of ASME Section XI should provide for timely detection of corrosion. 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 and hydrostatic tests are evaluated in accordance with IWC-3100 and acceptance standards of IWC-3400 and IWC-3516 for Class 2 components. Any evidence of aging effects or unacceptable results are evaluated. ***(7) Corrective Actions:*** Prior to service, corrective measures are needed to meet the requirements of IWB-3142 and IWA-5250. Repair is in conformance with IWA-4000 and replacement according to 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:*** Localized corrosion is likely to occur at crevice geometry where buildup of impurities can occur. Failure of steam turbine governor valves due to corrosion scale buildup has occurred in RCIC and AFW systems (NRC IN 98-24).	
Program delineated in NUREG-0313, Rev. 2 and implemented through NRC Generic letter (GL) 88-01, and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWC 2500-1, examination category C-G for pressure retaining welds in Class 2 valves, and category C-H for system leakage. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-103515 and BWRVIP-29 to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes implementing counter measures to mitigate stress corrosion cracking (SCC) of stainless steel (SS) and combination of inservice inspection (ISI) to monitor SCC and its effects on the intended function of valves. NUREG-0313 and GL 88-01, respectively, describe the technical basis and staff guidance regarding mitigating IGSCC in BWRs. ***(2) Preventive Actions:*** 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. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors SCC of valves by detection and sizing of cracks by implementing the inspection schedule, methods, personnel, sample expansion, and leak detection requirements of GL 88-01. In a group of multiple valves of similar design, size, function, and service in a system, examination of only one valve is required. Coolant water chemistry is monitored and maintained in accordance with the EPRI guidelines in BWRVIP-29 to minimize the potential of crack initiation and growth. ***(4) Detection of Aging Effects:*** Degradation of valves due to SCC can not occur without crack initiation and growth; ISI schedule delineated in the AMP is adequate and will assure detection of cracks or degradation of valve performance before the loss of intended function of 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	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D2.4.1 thru D2.4.4	Heat Exchangers (RHR & LPCI) (Serviced by Closed Cycle Cooling Water)	Tubes, Tubesheet, Channel & Head, Shell	Carbon Steel (CS)	Oxygenated Water on one side; Closed Cycle Cooling Water (Treated Water) on the other side	Loss of Material	Crevice and Pitting Corrosion	ASME Section XI, 1989 Edition. ASME OM S/G, Pt 2. NRC GL 89-13. Plant Technical Specifications.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) and performing similar functions in the system. Visual examination is performed only when the valve is disassembled for maintenance, repair, or volumetric examination, but at least once during the period. ***(6) Acceptance Criteria:*** Any SCC degradation is evaluated in accordance with IWC-3100 by comparing ISI results with the acceptance standards of IWC-3400 and IWC-3515. ***(7) Corrective Actions:*** Repair is in conformance with IWA-4000 and replacement is in 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:*** The comprehensive AMP outlined in GL 88-01 addresses improvements in all three elements that cause SCC, e.g., a susceptible material, significant tensile stress, and an aggressive environment, and has provided effective means of ensuring structural integrity of BWR components.	
Detection of reactor coolant leakage by radiation and temperature monitors in component cooling system; inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWC 2500-1, examination category C-H for pressure retaining Class 2 components; performance testing in accordance with ASME OM Standards and Guides, Part 2 to ensure that the heat exchanger serviced by the closed-cycle cooling water system is performing its function acceptably; and control of system water chemistries in suppression pool and component cooling water based on the plant technical specifications to minimize exposure to aggressive environments.	***(1) Scope of Program:*** The program includes monitoring and control of ECCS water chemistry to minimize exposure to aggressive environments, and performance testing in accordance with ASME OM-Standards and Guides, Part 2 provides assurance that the heat exchanger serviced by the closed-cycle cooling water system is performing its function acceptably. ***(2) Preventive Actions:*** Monitor and control of suppression pool and component cooling system water chemistries based on the plant technical specifications to minimize impurities, and timely corrective action prevent or mitigate corrosion. The parameters monitored include halogens, sulfates, oxygen, and pH in the suppression pool water, and in addition to these, dissolved copper and iron, suspended solids, and gamma activity in the component cooling water. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of corrosion by surveillance program to detect coolant leakage and inservice testing to evaluate component performance. The parameters monitored are directly related to corrosion, e.g., dissolved iron and copper, and by detection of leakage by radiation and temperature monitors, in the component cooling system. Also, inspection requirements of ASME Section XI specify visual VT-2 (IWA-5240) examination during system leakage test and hydrostatic test of all pressure retaining Class 2 components according to Table IWC 2500-1 category C-H. Performance testing is conducted in accordance with ASME OM S/G, Part 2. ***(4) Detection of Aging Effects:*** Degradation of component due to corrosion would result in leakage of coolant or degradation of component performance. Monitoring of radiation and suspended solids would detect leakage; extent and schedule of	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D2.4.1 thru D2.4.4	Heat Exchangers (RHR & LPCI) (Serviced by Open Cycle Cooling Water)	Tubes, Tubesheet, Channel & Head, Shell	Carbon Steel (CS), Stainless Steel (SS)	Oxygenated Water on one side; Open Cycle Cooling Water (Raw Water) on the other side	Loss of Material	General Corrosion and Microbio-logically Influenced Corrosion (MIC)	NRC GL 89-13. NRC IN 81-21. NRC IN 85-24. NRC IN 85-30. NRC IN 86-96. NRC IN 94-03. Plant Technical Specification.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) inspection/testing assure detection of corrosion before the loss of intended function of the component. ***(5) Monitoring and Trending:*** Results from performance tests to verify the heat transfer capabilities are trended. Also, based on the recommendations of NRC GL 89-13 or its equivalent, if adequacy of cooling water chemistry control can not be confirmed, implement Action III of GL 89-13 to include inspection and maintenance program for closed-cycle cooling water system to ensure that corrosion, erosion, and protective coating failure can not degrade the performance of safety-related systems serviced by closed-cycle cooling water. ***(6) Acceptance Criteria:*** Heat exchanger performance test results are evaluated in accordance with the guidelines of ASME OM S/G Part 2. Any relevant conditions related to corrosion causing leakage of ECCS water are compared with established acceptable limits. Results of Section XI leakage tests are evaluated in accordance with IWC-3100 and acceptance standards of IWC-3400 and IWB-3516. ***(7) Corrective Actions:*** Root cause evaluation and appropriate corrective action taken when acceptable limits are exceeded or leakage is detected. Repair is in conformance with IWA-4000 and replacement is in 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:*** Operating plant experience with this AMP indicate timely detection of corrosion in ECCS heat exchangers.	
Implementation of the recommendations of Generic Letter 89-13 or an equally effective program to ensure that open-cycle cooling water system is in compliance with General Design Criteria and Quality Assurance requirements. Mitigation is by control of ECCS water chemistry and the use of appropriate materials and lining/coating to protect the underlying metal surfaces from being exposed to environments causing MIC and biofouling.	***(1) Scope of Program:*** The program includes monitoring and control of water chemistry to minimize exposure to environments causing MIC and biofouling, and staff recommendations of Generic Letter (GL) 89-13 or an equivalent program provide assurance that the service cooling water system is in compliance with General Design Criteria and Quality Assurance requirements. Guidelines of GL 89-13 include (a) surveillance and control of biofouling, (b) test program to verify heat transfer capabilities, (c) routine inspection and maintenance program to ensure that corrosion, erosion, protective coating failure, silting, and biofouling, can not degrade the performance of safety-related systems serviced by the service water, (d) system walkdown inspection to ensure compliance with licensing basis, and (e) review of maintenance, operating, and training practices and procedures. ***(2) Preventive Actions:*** The component is constructed of appropriate materials, control of water chemistry is based on EPRI guidelines of TR-103515, and lining or coating protect the underlying metal surfaces from being exposed to aggressive cooling water environment. Based on GL 89-13 cooling water	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D2.4.1 thru D2.4.4	Heat Exchangers (RHR & LPCI) (Serviced by Open Cycle Cooling Water)	Tubes, Tubesheet, Channel & Head, Shell	Carbon Steel (CS), Stainless Steel (SS)	Oxygenated Water on one side; Open Cycle Cooling Water (Raw Water) on the other side	Buildup of Deposit	Biofouling	*Same as for General Corrosion and MIC of Items D2.4.1 thru D2.4 RHR and LPCI heat exchanger components*
D2.5.1 thru D2.5.4	Containment Spray Header and Spray Nozzles System	Piping and Fittings, Flow Orifice, Headers, Spray Nozzles	CS, SS	Air	Loss of Material	General, Pitting, and Crevice Corrosion	Plant Technical Specifications.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) system is continuously chlorinated or treated with biocide whenever the potential for biological fouling species exists. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of corrosion by surveillance program to detect coolant leakage and inservice testing to evaluate component performance. Based on recommendations of GL 89-13 or its equivalent, cooling water system is inspected for biofouling organisms, sediment, protective coating failure, and corrosion; and cooling water flow and temperature are monitored for component performance evaluation to ensure that flow blockage or excessive fouling accumulation does not exist. ***(4) Detection of Aging Effects:*** Degradation of component due to corrosion would result in leakage of coolant or degradation of component performance; extent and schedule of inspection/testing assure detection of corrosion before the loss of intended function of the component. ***(5) Monitoring and Trending:*** Results from performance tests to verify heat transfer capabilities are trended. ***(6) Acceptance Criteria:*** Any relevant conditions related to corrosion or leakage are compared to established acceptable limits. Maximum levels for various impurities in secondary side water and cooling water are specified. ***(7) Corrective Actions:*** If the heat exchanger fails to perform adequately, corrective actions are taken. Root cause evaluation is performed when acceptable limits are exceeded or leakage is detected. ***(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:*** Significant microbiologically influenced corrosion [NRC Information Notice (IN) 85-30], failure of protective coatings (IN 85-24), and fouling (IN 81-21, 86-96) has been observed in a number of heat exchangers. Although the AMP provides an effective means to manage the effects of corrosion on the intended function of ECCS heat exchangers, results of service water system operational performance inspections (IN 94-03) indicate that deficiencies still exist in implementation of GL 89-13.	
Same as for General Corrosion and MIC of Items D2.4.1 thru D2.4 RHR and LPCI heat exchanger components	*Same as for General Corrosion and MIC of Items D2.4.1 thru D2.4 RHR and LPCI heat exchanger components*	No
Plant specific aging management program.	Plant specific aging management program is to be evaluated.	Yes, no AMP

