NUCLEAR ENERGY INSTITUTE

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Mr. Christopher I. Grimes
Chief, License Renewal and Standardization Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: Generic Aging Lessons Learned Report Comments

PROJECT NUMBER: 690

Dear Mr. Grimes:

Enclosed are comments on Generic Aging Lessons Learned (GALL) Report Chapter IV, Sections A1, B1, and C1. Three documents are provided under each enclosure. One document is a pen-and-ink mark-up of the existing GALL pages to reflect our comments. Each comment is identified by number. The second document is a table containing our comments, numbered consistent with the marked-up pages. The third document is a clean copy of the GALL pages to reflect how GALL reads with our comments incorporated. In addition, we have included a copy of generic technical issues applicable to various sections, a copy of insertions keyed to the pen-and-ink mark-ups, and sections A.8, A.10, and A.15 of Chapter XI, titled Existing Aging Management Programs (AMP) and Activities.

We look forward to discussing the enclosed comments with the NRC staff. If you have any questions please cal Bob Evans (202) 739-8101 or me at (202) 739-8093.

Sincerely,

Douglas J. Walters

Douglas J. Walters

Enclosures

c: Mr. Sam Lee
Mr. P.T. Kuo

D042

A1. Reactor Vessel (Boiling Water Reactor)

A1.1 Top Head Enclosure

A1.1.1 Top Head

A1.1.2 Nozzles (Vent, Top Head Spray or RCIC, and Spare)

A1.1.3 Head Flange

A1.1.4 Closure Studs and Nuts

A1.1.5 Vessel Flange Leak Detection Line

A1.2 Vessel Shell

A1.2.1 Vessel Flange

A1.2.2 Upper Shell

A1.2.3 Intermediate (Nozzle) Shell

A1.2.4 Intermediate (Beltline) Shell

A1.2.5 Lower Shell

A1.2.6 Beltline Welds

A1.2.7 Attachment Welds

A1.3 Nozzles

A1.3.1 Main Steam

A1.3.2 Feedwater

A1.3.3 High Pressure Coolant Injection (HPCI)

A1.3.4 High Pressure Core Spray (HPCS)

A1.3.5 Low Pressure Core Spray (LPCS)

A1.3.6 CRD Return Line

A1.3.7 Recirculating Water (Inlet & Outlet)

A1.3.8 Low Pressure Coolant Injection (LPCI) or RHR Injection Mode

A1.3.9 Isolation Condenser Supply

A1.4 Nozzles Safe Ends

A1.4.1 High Pressure Core Spray (HPCS)

A1.4.2 Low Pressure Core Spray (LPCS)

A1.4.3 CRD Return Line

A1.4.4 Recirculating Water (Inlet & Outlet)

A1.4.5 Low Pressure Coolant Injection (LPCI) or RHR Injection Mode

A1.5 Penetrations

A1.5.1 CRD Stub Tubes

A1.5.2 Instrumentation

A1.5.3 Jet Pump Instrument

A1.5.4 Standby Liquid Control

A1.5.5 Flux Monitor

A1.5.6 Drain Line

A1.6 Bottom Head

A1.7 Control Rod Drive Mechanism

A1.7.1 Housing

A1.8 Support Skirt and Attachment Welds

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 (Boiling 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 or later approved Eidtions. EPRI-103515, R2 (BWRVIP-79) or later version.

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
See Chapter 11 for: ISI AMA Primary Water Chemistry AMA	***See Chapter 11***	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.
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 or later approved Editions. RICSIL 055R1. Reg. Guide 1.65. ASME Section III, 1989 edition.

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

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.	Yes, TLAA Move to Chapter 11
See Chapter 11 for: ISI AMA Primary Water Chemistry AMA	***See Chapter 11***	No

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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A1.1.5	Top Head Enclosure	Vessel Flange Leak Detection Line	Stainless Steel	Leaking Oxygenated Water and/or Steam up to 288°C	Crack Initiation and Growth	SCC, IGSCC	Plant Technical Specifications. EPRI-103515, R2 (BWRVIP-79) or later version.
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.
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.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
See Chapter 11 for: ISI AMP	***See Chapter 11***	
See Chapter 11 for: ISI AMP ISI AMA for leak test.	***See Chapter 11***	No, unless leak detected in ISI Program
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.	Move to TLAA Chapter 11
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.	Move to TLAA Chapter 11

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. BWRVIP-74

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. Upper shelf energy (USE) is maintained at acceptable levels for 54 effective full power years (EFPY) per BWRVIP-74.	The TLAA should evaluate the impact of neutron embrittlement on: (a) the adjusted reference temperature, the plant's pressure temperature limits, and the need for inservice inspection of circumferential and axial reactor vessel welds, (b) the Charpy upper shelf energy, 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 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. BWRVIP-74
A1.2.7	Vessel Shell	Attachment Welds	SS, Inconel 182	288°C, Oxygenated Water	Crack Initiation and Growth	SCC, IGSCC	ASME Section XI, 1989 Editions or later approved Editions. EPRI–103515, R2 9BWRVIP-79) or later version.

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. Upper shelf energy (USE) is maintained at acceptable levels for 54 effective full power years.	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 Move to TLAA Chapter 11
See Chapter 11 for: ISI AMA Primary Water Chemistry AMA	***See Chapter 11*** Inspect per BWRVIP-48	No

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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A1.3.1	Nozzles	Main Steam	SA508-Cl2 with or without SS Cladding	288°C Steam	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition.
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	ASME Section XI, 1989 Edition or later approved Editions. EPRI-103515, R2 (BWRVIP-79) or later version.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	See Chapter 11	
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.	Yes, TLAA Move TLAA to Chapter 11
See Chapter 11 for : ISI and Primary Water Chemistry AMA	***See Chapter 11***	No

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

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	See Chapter 11	

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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A1.3.2 A1.3.6	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 or later approved Editions. NUREG-0619; BWRVIP-74 GE-NE-523-A71-0594
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. BWRVIP-74
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 or later approved Editions. Reg. Guide 1.45. EPRI-103515, R2 (BWRVIP-79) or later version BWRVIP-75 *Operating Experience* NRC IN 82-39. NRC IN 84-41.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
The feedwater and CRDRL nozzles are inspected in accordance with NUREG-0619 or the BWROG alternative to the NUREG which received final NRC approval on March 10, 2000. In either case, the inspection program was specifically designed to monitor the components for potentially detrimental effects associated with fatigue. NUREG-0619 is the NRC's provision to address the concern and the BWROG alternative provides an option that is acceptable to NRC. Therefore, as long as either of these programs is implemented, fatigue of these locations is adequately managed.	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. The ISI AMA as augmented by NUREG-0619 or the NRC approved (3/10/00) BWROG alternative (GE-NE-523-A71-0594) is sufficient to manage fatigue and ensure component integrity. No additional analysis needed.	No
For a 40 y design life, pressure vessel integrity is assured by fracture toughness and material surveillance program requirements set forth in 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 (GSs) 88-11 and 92-01 Rev.1 and Supplement 1. to predict effects of neutron irradiation on reactor vessel materials.		Yes, TLAA Move to TLAA Chapter 11
Program delineated in NUREG-0313, Rev. 2 and implemented through NRC Generic letter (GL) 88-01 and its Supplement 1. ***See Chapter 11 for:*** ISI and Primary Water Chemistry AMP	***(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 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. ***(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 (Boiling 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. 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. ***(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 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 or later approved Editions. ANSI B31.1
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 ASME Section XI, 1989 Edition or later approved Editions. EPRI-103515, R2 (BWRVIP-79) or later version. BWRVIP-49 BWRVIP-27 BWRVIP-47 *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.	Yes, TLAA Move to Chapter 11
See Chapter 11 for: ISI AMA Primary Water Chemistry AMA	***See Chapter 11***	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. BWRVIP-49 BWRVIP-27 BWRVIP-47
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 or later approved Editions. EPRI-103515, R2 or later version. BWRVIP-58.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	See Chapter 11	
All of these components were evaluated as part of the BWRVIP program. The evaluations are documented in BWRVIP-49 for instrument penetrations, BWRVIP-27 for the standby liquid control penetration and BWRVIP-47 for lower plenum components (bottom head components and penetrations). The BWRVIP evaluations considered the degradation mechanisms that would affect the components and required inspections to ensure continued integrity	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. The BWRVIP program considered all modes of degradation for the covered components. The insurance program ensures the component will be able to perform its intended function. No additional analysis is required.	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 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
See Chapter 11 for: ISA AMA Primary Water Chemistry AMA	***See Chapter 11***	No

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. BWRVIP-47
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 or later approved Editions.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	See Chapter 11	
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 Move to TLAA Chapter 11
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 Move to TLAA Chapter 11

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IVA.1

SECTION-COMMENT #	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT/CHANGES
IV A1-1	Reactor Vessel (BWR)	All	All		The heading for all pages reads "Pressurized Water Reactor" and should be changed to "Boiling Water Reactor."
IV A1-2 **	Reactor Vessel (BWR)	Entire GALL	multiple		See words in insert 4. The subsections referenced should be deleted. The reason for the change is to have the GALL be consistent with current regulations that require licensees at their next 120 month update to use a different edition. Since the NRC is considering the position that once a plant is baselined at the 1996 Addenda any other update performed by a licensee will be voluntary, it seems sensible that the NRC position on license renewal would be based on the code edition most likely to be in place. However, it should be noted that for almost all the references, the inspection requirements referenced in GALL still exist in the later editions, although they may be located in different paragraphs.
IV A1-3 **	Reactor Vessel (BWR)	A1.1.1 through A1.1.3	IV A1-4 and IV A1-5		1. NUREG-0313 and GL 88-01 are appropriate as references for the top head and head flange only to the extent that the NRC position on IGSCC and materials is documented in them. As such it provides a technical basis for the NRC to conclude that cladding cracking, and the potential for cladding cracking to propagate into the RPV/head is not a concern. 2. In every case in the report BWRVIP-29 and EPRI TR-103515 are listed as separate documents. They in fact are the same document and the TR-103515 should have been shown in parentheses. Also, the water chemistry guidelines have been updated so that the correct reference for water chemistry should now be shown as BWRVIP-79 (TR-103515,R2). See insert 3. 3. Under the AMP column the sentence that begins with "Prevention is by material selection.." should have the phrase "NUREG-0313, Rev.2 and of" deleted. It should now read: Prevention is by material selection in accordance with guidelines of Regulatory Guide 1.43 for control of stainless steel cladding of low-allow steels. The reason for the change is that NUREG-0313 applies only to piping and nozzle welds. Further, since the provisions in the NUREG were developed in

SECTION-COMMENT #	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT/CHANGES
					the mid-1980's, the only materials that would have been selected using those criteria would have been pipe replacement materials. Note that the "Preventative Actions" item for the 10 criteria is correct. 4. BWRVIP-74 should be added under the "References" column. It addresses vessel issues for the renewal period. 5. The existing AMP column should specify ISI AMA and Primary Water Chemistry AMA. See GALL Insert #s 8 & 9.
IV A1-4	Reactor Vessel (BWR)	A1.1.5	IV A1-8 and A1-9		The NRC states under AMP that existing programs may not be capable of detecting SCC of vessel flange leak detection line. However, some licensees have this line in their Class 1 boundary and as such look at it as part of the Class 1 pressure test each outage. I believe that the AMP column should reflect that Section XI pressure testing is adequate for the finding SCC in the leak detection line outboard of the RPV flange.
IV A1-5	Reactor Vessel (BWR)	A1.2.3 through A1.2.6	IV A1-8 et seq.		A maximum neutron flux of $5x10^9$ n/cm^2-s is given in the "Environment" column. Such environmental limits are useful for license renewal applicants, and similar types of information should be included in the "Material" and "Environment" columns elsewhere in the report. For example, for Item A1.1.1 through A1.1.3, maximum carbon content and minimum ferrite content should be identified in the "Material" column, rather than in the "Evaluation and Technical Basis" column. Similarly, material selection in accordance with NUREG-0313, Rev. 2, and satisfaction of material requirements in Regulatory Guide 1.43 should be identified in the "Material" column, rather than in the "Aging Management Program" column. All of these items are related to a determination by the license renewal applicant of the potential significance of the aging degradation effects, rather than a determination of the adequacy of an aging management program. **This is also listed as a generic technical issue 2.**

