NRC FORM 658
(9-1999)

U.S. NUCLEAR REGULATORY COMMISSION

TRANSMITTAL OF MEETING HANDOUT MATERIALS FOR IMMEDIATE PLACEMENT IN THE PUBLIC DOMAIN

This form is to be filled out (typed or hand-printed) by the person who announced the meeting (i.e., the person who issued the meeting notice). The completed form, and the attached copy of meeting handout materials, will be sent to the Document Control Desk on the same day of the meeting; under no circumstances will this be done later than the working day after the meeting.
Do not include proprietary materials.

DATE OF MEETING

06/06/2000

The attached document(s), which was/were handed out in this meeting, is/are to be placed in the public domain as soon as possible. The minutes of the meeting will be issued in the near future. Following are administrative details regarding this meeting:

Docket Number(s): **PROJ690**

Plant/Facility Name: **License Renewal**

TAC Number(s) (if available):

Reference Meeting Notice: **5/25/2000**

Purpose of Meeting (copy from meeting notice): **To discuss NEI comments on the draft "Generic Aging Lessons Learned" (GALL) report - Mechanical systems.**

NAME OF PERSON WHO ISSUED MEETING NOTICE

Jerry Dozier

TITLE

General Engineer

OFFICE

NRR

DIVISION

DRIP

BRANCH

RLSB

<u>Distribution of this form and attachments:</u>
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Jerry Dozier	**General Engineer**

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GALL report
WORKING DRAFT
6/6/00

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

<u>A1.7.2 Withdrawal Line</u>

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	Environment	Aging Effect	Aging Mechanism	
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)	

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
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. [Supporting documents BWRVIP-03 for reactor pressure vessel internals examination guidelines; BWRVIP-14, -59, and -60 for evaluation of crack growth; and BWRVIP-62 for technical basis for inspection relief for internal components with hydrogen injection.]	***(1) Scope of Program:*** The program is focused on managing the effects of stress corrosion cracking (SCC) of SS cladding on the intended function of top head enclosure. NUREG-0313, Rev. 2 and Generic Letter (GL) 88-01, respectively, describe the technical basis and staff guidance regarding the problem of IGSCC in BWRs. However, SCC is not anticipated to be an issue for the top head enclosure because analytical evaluations indicate that cracks in the SS cladding will stop growing in the ferritic base metal. ***(2) Preventive Actions:*** Selection of material considered resistant to IGSCC, e.g., grades of weld metal with a maximum carbon of 0.035% and minimum 7.5% ferrite, prevent or mitigate IGSCC, and Regulatory Guide (RG) 1.43 provides assurance that production cladding complies with ASME Section III and XI guidelines to prevent underclad cracking. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in BWRVIP-29 and TR-103515 to minimize the potential of crack initiation and growth. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of IGSCC on the intended function of top head enclosure by detection and sizing of cracks by inservice inspection (ISI). Inspection requirements of Table IWB 2500-1, examination category B-A specifies volumetric inspection of all circumferential and meridian 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 (Boiling Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environment	Aging Effect	Aging Mechanism	
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	
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	