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D2.6.1 thru D2.6.4	Isolation Condenser	Tubing, Tubesheet, Channel Head, Shell	Tubes: SS; Tubesheet: CS; Channel Head: CS; Shell: CS	Tube side: Steam at 288°C; Shell side: deminera-lized water at 288°C	Loss of Material	Crevice and Pitting Corrosion	ASME Section XI, 1989 Edition.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Same as effect of Crevice and Pitting Corrosion of Items D2.4.1 thru D2.4.4 RHR and LPCI heat exchanger components	*Same as effect of Crevice and Pitting Corrosion of Items D2.4.1 thru D2.4.4 RHR and LPCI heat exchanger components*	No

E Fan Cooler System

E.1 Fan Coolers

E.1.1 Cooling Coils

E.1.2 Fan Housing

E.1.3 Blades

E.1.4 Fasteners

E.1.5 Piping

E.1.6 Fittings

E. Fan Cooler System

System, Structures, and Components

The system, structures, and components included in this table comprise the fan cooler system found in pressurized water reactors (PWR) and consist of cooling coils; fan housing, blades and fasteners; and piping and fittings. 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," all components in the fan cooler system are classified as Group B Quality Standards.

System Interfaces

The systems that interface with the fan cooler system include the containment spray system (Table V A) and the containment isolation components (Table V C).

V ENGINEERED SAFETY FEATURES

E. FAN COOLER SYSTEM

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
E.1.1	Fan Coolers	Cooling coils	Cu-Ni alloy (typically 90% Cu-10% Ni)	Treated water from service water system	Loss of Material	General Corrosion	NRC IN 80-37
E.1.2 E.1.3 E.1.4	Fan Coolers	Fan housing, blades, and fasteners	Carbon Steel	Air, possible steam conden-sation	Cumulative fatigue damage	Fatigue	ASME Section III, 1969 Edition,
E.1.2 E.1.3 E.1.4	Fan Coolers	Fan housing, blades, and fasteners	Carbon Steel	Air, possible steam conden-sation	Loss of material	General Corrosion	
E.1.5 E.1.6	Fan Coolers	Piping and fittings	Carbon and Stainless Steel	Treated water from service water system	Loss of Material	General Corrosion	

V ENGINEERED SAFETY FEATURES
E. FAN COOLER SYSTEM

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Plant-specific aging management program.	Plant-specific aging management program will be evaluated. One incident of containment cooler heat exchanger leakage caused by corrosion has been reported (NRC IN 80-37).	Yes. No generic AMP
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal.	Yes. TLAA
Plant-specific aging management program, typically including periodic visual inspection	Plant-specific aging management program to be evaluated.	Yes. No generic AMP
Plant-specific aging management program.	Plant-specific aging management program will be evaluated.	Yes. No generic AMP

References

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ASME OM Code, Appendix I, *Inservice Testing of Pressure Relief Devices in Light-Water Reactor Power Plants*, Code for Operation and Maintenance of Nuclear Power Plants, The American Society of Mechanical Engineers, New York, NY, 1990.

ASME OM Code, Subsection ISTB, *Inservice Testing of Pumps in Light-Water Reactor Power Plants*, Code for Operation and Maintenance of Nuclear Power Plants, The American Society of Mechanical Engineers, New York, NY, 1990.

ASME OM Code, Subsection ISTC, *Inservice Testing of Valves in Light-Water Reactor Power Plants*, Code for Operation and Maintenance of Nuclear Power Plants, The American Society of Mechanical Engineers, New York, NY, 1990.

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IE Bulletin No. 82-02, *Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants*, June 2, 1982.

NRC Bulletin 89-02, *Stress Corrosion Cracking of High-Hardness Type 410 Stainless Steel Internal Preloaded Bolting in Anchor Darling Model S350W Swing Check Valves or Valves of Similar Design*, July 19, 1989.

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CHAPTER VI

ELECTRICAL COMPONENTS

Major Electrical Components

A. Electric Cables

B. Electrical Connectors

C. Electrical Penetration Assemblies

D. Electrical Buses

E. Electrical Insulators

F. Transmission Conductors

G. Ground Conductors

A. Electric Cables

A.1 Power, Instrumentation and Control Cables

- A.1.1 Conductor
- A.1.2 Shield Wire
- A.1.3 Insulation
- A.1.4 Jacket

Systems, Structures and Components

This review table addresses electric cables, including power, instrumentation and control (I&C) cables. The power cables addressed are low-voltage (< 1000 V) and medium voltage (2 kV to 15 kV), which have similar constructions to I&C cables. High voltage power cables (>15 kV) have unique, specialized construction and must be evaluated on an application specific basis. Since the cable types addressed herein are very similar in construction and aging effects, they are grouped together in the table. Individual sub-components for a typical cable are addressed in terms of aging mechanisms and effects.