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IVA.1

SECTION-COMMENT #	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT/CHANGES
IV A1-6 *	Reactor Vessel (BWR)	A1.2.3 through A1.2.6	IV A1-9 and A1-10		1. The "Further Evaluation" column should be revised to say "No". 2. **This line item should be deleted.** NUREG-0313 and GL 88-01 are appropriate as references for the RPV only to the extent that the NRC position on IGSCC and materials is documented in them. As such it provides a technical basis for the NRC to conclude that cladding cracking, and the potential for cladding cracking to propagate into the RPV is not a concern. This is correctly shown under "Scope of Program". BWRVIP-05 evaluated the postulated case where a cladding crack is aligned perfectly with a manufacturing flaw in RPV base metal and showed that growth by SCC is not a concern. No structural or pressure boundary credit is taken for the cladding layer and as such there is no reason to manage this issue. 3. BWRVIP-74 should be added to the "References" column. It addresses vessel issues for the renewal period.
IV A1-7	Reactor Vessel (BWR)	A1.2.4	IV A1-11		**This is a TLAA and should be moved to a separate section (see generic technical issue list.)** This is one of the 17 open license renewal technical issues identified and described in EPRI TR-107521 (NEI/NRC 98-0036; Neutron Irradiation Embrittlement – Definition of the Reactor Vessel Beltline) – **see generic technical issue list 4.** 1. The "Evaluation and Technical Basis" column contains language that is repeated in many locations throughout the report. In all locations, the text should be modified to read "Neutron irradiation embrittlement is a time-limited aging analysis (TLAA) to be evaluated 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. For the limiting ferritic materials, as defined by Appendices G and H of 10 CFR Part 50, the TLAA should evaluate ….." This same change should be made at all of the affected locations in the report. This same change for definition of beltline should also be

SECTION-COMMENT #	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT/CHANGES
					made in the "Existing AMP" column 2. BWRVIP-74 should be added to the "References" column. It addresses vessel issues for the renewal period. 3. Under AMP it should be noted that BWRVIP-74 contains an updated Appendix G equivalent margins analysis for upper shelf energy (USE). The analysis shows that USE is maintained at acceptable levels for 54 effective full power years (EFPY).
IV A1-8 **	Reactor Vessel (BWR)	A1.2.6	IV A1-12 and A1-13		1. BWRVIP-74 should be added to the "References" column. It addresses vessel issues for the renewal period. 2. Under AMP it should be noted that BWRVIP-74 contains an updated Appendix G equivalent margins analysis for upper shelf energy (USE). The analysis shows that USE is maintained at acceptable levels for 54 effective full power years (EFPY). 3. See also comment IVA1-7
IV A1-9 *	Reactor Vessel (BWR)	A1.2.7	IV A1-12		1. The "Further Evaluation" column should be revised to say "No". 2. BWRVIP-48 and the NRC SER should be added to the "References" column, "existing AMP" column and "Evaluation and Technical Basis" column. 3. Element 4, "Detection of Aging Effects should be revised to read: "Aging effects degradation of the attachments cannot occur without crack initiation: extent and schedule of inspection assures detection of cracks before the loss of intended function." The basis follows: The BWRVIP program developed examination requirements for vessel attachments. The BWRVIP program incorporated code required examinations except where more stringent examinations were appropriate. The results are documented in BWRVIP-48 which was approved by the NRC for use in the renewal period in an SER issued September 29, 1999. The

SECTION-COMMENT #	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT/CHANGES
					shroud support attachment weld is addressed in BWRVIP-38. That is under NRC review at this time. Note: In addition to the VT-3 that is listed in the GALL table, the Code requires a more stringent VT-1 of the attachments in the beltline. 1. NUREG-0313 and GL 88-01 are applicable to piping only, not vessel attachment welds and should not be listed under "References" or "AMP". The discussion as part of element 1 "Scope of Program" is correct information but could be deleted since the BWRVIP program implements or supercedes any piping criteria that would be used on internals. 2. The existing AMP column should specify ISI AMA and Primary Water Chemistry AMA. See GALL Insert #s 8 & 9.
IV A1-10	Reactor Vessel (BWR)	A1.3.2 through A1.3.9	IV A1-14 and A1-15		4. The existing AMP column should specify ISI AMA and Primary Water Chemistry AMA. See GALL Insert #s 8 & 9.
IV A1-11 *	Reactor Vessel (BWR)	A1.3.2 A1.3.6	IV A1-18 and A1-19		5. BWRVIP-74 should be added to the "References" column. It addresses vessel issues for the renewal period, including the inspection of feedwater and CRDRL nozzles. 6. The BWROG alternative to NUREG-0619 should be added to the reference list. It is "Alternate BWR Feedwater Nozzle Inspection Requirements", GE-NE-523-A71-0594. 7. The following information should be added under "AMP" The feedwater and CRDRL nozzles are inspected in accordance with NUREG-0619 or the BWROG alternative to the NUREG which received final NRC approval on March 10, 2000. In either case, the inspection program was specifically designed to monitor the components for potentially detrimental effects associated with fatigue. NUREG-0619 is the NRC's provision to address the concern and the BWROG alternative provides an option that is acceptable to NRC. Therefore, as long as a either of these programs are implemented, fatigue of these locations is adequately managed.

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IVA.1

SECTION-COMMENT #	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT/CHANGES
IV A1-12 *	Reactor Vessel (BWR)	A1.3.8	IV A1-18		The nozzles are part of the reactor vessel surveillance program if the materials are limiting. As such, this location will be monitored if necessary and a separate listing is not necessary. If NRC chooses to maintain a separate listing the following items should be addressed. 1. BWRVIP-74 should be added to the "References" column. It addresses vessel issues for the renewal period. 2. Under AMP it should be noted that BWRVIP-74 contains an updated Appendix G equivalent margins analysis for upper shelf energy (USE). The analysis shows that USE is maintained at acceptable levels for 54 effective full power years (EFPY.)
IV A1-13 **	Reactor Vessel (BWR)	A1.4.1 through A1.4.5	IV A1-18 and A1-19		1. The existing AMP for this component is GL 88-01 or BWRVIP-75; ISI AMA and Primary Water Chemistry AMA. 2. The existing AMP column should specify ISI AMA and Primary Water Chemistry AMA. See GALL Insert #s 8 & 9. 3. The Evaluation and Technical Basis column provides a good description of GL 88-01 or BWRVIP-75 program with changes as identified..
IV A1-14 *	Reactor Vessel (BWR)	A1.4.3	IV A1-22 and A1-23		All but 2 plants have cut and capped the CRD return lines. Fatigue should not be an issue for these plants at this location. For the 2 plants that have not capped those lines, the inspections required by NUREG-0619 and GL 88-01 are sufficient to ensure integrity of the safe ends. NRC can apply the 10 elements from their GL 88-01 program to the location that contains IGSCC susceptible materials. Therefore a TLAA should not be required and the "Further Evaluation" column should be revised to say "No". 1. BWRVIP-74 should be added to the "References" column. It addresses vessel issues for the renewal period, including the inspection of feedwater and CRDRL nozzles. 2. The following information should be added under "AMP:" The CRDRL nozzles are inspected in accordance with NUREG-

SECTION-COMMENT #	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT/CHANGES
					0619. The inspection program was specifically designed to monitor the components for potentially detrimental effects associated with fatigue. NUREG-0619 is the NRC's provision to address the fatigue concern at this location. Therefore, as long as the program is implemented, fatigue of these locations is adequately managed.
IV A1-15 **	Reactor Vessel (BWR)	A1.5.1 through A1.5.6	IV A1-22 and A1-23		1. Add the following information to the "AMP" column: All of these components were evaluated as part of the BWRVIP program. The evaluations are documented in BWRVIP-49 for instrument penetrations, BWRVIP–27 for the standby liquid control penetration and BWRVIP-47 for lower plenum components (bottom head components and penetrations). The BWRVIP evaluations considered the degradation mechanisms that would affect the components and required inspections to ensure continued integrity. (Note: NRC safety evaluations issued 09/01/99 and 12/20/99 approve the use of BWRVIP-49 and BWRVIP-27 for use in the renewal period. NRC has issued a safety evaluation for the technical adequacy of BWRVIP-47. The license renewal review is being finalized.) 2. Add BWRVIP-27, -47, and –49 to references. 1. The components listed, with the possible exception of the jet pump instrumentation nozzle, are not covered by GL 88-01. They are either not IGSCC susceptible material or they are less than 4in. NRC is better served and so is the industry if they refer to the BWRVIP program. 2. The existing AMP column should specify ISI AMA and Primary Water Chemistry AMA. See GALL Insert #s 8 & 9.
IV A1-16 **	Reactor Vessel (BWR)	A1.5.1 through A1.5.6	IV A1-24 and A1-25		1. Revise the "Further Evaluation" column to read "No". 2. Add the following information to the "AMP" column: All of these components were evaluated as part of the BWRVIP program. The evaluations are documented in BWRVIP-49 for

SECTION-COMMENT #	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT/CHANGES
					instrument penetrations, BWRVIP–27 for the standby liquid control penetration and BWRVIP-47 for lower plenum components (bottom head components and penetrations). The BWRVIP evaluations considered the degradation mechanisms that would affect the components and required inspections to ensure continued integrity. (Note: NRC safety evaluations issued 09/01/99 and 12/20/99 approve the use of BWRVIP-49 and BWRVIP-27 for use in the renewal period. NRC has issued a safety evaluation for the technical adequacy of BWRVIP-47. The license renewal review is being finalized.) 3. Add BWRVIP-27, -47, and –49 to references.
IV A1-17 *	Reactor Vessel (BWR)	A1.7.1	IV A1-24 and A1-25		1. Revise the "Further Evaluation" column to read "No". This component was evaluated by the BWRVIP. The evaluations are documented in BWRVIP-47. The evaluations considered the degradation mechanisms that would affect the components and required inspections to ensure continued integrity. NRC has issued a safety evaluation for the technical adequacy of BWRVIP-47. The license renewal review is being finalized. 2. Element 7, "Corrective Actions", needs to be revised. The BWRVIP program includes inspection and flaw evaluation criteria. When specified safety margins cannot be met corrective action is required. The component is part of the reactor coolant pressure boundary and is a welded core support structure. Should repairs be needed, they would have to meet the provisions of Section XI which requires meeting the original Construction Code or later approved versions of Section III. Alternatives may be used if approved by NRC. Since the provisions for corrective measures developed by the BWRVIP are alternatives to the original construction Code and would require NRC approval prior to use, there is no reason to require

SECTION-COMMENT #	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT/CHANGES
					additional evaluation as part of a renewal application. When the NRC approves the BWRVIP proposed corrective measures, the BWRVIP measures would be an approved alternative. 3. The existing AMP column should specify ISI AMA and Primary Water Chemistry AMA. See GALL Insert #s 8 & 9.
IV A1-18 *	Reactor Vessel (BWR)	A1.7.1	IV A1-26 and A1-27		1. Revise the "Further Evaluation" column to read "No". This component was evaluated by the BWRVIP. The evaluations are documented in BWRVIP-47. The evaluations considered the degradation mechanisms, including fatigue, that would affect the components and required inspections to ensure continued integrity. NRC has issued a safety evaluation for the technical adequacy of BWRVIP-47. The license renewal review is being finalized. 2. Add the reference to BWRVIP-47.
IV A1-19 *	Reactor Vessel (BWR)	A1.8	IV A1-26 and A1-27		1. It should be noted that environmental fatigue concerns are not relevant to the skirt since the environment is air.