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or Section I (Power Boilers) and Section VIII, Division 1 (Unfired Pressure Vessel).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed. Insert #8	Yes TLAA
Inservice inspection in conformance with ASME Section XI , edition specified in 10 CFR 50.55a, Subsection IWB, Table IWB 2500-1, examination category B-G-1, and testing category B-P for system leakage, and additional recommendations of GE Rapid Information Communication Service Information Letter (RICSIL) 055 Revision 1, Supplement 1. Prevention and replacement in accordance with Regulatory Guide 1.65.	***(1) Scope of Program:*** The program is focused on managing the effects of IGSCC on the intended function of reactor vessel closure stud bolting. ***(2) Preventive Actions:*** Design requirements of ASME Section III, Subsection NB, and additional guidance of Regulatory Guide (RG) 1.65 on material selection, preservice inspection, and protection against corrosion, prevent or mitigate IGSCC. High-strength low-alloy steels with controlled tempering procedures are used. Maximum tensile strength is limited to <1172 MPa (<170 ksi) to provide resistance to SCC, and Charpy V energy requirements of Appendix G to 10 CFR Part 50 provide adequate toughness to provide resistance to crack growth in the stud threads. Metal-plated stud bolting is avoided to prevent degradation due to corrosion or hydrogen embrittlement. Manganese phosphate or other acceptable surface treatment, or stable lubricants are permissible. Preservice inspection in conformance with NB-2580 of Section III of the Code requires ultrasonic examination of stud bolting over the entire surface prior to threading. During refueling and while the head is removed, the stud bolts and holes are protected from corrosion and contamination in accordance with RICSIL 055 R1 S1. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of IGSCC on the intended function of closure stud bolting by detection and sizing of cracks by inservice inspection (ISI). Inspection requirements of ASME Section XI, Table IWB 2500-1, examination category B-G-1, specify the following for all closure stud bolting: volumetric examination of studs in place, from top of the nut to bottom of the flange hole, and surface and volumetric examination of studs when removed; volumetric examination of flange threads; and visual VT-1 examination of surfaces of nuts, washers, and bushings. RICSIL Rev. 1 and its Supplement 1 provide additional recommendations regarding inspection and evaluation of the data. Requirements for training and qualification of personnel and performance demonstration for procedures and equipment is in conformance with Appendices VII and VIII of ASME Section XI. Inspection requirements of testing category B-P conducted according to IWA-5000 specify visual VT-2 (IWA-5240) examination of all pressure retaining components extending to and including the second closed valve at the boundary extremity, during system leakage test (IWB-5221) and system hydrostatic test (IWB-5222). ***(4) Detection of Aging Effects:*** Aging effects degradation of the closure stud bolting can not occur without crack initiation, the extent and schedule of inspection assure detection of cracks before the loss of intended function of closure stud bolting. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI	No

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	
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	
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	
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	

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) and, based on operating experience, additional requirements of RICSIL 055 Rev. 1, are effective and adequate for timely detection of cracks. All BWRs are inspected in accordance with Program B IWB-2412 which requires 100% inspection every 10 y, at least 16% in 3 y and 50% in 7 y. Recommendations of RICSIL 055 include expansion of sample size and ultrasonic examination from the center drilled hole of studs in compliance with ASME Code Case N-307-1. System leakage test is conducted prior to plant startup following each refueling outage, and hydrostatic test at or near the end of each inspection interval. ***(6) Acceptance Criteria:*** Any cracks in closure stud bolting are evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3515/17. ***(7) Corrective Actions:*** Repair and replacement is in conformance with IWB-4000 and material and inspection guidance of RG 1.65. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** SCC has occurred in BWR pressure vessel head studs. The AMP based on ASME Section XI and industry guidelines of RICSIL 055 Revision 1 and its Supplement 1, provides recommendations regarding inspection techniques and evaluation, material specifications, corrosion prevention, and other aspects of reactor pressure vessel head stud cracking, and is effective in managing the effects of SCC to maintain the intended function of closure studs and nuts during the period of license renewal.	
Plant-specific aging management program; existing programs may not be capable of mitigating or detecting SCC of vessel flange leak detection line.	Plant-specific aging management program is to be evaluated.	Yes, no AMP
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or Section I (Power Boilers) and Section VIII, Division 1 (Unfired Pressure Vessel).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed. Insert #8.	Yes TLAA
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or Section I (Power Boilers) and Section VIII, Division 1 (Unfired Pressure Vessel).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed. Insert #1.	Yes TLAA

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

A1. REACTOR VESSEL (Boiling Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environment	Aging Effect	Aging Mechanism	
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 Embrittlement	
A1.2.3 thru A1.2.6	Vessel Shell	Intermediate (Nozzle) Shell, Intermediate (Beltline) Shell, Lower Shell, Beltline Welds	SA302-Gr B, SA533-Gr B with 308, 309, 308L, 309L Cladding	288°C, Oxygenated Water, $5x10^8$ - $5x10^9$ $n/cm^2 \cdot s$	Crack Initiation and Growth	SCC, IGSCC	