System Interfaces

Electric cables functionally interface with all plant systems that rely on electric power and/or instrumentation and control. Physical interfaces include routing in cable trays and conduits.

VI. ELECTRICAL COMPONENTS
A. Electric Cables

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A.1.1	Power, Control, & Instrument Cables	Conductor	Copper • coated or non-coated • stranded or solid	Humid, Chemical Exposure	Increased circuit resistance, heating, signal noise, circuit failure	Corrosion	IE Bulletin 79-1B (DOR Guideline) NUREG-0588 IEEE Standards • 323-1971 • 323-1974 • 383-1974 • 317-1976 • 338-1987 • 1205-1993 Regulatory Guide 1.89, Rev. 1 10CFR50.49 EQ Rule

VI. ELECTRICAL COMPONENTS
A. Electric Cables

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
A. Environmentally Qualified Equipment For electrical equipment that is environmentally qualified for use in nuclear power plants, the environmental qualification program may be applicable as a tool for aging management. **Environmental Qualification (10CFR50.49: EQ Rule)** EQ requirements have evolved over the years; therefore, plants of various vintages are licensed based on different EQ requirements. There are three main documents that chronicle the EQ requirements, starting with the IE Bulletin 79-01B (DOR guidelines) issued in 1979. This was followed by NUREG-0588, which specifies two categories of qualifications, and finally the current EQ Rule (10 CFR 50.49). The DOR Guidelines and NUREG-0588 Category II are consistent with the original IEEE Standard for qualifying Class 1E equipment (IEEE Std 323-1971), while NUREG-0588 Category I and 10 CFR 50.49 endorse a later version of the standard (IEEE Std 323-1974). IEEE Standard 323-1974 includes more stringent requirements than the 1971 version, including the application of margins to test parameters and pre-aging of equipment prior to accident testing. It should be noted that the NRC has not endorsed a later version of the standard (IEEE Std 323-1983). While many of the older vintage plants were licensed based on the DOR Guidelines/NUREG-0588, Category II, many of the electric cables inside containment (over 70%) included pre-aging as part of their original qualification, or have been re-qualified to Category I criteria. Many older plants still utilize cable connections and electrical penetrations that were environmentally qualified in accordance with the DOR Guidelines and/or the NUREG-0588, Category II requirements. The original qualification of many of these components might not have included pre-aging prior to exposing them to accident conditions.	**A. Environmentally Qualified Equipment** In general, the EQ process accounts for aging through the use of a Time Limited Aging Analysis (TLAA) for the equipment to be qualified. It does not require the use of prevention or mitigation measures, or the use of condition/performance monitoring. Therefore, EQ cannot be considered a typical aging management program. However, the TLAA does provide some assurance that the effects of aging will not be problematic during the qualified life of the equipment. As such, EQ can be considered part of an aging management program for license renewal if the licensee can show i) the TLAA remains valid for the period of extended operation, ii) the TLAA is projected to the end of the period of extended operation through re-analysis, or iii) the effects of aging on the intended function(s) will be adequately managed during the period of extended operation. For case (i), the existing qualification is acceptable for extended life and no further evaluation is necessary. For case (ii), a re-analysis is necessary to extend the qualified life of the equipment. In the re-analysis, attributes that should be addressed include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria, corrective actions if acceptance criteria are not met, and the period of time prior to the end of qualified life when the re-analyses will be completed In light of case (iii), the EQ process was evaluated as an aging management program based on the 10 criteria identified in the draft SRP-LR. The following summarize this evaluation: ***(1) Scope of Program:*** The EQ requirements apply to electric equipment important to safety, which includes those electrical components within the scope of license renewal (i.e., cables, connectors, and penetration assemblies). ***(2) Preventive Actions:*** EQ does not require the use of preventive actions to manage the effects of aging. Aging is addressed through the use of a TLAA. As such, the EQ process identifies no preventive actions. ***(3) Parameter Monitored/Inspected:*** EQ is not a condition or performance monitoring program. As such, it does not identify any parameters to be monitored to manage the effects of aging. Aging is addressed through the use of a TLAA. ***(4) Detection of Aging Effects:*** In general, EQ does not require the detection of aging effects for equipment while in service. When the qualified life is less than the current plant license period, EQ requires a program to replace or refurbish the component at the end of its qualified life. ***(5) Monitoring and Trending:*** EQ does not rely on monitoring and trending of condition or performance parameters of equipment while in service to manage the effects of aging. As such, no monitoring or trending activities for assessing	**A. Environmentally Qualified Equipment** Yes. In the case where the TLAA is projected to the end of the period of extended operation, the analysis attributes identified should be addressed

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A. Electric Cables

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

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A. Electric Cables

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	the impact on equipment condition due to aging are identified by the EQ process. It should also be noted that currently, there are no recognized in situ condition monitoring methods that are effective for monitoring the condition of electric cables. Research is ongoing to determine if acceptable methods exist. ***(6) Acceptance Criteria:*** EQ does not rely on monitoring and trending of condition or performance parameters to manage the effects of aging. As such, no acceptance criteria are established for equipment operation while in service. ***(7) Corrective Actions:*** As part of the EQ process, a qualified life is established for the equipment being qualified. Once the equipment reaches the end of its qualified life, the only acceptable corrective action is refurbishment or replacement. ***(8 & 9) Confirmation process and Administrative Controls:*** EQ does not rely on preventive or corrective actions to address the effects of aging. As such, the EQ process identifies no confirmation process. EQ documentation for each qualified component is maintained at the plant site in an auditable form for the duration of the installed life of the equipment. ***(10) Operating Experience:*** Passive electrical components are typically reliable devices under normal plant conditions and have very little evidence of significant failures. In a study performed by Sandia (SAND96-0344, 9/96), a database of nuclear plant component failure records was reviewed to identify relative number of failures, as well as typical failure modes and causes for electrical cables and terminations. The review covered data for the time period from 1975 to 1994, and generated 1,458 reports applicable to low and medium voltage cables and terminations. An analysis of these records showed the following: - In general, these components have good reliability. However, aging degradation does occur and has led to failures. - For low-voltage components, connectors accounted for the highest percentage of failures (30%). Cables (14.5%), terminal blocks (3.5%) and splices (2.5%) had relatively fewer failures. - For medium voltage components cables had the highest percentage of failures (69%), followed by connectors (11%) and splices (17%). - Most of the failures are detected by operation of the component; relatively few are detected by maintenance or surveillance. Another EPRI study on low-voltage environmentally qualified cables presented in an industry report (EPRI TR-103841, 6/94) analyzed Licensee Event Reports for the period from 1968 to June 1992. Only 87 LERs related to cables were considered attributable to aging and the failures were categorized as follows: thermal degradation (13 reports), mechanical damage (23 reports), misapplication (11 reports), and unknown (40 reports). Roughly half of these failures occurred in the first 6 years of operation, and the number of failures decreased significantly after 10 years of operation.	