5/2

A1. Reactor Vessel (Boiling Water Reactor)

A1.1 Top Head Enclosure

A1.1.1 Top Head

A1.1.2 Nozzles (Vent, Top Head Spray or RCIC, and Spare)

A1.1.3 Head Flange

A1.1.4 Closure Studs and Nuts

A1.1.5 Vessel Flange Leak Detection Line

A1.2 Vessel Shell

A1.2.1 Vessel Flange

A1.2.2 Upper Shell

A1.2.3 Intermediate (Nozzle) Shell

A1.2.4 Intermediate (Beltline) Shell

A1.2.5 Lower Shell

A1.2.6 Beltline Welds

A1.2.7 Attachment Welds

A1.3 Nozzles

A1.3.1 Main Steam

A1.3.2 Feedwater

A1.3.3 High Pressure Coolant Injection (HPCI)

A1.3.4 High Pressure Core Spray (HPCS)

A1.3.5 Low Pressure Core Spray (LPCS)

A1.3.6 CRD Return Line

A1.3.7 Recirculating Water (Inlet & Outlet)

A1.3.8 Low Pressure Coolant Injection (LPCI) or RHR Injection Mode

A1.3.9 Isolation Condenser Supply

A1.4 Nozzles Safe Ends

A1.4.1 High Pressure Core Spray (HPCS)

A1.4.2 Low Pressure Core Spray (LPCS)

A1.4.3 CRD Return Line

A1.4.4 Recirculating Water (Inlet & Outlet)

A1.4.5 Low Pressure Coolant Injection (LPCI) or RHR Injection Mode

A1.5 Penetrations

A1.5.1 CRD Stub Tubes

A1.5.2 Instrumentation

A1.5.3 Jet Pump Instrument

A1.5.4 Standby Liquid Control

A1.5.5 Flux Monitor

A1.5.6 Drain Line

A1.6 Bottom Head

A1.7 Control Rod Drive Mechanism

A1.7.1 Housing

A1.8 Support Skirt and Attachment Welds

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).

Boiling applies to all pages. Comment IV A1-1

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

A1. REACTOR VESSEL ([redacted] Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environment	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 1.45~~ ~~BWRVIP-29.~~ EPRI TR-103515. BWRVIP-74

Insert 4

Insert 3

See ISI AMA writeup

See Prim Wtr Chem AMA writeup

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

A1. REACTOR VESSEL (Pressurized Water Reactor)

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. ISI AMA Prim Water Chemistry AMA SEE CHAPTER II FOR	*(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 [illegible]-01, respectively, describe the technical basis and staff guidance regarding the problem of SCC 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

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
A1.1.3	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. [redacted]
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.

Insert 4

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*, [redacted]. Move to TLAA Section X.	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. 151 AMA Prim. wtr Chemistry AMA SEE CHAPTER 11 FOR	***(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 [illegible] and additional guidance of Regulatory Guide (RG) [illegible] on material selection, preservice inspection, and protection against corrosion, prevent or mitigate SCC. High-strength low-alloy steels with controlled tempering procedures are used. Maximum tensile strength is limited *to <1172 MPa (<170 ksi) to provide resistance to SCC*, and Charpy V energy requirements of Appendix G to 10 CFR Part 50 provide adequate toughness to 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 suitable 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 Rl 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

See 151 AMA write up
See prim wtr chemistry write up

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
A1.1.5	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. INSERT 3
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. [redacted]
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-s	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. [redacted]

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
SEE CHAPTER II FOR ISI AMP	*(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 BWR are inspected in accordance with Program B IWB-2400 which requires 100% inspection every 10 y, at least 15% 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 is done near the end of each inspection interval. *(6) Acceptance Criteria:* Any cracks in closure stud bolts 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 replacements 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.~~ ISI AMA for leak test	~~Plant-specific aging management program is to be evaluated.~~ See ISI AMA SEE CHAPTER II	~~Yes, no AMP~~ unless leak detection line not in ISI program
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. [redacted]	~~Yes TLAA~~ Move to TLAA Section X
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. [redacted]	~~Yes TLAA~~

Comment IV A1-4

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
A1.2.4	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-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. [redacted] BWRVIP-74
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-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.

DELETE

Comment IV A-1-6

The line item for cladding SCC/IGSCC can be deleted. Cladding has no structural significance. SCC of LAS vessel is not a concern.

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. [redacted]	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 [redacted]). [redacted]	***(1) Scope of Program:*** The program is based on managing the effects of stress corrosion cracking (SCC) of the SS cladding on the intended function of the reactor vessel shell. NUREG-0313 and GL 88-[redacted] effectively 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 [redacted] shell based on analytical evaluations indicate that [redacted] of the [redacted] SS cladding into the ferritic [redacted] will be [redacted]. ***(2) Preventive Actions:*** [redacted] section of [redacted] considers [redacted] to IGSCC, e.g., grades [redacted] metal with [redacted] carbon of [redacted].035% and minimum [redacted].5% ferrite, prevent or mitigate IGSCC [redacted] Regulatory Guide (RG [redacted]) provides assurance that production cladding complies with ASME Section III and XI guidelines to prevent [redacted] cracking. Coolant water chemistry is monitored [redacted] maintained in accordance with EPRI guidelines in BWRVIP-29 and TR-103515 to minimize the potential of crack initiation and growth. Also, [redacted] water chemistry and stringent control of conductivity [redacted] 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

Comment IV A1-7

INSERT 6

Move to TLAA Section

delete

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

A1. REACTOR VESSEL (Pressurized Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environment	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-s	*Loss of* Fracture Toughness	*Neutron* Irradiation Embrittlement	*Appendices G & H* to 10 CFR Part 50, Reg. Guide 1.99, NRC GL 88-11, NRC GL 92-11, Rev. 1. [redacted] BWRVIP-74
A1.2.7	Vessel Shell	Attachment Welds	SS, Inconel 182	288°C, Oxygenated Water	Crack Initiation and Growth	SCC, IGSCC	[redacted] ASME Section XI, 1989 Edition, ~~BWRVIP-29~~, EPRI TR-103515.

Comment IV A1-8

Insert 4

Insert 3

Comment IV A1-9

Existing Aging Management Program (AMP)	Evaluation and [illegible] Basis	Further Evaluation
	(continued from previou[illegible] *(7) Corrective Actio[illegible]* [illegible]ction pro[illegible] the BWRVIP is u[illegible] *(9) Con[illegible] Process and [illegible]:* Site [illegible]res, review a[illegible]val [illegible] and a[illegible] control [illegible] implemen[illegible] accorda[illegible] requirements of Appendix B to 10 CFR [illegible] and will continue to be adequate for the [illegible] license renewal. *(10) Operating Experience:* The [illegible]P is effective in mana[illegible] crack initiation and g[illegible] due to SCC, however, because of inaccessibil[illegible], 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. [redacted] Subsection IWB, examination category B-A of all pressure-retaining welds in the vessel and [redacted] beltline region, defi[redacted] the length of the minimal shield or effective length of [redacted] elements. A NRC Generic Letter [redacted] regard[redacted]ions from [redacted] for circum[redacted]grated surveillance [redacted] 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 Move to TLAA Section X
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. BWRVIP-48 ISI AMA Prim. Wtr Chemistry AMA	*(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 implement 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. NO

Insert 6

SEE CHAPTER

SEE CHAP. 11 FOR

See BWR VIP-48, SER 9/29/98
See ISI AMA writeup
See Prim wtr Chem writeup

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
A1.3.1	Nozzles	Main Steam	SA508-Cl2 with or without SS Cladding	288°C Steam	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. [redacted]
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	[redacted] Rev. 2. [redacted] ~~Sup. 1.~~ ASME Section XI, 1989 Edition. ~~Reg. Guide 1.43.~~ BWRVIP-29. EPRI TR-103515. Insert 4 Insert 3

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 may not be adequate to inspect tight cracks and crevice condition. Applicant should determine whether an augmented inspection program is required. Site 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. [redacted] Move to TLAA Section X	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. ISI AMA Prim Water Chemistry AMP	***(1) Scope of Program:*** The program is focused on managing the effects of stress corrosion cracking [illegible] cladding on the intended function of reactor vessel nozzles. NUREG-0313 and GL 88-01, respectively, describe the technical basis and surveillance regarding the problem of IGSCC in BWRs. ***(2) Preventive Actions:*** Selection of materials considered resistant to IGSCC, e.g., grades of wrought metals with a maximum carbon of 0.035% and minimum 7.5% ferrite, prevent or mitigate IGSCC, and Regulatory Guide (RG) 1.44 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

See ISI AMA writeup
See Prim Wtr Chemistry AMA writeup

See Chap. 11 ISI & Prim. Water Chemistry AMP

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) 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. Visual VT-2 (IWA-5240) examination performed on 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. Systems hydrostatic test, IWB-5222, is conducted at least at 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.	

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
A1.3.2 A1.3.6	Nozzles	Feedwater, CRDRL	SA508-Cl2 with or without SS cladding	Up to 288°C, Oxygenated Water	Cumulative Fatigue Damage Comment IV A1-11	Fatigue	ASME Section III, 1989 Edition; NUREG-0619; [redacted] BWRVIP-74 GE-NE-523-A71-0594
A1.3.8	Nozzles	LPCI (or RHR Injection Mode)	SA508-Cl2	Up to 288°C, Oxygenated Water, $5x10^8$ - $5x10^9$ n/cm^2-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. BWRVIP-74
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, ~~BWRVIP-75~~ *Operating Experience* NRC IN 82-39, NRC IN 84-41.

Insert 4

Insert 4

Insert 3

Insert text from last item on Comment IV A1-11

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. ISI AMA	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. The ISI AMA As augmented by NUREG-0619 or the NRC approved (3/10/00) BWROG alternative (GE-NE-523-A71-0594) is sufficient to manage fatigue and ensure component integrity. No additional analysis needed.	Yes TLAA No
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. [redacted] 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 Move to TLAA Section X
Program delineated in NUREG-0313, Rev. 2 and implemented through NRC Generic letter (GL) 88-01 and its Supplement 1, [redacted] leakage, and additional recommendations of [redacted] (SI) [redacted] 1. [redacted] monitored [redacted] in accordance with EPRI [redacted] and TR-103515 to minimize the potential of [redacted] AMA	*(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, [redacted] 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. [redacted] *(3) Parameters Monitored/Inspected:* The AMP monitors the effects of IGSCC on the intended	No AND

or BWRVIP-75

SEE CHAP. 11

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. [redacted] 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. [redacted] ***(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. [redacted] 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	or BWRVIP-75

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 [redacted] Insert 4
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. [redacted] ASME Section XI, 1989 Edition. Insert 4 ~~BWRVIP-29.~~ EPRI TR-103515. Insert 3 *Operating Experience* NRC IN 82-39. NRC IN 84-41. BWRVIP-49 BWRVIP-27 BWRVIP-47

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, [redacted]	Yes TLAA Move to TLAA Section X
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. ISI AMA Prim Water Chemistry AMP References BWRVIP documents + SERs	***(1) Scope of Program:*** NUREG-0313 and GL 88-01, respectively, describe the technical basis for 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 See ISI AMA write up See Prim Water chemistry write up Comment IV A1-16

LEE CHAP. II FOR

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. [redacted] BWRVIP 49 BWRVIP 27 BWRVIP 47 Comment IV A1-16
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. Insert 4 BWRVIP-29. EPRI TR-103515. Insert 3 BWRVIP-38. Comment IV A1-17

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 [illegible] and schedule are enhanced for welds of nonresistant materials. ***(6) Acceptance Criteria:*** Any IGSCC detected is evaluated in accordance with IWB-3600 by comparing ISI results with the acceptance standards of IWB-3500 and IWB-3522. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWA-4000 and 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 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.	
[redacted] Insert text from Comment IV A1-16(2)	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. Reference SERs for BWRVIP document and add insert 7	Yes TLAA 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
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 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. ISI AMA	***(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. [illegible] 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 NO

SEE CHAP. 11 of Primary Water Chemistry AMA

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. [redacted] BWRVIP 47
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. Insert 4

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
See ISI AMR writeup See Prim Wtr Chem writeup	*(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 in-service 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:*** AMP detects degradation of the CRD mechanism housing that not occur without crack initiation. Current 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. [redacted]	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

Move to TLAA Section.