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, Supplement 1, to predict effects of neutron irradiation on reactor vessel materials. In addition, inservice inspection of ASME Section XI, edition specified in 10CFR50.55a, Subsection IWB, examination category B-A of all pressure retaining welds in the vessel and repair welds in beltline region, defined as the region extending for the length of the thermal shield or effective length of reactor fuel elements. NRC Generic Letter 98-05 covers exemptions from inspection requirements for circumferential welds. [Supporting documents BWRVIP-05, -29, -74, and -78]	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 10CFR50.55a, Codes and Standards), Subsection IWB, Table IWB 2500-1, examination categories B-N-1 for vessel interior and B-A for shell welds. Prevention is by material selection in accordance with guidelines of NUREG-0313, Rev. 2, and of Regulatory Guide 1.43 for control of stainless steel weld cladding of low-alloy steels. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in BWRVIP-29 and TR-103515 to minimize the potential of crack initiation and growth. NRC Generic Letter 98-05 covers exemptions from inspection requirements for circumferential welds. BWRVIP-74 for reactor pressure vessel inspection and flawevaluation guidelines is under staff review. [Supporting documents BWRVIP-03 for reactor pressure vessel internals examination guidelines; BWRVIP-14, -59, and -60 for evaluation of crack growth; BWRVIP-44 for weld repair of Ni-alloys; BWRVIP-45 for weldability of irradiated structural components; BWRVIP-62 for technical basis for inspection relief for internal components with hydrogen injection; and BWRVIP-78 BWR integrated surveillance program.]	***(1) Scope of Program:*** The program is focused on managing the effects of stress corrosion cracking (SCC) of SS cladding on the intended function of reactor vessel shell. NUREG-0313 and GL 88-01, respectively, describe the technical basis and staff guidance regarding the problem of IGSCC in BWRs. However, SCC is not anticipated to be an issue for the vessel shell because analytical evaluations and experimental data indicate that growth of the cracks in ferritic base metal will be very slow. ***(2) Preventive Actions:*** Selection of material considered resistant to IGSCC, e.g., grades of weld metal with a maximum carbon of 0.035% and minimum 7.5% ferrite, prevent or mitigate IGSCC, and Regulatory Guide (RG) 1.43 provides assurance that production cladding complies with ASME Section III and XI guidelines to prevent underclad cracking. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in BWRVIP-29 and TR-103515 to minimize the potential of crack initiation and growth. Also, hydrogen water chemistry and stringent control of conductivity is used to inhibit IGSCC. ***(3) Parameters Monitored/Inspected:*** Inspection and flaw evaluation are to be performed in accordance with referenced BWRVIP guideline, as approved by the NRC staff. ***(4) Detection of Aging Effects:*** Aging effects degradation of the reactor vessel shell can not occur without crack initiation. However, because of inaccessibility, the extent and size of inspection may not be adequate to assure detection of cracks in the SS cladding before the loss of intended function of the reactor vessel. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with applicable approved BWRVIP guideline. ***(6) Acceptance Criteria:*** Any IGSCC degradation is evaluated in accordance with applicable approved BWRVIP guideline.	Yes BWRVIP Guideline

Item	Structure and Component	Region of Interest	Material	Environment	Aging Effect	Aging Mechanism	
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 Irradiation Embrittlement	
A1.2.7	Vessel Shell	Attachment Welds	SS, Inconel 182	288°C, Oxygenated Water	Crack Initiation and Growth	SCC, IGSCC	

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) ***(7) Corrective Actions:*** The corrective action proposed by the BWRVIP is under staff review. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** The present AMP is effective in managing crack initiation and growth due to SCC, however, because of inaccessibility, the extent and size of inspection may not be adequate to assure detection of cracks.	
Same as for the effect of Neutron Irradiation Embrittlement on Item A2.1.4 intermediate (beltline) shell.	*Same as for the effect of Neutron Irradiation Embrittlement on Item A2.1.4 intermediate (beltline) shell.*	Yes TLAA
Inservice inspection in conformance with the guidelines of BWRVIP-48 and ASME Section XI, edition specified in 10CFR50.55a, Codes and Standards), Subsection IWB, Table IWB 2500-1, examination categories B-N-2 for integrally welded core support structure. Prevention is by material selection in accordance with guidelines of NUREG-0313, Rev. 2, Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in BWRVIP-29 and TR-103515 to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes preventive measures to mitigate stress corrosion cracking (SCC) and inservice inspection (ISI) to monitor the effects of SCC on the intended function of the component. NUREG-0313 and GL 88-01, respectively, describe the technical basis and staff guidance regarding the problem of IGSCC in BWRs. ***(2) Preventive Actions:*** Mitigation is by selection of materials resistant to IGSCC and control of coolant water chemistry in accordance with EPRI guidelines in BWRVIP-29 and TR-103515 including stringent control of conductivity (many BWRs now operate at <0.15 $\mu S/cm^2$). 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	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	
A1.3.1	Nozzles	Main Steam	SA508-Cl2 with or without SS Cladding	288°C Steam	Cumulative Fatigue Damage	Fatigue	
A1.3.2 thru A1.3.9	Nozzles	HPCI, HPCS, LPCS, Recirculating Water, LPCI, Isolation Condenser Supply	SA508-Cl2 with SS Cladding	Up to 288°C, Oxygenated Water	Crack Initiation and Growth	Unantici-pated Cyclic Loading	