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A. Electric Cables

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

VI. ELECTRICAL COMPONENTS
A. Electric Cables

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	NRC's aging assessment on cables, connections, and electrical penetration assemblies analyzed LER/NPE data for the period from mid-1980 to 1988 (NUREG/CR-5461, 6/90). An analysis of these failure data showed the following: • Out of 151 reported events on cables, more than 70% involved some type of electrical failure, either shorting, open circuit, or grounding faults. • Out of 196 reported events on connections, almost 80% involved shorted, grounded, loose, or open connections. • Out of 39 reported events on EPAs, pressure leakage (41%) and electrical failure (26%) caused the most events. Based on the results presented by these studies, it is seen that qualified electrical equipment does have good reliability, and aging degradation is usually well managed. These components receive little or no preventative maintenance. Under accident conditions, however, the reliability of these components is relatively unknown. Many of the causes of failures in accident conditions would not be detected during normal operation because of the absence of high temperatures and humidity. Note that not all degradation is detected and mitigated before it results in failure. Therefore, additional aging management practices are needed to completely manage the effects of aging for these electrical components. As discussed in SECY-93-049, during the staff's review of license renewal issues, the EQ process was found to be a significant issue. Of particular concern was whether the EQ requirements for older plants (i.e., DOR guidelines, NUREG-0588 Cat. II), whose licensing bases differ from newer plants, are adequate for license renewal. Further, a question was raised as to whether the EQ requirements for older plants should be reassessed for the current licensing term. Upon subsequent review, additional concerns were raised related to the EQ process, and it was concluded that differences in EQ requirements constituted a potential generic issue that should be evaluated for backfit, independent of license renewal. This came to be identified as Generic Issue 168. Key items to be addressed in GSI-168 are: • The adequacy of older EQ requirements for license renewal, as well as for the current licensing term • The adequacy of accelerated aging techniques to simulate long-term natural service aging • The possibility that unique failure mechanisms exist for bonded jacket and multi-conductor cable configurations that are not adequately addressed in EQ • The feasibility of using condition monitoring (CM) techniques to monitor current cable condition in situ as a means of offsetting uncertainties in the process used to predict long-term service aging Presently, GSI-168 is an open generic issue related to license renewal, and research is ongoing to provide information to resolve it. Specific issues being addressed in this research are presented in	

VI. ELECTRICAL COMPONENTS
A. Electric Cables

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	NUREG/CR-6384. Once this generic issue is resolved, guidance will be provided as to the impact on license renewal. In the interim, NRC letter dated June 2, 1998 "Guidance on Addressing GSI-168 for License Renewal," (C. Grimes, NRC to D. Walters, NEI) provides guidance on addressing GSI-168 in license renewal applications. It states that, until the generic issue is resolved, "*...an acceptable approach described in the SOC is to provide a technical rationale demonstrating that the current licensing basis for EQ, pursuant to 10 CFR 50.49 will be maintained in the period of extended operation.*" It should be noted that, currently, there are no acceptable non-destructive CM techniques to measure the integrity of electric cables in situ. It does not appear that utilities can take credit for current functional testing of cables by periodic system or circuit testing as a means of satisfying the criteria for an item to be considered a replacement item. The effectiveness of several promising CM techniques for monitoring degradation of cables is the subject of an ongoing NRC research program. The results of this program will be part of the resolution of GSI-168.	
B. <u>Non-environmentally Qualified Equipment</u> In many applications, electrical equipment may not be environmentally qualified, and other aging management programs may be applicable. The following are examples. **Aging Inspection Program** For those electrical components that are accessible, a visual inspection can be used to provide some indication of aging degradation. The visual inspection can check for surface anomalies, such as discoloration, cracking or surface contamination that would indicate the presence of active aging degradation. For cables, if the jacket or insulation can be touched, a qualitative indication of material hardening can be made. Observation of aging degradation would indicate the need for further investigation of the component. **Instrument Calibration Program** Instrument calibration programs, including technical specification surveillance, may be used to provide an indirect indication of the condition of various electrical components. If calibration drift is noted for instruments, this could be an indication that aging degradation is affecting the electrical circuit. Further investigation could then be initiated to determine the nature of the degradation and the component affected.	**B. <u>Non-environmentally Qualified Equipment</u>** The aging management programs discussed are generic in nature and should be developed based on specific plant applications. These programs will be evaluated on a plant specific basis.	**B. <u>Non-environmentally Qualified Equipment</u>** Yes. A plant specific evaluation is required.

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A. Electric Cables

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A.1.2	Power, Control, & Instrument Cables	Shield Wires	Braided copper, Aluminum Foil, Metallized mylar tape	Humid, Chemical Exposure	Signal noise or error in control and instrumnt. cable	Corrosion	*Same as effect of corrosion on conductor for cables (A.1.1).*
A.1.3	Power, Control, & Instrument Cables	Insulation	Polymers such as XLPE, EPR, SR	Humid, High voltage gradient (Power Cable)	Loss of dielectric strength, signal noise/ error, leakage current	Moisture diffusion/ absorption; Formation of water trees in power cables	*Same as effect of corrosion on conductor for cables (A.1.1).*

VI. ELECTRICAL COMPONENTS
A. Electric Cables

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
A. Environmentally Qualified Equipment *Same as effect of corrosion on conductor for cables (A.1.1).*	**A. Environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*	**A. Environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*
B. Non-environmentally Qualified Equipment *Same as effect of corrosion on conductor for cables (A.1.1).*	**B. Non-environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).* Note: The most probable location for shield wire corrosion is at exposed sites, such as terminations on equipment or terminal strips	**B. Non-environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*
A. Environmentally Qualified Equipment *Same as effect of corrosion on conductor for cables (A.1.1).*	**A. Environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*	**A. Environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*
B. Non-environmentally Qualified Equipment *Same as effect of corrosion on conductor for cables (A.1.1)*	**B. Non-environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).* Note: Underwater cables or cables with prolonged exposure to humid environment, should be specifically designed for such applications.	**B. Non-environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*

VI. ELECTRICAL COMPONENTS
A. Electric Cables

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A.1.3	Power, Control, & Instrument Cables	Insulation	Polymers such as XLPE, EPR, SR	High temp., Radiation, Oxygen, and Internal Ohmic heating (Power Cables)	Loss of dielectric strength, leakage current, signal noise/ error, circuit failure	Hardening, Cracking	*Same as effect of corrosion on conductor for cables (A.1.1).*

VI. ELECTRICAL COMPONENTS
A. Electric Cables

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
A. Environmentally Qualified Equipment *Same as effect of corrosion on conductor for cables (A.1.1).*	**A. Environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*	**A. Environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*
B. Non-environmentally Qualified Equipment *Same as effect of corrosion on conductor for cables (A.1.1).*	**B. Non-environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).* Note: Some applications use different insulation materials, such as mineral insulation and polyimides (e.g., Kapton) which may be susceptible to different aging mechanisms. Cracking can be initiated in a an embrittled cable by any movement of the cable, such as a seismic event, maintenance activities, or vibration from nearby operating equipment. While embrittlement and cracking of cable insulation may not affect cable performance under normal, dry conditions, the aging effects noted would be probable when cables with cracks are exposed to moisture, such as in a design basis event. Moisture intrusion through the cracks could lead to shorting and possible circuit failure.	**B. Non-environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*

VI. ELECTRICAL COMPONENTS
A. Electric Cables

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A.1.4	Power, Control, & Instrument Cables	Jacket	Polymers such as Neoprene, CSPE, PVC	High temp., Radiation, Oxygen	Loss of mechanical and environ-mental protection to underlying insulation. Exposure of insulation to outside conditions.	Hardening, Cracking	*Same as effect of corrosion on conductor for cables (A.1.1).*