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
B1.1.2	Core Shroud, Shroud Head and Core Plate	Core Shroud (Upper, Central, Lower)	SS	288°C, High-Purity Water	Crack Initiation and Growth	SCC IASCC Fatigue	ASME Section XI, 1989 Edition or later approved Editions BWRVIP-29. EPRI-103515, R2 (BWRVIP-79) or later version. BWRVIP-01 BWRVIP-07. BWRVIP-63. BWRVIP-76 *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
		Yes, BWRVIP Guideline
See Chapter 11 for ISI AMA Primary Water Chemistry AMA	***See Chapter 11***	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
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)	*(continued from previous page)* *Supporting BWRVIP:* BWRVIP-02 BWRVIP-03. BWRVIP-06. BWRVIP-14. BWRVIP-44. BWRVIP-45. BWRVIP-59. BWRVIP-60. BWRVIP-62. *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 IASCC Fatigue	EPRI-103515, R2 (BWRVIP-79) or later version. BWRVIP-25. *Supporting BWRVIP:* BWRVIP-03. BWRVIP-06. BWRVIP-14. BWRVIP-44. BWRVIP-45. BWRVIP-50. BWRVIP-59. BWRVIP-60. BWRVIP-62.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
See Chapter 11 for: BWRVIP-76 or ISI AMA Primary Water Chemistry AMA	***See Chapter 11***	
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
See Chapter 11 BWRVIP-25 Primary Water Chemistry AMA	***See Chapter 11***	Yes, BWRVIP Guideline

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
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	Cumulative Fatigue Damage Crack Initiation and Growth	Fatigue IASCC	*(continued from previous page)* *Operating Experience* NRC GL 94–03. NRC IN 95–17. NUREG–1544. Original Licensing Criteria or ASME Section III, 1989 Edition GSI-190. *Same as for the effect of Stress Corrosion Cracking on Item B1.1.3 core plate.*
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 or later approved Edition. GE SIL 462 Sup. 3. EPRI-103515, R2 (BWRVIP-79) or later version. *Operating Experience* NRC IN 88-03. NRC IN 92-57.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	See Chapter 11	
See Chapter 11 for: Visual inspection VT–3 per ISI AMA Primary Water Chemistry AMA	***See Chapter 11***	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B1.1.5							
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 IASCC	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
	See Chapter 11	
See Chapter 11 for Primary Water Chemistry AMA	***See Chapter 11*** Inspection per BWRVIP-38 and BWRVIP-76 as applicable. Alternately, visual inspection (VT-3) per ISI AMA.	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B1.1.8	Core Shroud, Shroud Head and Core Plate	LPCI Coupling	SS	288°C, High-Purity Water	Crack Initiation and Growth	SCC IASCC	EPRI-103515, R2 (BWRVIP-79) or later version. BWRVIP-42. *Supporting BWRVIP:* BWRVIP-03. BWRVIP-56.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
See Chapter 11 for Primary Water Chemistry Program AMA ISI AMA	***See Chapter 11***	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B1.1.8	Core Shroud, Shroud Head and Core Plate	LPCI Coupling	SS	288°C, High-Purity Water		IASCC	Same as for the effect of Stress Corrosion, Cracking on Item B1,1,1,8 LPCI coupling. *(continued from previous page)* BWRVIP-06. BWRVIP-14. BWRVIP-44. BWRVIP-45. BWRVIP-59. BWRVIP-60. BWRVIP-62.
B1.2	Top Guide	Top Guide	SS	288°C, High-Purity Water	Crack Initiation and Growth	SCC IASCC Fatigue	EPRI-10351, R2 (BWRVIP-79) or later version. BWRVIP-26. *Supporting BWRVIP:* BWRVIP-03. BWRVIP-06. BWRVIP-14. BWRVIP-44. BWRVIP-45. BWRVIP-50. BWRVIP-59. BWRVIP-60. BWRVIP-62. *Operating Experience* NRC GL 94–03. NRC IN 95–17. NUREG–1544

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
See Chapter 11 for Primary Water Chemistry AMA	***See Chapter 11*** Inspect per BWRVIP-42	
See Chapter 11 for Primary Water Chemistry AMA	***See Chapter 11*** Inspect per BWRVIP-26	No

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 Crack Initiation and Growth	Fatigue IASCC	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190. *Same as for the effect of Stress Corrosion Cracking on Item B1.2 top guide.*

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	See Chapter 11	
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. *NOTE: Combine all B1.2 (see comment # IV B1-8, number 3)*	*Same as for the effect of Stress Corrosion Cracking on Item B1.2 top guide.*	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	Crack Initiation and Growth Cumulative Fatigue Damage	SCC IASCC Fatigue	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. *Same as for the effect of Stress Corrosion Cracking on Items B1.4.1-B1.4.4 core spray line components.* Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
See Chapter 11 for Primary Water Chemistry AMA	***See Chapter.11*** Inspection per BWRVIP-18	

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), Castings: Cast Austenitic Stainless Steel (CASS), Others: SS	288°C, High-Purity Water	Crack Initiation and Growth Cumulative Fatigue Damage	SCC IASCC Fatigue	EPRI-103515, R2 (BWRVIP-79) or later version. 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. *Same as for the effect of Stress Corrosion Cracking on Items B1.5.1-B1.5.8 jet pump assembly components.* *Original licensing Criteria or ASME section III 1989 Edition GSI-190*

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
See Chapter 11 for Primary Water Chemistry AMA	***See Chapter 11*** Inspect per BWRVIP-41	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	SS, CASS	288°C, High-Purity Water	Crack Initiation and Growth Cumulative Fatigue Damage	Thermal Aging and Neutron Irradiation Embrittle-ment Fatigue	EPRI TR-106092. BWRVIP-47 BWRVIP-25 BWRVIP-56 Original Licensing Criteria or ASME Section III, 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 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. NOTE: Combine all B1.6.1 (see comment IV B1-13)	No

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B1.7.1, B1.7.2, B1.7.3	Instrument Housings	Intermediate Range Monitor (IRM) Dry Tubes, Source Range Monitor (SRM) Dry Tubes IRM Dry Tubes, SRM Dry Tubes IRM Dry Tubes, Low Power Range Monitor (LPRM) Dry Tubes, SRM Dry Tubes LPRM Dry Tubes	SS	288°C, High-Purity Water	Crack Initiation and Growth Cumulative Fatigue Damage	SCC Fatigue IASCC	EPRI-103515, R2 (BWRVIP-79) or later version. BWRVIP-49. BWRVIP-57. *Supporting BWRVIP:* BWRVIP-03. BWRVIP-06. BWRVIP-14. BWRVIP-44. BWRVIP-45. BWRVIP-47 BWRVIP-59. BWRVIP-60. BWRVIP-62. Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190. *Same as for the effect of Stress Corrosion Cracking on Items B1.7.1 IRM dry tubes and B1.7.3 SRM dry tubes.* Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
See Chapter 11 for: Primary Water Chemistry AMA	***See Chapter 11*** Inspection per BWRVIP-47	No

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IVB.1

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV B1-1 *	Reactor Vessel Internals (BWR)	B1.1.1	IV B1-4 and B1-5		1. This item should be deleted form the GALL report. As noted in BWRVIP-06, the shroud head and the bolts are not safety-related. Their failure would not hinder core coolant flow nor the ability to put the reactor in a safe shutdown condition. BWRVIP-06 was reviewed by NRC. A safety evaluation was issued 09/15/98. 2. Fatigue is not a TLAA for the shroud head bolts since the shroud head and the bolts are not safety-related. Their failure would not hinder core coolant flow nor the ability to put the reactor in a safe shutdown condition 3. Section XI does not apply to core shroud head bolts. The code requirement B-N-2 is for the "Integrally Welded Core Support Structures and Interior Attachments to Reactor Vessels" and the bolts are not welded nor attached to the reactor vessel.
IV B1-2 *	Reactor Vessel Internals (BWR)	B1.1.2	IV B1-4 through IV B1-7		1. The "Further Evaluation" column should be revised to read "No". 2. BWRVIP-01 is not listed as a reference, but should be. BWRVIP-01 is the original shroud inspection document. BWRVIP-07 contains reinspection criteria and BWRVIP-63 contains vertical weld criteria. All 3 documents have been combined into BWRVIP-76 which should also be referenced. 3. The 7th element, "Corrective Actions", should be revised. Since the core shroud is a core support structure, repairs to it must meet the criteria of Section XI or NRC must approve the alternative. Therefore, corrective action will either restore the component to a condition that meets the original design specification and Construction Code or NRC will approve the alternate. BWRVIP-02 should be listed as a reference. It contains repair criteria that has been used at several plants with NRC approval. The BWRVIP-02 criteria provide an alternative to the Code and as such constitute a generic non-

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
					code repair method for a Code component. Inspection criteria for the repaired shroud and the repair hardware are contained in the BWRVIP program. No additional evaluations are necessary if the BWRVIP criteria are followed. 4. The separate listing for this item dealing with IASCC should be deleted. The BWRVIP criteria considered all plausible modes of degradation for each component. The inspection program is such that the effects of IGSCC, IASCC and fatigue are appropriately accounted for with the prescribed inspections. No other evaluation is needed. 5. The existing AMP column should specify BWRVIP-76 or ISI AMA and Primary Water Chemistry AMA. See GALL Insert #s 8 & 9.
IV B1-3 *	Reactor Vessel Internals (BWR)	B1.1.3 and B1.1.4	IV B1-6 through IV B1-9		1. The "Further Evaluation" column should be revised to read "No" 2. .The separate listings for this item dealing with SCC, IASCC and Fatigue should be combined. The BWRVIP criteria considered all plausible modes of degradation for each component. The inspection program is such that the effects of IGSCC, IASCC and fatigue are appropriately accounted for with the prescribed inspections. No other evaluation is needed. 3. Section XI reference should be deleted since it does not apply to the core plate or the core plate bolts. The code requirement B-N-2 is for the "Integrally Welded Core Support Structures and Interior Attachments to Reactor Vessels" and the bolts are not welded nor are they attached to the reactor vessel. 4. BWRVIP-25 contains inspection requirements for the core plate and bolting necessary to assure continued integrity and safe operation. NRC issued a safety evaluation on 12/19/99. 5. The existing AMP column should specify BWRVIP-25 (SER dated 12/19/99), ISI AMA and Primary Water Chemistry AMA. See GALL Insert #s 8 & 9.