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. Attachment weld inspection and flaw evaluation guidelines are provided in BWRVIP-48. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 and BWRVIP-48 is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with IWB-3520 and BWRVIP-48. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWB-3140. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** IGSCC has occurred BWR components. The program addresses improvements in all three of the elements, viz., a susceptible (sensitized) material, significant tensile stress, and an aggressive environment, that cause IGSCC.	
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or Section I (Power Boilers) and Section VIII, Division 1 (Unfired Pressure Vessel).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed. Insert #8.	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-D for nozzle-to-vessel welds, and testing category B-P for system leakage. Selection of materials considered resistant to enhanced crack growth is 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. [Supporting documents BWRVIP-74 for reactor pressure vessel inspection and flaw evaluation guidelines; BWRVIP-14, -59, and -60 for evaluation of crack growth; BWRVIP-62 for technical basis for inspection relief for internal components with hydrogen injection; BWRVIP-75 for technical basis for revisions to GL 88-01 inspection schedule; and BWRVIP-78 BWR integrated surveillance program.]	***(1) Scope of Program:*** The program is focused on managing the effects of crack initiation and growth due to unanticipated cyclic loading by inservice inspection (ISI). ***(2) Preventive Actions:*** Selection of material considered resistant to to enhanced crack growth is in accordance with guidelines of NUREG-0313, Rev. 2, 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. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of crack initiation and growth 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

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	

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) Section XI, or any other formal program approved by the NRC. System leakage test, IWB-5221, is conducted prior to plant startup following each refueling outage and visual VT-2 (IWA-5240) examination performed for all pressure retaining components extending to and including the second closed valve at the boundary extremity. System hydrostatic test, IWB-5222, is conducted at or near the end of each inspection interval and visual VT-2 examination performed for all class 1 components within the boundary. ***(4) Detection of Aging Effects:*** Aging effects degradation of the reactor vessel nozzles can not occur without crack initiation; extent and schedule of inspection assure detection of cracks before the loss of intended function of the reactor vessel nozzles. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI should provide for timely detection of cracks. All BWRs are inspected in accordance with Program B IWB-2412 which requires 100% inspection every 10 y; for reactor vessel nozzles at least 25% but not more than 50% shall be examined by the end of 1st inspection interval. ***(6) Acceptance Criteria:*** Any 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:*** NUREG-0619 summarizes work performed by the NRC to resolve Generic Technical Activity A-10, "BWR Nozzle Cracking" and the industry testing and analysis program is described in GE NEDE-21821-A.	

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

A1. REACTOR VESSEL (Boiling Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environment	Aging Effect	Aging Mechanism	
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	
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 Embrittlement	
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	