VI. ELECTRICAL COMPONENTS
A. Electric Cables

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
A. Environmentally Qualified Equipment *Same as effect of corrosion on conductor for cables (A.1.1).*	**A. Environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*	**A. Environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*
B. Non-environmentally Qualified Equipment *Same as effect of corrosion on conductor for cables (A.1.1).*	**B. Non-environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).* Note: Jackets provide some degree of protection to underlying insulation from exposure to outside stressors, such as radiation, oxygen, moisture, dirt, dust and other contaminants. For bonded jacket cables, in which the jacket is bonded to the insulation, cracking in the jacket has been found to propagate through to the insulation in some cases.	**B. Non-environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*

VI. ELECTRICAL COMPONENTS
A. Electric Cables

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A.1.4	Power, Control, & Instrument Cables	Jacket	Polymers such as Neoprene, CSPE, PVC	High temp., Radiation, Oxygen	Loss of fire protection	Loss of fire retardant	*Same as effect of corrosion on conductor for cables (A.1.1).*

VI. ELECTRICAL COMPONENTS
A. Electric Cables

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
A. Environmentally Qualified Equipment *Same as effect of corrosion on conductor for cables (A.1.1).*	**A. Environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*	**A. Environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*
B. Non-environmentally Qualified Equipment *Same as effect of corrosion on conductor for cables (A.1.1).*	**B. Non-environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).* Note: The primary purpose of the jacket is to protect the insulated conductors from fire and environmental stressors. No known condition monitoring method is available to determine the amount of fire retardant lost with the age of the jacket material.	**B. Non-environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*

VI. ELECTRICAL COMPONENTS
A. Electric Cables

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A.1.4	Power, Control, & Instrument Cables	Jacket	Polymers such as Neoprene, CSPE, PVC	Vibration, maint-enance abuse	Exposure of insulation to outside conditions	Wear and tear	*Same as effect of corrosion on conductor for cables (A.1.1).*

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A. Electric Cables

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
A. Environmentally Qualified Equipment *Same as effect of corrosion on conductor for cables (A.1.1).*	**A. Environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*	**A. Environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*
B. Non-environmentally Qualified Equipment *Same as effect of corrosion on conductor for cables (A.1.1).*	**B. Non-environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).* Note: Wear due to vibration is most probable in locations where jacket is adjacent to rough or sharp objects capable of causing cutting, chafing or abrasion. Jackets provide some degree of protection to underlying insulation from exposure to outside stressors, such as radiation, oxygen, moisture, dirt, dust and other contaminants.	**B. Non-environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*

B. Electrical Connectors

B.1 Splices
- B.1.1 Jackets
- B.1.2 Seals
- B.1.3 Insulators

B.2 Mechanical Connectors
- B.2.1 Terminal Lugs, compression fittings, fusion connectors, contact pins

B.3 Terminal Blocks
- B.3.1 Block Assembly

Systems, Structures and Components

This review table addresses the electrical connectors that are used in electrical circuits to join the various components electrically. This includes splices, mechanical connectors and terminal blocks. Individual sub-components for each connector are addressed in terms of aging mechanisms and effects.

System Interfaces

Electrical connectors are used in all electrical circuits, therefore, they functionally interface with all plant systems that rely on electric power and/or instrumentation and control. Physical interfaces include installation in junction boxes and various control panels.

VI. ELECTRICAL COMPONENTS
B. Electrical Connectors

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B.1.1	Splices	Jackets	Polymers	High temp., Radiation, Oxygen	Exposure of insulation and internal parts to outside conditions	Hardening and Cracking	*Same as effect of corrosion on conductor for cables (A.1.1).*
B.1.2	Splices	Seals (potting) Compounds (gaskets, sealant)	Organic Compounds or cement, Rubber	High temp., Radiation, Oxygen	Moisture intrusion, leakage current, Signal noise/ Error, circuit failure	Hardening and Cracking	*Same as effect of corrosion on conductor for cables (A.1.1).*

VI. ELECTRICAL COMPONENTS
B. Electrical Connectors

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
A. Environmentally Qualified Equipment *Same as effect of corrosion on conductor for cables (A.1.1).*	**A. Environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*	**A. Environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*
B. Non-environmentally Qualified Equipment *Same as effect of corrosion on conductor for cables (A.1.1).*	**B. Non-environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*	**B. Non-environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*
A. Environmentally Qualified Equipment *Same as effect of corrosion on conductor for cables (A.1.1).*	**A. Environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*	**A. Environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*
B. Non-environmentally Qualified Equipment *Same as effect of corrosion on conductor for cables (A.1.1).*	**B. Non-environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*	**B. Non-environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*

VI. ELECTRICAL COMPONENTS
B. Electrical Connectors

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B.1.2	Splices	Seals (potting) Compounds (gaskets, sealant)	Organic Compounds or cement, Rubber	High temp. hmidity, Mech. Stress	Moisture intrusion, leakage current, Signal noise/ Error, circuit failure	Creep, distortion	*Same as effect of corrosion on conductor for cables (A.1.1).*
B.1.3	Splices	Insulators (Heat shrink, Tape)	Organic materials, rubber, specialty tapes	High temp., Radiation, Oxygen	Leakage current, Signal noise/ Error, circuit failure	Hardening and Cracking	*Same as effect of corrosion on conductor for cables (A.1.1).*

VI. ELECTRICAL COMPONENTS
B. Electrical Connectors

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
A. Environmentally Qualified Equipment *Same as effect of corrosion on conductor for cables (A.1.1).*	**A. Environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*	**A. Environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*
B. Non-environmentally Qualified Equipment *Same as effect of corrosion on conductor for cables (A.1.1).*	**B. Non-environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*	**B. Non-environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*
A. Environmentally Qualified Equipment *Same as effect of corrosion on conductor for cables (A.1.1).*	**A. Environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*	**A. Environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*
B. Non-environmentally Qualified Equipment *Same as effect of corrosion on conductor for cables (A.1.1).*	**B. Non-environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*	**B. Non-environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*

VI. ELECTRICAL COMPONENTS
B. Electrical Connectors

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B.2.1	Mechanical Connectors	Terminal lugs, compression fittings, Fusion connectors Contacts/ pins	Copper (plated/ Nonplated)	Moisture, chemicals, oxygen	Increased circuit resistance, leakage current, signal noise/ error	Corrosion, oxidation	*Same as effect of corrosion on conductor for cables (A.1.1).*
B.2.1	Mechanical Connectors	Terminal lugs, compression fittings, fusion connectors, Contacts/ pins	Copper (plated/ Nonplated)	Vibration, thermal cycling, repeated connect/ disconnect	Increased circuit resistance, leakage current, signal noise/ error	Distortion, cracking, work hardening	*Same as effect of corrosion on conductor for cables (A.1.1).*

VI. ELECTRICAL COMPONENTS
B. Electrical Connectors

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
A. Environmentally Qualified Equipment *Same as effect of corrosion on conductor for cables (A.1.1).*	**A. Environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*	**A. Environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*
B. Non-environmentally Qualified Equipment *Same as effect of corrosion on conductor for cables (A.1.1).*	**B. Non-environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*	**B. Non-environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*
A. Environmentally Qualified Equipment *Same as effect of corrosion on conductor for cables (A.1.1).*	**A. Environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*	**A. Environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*
B. Non-environmentally Qualified Equipment *Same as effect of corrosion on conductor for cables (A.1.1).*	**B. Non-environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*	**B. Non-environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*