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV B1-4 **	Reactor Vessel Internals (BWR)	B1.1.5	IV B1-8 and B1-9		1. The second entry for Access Hole Cover on page IV B1-10 is a duplicate of the information on the previous page and should therefore be deleted.
IV B1-5 *	Reactor Vessel Internals (BWR)	B1.1.6	IV B1-10 through IV B1-13		1. The "Further Evaluation" column should be revised to read "No". 2. BWRVIP-01, -07, -63, and-76 should be referenced. BWRVIP-01 is the original shroud inspection document. BWRVIP-07 contains reinspection criteria and BWRVIP-63 contains vertical weld criteria. All 3 documents have been combined into BWRVIP-76. 3. Section XI does not require examination of the shroud support legs or any other component below the core plate. However, any repair or replacement would be activity would be within the jurisdiction of Section XI. 4. Element 7, "Corrective Actions", needs to be revised. The BWRVIP program includes inspection and flaw evaluation criteria. When specified safety margins cannot be met corrective action is required. The component is part of the welded core support structure. Should repairs be needed, they would have to meet the provisions of Section XI which requires meeting the original Construction Code or later approved versions of Section III. Alternatives may be used if approved by NRC. Since the provisions for corrective measures developed by the BWRVIP are alternatives to the original construction Code and would require NRC approval prior to use, there is no reason to require additional evaluation as part of a renewal application. When the NRC approves the BWRVIP proposed corrective measures, the BWRVIP measures would be an approved alternative. 5. BWRVIP-38 contains inspection requirements for the shroud support structure necessary to assure continued integrity and safe operation. NRC issued an initial safety evaluation on

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
					09/16/99. 6. The listings for this item for SCC and IASCC should be combined. The BWRVIP criteria considered all plausible modes of degradation for each component. The inspection program is such that the effects of IGSCC, IASCC and fatigue are appropriately accounted for with the prescribed inspections. No other evaluation is needed. 7. The existing AMP column should specify inspection per BWRVIP-38 and BWRVIP-76 as applicable or alternatively, visual inspection (VT-3) per ISI AMA and Primary Water Chemistry AMA. See GALL Insert #s 8 & 9.
IV B1-6 *	Reactor Vessel Internals (BWR)	B1.1.7	IV B1-12 and B1-13		1. This item should be deleted from the GALL. BWRVIP-06 and BWRVIP–27 both indicate that the standby liquid control piping inside the vessel is not needed to shutdown the reactor or maintain it in a safe shutdown mode. NRC issued an SER for BWRVIP-06 on 9/15/98. on both BWRVIP documents. BWRVIP-27 contains the inspection requirements necessary to assure continued integrity and safe operation of the standby liquid control line. NRC issued a safety evaluation 12/20/99 approving the use of BWRVIP-27 in the renewal period. No other evaluation is needed. 2. The separate listings for this component addressing SCC and IASCC can be combined/deleted. The BWRVIP criteria considered all plausible modes of degradation for each component. The inspection program is such that the effects of IGSCC, IASCC and fatigue are appropriately accounted for with the prescribed inspections. Further, IASCC is not a plausible degradation mechanism for the standby liquid control line; its location relative to the core eliminates the concern. No other evaluation is needed.

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IVB.1

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV B1-7 *	Reactor Vessel Internals (BWR)	B1.1.8	IV B1-12 through IV B1-14		1. Revise the "further Evaluation" column to read "No". BWRVIP-42 contains the inspection requirements necessary to assure continued integrity and safe operation of the LPCI coupling. NRC issued an SER on 06/14/99. No other evaluation is needed. 2. The BWRVIP criteria considered all plausible modes of degradation for each component. The inspection program is such that the effects of IGSCC, IASCC and fatigue are appropriately accounted for with the prescribed inspections. No other evaluation is needed. 1. The existing AMP column should specify BWRVIP-42 (SER dated 6/14/99) and Primary Water Chemistry AMA. See GALL Insert # 9.
IV B1-8 *	Reactor Vessel Internals (BWR)	B1.2	IV B1-14 through B1-17		3. Correct the water chemistry reference to BWRVIP-79 (TR-103515,R2). 4. Revise the "Further Evaluation" column to read "No". BWRVIP-26 contains the inspection requirements necessary to assure continued integrity and safe operation of the top guide. NRC issued an SER on 09/29/99. No other evaluation is needed. 5. All three listings addressing SCC, IASCC, and fatigue can be combined into one item. The BWRVIP criteria considered all plausible modes of degradation for each component. The inspection program is such that the effects of IGSCC, IASCC and fatigue are appropriately accounted for with the prescribed inspections. No other evaluation is needed. 6. The existing AMP column should specify inspect per BWRVIP-26 (SER dated 9/29/99) and Primary Water Chemistry AMA. See GALL Insert # 9.
IV B1-9 *	Reactor Vessel Internals (BWR)	B1.3.1 through B1.3.3	IV B1-16 through B1-19		These items can be deleted from GALL. The feedwater sparger and flow nozzles are not safety-related and their failure would not prevent the injection of coolant makeup. They are not required to

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
					place the plant in safe shutdown. This is documented in BWRVIP-06. NRC issued a safety evaluation for BWRVIP-06 09/15/98. (Note: The feedwater nozzle inner radius is the limiting location for fatigue of the feedwater components. It is inspected in accordance with NUREG-0619, or the BWROG alternative, and would bound any fatigue concerns for the sparger.)
IV B1-10 *	Reactor Vessel Internals (BWR)	B1.4.1 through B1.4.4	IV B1-18 through IV B1-21		1. Revise the "Further Evaluation" column to read "No". BWRVIP-18 contains the inspection requirements necessary to assure continued integrity and safe operation of the core spray piping and sparger. NRC issued an SER on 12/02/99. No other evaluation is needed. 2. The BWRVIP criteria considered all plausible modes of degradation for each component. The inspection program is such that the effects of IGSCC, IASCC and fatigue are appropriately accounted for with the prescribed inspections. No other evaluation is needed. 3. The existing AMP column should specify inspection per BWRVIP-18 (SER dated 12/02/99) and Primary Water Chemistry AMA. See GALL Insert # 9.
IV B1-11 *	Reactor Vessel Internals (BWR)	B1.5.1 through B1.5.8	IV B1-20 through IV B1-23		1. Revise the "Further Evaluation" column to read "No". BWRVIP-41 contains the inspection requirements necessary to assure continued integrity and safe operation of the jet pump assembly and its sub-components. NRC review is in progress. No other evaluation is needed. 2. The BWRVIP criteria considered all plausible modes of degradation for each component. The inspection program is such that the effects of IGSCC, IASCC and fatigue are appropriately accounted for with the prescribed inspections. No other evaluation is needed. 3. The existing AMP column should specify inspect per BWRVIP-41 and Primary Water Chemistry AMA. See GALL Insert # 9.
IV B1-12	Reactor Vessel Internals (BWR)	B1.5.4	IV B1-23 through IV		BWRVIP-41 program accounts for this aging effect. This item could be deleted or combined with B1.5.1 thru 1.5.8.

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IVB.1

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
			B1-25		
IV B1-13 *	Reactor Vessel Internals (BWR)	B1.6.1	IV B1-24 and B1-25		1. Revise the "Further Evaluation" column to read "No". The orificed fuel support (OFS) was evaluated in BWRVIP-47, except for the peripheral OFS which was evaluated in BWRVIP-25 . NRC issued a safety evaluation 10/13/99 for BWRVIP-47 and 12/19/99 for BWRVIP-25. The BWRVIP criteria considered all plausible modes of degradation for each component. The inspection program is such that the effects of IGSCC, IASCC and fatigue are appropriately accounted for with the prescribed inspections. No other evaluation is needed. 2. The existing AMP column and Evaluation and technical Basis should specify BWRVIP-47 (SER dated 10/13/99 and BWRVIP-25 (SER dated 12/19/99).
IV B1-14 *	Reactor Vessel Internals (BWR)	B1.7.1 through B1.7.3	IV B1-26 and B1-27		1. Revise the "Further Evaluation" column to read "No". BWRVIP-47 contains the inspection requirements necessary to assure continued integrity and safe operation of the instrument housings. NRC issued an SER on 10/13/99. No other evaluation is needed. 2. The BWRVIP criteria considered all plausible modes of degradation for each component. The inspection program is such that the effects of IGSCC, IASCC and fatigue are appropriately accounted for with the prescribed inspections. No other evaluation is needed. 3. The existing AMP column should specify inspection per BWRVIP-47 (SER dated 10/13/99) and Primary Water Chemistry AMA. See GALL Insert # 9.

OK T/2

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	Environment	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 /ASCC Fatigue	ASME Section XI, 1989 Edition. BWRVIP-29. EPRI TR-103515. BWRVIP-07. BWRVIP-63. BWRVIP-76 *Operating Experience* NRC IN 94-42. NRC IN 97-17. NRC GL 94-03. NUREG-1544.

DELETE

Delete

Comment IV B-1
The shroud head bolts are not safety related per BWR-VIP-06, NRC SER 9/15/98

DELETE

BWRVIP-01

Insert 4

Insert 3

Comment IV B1-2

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 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 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 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 year 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. Guidelines 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 of 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 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	Yes, BWRVIP Guideline No

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~~ IASCC fatigue	~~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.

Combine with above

Comment IVB1-2(5)

BWRVIP-02

Comment IV B1-2

Issue 3

Comment IVB1-3

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
(continued from previous page) platinum to reduce electrochemical potential. Future 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. SEE CHAP. 11 FOR BWRVIP-76 or ISI AMA Primary Water Chemistry AMA See ISI AMA writeup See Prim Wtr Chem writeup	*(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) 94-42] 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 VT 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 control 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 (manufacturer's specification at <0.1 µS/cm²) reduces susceptibility to SCC. Hydrogen additions are effective in reducing electrochemical potentials in the recirculation piping 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 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

BWRVIP-25 → Replace with reference to BWRVIP-25, SER 12/19/99

Primary Wtr Chemistry AMA → See AMA writeup

Item	Structure and Component	Region of Interest	Material	Environment	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. Insert 4 GE SIL 462 Sup. 3. ~~BWRVIP-29.~~ EPRI TR-103515. Insert 3 *Operating Experience* NRC IN 88-03. NRC IN 92-57.

Combine with previous item

Cumulative IV B1-3

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 guidelines. ***(7) Corrective Actions:*** The corrective action proposed by the BWRVIP is under staff review. ***(8) Confirmation Process and Administrative Controls:*** Standard 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 40 year design life, according to the requirements of the original licensing criteria or ASME Section III (equivalent 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 chemistry is monitored and maintained in accordance with EPRI guidelines 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. Visual inspection VT-3 per ISI AMA Prism with chemistry AMA	***(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 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/cm2) reduces susceptibility to SCC. Hydrogen additions are effective in reducing electrochemical potential 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
See ISI AMA writeup See Prim-Wtr Chemistry writeup	*(continued from previous page)* of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with IWB-3640. ***(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.* Comment IV B1-4	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 controlled in accordance with EPRI guidelines 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. Comment IV B1-5 Inspection per BWRVIP-38 and BWRVIP-76 AS applicable. Alternatively, visual inspection (VT-3) per ISI AMA. Prim Wtr chemistry. AMA	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC, in-service 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) reduces susceptibility to SCC. Hydrogen additions are effective in reducing the electrochemical potential in the recirculation piping system, but not as 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. See ISI AMA writeup See prim-wtr chemistry writeup	Yes, BWRVIP Guideline No

Item	Structure and Component	Region of Interest	Material	Environment	Aging Effect	Aging Mechanism	References
B1.1.6	Core Shroud, Shroud Head and Core Plate	Shroud Support Structure Combine with previous item	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~~ IASCC	[redacted] BWRVIP-29. EPRI TR-103515. BWRVIP-42. BWRVIP-56. *Supporting BWRVIP:* BWRVIP-03.