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or Section I (Power Boilers) and Section VIII, Division 1 (Unfired Pressure Vessel). The technical basis and staff guidance regarding the problem of feedwater nozzle cracking due to thermal cycling is described in NUREG-0619.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed. Insert #1.	Yes TLAA
Same as for the effect of Neutron Irradiation Embrittlement on Item A2.1.4 intermediate (beltline) shell.	*Same as for the effect of Neutron Irradiation Embrittlement on Item A2.1.4 intermediate (beltline) shell.*	Yes TLAA
Program delineated in NUREG-0313, Rev. 2 and implemented through NRC Generic letter (GL) 88-01 and its Supplement 1, and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination category B-F for pressure retaining dissimilar metal welds in vessel nozzles and testing category B-P for system leakage, and additional recommendations of Nuclear Services Information Letter (SIL) No. 455, Rev. 1 and Supplement 1. BWRVIP guideline is under staff review. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in BWRVIP-29 and TR-103515 to minimize the potential of crack initiation and growth. BWRVIP-75 technical basis for revisions to GL 88-01 inspection schedule are under staff review.	***(1) Scope of Program:*** The program is focused on managing the effects of IGSCC on the intended function of austenitic stainless steel (SS) piping 4 in. or larger in diameter, and reactor vessel attachments and appurtenances. Although these guidelines primarily address austenitic SS components, they are also applied to nickel alloys. NUREG-0313 and GL 88-01, respectively, describe the technical basis and staff guidance regarding the problem of IGSCC in BWRs. ***(2) Preventive Actions:*** Mitigation of IGSCC is by selection of material considered resistant to sensitization and IGSCC, e.g., low-carbon grades of austenitic SSs and weld metal, with a maximum carbon of 0.035% and minimum 7.5% ferrite in weld metal, and by special processing such as solution heat treatment, heat sink welding, and induction heating or mechanical stress improvement (SI). Inconel 82 is the only nickel base weld metal considered to be resistant to IGSCC. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in BWRVIP-29 and TR-103515. Also, hydrogen water chemistry and stringent control of conductivity is used to inhibit IGSCC. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of IGSCC on the intended	No

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) function of reactor vessel nozzle safe ends by detection and sizing of cracks by inservice inspection (ISI). Inspection requirements of ASME Section XI, Subsection IWB, Table IWB 2500-1, examination category B-F specifies for all nozzle-to-safe end butt welds NPS 4 or larger, volumetric and surface examination of ID region extending 1/4 in. on either side of the weld and 1/3 wall thickness deep, and surface examination of OD surface extending 1/2 in. on either side. Only surface examination is conducted for all butt welds less than NPS 4. For all nozzle-to-safe end socket welds, surface examination is specified of OD surface extending 1 in. on the buttered side and 1/2 in. on the other. Requirements for training and qualification of personnel and performance demonstration for procedures and equipment is in conformance with Appendices VII and VIII of ASME Section XI, or any other formal program approved by the NRC. SIL No. 455 and Supplement 1 contain specific recommendations regarding ultrasonic testing (UT) methods for dissimilar metal welds, i.e., the use of 45-degree and 60-degree refracted longitudinal wave transducers for detecting IGSCC cracks in alloy 182 and low-alloy materials. Visual VT-2 (IWA-5240) examination is performed for all pressure retaining components during system leakage test (IWB-5221), conducted prior to plant startup following each refueling outage, and during system hydrostatic test (IWB-5222) conducted at or near the end of each inspection interval. Leakage detection is in conformance with Position C of Regulatory Guide 1.45 and additional guidelines of GL 88-01, Supplement 1. ***(4) Detection of Aging Effects:*** Aging effects degradation of the nozzle safe ends can not occur without crack initiation; extent and schedule of inspection assure detection of cracks before the loss of intended function of the reactor vessel nozzle safe ends. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI should provide for timely detection of cracks. Inspection schedule and sample size specified in Table 1 of GL 88-01 are based on the condition of each weld and are adequate for timely detection of cracks. Welds of resistant material are as a minimum examined according to an extent and frequency comparable to those of ASME Section XI, e.g., 25% are examined every 10 y, at least 12% in 6 y. Inspection extent and schedule are enhanced for welds of non-resistant materials, or welds that have been treated by stress improvement (SI) or reinforced by weld overlay. ***(6) Acceptance Criteria:*** Any IGSCC degradation is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3514. Planar and liner flaws are sized according to IWA-3300 and -3400. ***(7) Corrective Actions:*** Repair and reexaminations are in conformance with IWB-4000. Continued operation without repair requires that crack growth calculation be performed according to the guidance of GL 88-01 or other approved procedures. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and	

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	
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	
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, <u>Unantici-pated Cyclic Loading</u>	