VI. ELECTRICAL COMPONENTS
B. Electrical Connectors

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B.3.1	Terminal Blocks	Block assembly	Organic Compounds	High temp., Radiation, Oxygen	Shorting	Hardening, Cracking	*Same as effect of corrosion on conductor for cables (A.1.1).*
B.3.1	Terminal Blocks	Block assembly	Organic Compounds	Moisture, Contami-nants	Shorting	Loss of insulating properties	*Same as effect of corrosion on conductor for cables (A.1.1).*

VI. ELECTRICAL COMPONENTS
B. Electrical Connectors

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
A. Environmentally Qualified Equipment *Same as effect of corrosion on conductor for cables (A.1.1).*	**A. Environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*	**A. Environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*
B. Non-environmentally Qualified Equipment *Same as effect of corrosion on conductor for cables (A.1.1).*	**B. Non-environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*	**B. Non-environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*
A. Environmentally Qualified Equipment *Same as effect of corrosion on conductor for cables (A.1.1).*	**A. Environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*	**A. Environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*
B. Non-environmentally Qualified Equipment *Same as effect of corrosion on conductor for cables (A.1.1).*	**B. Non-environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*	**B. Non-environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*

C. Electrical Penetration Assemblies (EPA)

C.1 Modular EPA

- C.1.1 O-ring seals
- C.1.2 Conductor-to-insulator seals
- C.1.3 Cable lead wires
- C.1.4 Interface connectors

Systems, Structures and Components

This review table addresses electric penetration assemblies (EPA). EPAs are used to route electric cable circuits through the containment wall. They provide electrical continuity for the circuit, as well as a pressure boundary for the containment. Individual sub-components for a typical modular type EPA are addressed in terms of aging mechanisms and effects.

System Interfaces

Electric penetration assemblies functionally interface with all electric circuits that must be routed through the containment wall.

VI. ELECTRICAL COMPONENTS
C. Electrical Penetration Assemblies

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C.1.1	Modular Electrical Penetration Assemblies	O-ring seals	Organic compound	High temp., Radiation, Oxygen	Loss of pressure boundary	Hardening, oxidation	*Same as effect of corrosion on conductor for cables (A.1.1).*
C.1.2	Modular Electrical Penetration Assemblies	Conductor-to-insulator seals	Fused glass/metal Fused epoxy/ metal Mechanical swage	Moisture, Contami-nants	Loss of pressure boundary	Corrosion	*Same as effect of corrosion on conductor for cables (A.1.1).*

VI. ELECTRICAL COMPONENTS
C. Electrical Penetration Assemblies

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
A. Environmentally Qualified Equipment *Same as effect of corrosion on conductor for cables (A.1.1).*	**A. Environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*	**A. Environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*
B. Non-environmentally Qualified Equipment *Same as effect of corrosion on conductor for cables (A.1.1).*	**B. Non-environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*	**B. Non-environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*
A. Environmentally Qualified Equipment *Same as effect of corrosion on conductor for cables (A.1.1).*	**A. Environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*	**A. Environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*
B. Non-environmentally Qualified Equipment *Same as effect of corrosion on conductor for cables (A.1.1).*	**B. Non-environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*	**B. Non-environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*

VI. ELECTRICAL COMPONENTS
C. Electrical Penetration Assemblies

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C.1.3	Modular Electrical Penetration Assemblies	Cable lead wires	*Same as effect of corrosion on conductor for cables (A.1.1).*	*Same as effect of corrosion on conductor for cables (A.1.1).*	*Same as effect of corrosion on conductor for cables (A.1.1).*	*Same as effect of corrosion on conductor for cables (A.1.1).*	*Same as effect of corrosion on conductor for cables (A.1.1).*
C.1.4	Modular Electrical Penetration Assemblies	Interface connectors	Copper, Polymers, Organic Compounds	*Same as effect of corrosion on conductor for cables (A.1.1).*	*Same as effect of corrosion on conductor for cables (A.1.1).*	Insulation hardening/ cracking, wear, distortion, metal corrosion	*Same as effect of corrosion on conductor for cables (A.1.1).*

VI. ELECTRICAL COMPONENTS
C. Electrical Penetration Assemblies

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
A. Environmentally Qualified Equipment *Same as effect of corrosion on conductor for cables (A.1.1).*	**A. Environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*	**A. Environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*
B. Non-environmentally Qualified Equipment *Same as effect of corrosion on conductor for cables (A.1.1).*	**B. Non-environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*	**B. Non-environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*
A. Environmentally Qualified Equipment *Same as effect of corrosion on conductor for cables (A.1.1).*	**A. Environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*	**A. Environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*
B. Non-environmentally Qualified Equipment *Same as effect of corrosion on conductor for cables (A.1.1).*	**B. Non-environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*	**B. Non-environmentally Qualified Equipment** *Same as effect of corrosion on conductor for cables (A.1.1).*

D. Electrical Buses

D.1 Isolated Phase Bus
- D.1.1 Bus assembly
- D.1.2 Bus support assembly
- D.1.3 Bus enclosure assembly
- D.1.4 Baffle bushing assembly

D.2 Non-segregated Phase Bus
- D.2.1 Bus assembly
- D.2.2 Bus support assembly
- D.2.3 Bus enclosure assembly
- D.2.4 Baffle bushing assembly

D.3 Segregated Phase Bus
- D.3.1 Bus assembly
- D.3.2 Bus support assembly
- D.3.3 Bus enclosure assembly

D.4 Switchyard Bus
- D.4.1 Bus Enclosure

Systems, Structures and Components

This review table addresses electric buses, including isolated phase buses, non-segregated phase buses, segregated phase buses and switchyard buses. Electrical bus assemblies serve as the interconnecting means for distribution of bulk electric power within the plant, from offsite power sources to the plant power system, and from the main generator to the utility power grid.

System Interfaces

Electrical bus assemblies interface with high voltage switchgear, power transformers, power circuit breakers, disconnect switches, and generators; bus penetrations route electrical bus bars through structural walls, equipment enclosures and other boundaries

VI. ELECTRICAL COMPONENTS
D. Electrical Buses

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D.1.1	Isolated-Phase Bus	Bus assembly	Aluminum, Bronze, Copper, Stainless Steel	Normal Service Conditions, including elevated temperature and oxygen	Increased electrical resistance and heating	Oxidation, contact surface corrosion	Bulletin 79-27 Generic Letter 91-11 IN 86-87 IN-86-100 IN 88-55 IN-89-64 IN 91-57 IN 92-09 IN 92-40 IN 93-28

VI. ELECTRICAL COMPONENTS
D. Electrical Buses

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Electrical Bus Inspection Program While no requirement currently exists for such a program, periodic visual inspection of electrical buses is a potential method of managing aging degradation for these components. The inspection program should check for indications of any of the identified aging mechanisms, such as surface contamination, oxidation, or corrosion. In addition, infrared thermography can be used to identify hot spots. The visual inspections should also check for settling of support structures, which might place mechanical stress on these components. If no indications are found, this would provide some assurance that aging degradation is not adversely impacting the ability of the components to perform their intended function. If indications of aging degradation are noted, corrective actions can be taken prior to failure occurring.	As one potential means of managing aging of electrical buses, an inspection program can be implemented in which periodic visual inspections of the components are performed. The 10 criteria identified in the draft SRP-LR are discussed for such a program below: ***(1) Scope:*** The inspection program should include all electrical buses that are important to safety. ***(2) Preventive Actions:*** Any preventive actions that can be taken to mitigate aging degradation should be identified. ***(3) Parameter Monitored/Inspected:*** The parameters to be monitored/inspected should be determined based on the aging mechanisms identified as important for these components. Each of the aging mechanisms presented in this table should be addressed by identifying a parameter or indicator that can be observed during the inspection. ***(4) Detection:*** Each of the parameters/indicators should be observed during the inspection to provide some assurance that aging degradation is detected prior to failure. ***(5) Monitoring and Trending:*** Any aging indicators noted during the inspection should be quantified, to the extent possible, to allow trending in future inspections. ***(6) Acceptance Criteria:*** An acceptance criteria should be established for each of the parameters/indicators identified such that once the criteria is exceeded, corrective actions must be taken to refurbish or replace the component. ***(7) Corrective Actions:*** Based on the acceptance criteria established, corrective actions should be implemented to refurbish or replace components not meeting the minimum acceptance criteria. ***(8 & 9) Confirmation Process and Administrative controls:*** A process should be included to ensure that inspection results are reviewed and compared against acceptance criteria, and that corrective actions are implemented, when necessary. Appropriate administrative controls should be in place to ensure that the inspections are performed in a standardized manner and at the proper frequency, and that results are properly documented. ***(10) Operating experience:*** Past operating experience should be reviewed and evaluated to identify any plant specific aging issues that should be addressed for these components in the program.	No