Delete Not safety related per BWR VIP-06 per comment IV B1-6 NRC SER At 9/15/98

Comment IV B1-5

Insert 3

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 visual inspections and UT inspections for specific programs. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines TR-103515 and BWRVIP-29 to minimize the potential of crack initiation and growth. Plant chemistry measures to reduce water contaminants or induce hydrogen addition and electrochemical 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 replacement as needed to maintain the capability of the intended function. *(2) Preventive Actions:* Maintaining high water purity (many BWRs now operate at <0.15 µS/cm) reduces susceptibility to SCC. Hydrogen addition is effective in reducing electrochemical potentials in the recirculation piping system, but is less effective in the core region. Noble metal additions appear to increase the effectiveness of hydrogen additions in the core region. *(3) Parameters Monitored/Inspected:* Inspection and flaw evaluations are to be performed in accordance with referenced BWRVIP guidelines, subject to NRC staff. *(4) Detection of Aging Effects:* Degradation due to SCC can not occur without cracks initiation and growth. *(5) Monitoring and Trending:* Inspection frequency 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 guidelines. *(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 visual inspections and UT inspections for specific programs. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines 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 replacement as needed to maintain the capability of the intended function. *(2) Preventive Actions:* Maintaining high water purity (many BWRs now operate at <0.15 µS/cm) 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 No

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 Combine with above	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 IASCC Fatigue	[redacted] ~~1999 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. *Operating Experience* NRC GL 94-03. NRC IN 95-17. NUREG-1544

Insert 3

approved SER 6/14/99

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 control on conductivity, hydrogen addition, and use of noble metal additions to reduce electrochemical potential. BWRVIP guideline is under staff review. Inspect per BWRVIP-42 Primary WW Chemistry AMA see Primary WW Chemistry AMA	*(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 (IWB-3) is performed according to ASME Section XI, IWB-2500, category B-N-2. Acceptance for enhanced VT-1 inspections and UT inspections are plant specific programs. Cooling 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 control 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. Inspect per BWRVIP-26 Primary WW Chemistry AMA	***(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 materials and replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** Maintaining high water purity in many BWRs normally at <0.15 µS/cm 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 NO Comment IV B1-8

Approved NRC SER 9/29/99

See AMA write up

Combine with above

Item	Structure and Component	Region of Interest	Material	Environment	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.

Delete

Not safety related per BWR VIP-06 SER 9/15/98 Comment IVB-9

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 Comment IV B1-8 (#3)
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 requirements (volumetric, visual, liquid penetrant inspection) that depend upon specific plant design, other plant actions (monitoring, etc.). An alternate to NUREG-0619 with detailed inspection methods has been approved by the NRC staff. Plant water chemistry is monitored and maintained in accordance with EPRI guidelines TR-103515 and BWRVIP guidelines to minimize the potential of crack initiation and growth. Plant programs also may include water chemistry measures such as strict control of conductivity, hydrogen injection and use of noble metals 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 with < 0.1 µS/cm) reduces susceptibility to SCC. Hydrogen additions are effective in reducing electrochemical potential in the recirculation piping system, but are less effective in the core region. Design features are used at mitigating thermal fatigue cracking, which is the primary source of degradation in 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 occur without crack initiation and growth in accordance to NUREG-0619 with qualified inspection methods has been approved by the NRC staff. ***(5) Monitoring and Trending:*** Inspection schedule in NUREG-0619 is adequate for timely detection. ***(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
		Not Selected					
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. GEI-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 IASCC Fatigue	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.
		Comment IV B1-10					
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
Comp[illegible] have [illegible] evalua[illegible] fatig[illegible] 40 [illegible] life, according to the required [illegible] the original licensing criteria [illegible] ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	[illegible] period [illegible] 0 is [illegible]	Yes [illegible]
Implementation of the program delineated in the NRC Inspection and Enforcement Bulletin (IEB) [illegible]-13 including enhanced visual [illegible] section to inspect, to supplement or replace visual [illegible] requirement of GE Service Information Letter (SIL) 289. Core 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. Inspection per BWRVIP-18 Prime Wtr Chemistry AMA See AMA writeup	***(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) 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 guidelines 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~~ NRC SER 12/02/99
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.* Combined with above	Yes BWRVIP Guideline

Combined with previous Item Comment IV B1-10

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

Item	Structure and Component	Region of Interest	Material	Environment	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 IASCC Fatigue	[redacted] 1. BWRVIP-29. EPRI TR-108515. 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.

Comment IV B1-11

Combine with above

Item 3

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, GE Services Information Letter (SIL) 530 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. Reactor 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. Inspect per BWRVIP-41 Primary w/ Chemistry AMA See AMA Write up	*(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:* Maintain reactor water purity (many BWRs operate at <0.5 μS/cm²) reduces susceptibility to SCC. Hydrogen additions are effective in reducing electrochemical potentials in the recirculation piping system and are effective in the core region. *(3) Parameters Monitored/Inspected:* Inspection and flaw evaluation are to be performed in accordance with referenced BWRVIP guidelines or 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~~ NRC review in progress for BWRVIP-41
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.* Combined with above	Yes BWRVIP Guideline

Item	Structure and Component	Region of Interest	Material	Environment	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.
					Combined with above		
B1.5.4	Jet Pump Assemblies	Castings	CASS	288°C, High-Purity Water	Loss of Fracture Toughness	Thermal Aging and Neutron Irradiation Embrittlement	EPRI TR-106092. ASME Section XI, 1989 Edition.

Delete this item or combine with B1.5.1 thru B1.5.8. BWR VIP-41 program accounts for this aging effect. No separate listing needed.

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. Combine with above See Comment IV B1-11	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 influence of the component in [illegible] either a supplemental examination of the affected components as part of the application 10-year in-service inspection (ISI) program during the license renewal term for a component-specific evaluation to determine the susceptibility to loss of fracture toughness. DELETE	*(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 plastic loss of fracture toughness due to neutron irradiation and thermal aging embrittlement either a supplemental examination of components as part of the 10-year in-service (ISI) program during the license renewal term, component-specific evaluation to determine susceptibility to loss of fracture toughness. *(2) Preventive Actions:* The program includes no mitigation 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. [illegible] the criteria in EPRI TR-106092, with [illegible] modifications, the susceptibility to thermal aging embrittlement of CASS components is determined in terms of casting method, Mo content, and [illegible] 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.~~ BWRVIP-47 BWRVIP-25 BWRVIP-56
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.~~

Combine See LVB1-13

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) r(lighting and surface cle[illegible] the IS bounded by th[illegible]se used to [illegible] resolution of the inspection [illegible]chnique[illegible] perform [illegible]mponent-specific evalu[illegible] mechanical loading assessment to determ[illegible]um tensile loading on the comp[illegible] ASME Code Level A, B, C, and D condi[illegible] loading is compressive or low enough [illegible] then [illegible]emental inspection if the component is not require[illegible] to meet this criteria requires continued [illegible]emental inspection program. For all CASS [illegible] that have a neutron fluence of [illegible] (E>1 MeV), implement the supplement inspe[illegible] if they are "potentially susceptible" [illegible] the existing ASME Section XI inspection requi[illegible] are considered adequate if the com[illegible] "not susceptible" to thermal aging. ***(5) [illegible] Trending:*** Inspection schedule in accor[illegible] IWB-2400 should provide timely detection [illegible]. ***(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. BWRVIP-47 BWRVIP-25	*Same as for the effect of Thermal Aging and Neutron Irradiation Embrittlement on Item B1.5.8 jet pump castings.* SER approval 10/13/99 SER approval 12/19/99	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. See Comment IVB1-13	~~Yes~~ ~~TLAA~~ No

Item	Structure and Component	Region of Interest	Material	Environment	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	[redacted] BWRVIP-29, EPRI TR-103515. [redacted] Insert 3 BWRVIP-4 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.

Combine Component IV B1-14
BWRVIP-47 considered all mode of degradation. No other analysis required

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Implementation of aging management program recommended by GE Service Information Letter (SIL) 409 Rev. 1. Reactor 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 include water chemistry such as strict controls on conductivity, and hydrogen additions. BWRVIP guideline is under staff review. BWR VIP-47 Primary Water Chemistry AMA See AMA Writeup	***(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 SCC-resistant materials which are free of stress. Maintaining high water purity (maintaining KRs power rate at <0.15 μS/cm²) 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 be occurring 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.~~ No ~~BWRVIP Guideline~~ SER approved 10/13/99
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. Combine with above	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

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-103515, R2 (BWRVIP-79) or later version *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
See Chapter 11 for: Flow Accelerated Corrosion Aging Management Activities, and Primary Water Chemistry AMP	***See Chapter 11***	No

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.
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. .
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.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	See Chapter 11	
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.	Yes TLAA Move to TLAA Section XI
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.	Yes TLAA Move to TLAA Section XI
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.	Yes TLAA Move to TLAA Section XI

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
C1. REACTOR COOLANT PRESSURE BOUNDARY (Boiling Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C1.1.5 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 or later approved Editions. EPRI-103515. R2 (BWRVIP-79) or later version. BWRVIP-42. BWRVIP-56. BWRVIP-75. *Supporting BWRVIP:* BWRVIP-03. BWRVIP-61. BWRVIP-62.
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 or later approved Editions.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
See Chapter 11 for: ISI Primary Water Chemistry AMP	***See Chapter 11***	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
See Chapter 11 for: Primary Water Chemistry AMA	***See Chapter 11***	

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.
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.
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.
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 or later approved Editions.
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. EPRI-103515, R2 (BWRVIP-79) or later version.

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.	Yes TLAA Move to Section XI
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.	Yes TLAA Move to Section XI
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.	Yes TLAA Move to Section XI
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
See Chapter 11 for: The effect of SCC, IGSCC on piping and fitting, in various reactor coolant pressure boundary systems C1.1.5 thru C1.11	***See Chapter 11***	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 or later approved Editions.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	See Chapter 11	
See Chapter 11 for: ISI AMP	***See Chapter 11***	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 or later approved Editions.
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.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	See Chapter 11	
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.	Yes TLAA Move to Section XI

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
C1. REACTOR COOLANT PRESSURE BOUNDARY (Boiling Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C1.3.1	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-103515,R2 (BWRVIP-79) or later version.
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 or later approved Editions.
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 or later approved Editions. NRC GL 88-01. NRC GL 88-01, S 1. EPRI-103515, R2 (BWRVIP-79) or later version. BWRVIP-75

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
See Chapter 11 for: The effects of SCC, IGSCC on piping and fittings in various reactor coolant pressure boundary systems	***See Chapter 11***	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 or later approved Editions.
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.
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 or later approved Editions.
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 Editions. ANSI B31.1.
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 or later approved Editions. ASME OM S/G, Pt 2. NRC GL 89-13. Plant Technical Specifications.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	See Chapter 11	
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.	Yes TLAA Move to Section XI
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 Move to Section XI
Detection of reactor coolant leakage by radiation and temperature monitors in component cooling system; inservice inspection program in conformance with Edition of ASME Section XI specified in the licensees' CLB per 10 CFR 50.55a; 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 examination during system leakage test and hydrostatic test of all pressure retaining Class 2 components. 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.	