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** IGSCC has occurred in small- and large-diameter BWR piping safe end-to-nozzle welds (IN 82-39 & IN 84-41). The present AMP has provided effective means of ensuring structural integrity of the primary coolant pressure boundary.	
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or ANSI B31.1, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed. <u>Insert #1.</u>	Yes TLAA
Program delineated in NUREG-0313, Rev. 2 and implemented through NRC Generic letter 88-01 and its Supplement 1, and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination category B-E for pressure retaining partial penetration welds and testing category B-P for system leakage. 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. <u>Inspection and flaw evaluation guidelines for instrument penetration (BWRVIP-49) and for standby liquid control system/core plate AP (BWRVIP-27) are under staff review.</u> <u>[Supporting documents for repair design criteria BWRVIP-57 for instrumentation penetrations and BWRVIP-53 for standby liquid control line; BWRVIP-14, -59, and -60 for evaluation of crack growth; BWRVIP-62 for technical basis for inspection relief for internal components with hydrogen injection; and BWRVIP-75 for technical basis for revisions to GL 88-01 inspection schedule.]</u>	***(1) Scope of Program:*** NUREG-0313 and GL 88-01, respectively, describe the technical basis and staff guidance regarding the problem of IGSCC in BWRs. The program is focused on managing the effects of IGSCC on the intended function of austenitic stainless steel (SS) piping 4 in. or larger in diameter, and reactor vessel attachments and appurtenances. Although these guidelines primarily address austenitic SS components, they are also applied to nickel alloys. ***(2) Preventive Actions:*** Mitigation of IGSCC is by selection of material considered resistant to sensitization and IGSCC, e.g., low-carbon grades of austenitic SSs and weld metal, with a maximum carbon of 0.035% and minimum 7.5% ferrite in weld metal, and by special processing such as solution heat treatment, heat sink welding, and induction heating or mechanical stress improvement. Inconel 82 is the only nickel base weld metal considered to be resistant to IGSCC. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in BWRVIP-29 and TR-103515. Also, hydrogen water chemistry and stringent control of conductivity is used to inhibit IGSCC. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of IGSCC on the intended function of reactor vessel penetrations by detection and sizing of cracks by inservice inspection (ISI). System leakage test, IWB-5221, is conducted prior to plant startup following each refueling outage and visual VT-2 (IWA-5240) examination performed for all pressure retaining components extending to and including the second closed valve at the boundary extremity. Leakage detection is in conformance with Position C of Regulatory Guide 1.45 and additional guidelines of GL 88-01, Suppl. 1. System hydrostatic test, IWB-5222, is conducted at or near the end of each inspection interval and visual VT-2 examination performed for all class 1 components within boundary. Inspection requirements of examination category B-E focus on visual VT-2 examination of partial penetration welds during the hydrostatic test. ***(4) Detection of Aging Effects:*** Aging effects degradation of the reactor vessel penetrations can not occur without crack initiation; extent and schedule of inspection assure detection of cracks before loss of intended function of the reactor vessel penetrations. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI should provide for timely detection of	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	
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	
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	
A1.7.1	Control Rod Drive (CRD) Mechanism	Housing	SS	Up to 288°C, Oxygenated Water	Crack Initiation and Growth	SCC, IGSCC	

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) cracks. Inspection schedule and sample size specified in Table 1 of GL 88-01 are based on the condition of each weld and are adequate for timely detection of cracks. Welds of resistant material are as a minimum examined according to an extent and frequency comparable to those of ASME Section XI. Inspection extent and schedule are enhanced for welds of non-resistant materials. ***(6) Acceptance Criteria:*** Any IGSCC degradation is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3522. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWA-4000 and IWB-4000, and reexamination in accordance with requirements of IWA-2200. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** The program addresses improvements in all three of the elements, viz., a susceptible (sensitized) material, significant tensile stress, and an aggressive environment, that cause IGSCC, and has provided effective means of ensuring structural integrity of the primary coolant pressure boundary.	
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. Insert #1.	Yes TLAA
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or Section I (Power Boilers) and Section VIII, Division 1 (Unfired Pressure Vessel).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed. Insert #1.	Yes TLAA
Inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination categories B-O for pressure retaining welds in control rod housings and testing category B-P for system leakage, and BWRVIP-27. 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 guideline is under staff review.	***(1) Scope of Program:*** The program is focused on managing the effects of stress corrosion cracking (SCC) on the intended function of CRD mechanism housing. ***(2) Preventive Actions:*** Mitigation of IGSCC is by selection of material considered resistant to sensitization and IGSCC, e.g., low-carbon grades of austenitic SSs and weld metal, with a maximum carbon of 0.035% and minimum 7.5% ferrite in weld metal, and by special processing such as solution heat treatment, heat sink welding, and induction heating or mechanical stress improvement. Inconel 82 is the only nickel base weld metal considered to be resistant to IGSCC. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in BWRVIP-29 and TR-103515. Also, hydrogen water chemistry and stringent control of	Yes, BWRVIP Guideline (Element 7)