VI. ELECTRICAL COMPONENTS
D. Electrical Buses

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D.1.2	Isolated phase bus	Bus support assembly	Aluminum, galvanized metals, porcelain, steel	Elevated temperature, oxygen	Loss of function, change in bus geometry, cracking, leakage current	Corrosion, vibration, contamination	*Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).*
D.1.3	Isolated phase bus	Bus enclosure assembly	Aluminum, steel	Elevated temperature, moisture, oxygen	Loss of function, change in bus geometry	Corrosion	*Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).*
D.1.4	Isolated phase bus	Baffle bushing assembly	Aluminum, brass, grout, porcelain, silicone caulk, steel	Elevated temperature, radiation, moisture, dust	Loss of function, change in bus geometry, increased electrical resistance and heating, leakage current, change in material properties	Oxidation of contact surfaces, corrosion, contamination	*Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).*
D.2.1	Non-segregated phase bus	Bus assembly	Aluminum, Bronze, Copper, Stainless Steel	Normal Service Conditions, including elevated temperature and oxygen	Increased electrical resistance and heating	Oxidation, contact surface corrosion	*Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).*
D.2.2	Non-segregated phase bus	Bus support assembly	Aluminum, galvanized metals, porcelain, steel	Elevated temperature, oxygen	Loss of function, change in bus geometry, cracking, leakage current	Corrosion, vibration, contamination	*Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).*

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).	*Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).*	*Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).*
Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).	*Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).*	*Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).*
Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).	*Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).*	*Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).*
Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).	*Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).*	*Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).*
Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).	*Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).*	*Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).*

VI. ELECTRICAL COMPONENTS
D. Electrical Buses

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D.2.3	Non-segregated phase bus	Bus enclosure assembly	Aluminum, steel	Elevated temperature, moisture, oxygen	Loss of function, change in bus geometry	Corrosion	*Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).*
D.2.4	Non-segregated phase bus	Baffle bushing assembly	Aluminum, brass, grout, porcelain, silicone caulk, steel	Elevated temperature, radiation, moisture, dust	Loss of function, change in bus geometry, increased electrical resistance and heating, leakage current, change in material properties	Oxidation of contact surfaces, corrosion, contamination	*Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).*
D.3.1	Segregated phase bus	Bus assembly	Aluminum, Bronze, Copper, Stainless Steel	Normal Service Conditions, including elevated temperature and oxygen	Increased electrical resistance and heating	Oxidation, contact surface corrosion	*Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).*

VI. ELECTRICAL COMPONENTS
D. Electrical Buses

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).	*Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).*	No
Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).	*Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).*	No
Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).	*Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).*	No

VI. ELECTRICAL COMPONENTS
D. Electrical Buses

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D.3.2	Segregated phase bus	Bus support assembly	Aluminum, galvanized metals, porcelain, steel	Elevated temperature, oxygen	Loss of function, change in bus geometry, cracking, leakage current	Corrosion, vibration, contaminati on	*Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).*
D.3.3	Segregated phase bus	Bus enclosure assembly	Aluminum, steel	Elevated temperature, moisture, oxygen	Loss of function, change in bus geometry	Corrosion	*Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).*
D.4.1	Switchyard bus	Bus assembly	Aluminum	Moisture, dirt, dust, salt, wind, high temperature	Increased electrical resistance and heating, fatigue	Oxidation of contact surfaces, corrosion, vibration, contaminati on	*Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).*

VI. ELECTRICAL COMPONENTS
D. Electrical Buses

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).	*Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).*	No
Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).	*Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).*	No
Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).	*Same as effect of surface oxidation on the bus assembly in the isolated phase bus (D.1.1).*	No

E. Electrical Insulators

E.1 Station Post Insulators
E.1.1 Assembly

E.2 Strain/suspension Insulators
E.2.1 Assembly

Systems, Structures and Components

This review table addresses electric insulators, including station post insulators and suspension insulators. Station post insulators and suspension insulators form an integral part of the utility transmission system connecting the power station to offsite power sources, and tying the main generator output to the utility's power grid. Station post insulators provide electrical insulation, spacing, and support between sub-station and switchyard electrical buses and their support structures. Similarly, suspension insulators provide electrical insulation, spacing, and support between transmission line conductors and their transmission structures.

System Interfaces

Electric insulators functionally interface with the utility transmission system connecting the power station to offsite power sources, and tying the main generator output to the utility's power grid

VI. ELECTRICAL COMPONENTS
E. Electrical Insulators

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
E.1.1	Station post insulator	Assembly	Porcelain, Galvanized Metals, Stainless Steel, Cement	High temperature, dirt, dust, salt, vibration, and humidity	Leakage current, loss of function, cracking	Surface contaminati on or oxidation, loss of material due to wear, corrosion, mechanical stress	IN 93-95

VI. ELECTRICAL COMPONENTS
E. Electrical Insulators

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Insulator Inspection Program While no requirement currently exists for such a program, periodic visual inspection of insulators is a potential method of managing aging degradation for these components. The inspection program should check for indications of any of the identified aging mechanisms, such as cracking or surface contamination. If no indications are found, this would provide some assurance that aging degradation is not adversely impacting the ability of the components to perform their intended function. If indications of aging degradation are noted, corrective actions can be taken prior to failure occurring.	As one potential means of managing aging of insulators, an inspection program can be implemented in which periodic visual inspections of the components are performed. The 10 criteria identified in the draft SRP-LR are discussed for such a program below: ***(1) Scope:*** The inspection program should include all insulators that are important to safety. ***(2) Preventive Actions:*** Any preventive actions that can be taken to mitigate aging degradation should be identified. ***(3) Parameter Monitored/Inspected:*** The parameters to be monitored/inspected should be determined based on the aging mechanisms identified as important for these components. Each of the aging mechanisms presented in this table should be addressed by identifying a parameter or indicator that can be observed during the inspection. ***(4) Detection:*** Each of the parameters/indicators should be observed during the inspection to provide some assurance that aging degradation is detected prior to failure. ***(5) Monitoring and Trending:*** Any aging indicators noted during the inspection should be quantified, to the extent possible, to allow trending in future inspections. ***(6) Acceptance Criteria:*** An acceptance criteria should be established for each of the parameters/indicators identified such that once the criteria is exceeded, corrective actions must be taken to refurbish or replace the component. ***(7) Corrective Actions:*** Based on the acceptance criteria established, corrective actions should be implemented to refurbish or replace components not meeting the minimum acceptance criteria. ***(8 & 9) Confirmation Process and Administrative controls:*** A process should be included to ensure that inspection results are reviewed and compared against acceptance criteria, and that corrective actions are implemented, when necessary. Appropriate administrative controls should be in place to ensure that the inspections are performed in a standardized manner and at the proper frequency, and that results are properly documented. ***(10) Operating experience:*** Past operating experience should be reviewed and evaluated to identify any plant specific aging issues that should be addressed for these components in the program.	No