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IVC.1

SECTION-COMMENT #	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT/CHANGES
IV C1-1	Reactor Coolant Pressure Boundary (BWR)	C1.1.1, C1.1.12	IV C1-5		1. Delete last sentence of element 1 of "Evaluation and Technical Basis" column. EPRI document NSAC-202L-R2 is adequate as a guideline for the FAC program. 2. The existing AMP column should specify FAC AMA and Primary Water Chemistry AMA. See GALL Insert #s 9 & 10.
IV C1-2	Reactor Coolant Pressure Boundary (BWR)	C1.1.5, C1.1.11	IV C1-9, IV C1-11		This is one of the 17 open license renewal technical issues identified and described in EPRI TR-107521 (NEI/NRC 98-0030; Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components). The discussion in the "Aging Program Management" and "Evaluation and Technical Basis" columns is excellent. In particular, the discussion on inservice inspection or flaw tolerance evaluation that references ASME Nuclear Code Case N-481 is acknowledged under Element 4. The inclusion of the alternate mechanical loading assessment in Element 4 is also to be commended. This same comment applies to all other items in B2 (W), B3 (CE), and B4 (B&W) that deal with thermal aging embrittlement and loss of fracture toughness. This same comment applies to all other items in C1 that deal with thermal aging embrittlement and loss of fracture toughness.
IV C1-3	Reactor Coolant Pressure Boundary (BWR)	C1.1.5 through C1.1.11	IV C1-8		1. SCC and IGSCC are adequately controlled by existing programs. GL 88-01 programs have been implemented for approximately 10 years and have been shown to be effective in managing the issue. BWRVIP-75 provides new inspection frequencies based on the materials of construction, water chemistry and inspection history. The repair method using weld overlays proposed in GL 88-01 has since been incorporated into a Section XI Code Case that has been endorsed by NRC. The inspection methods and frequencies are adequate to assure safe operation and component integrity and the repair methods are acceptable for long term operation. No further evaluation for IGSCC is needed. This is consistent with the NRC conclusion on page V D2-7 for items D2.1.1 through D2.1.7 and is

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IVC.1

SECTION-COMMENT #	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT/CHANGES
					appropriate since the materials and service conditions are essentially the same. 2. The existing AMP column should specify GL 88-01 program, ISI AMA and Primary Water Chemistry AMA. See GALL Insert #s 8 & 9.
IVC1-4	Reactor Coolant Pressure Boundary (BWR)	C1.1.1 C1.1.2 C1.1.12 C1.3.1	IVC1-5		For the FAC program under evaluation element 3, remove reference to UT or RT of 10 locations. The parameters monitored are in accordance with individual plant calculations/CHECWORKS results. (SEE GENERIC PROGRAM WRITE-UP) Under element 5, delete lat sentence. GENERIC COMMENT: This program should be called out as FLOW ACCELARATED CORROSION instead of erosion/corrosion program.
IVC1-5	Reactor Coolant Pressure Boundary (BWR)	C1.2.1 C1.3.1 C1.3.2	IVC-12,13,15 IVC18-21		The existing AMP and Evaluation and Technical Basis should state " same as for the effect of SSC,IGSCC on piping and fittings in various reactor coolant pressure boundary systems C1.1.5 thru C1.1.11.
IVC1-6	Reactor Coolant Pressure Boundary (BWR)	C1.2.3, C1.2.4	IVC14-17		The existing AMP column should specify ISI AMA. See GALL Insert # 8.

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. [redacted]. INSERT 3 [redacted] *Operating Experience* NRC BI 87-01. NRC GL 89-06. 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.

1. See FAC AMA attached

2. See Primary Water Chemistry AMA attached

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 in EPRI guidelines in TR-103515 and BWRVIP-79 for water chemistry in BWRs.	***(1) Scope of Program:*** The AMP includes NUMARC 97-06 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, to determine the extent of thinning and repair/replace components, and (b) 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 No Comment IVC1-1

Comment IVC1-4

1. Flow Accelarated Corrosion Aging Management Activities – implementation of EPRI guidelines of NSAC-202L-R2 for effective erosion/corrosion program

2. Primary water chemistry AMA minimizes impurities by monitoring and maintaining the water chemistry conditions based on guidelines of EPRI TR-103515R2 (BWRVIP-79) for primary water chemistry.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

C1. REACTOR COOLANT PRESSURE BOUNDARY (Boiling Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C1.1.1	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. [redacted]
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. [redacted]
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. [redacted]

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 make long-term corrective actions such as adjustment of chemistry and operating parameters or selection of materials resistant to E/C. Now the NRC staff has identified weakness or errors in repositioning 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 [redacted]	Yes TLAA Move to TLAA Section X
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 [redacted]	Yes TLAA Move to TLAA Section X
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 [redacted]	Yes TLAA Move to TLAA Section X

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
C1. REACTOR COOLANT PRESSURE BOUNDARY (Boiling Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C1.1.5 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. [redacted] INSERT 4 ~~BWRVIP-29~~ EPRI ~~TR-~~103515. INSERT 3 BWRVIP-42. BWRVIP-56. BWRVIP-61. BWRVIP-75. *Supporting BWRVIP:* BWRVIP-03. BWRVIP-61 BWRVIP-62
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-108092. [redacted] INSERT 4

Add: See ISI AMA
See Primary Water Chemistry AMA

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Program delineated in NUREG-0313, Rev. 1 and NRC Generic Letter GL 88-01 and its Supplement 1. Service inspection conforms with ASME Section XI, as 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 inspection category B-P for system leakage. The 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 mitigation measures for 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 supports welds. NUREG-0313 and GL 88-01, respectively, describe the technical basis and staff guidance for managing 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 GL 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

or BWRVIP-75

ISI in conformance with ASME Section XI (edition specified in 10 CFR 50.55a) for all Class 1, 2 and 3 pressure retaining components, as amended by CLB.
Primary Water Chemistry AMA

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 required. 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 positions 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	

SEE ITEM C1.1

SEE ITEM C1.1

FOR PRIMARY WATER CHEMISTRY AMP

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. [redacted]
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. [redacted]
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. [redacted]
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. Insert 4
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. [redacted] Insert 3

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, [redacted]	Yes TLAA Move to Section X
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, [redacted]	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, [redacted]	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-2 for pump casing welds and B-L-3 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 for 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 grade of cast austenitic stainless steel with a maximum carbon of 0.035% and minimum 7.5% ferrite. Also, hydrogen water chemistry and stringent control of conductivity [illegible] 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 on 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

Same as for the effect of SCC, IGSCC on piping and fittings in various reactor coolant pressure boundary systems C1.1.5 thru C1.11

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. See Inlet 4

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 leak test is conducted prior to plant startup following a 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 improvement in all elements that cause IGSCC, e.g., susceptible material, significant tensile stress, and 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 are implemented through Generic Letter 91-17 and additional guidelines 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 of the 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 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

Same as for C1.1.5 Thru 1.11

Replace with ISI AMP

See ISI AMA writeup

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. Insert 4
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. [redacted]

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 studs, nuts, washers, bushings, and flanges. Examination category B-G-2 for bolting 2 in. or smaller specifies only visual VT-1 examination of studs, of bolts, studs, and nuts. However, based on most failures that occur 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 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 [redacted]	Yes TLAA Move to Section X

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

C1. REACTOR COOLANT PRESSURE BOUNDARY (Boiling Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C1.3.1	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, [redacted] Insert 3
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. Insert 4
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. Insert 3 BWRVIP-75

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 Section XI (ed[illegible] 10 CFR 50.55a) Subsection IWB Table IWB 2500-1, examination categories B-M-1 for valve body, B-M-2 for valve body, and 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. Same as for the effects of SCC, IASCC on piping and fittings in various reactor coolant pressure boundary system C1.1.5 thru C1.11	*(1) Scope of Program:* The program includes preventive measures to mitigate stress corrosion cracking (SCC) and inservice inspection (ISI) to monitor the effect 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 materials that are resistant to sensitization and IGSCC, e.g., low-carbon grades of cast SS 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

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

C1. REACTOR COOLANT PRESSURE BOUNDARY (Boiling Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C1.3.3, 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.

SEE INSERT 4

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) evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400; IWB-3518 for dimensional measurement of welds and -3519 for visual examination of pressure internal surfaces. *(7) Corrective Actions:* Repair and replacement are in conformance with IWA-4000 and IWB-4000 or GL 88-01, and examination in accordance with requirements of IWA-2200. Continued operation without repair require that crack growth calculations be performed according to the guidelines 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, [redacted]	Yes TLAA Move to Section X
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 Move to Section X
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

INSERT 1

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 ([redacted]) examination during system leakage test and hydrostatic test of all pressure retaining Class 2 components a[redacted]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.	

GALL REPORT

MECHANICAL DISCIPLINE TEAM REVIEW

Generic technical issues applicable to various sections:

1. Remove reference to GSI-190 relative to fatigue for Reactor vessel internals. GSI-190 has been closed by the NRC. The only open issue in GSI-190 is environmental fatigue, which may be a contributor for leakage paths. Since none of the reactor vessel internal components represent a pressure boundary, consideration of environmental effects should not be considered. **(NEI task force needs to address this generically with the NRC for Section IV.)**
2. Remove material and environment information out of the "Evaluation and Technical Basis" column and identify in the appropriate "Material" or "Environment" columns on the left hand page or delete completely. For example, for items IVA1.1.1 through IVA1.1.3 on page IVA1-5, maximum carbon content and maximum ferrite content should be deleted since it is a design/construction requirement rather than an aging management requirement. Similarly in the same section, material selection in accordance with NUREG-0313, Rev. 2, and satisfaction of material requirements in Regulatory Guide 1.43 should be identified in the "Material" column, if needed, rather than in the "Existing Aging Management Program" column.
3. TLAA is not necessarily considered an Aging Management Program. It should not be included in the GALL report. If the NRC wants to include it in the GALL report, then it should be removed from the individual sections, and moved to a separate section e.g. section X titled "TLAA". This section would then include all TLAA issues including Civil/Structural, Electrical, and Mechanical. Individual plants that use an AMP as one of the methods in evaluating TLAA's, then that plant specific application would include it in the appropriate section.
4. The generic License Renewal Technical Issues Summery report, EPRI document TR-107521, final report dated April 1998, represents an effective approach for cost-effective resolution of generic technical issues. This report has not been consistently applied in the GALL report. Some of these issues apply to mechanical sections and are addressed below:

   a. Issue 4.9 – Neutron Irradiation Embrittlement of Reactor Pressure Vessel Beltline Materials: The GALL report defines the reactor vessel beltline region differently than defined in 10 CFR 50 Appendices G and H. The GALL report should be modified to incorporate a definition of the reactor vessel beltline region that is consistent with regulations. The EPRI document contains the technical basis and industry recommendations for the issue. **(Furthermore, this is a TLAA issue and should be removed from Section IV and moved to the new TLAA section.)**
   b. Issue 4.17 – Void Swelling of Reactor Internals: The GALL report considers an aging effect of changes in dimension due to void swelling for reactor internals and considers this issue as a TLAA. (The GALL report does reference the EPRI report.) No time-dependent calculations of void swelling are contained in plant licensing bases. The industry position is that the effect (dimensional change) of potential void swelling in both BWR and PWR internal components is not considered to be significant, either during the current or an extended license term and will not compromise the intended safety function of the internals components. This aging effect should therefore be deleted from the GALL report.
5. The GALL report has not addressed BWR-VIP reports that have been approved by the NRC in Sections IVA-1 and IVB-1 for BWR vessels and internals. Comments on individual sections will address the various reports. The comments identify the various VIP references.

6. Programs such as App. J and IST, which are normally used to test active functions, should not be used for credit as AMP in the GALL report.
7. Oconee and CCNP SER information is not consistently used – see comments in individual sections
8. The GALL report is not consistent in addressing external environment. Maybe a separate section could be considered for external environment.
9. References call out IN's; which are Information Notices that do not require any response from utilities. Unless they are specifically used in the section, IN's should not be referenced. We do not credit aging management activities by IN's and do not intend to. IN's should only be used in the Operating Experience element 10 of the "Evaluation and Technical Basis" column and only these IN's should be referenced. Same information applies for GESIL information letters.

GALL INSERTION #'S FOR GALL TEXT MARK-UP

1. Inservice Inspection Program in conformance with the Edition of ASMESection XI specified in the licensees' CLB per 10CFR50.55a.