Item	Structure and Component	Region of Interest	Material	Environment	Aging Effect	Aging Mechanism	
A1.7.1	CRD Mechanism	Housing	SS	Up to 288°C Oxygenated Water	Cumulative Fatigue Damage	Fatigue	
A1.8	Support Skirt & Attachment Welds	-	SA533-Gr B (Welds SS or Inconel 182)	Ambient Temperature Air	Cumulative Fatigue Damage	Fatigue	
A1.7.2	CRD Mechanism	Withdrawal Line (Outside Surface)	SS	Air, Leaking Oxygenated Water up to 288°C	Crack Initiation and Growth	Stress Corrosion Cracking	

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
[Supporting documents [BWRVIP-58 for CRD internal access weld repair; BWRVIP-14, -59, and -60 for evaluation of crack growth; BWRVIP-62 for technical basis for inspection relief for internal components with hydrogen injection; and BWRVIP-53 for standby liquid control line repair design criteria.]	*(continued from previous page)* conductivity is used to inhibit IGSCC. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of IGSCC on the intended function of CRD mechanism housing by detection and sizing of cracks by inservice inspection (ISI). Inspection requirements of Table IWB 2500-1, examination category B-O specifies volumetric or surface examination extending 1/2 in. each side of the CRD housing welds, including weld buttering. ***(4) Detection of Aging Effects:*** Aging effects degradation of the CRD mechanism housing can not occur without crack initiation; the extent and schedule of inspection assure detection of cracks before the loss of intended function of the CRD housing. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with Program B IWB-2412 should provides timely detection of cracks. 10% peripheral CRD housings are examined each inspection interval. ***(6) Acceptance Criteria:*** Any IGSCC degradation is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3523. Planar and liner flaws are sized according to IWA-3300 and IWA-3400. ***(7) Corrective Actions:*** The corrective action proposed by the BWRVIP is under staff review. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** The program addresses improvements in all three of the elements, viz., a susceptible (sensitized) material, significant tensile stress, and an aggressive environment, that cause IGSCC, and has provided effective means of ensuring structural integrity of the primary coolant pressure boundary.	
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or other evaluations.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed. Insert #1.	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
The chlorides from insulation and other sources can cause externally-initiated transgranular stress corrosion cracking (TGSCC) in the stainless steel lines. Plant specific aging management program should be implemented.	Plant specific aging management program is to be evaluated.	Yes, no generic AMP

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. 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).

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	
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)	
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	
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	Stress Corrosion Cracking (SCC), Irradiation Assisted Stress Corrosion Cracking (IASCC)	

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. 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. BWRVIP-07 and -63 for inspection and evaluation of core shrouds and BWRVIP-76 for ??? are under staff review. [Supporting documents BWRVIP-03 for reactor pressure vessel internals examination guidelines; BWRVIP-14, -59, and -60 for evaluation of crack growth; BWRVIP-44 for weld repair of Ni-alloys; BWRVIP-45 for weldability of irradiated structural components; and BWRVIP-62 for technical basis for inspection relief for internal components with hydrogen injection.]	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC, inservice inspection (ISI) to monitor the effects of SCC on the intended function of the components, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** Maintaining high water purity (many BWRs now operate at <0.15 μS/cm^2) reduces susceptibility to SCC. Hydrogen additions are effective in reducing electrochemical potentials in the recirculation piping system, but are less effective in the core region. Noble metal additions through a catalytic action ~~appear to~~ increase the effectiveness of hydrogen additions in the core region. ~~but only limited data are available at present to demonstrate their effectiveness~~. GE Services Information Letter (SIL) 433 recommends replacement of bolts with crevice-free design. ***(3) Parameters Monitored/Inspected:*** Inspection and flaw evaluation are to be performed in accordance with referenced BWRVIP guideline, as approved by the NRC staff. ***(4) Detection of Aging Effects:*** Degradation due to SCC can not occur without crack initiation and growth, inspection schedule assures detection of cracks before the loss of intended function of the component. ***(5) Monitoring and Trending:*** Schedule in accordance with applicable, approved BWRVIP guideline is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with applicable, approved BWRVIP guideline. ***(7) Corrective Actions:*** The corrective action proposed by the BWRVIP is under staff review. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** The present AMP has been effective in managing the effects of SCC on the intended function of core shroud head bolts.	Yes, BWRVIP Guideline
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed. Insert #1.	Yes, TLAA
Visual inspection (VT-3) is performed according to ASME Section XI, IWB-2500, category B-N-2. Guidance for enhanced VT-1 and UT inspections in plant specific programs. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-103515 and BWRVIP-29 to minimize the potential of crack initiation and growth. Plant programs also may include water chemistry measures such as strict controls on conductivity, hydrogen addition, and use of noble metal additions such as palladium or	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC, inservice inspection (ISI) to monitor the effects of SCC on the intended function of the components, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** Maintaining high water purity (many BWRs now operate at <0.15 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.	Yes, BWRVIP Guideline