VI. ELECTRICAL COMPONENTS
E. Electrical Insulators

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
E.2.1	Strain and suspension insulator	Assembly	Porcelain, Galvanized Metals, Stainless Steel, Cement	High temperature, dirt, dust, salt, vibration, and humidity, wind	Leakage current, loss of function, cracking	Surface contamination or oxidation, loss of material due to wear, corrosion, mechanical stress, vibration	IN 93-95

VI. ELECTRICAL COMPONENTS
E. Electrical Insulators

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Same as effect of surface contamination on the assembly in the station post insulator (E.1.1).	*Same as effect of surface contamination on the assembly in the station post insulator (E.1.1).*	No

F. Transmission Conductors

F.1 Conductor

F.1.1 Assembly

Systems, Structures and Components

This review table addresses transmission conductors. Transmission conductors form an integral part of the utility transmission system connecting the power station to offsite power sources, and tying the main generator output to the utility's power grid.

System Interfaces

Transmission conductors functionally interface with the utility transmission system connecting the power station to offsite power sources, and tying the main generator output to the utility's power grid

VI. ELECTRICAL COMPONENTS
F. Transmission Conductors

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
F.1.1	Transmission conductors	Assembly	Aluminum, Steel	High temperature, vibration, dirt, dust, salt, wind, ice, oxygen, and humidity	Leakage current, fatigue	Surface contamination or oxidation, corrosion, material loss due to wear	None

VI. ELECTRICAL COMPONENTS
F. Transmission Conductors

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Transmission Conductor Inspection Program While no requirement currently exists for such a program, periodic visual inspection of transmission conductors is a potential method of managing aging degradation for these components. The inspection program should check for indications of any of the identified aging mechanisms, such as corrosion. In addition, infrared thermography can be used to identify hot spots. If no indications are found, this would provide some assurance that aging degradation is not adversely impacting the ability of the components to perform their intended function. If indications of aging degradation are noted, corrective actions can be taken prior to failure occurring.	As one potential means of managing aging of transmission conductors, an inspection program can be implemented in which periodic visual inspections of the components are performed. The 10 criteria identified in the draft SRP-LR are discussed for such a program below: ***(1) Scope:*** The inspection program should include all transmission conductors that are important to safety. ***(2) Preventive Actions:*** Any preventive actions that can be taken to mitigate aging degradation should be identified. ***(3) Parameter Monitored/Inspected:*** The parameters to be monitored/inspected should be determined based on the aging mechanisms identified as important for these components. Each of the aging mechanisms presented in this table should be addressed by identifying a parameter or indicator that can be observed during the inspection. ***(4) Detection:*** Each of the parameters/indicators should be observed during the inspection to provide some assurance that aging degradation is detected prior to failure. ***(5) Monitoring and Trending:*** Any aging indicators noted during the inspection should be quantified, to the extent possible, to allow trending in future inspections. ***(6) Acceptance Criteria:*** An acceptance criteria should be established for each of the parameters/indicators identified such that once the criteria is exceeded, corrective actions must be taken to refurbish or replace the component. ***(7) Corrective Actions:*** Based on the acceptance criteria established, corrective actions should be implemented to refurbish or replace components not meeting the minimum acceptance criteria. ***(8 & 9) Confirmation Process and Administrative controls:*** A process should be included to ensure that inspection results are reviewed and compared against acceptance criteria, and that corrective actions are implemented, when necessary. Appropriate administrative controls should be in place to ensure that the inspections are performed in a standardized manner and at the proper frequency, and that results are properly documented. ***(10) Operating experience:*** Past operating experience should be reviewed and evaluated to identify any plant specific aging issues that should be addressed for these components in the program.	No

G. Ground Conductors

G.1 Conductor

G.1.1 Assembly

Systems, Structures and Components

This review table addresses ground conductors. The electrical ground conductors make up the plant's electrical ground system. This system establishes the reference ground potential for electrical system voltages in the entire plant. Electric power system voltage measurements are referenced to the ground system, and all protective relaying, basic insulation levels, instrumentation, controls, and metering depend on the design integrity of the plant ground system. Personnel and equipment safety are also dependent on the ground system grid.

System Interfaces

Ground conductors functionally interface with all circuits that are electrically connected to ground.

VI. ELECTRICAL COMPONENTS
G. Ground Conductors

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
G.1.1	Ground conductor	Assembly	Copper, bronze	Humidity, salt, oxygen	Loss of function, increased electrical resistance	Surface contaminati on or oxidation, corrosion, mechanical stress	None

VI. ELECTRICAL COMPONENTS
G. Ground Conductors

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Ground Conductor Inspection Program Inspection of ground grid conductors may or may not be included in a plant preventive maintenance program. No generally accepted methods to monitor the integrity of cable ground conductors exist. Periodic visual inspection is one potential approach, however, the majority of the ground grid is inaccessible. Indirect indicators of ground integrity are provided through instrument calibration programs, periodic inspection, maintenance, and testing of protective relaying, and the monitoring of electric power system quality and operating parameters. While no requirement currently exists for such a program, periodic visual inspection of accessible ground conductors is a potential method of managing aging degradation for these components. The inspection program should check for indications of any of the identified aging mechanisms, such as corrosion. In addition, infrared thermography can be used to identify hot spots. Since the majority of the ground grid is inaccessible, indirect indicators of ground integrity should also be included. If no indications are found, this would provide some assurance that aging degradation is not adversely impacting the ability of the components to perform their intended function. If indications of aging degradation are noted, corrective actions can be taken prior to failure occurring.	As one potential means of managing aging of ground conductors, an inspection program can be implemented in which periodic visual inspections, along with indirect measurements of ground integrity are performed. The 10 criteria identified in the draft SRP-LR are discussed for such a program below: ***(1) Scope:*** The inspection program should include all ground conductors that are important to safety. ***(2) Preventive Actions:*** Any preventive actions that can be taken to mitigate aging degradation should be identified. ***(3) Parameter Monitored/Inspected:*** The parameters to be monitored/inspected should be determined based on the aging mechanisms identified as important for these components. Each of the aging mechanisms presented in this table should be addressed by identifying a parameter or indicator that can be observed during the inspection. ***(4) Detection:*** Each of the parameters/indicators should be observed during the inspection to provide some assurance that aging degradation is detected prior to failure. ***(5) Monitoring and Trending:*** Any aging indicators noted during the inspection should be quantified, to the extent possible, to allow trending in future inspections. ***(6) Acceptance Criteria:*** An acceptance criteria should be established for each of the parameters/indicators identified such that once the criteria is exceeded, corrective actions must be taken to refurbish or replace the component. ***(7) Corrective Actions:*** Based on the acceptance criteria established, corrective actions should be implemented to refurbish or replace components not meeting the minimum acceptance criteria. ***(8& 9) Confirmation Process and Administrative controls:*** A process should be included to ensure that inspection results are reviewed and compared against acceptance criteria, and that corrective actions are implemented, when necessary. Appropriate administrative controls should be in place to ensure that the inspections are performed in a standardized manner and at the proper frequency, and that results are properly documented. ***(10) Operating experience:*** Past operating experience should be reviewed and evaluated to identify any plant specific aging issues that should be addressed for these components in the program.	No

References

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