2. Coolant water Chemistry in accordance with the guidelines of EPRI TR-103515, R2 or later.

3. EPRI-103515, R2 (BWRVIP-79) or later version.

4. Replace with words "ASME Section XI 1989 Edition or later approved Editions."

5. Repair and Replacement are in conformance with IWA-4000 and IWB-4000 and reexaminations are in accordance with IWA-2200.

6. Upper shelf energy (USE) is maintained at acceptable levels for 54 effective fu;; power years (EFPY) per BWRVIP-74.

7. The BWRVIP program considered all modes of degradation for the covered components. The insurance program ensures the component will be able to perform its intended function. No additional analysis is required.

8. When ISI AMA is used: "Existing AMP" column should state "Inservice Inspection (ISI) in conformance with ASME Section XI (edition specified in 10CFR50.55a) for all Class 1, 2 and 3 pressure retaining components; as amended by CLB." "Evaluation and Technical Basis" column should state "See ISI AMA write-up."

9. When Primary Water Chemistry AMA is used: "Existing AMP" column should state "The primary water chemistry program minimizes impurities by monitoring and maintaining the water chemistry conditions based on guidelines of EPRI TR-105714 and EPRI TR-103515 for primary water chemistry." The EPRI TR-103515 is for BWR plants and EPRI TR-105714 is for PWR plants. "Evaluation and Technical Basis" column should state "See Primary Water Chemistry AMA write-up."

10. When Flow Accelarated Corrosion (FAC) AMA is used: "Existing AMP" column should state "Implementation of EPRI guidelines of NSAC-202L-R2 for effective erosion/corrosion program." " Evaluation and Technical Basis" column should state "See Flow Accelarated Corrosion (FAC) AMA write-up."

11. When Boric Acid Corrosion (BAC) AMA is used: "Existing AMP" column should state "Implementation of NRC Generic Letter 88-05." "Evaluation and Technical Basis" column should state "See Boric Acid Corrosion (BAC) AMA write-up."

12. When GL 89-13 AMA is used: "Existing AMP" column should state "In response to GL 89-13, the industry implemented surveillance and control program to manage flow blockage problems in open-cycle service water systems due to biofouling, silt, mud, and corrosion products. The AMA program generally includes (1) water sampling to determine biological species. (2) inspections of piping and components for biofouling, damaged coatings and degraded material condition, (3) removal of biofouling and corrosion products, (4) and monitoring of heat exchanger performance for all safety-related heat exchangers cooled by open-cycle service water." "Evaluation and Technical Basis" column should state "See GL 89-13 AMA write-up."

13. When Closed Cooling Water chemistry AMA is used: "Existing AMP" column should state "The closed cooling water chemistry program minimizes corrosion by maintaining corrosion inhibitors based on the guidelines of EPRI TR-107396 for closed cooling water systems." "Evaluation and Technical Basis" column should state "See closed cooling water chemistry AMA write-up."

14. When Secondary Water Chemistry AMA is used: "Existing AMP" column should state "The secondary water chemistry program minimizes impurities by monitoring and maintaining the water chemistry conditions based on guidelines of EPRI TR-102134 R4 for secondary water chemistry." "Evaluation and Technical Basis" column should state "See Secondary Water Chemistry AMA write-up."

15. When Fire Water System AMA is used: "Existing AMP" column should state "Water based fire protection system piping and components are tested in accordance with applicable NFPA commitments. NFPA testing assures functionality of the systems and assures adequate flows. In addition, selected portions of the system are inspected to address concerns of corrosion, erosion and biofouling. Also, system is normally maintained at required operating pressure and is monitored such that loss of system pressure is immediately detected and corrective actions initiated.

16. When Fuel Oil Chemistry AMA is used: "Existing AMP" column should state "The plant chemistry program maintains fuel oil quality through monitoring and controlling of fuel oil impurities using the guidance of ASTM Standards D975, D1796, D2276, and D2709. If excessive water, biological activity, or particulates are found corrective actions are taken. Additionally, new fuel oil deliveries are treated with a biocide and analyzed for water content to avoid contamination of existing fuel oil stock." "Evaluation and Technical Basis" column should state "See Fuel Oil Chemistry AMA."

CHAPTER XI

EXISTING AGING MANAGEMENT PROGRAMS (AMP) AND ACTIVITIES

A.8

Inservice Inspection Program

DESCRIPTION

Inservice Inspection (ISI) in conformance with ASME Section XI (edition specified in 10CFR50.55a) for all Class 1, 2 and 3 pressure retaining components; as amended by CLB.
See existing GALL for specific ASME XI examination category.

EVALUATION AND TECHNICAL BASIS

1) **Scope of Program:** The inservice inspection program manages the cracking of pressure retaining boundary of Class 1, 2 and 3 components. The AMA utilizes systematic inspections, and corrective actions to ensure that pressure boundary integrity is maintained.
2) **Preventive Actions:** ASME Section XI provides a method of detection; no preventative actions are provided.
3) **Parameters Monitored or Inspected:** The inservice inspection program detects and sizes cracks and detects leakage by ISI. Examination and inspection requirements are specified in Tables IWB-2500-1, IWC-2500-1, or IWD-2500-1
4) **Detection of Aging Effects:** Degradation of the pressure boundary due to cracks cannot occur without crack initiation and growth; the extent and schedule of ISI assure detection of cracks before loss of intended function. Degradation of the pressure boundary due to corrosion will result in leakage through the pressure boundary; the extent and schedule of ISI assure detection of leakage before loss of intended function.
5) **Monitoring and Trending:** The inspection schedule of IWB-2400, IWC-2400 or IWD-2400 should provide for timely detection of cracks and/or leaks.
6) **Acceptance Criteria:** Any relevant conditions detected are evaluated in accordance with IWB-3000, IWC-3000 or IWD-3000.
7) **Corrective Actions:** Repair is 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 the requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of extended operation.
10) **Operating Experience:** ISI provides assurance of detection of cracks and/or pitting for examined components.

A.10

Flow Accelerated Corrosion Program

DESCRIPTION

Implementation of EPRI guidelines of NSAC-202L-R2 for effective erosion/corrosion program

EVALUATION AND TECHNICAL BASIS

1) ***Scope of Program:*** The flow accelerated corrosion program predicts, detects, monitors, and mitigates Flow Accelerated Corrosion (FAC) in plant piping. The program is described in the EPRI guidelines of NSAC-202L-R2. The program includes the following activities: (a) conduct appropriate analysis and limited baseline inspection, (b) determine the extent of thinning and repair/ replace components as appropriate, and (c) perform follow-up inspections to confirm or quantify and take longer term corrective actions if necessary.
2) ***Preventive Actions:*** The rate of FAC is affected by piping material, geometry and hydrodynamic conditions, and operating conditions such as temperature, pH, steam quality, operating hours, and dissolved oxygen content.
3) ***Parameters Monitored/Inspected:*** The program monitors the effects of FAC on the intended function of piping by measuring wall thickness by nondestructive examination and by performing analytical evaluations. The inspection program delineated in NSAC-202L stipulates ultrasonic or radiographic testing of susceptible locations based on operating conditions. 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 FAC in piping systems based on specific plant data including material and hydrodynamic and operating conditions.
4) ***Detection of Aging Effects:*** Inspection schedule of the EPRI guidelines provides for timely detection of leakage. 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:*** Inspections and analytical evaluations are performed per the EPRI guidelines. Remaining life is recalculated after each inspection, and the next scheduled inspection is determined based on this. If degradation is detected such that the wall thickness is less than the minimum allowed wall thickness, additional examinations are performed in adjacent areas to bound the thinning.
6) ***Acceptance Criteria:*** Inspection results are used to calculate number of refueling or operating cycles remaining before the component reaches design Code minimum allowable wall thickness. If calculations indicate that an area will reach design Code minimum, the component must be repaired, replaced, or reevaluated.
7) ***Corrective Actions:*** Prior to service, reevaluate, repair or replace degraded areas as required. Follow-up inspections are performed to confirm or quantify thinning and take longer-term corrective actions such as adjustment of operating parameters, or selection of materials resistant to FAC.
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 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, IN 81-28, 92-35, 95-11), and in two-phase piping in extraction steam lines (IN 89-53, 97-84) and moisture separation reheater and feedwater heater drains (IN 89-53, 91-18, 93-21, 97-84). This AMA was originally outlined in NUREG-1344 and implemented through GL 89-08. The program has evolved through industry experience and is now described in NSAC-202L. Application of this AMA has resulted in the replacement of piping identified as being subject to FAC before this degradation has challenged the pressure boundary integrity. This AMA has provided effective means of ensuring the structural integrity of high-energy carbon steel systems. NRC has audited programs based on the EPRI methodology at several plants and has determined that these activities can provide a good prediction of the onset of FAC so that timely corrective actions can be undertaken.

A.15

Primary Water Chemistry Program

DESCRIPTION

The primary water chemistry program minimizes impurities by monitoring and maintaining the water chemistry conditions based on guidelines of EPRI TR-105714 and EPRI TR-103515 for primary water chemistry.

Evaluation and technical basis

1) ***Scope Of Program***: This program relies on preventive measures to preclude loss of material due to general, crevice, and pitting corrosion and cracking due to stress corrosion cracking. The preventive measures are accomplished through periodic monitoring and controlling of know detrimental contaminants such as chlorides, fluorides (pwrs only), dissolved oxygen, and sulfate concentrations below the levels known to result in the onset and propagation of loss of material and/or cracking.
2) ***Preventive Actions***: Maintaining system chemistry to minimize contaminant concentrations to preclude loss of material due to general, crevice, and pitting corrosion and cracking due to stress corrosion cracking.
3) ***Parameters Monitored***: Environmental contaminates such as chlorides, fluorides (pwrs only), sulfates, and dissolved oxygen/hydrogen peroxide are directly or indirectly (electrochemical corrosion potential [ecp] for o_2 and h_2o_2) monitored to preclude loss of material due to corrosion and cracking. water quality is maintained per the guidance outlined in the EPRI water chemistry guidelines for primary water systems.
4) ***Detection Of Aging Effects***: The chemistry program precludes aging and requires no detection of aging effects.
5) ***Monitoring And Trending***: Contaminants such as chlorides, fluorides (pwrs only), sulfates, and dissolved oxygen/hydrogen peroxide concentrations are monitored and trended on a system specific periodic based upon the guidance outlined in the EPRI water chemistry guidelines for primary water systems.
6) ***Acceptance Criteria***: Contaminants such as chlorides, fluorides (pwrs only), sulfates, and dissolved oxygen/hydrogen peroxide concentrations are maintained below the system specific limits based on the limits specified in the epri water chemistry guidelines for primary water systems.
7) ***Corrective Actions***: Contaminants such as chlorides, fluorides (pwrs only), sulfates, and dissolved oxygen/hydrogen peroxide concentrations in excess of allowable limits must be returned to the acceptable range within the time periods specified in the EPRI water chemistry guidelines for primary water systems. Specific corrective actions to return out-of-specification contaminant levels to acceptable levels are site specific.
8) ***Confirmation Process***: Following corrective actions, additional samples are taken and analyzed to verify that the corrective actions were effective in returning contaminants such as chlorides, fluorides (pwrs only), sulfates, and dissolved oxygen/hydrogen peroxide concentrations to acceptable ranges.
9) ***Administrative Controls***: The chemistry program is controlled by a company, department, or station directive or program manual and is implemented by controlled plant procedures.
10) ***Operating Experience***: No incidents of loss of material attributed to an inadequate chemistry program for primary water have been reported. Several instances of cracking have been reported that were attributed to improper construction, fatigue, or the accidental introduction of contaminants. this good operating experience demonstrates that the chemistry program for primary water systems is effective in maintaining an environment that precludes loss of material and cracking.