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	
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	

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
(continued from previous page) platinum to reduce electrochemical potential. Either preventive or restorative mechanical repairs may be made to the shroud. Possible inspection relief based on hydrogen injection is currently under staff review. <u>BWRVIP-07 and -63 for inspection and evaluation of core shrouds and BWRVIP-76 for ??? are under staff review. [Supporting documents BWRVIP-03 for reactor pressure vessel internals examination guidelines; BWRVIP-14, -59, and -60 for evaluation of crack growth; BWRVIP-44 for weld repair of Ni-alloys; BWRVIP-45 for weldability of irradiated structural components; and BWRVIP-62 for technical basis for inspection relief for internal components with hydrogen injection.]</u>	*(continued from previous page)* ***(3) Parameters Monitored/Inspected:*** Inspection and flaw evaluation are to be performed in accordance with referenced BWRVIP guideline, as approved by the NRC staff. ***(4) Detection of Aging Effects:*** Degradation due to SCC can not occur without crack initiation and growth. Extensive cracking has been observed at both horizontal [NRC Generic Letter (GL) 94-03] and vertical [NRC Information Notice (IN) 97-17] welds. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with applicable, approved BWRVIP guideline is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with applicable, approved BWRVIP guideline. ***(7) Corrective Actions:*** The corrective action proposed by the BWRVIP is under staff review. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Cracking has occurred in a number of BWRs. It has affected shrouds fabricated from Type 304 SS and Type 304L SS, which is generally considered to be more resistant to SCC. Weld regions are most susceptible, although it is not clear whether this is due to sensitization and/or impurities associated with the welds or the high residual stresses in the weld regions. This experience is reviewed in GL 94-03 and NUREG-1544. Some experiences with visual inspections are discussed in IN 94-42.	
Visual inspection (VT-3) is performed according to ASME Section XI, IWB-2500, category B-N-2 <u>or BWRVIP-03 guidelines (EVT-1)</u>. Guidance for enhanced VT-1 and UT inspections in plant specific programs. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-103515 and BWRVIP-29 to minimize the potential of crack initiation and growth. Plant programs also may include water chemistry measures such as strict controls on conductivity, hydrogen addition, and use of noble metal additions such as palladium or platinum to reduce electrochemical potential. Possible inspection relief based on hydrogen injection is currently under staff review. BWRVIP-25 for core plate inspection and flaw evaluation guidelines is under staff review.	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC, inservice inspection (ISI) to monitor the effects of SCC on the intended function of the components, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** Maintaining high water purity (many BWRs now operate at <0.15 $\mu S/cm^2$) reduces susceptibility to SCC. Hydrogen additions are effective in reducing electrochemical potentials in the recirculation piping system, but are less effective in the core region. Noble metal additions through a catalytic action ~~appear to~~ increase the effectiveness of hydrogen additions in the core region, ~~but only limited data are available at present to demonstrate their effectiveness~~. ***(3) Parameters Monitored/ Inspected:*** Inspection and flaw evaluation are to be performed in accordance with referenced BWRVIP guideline, as approved by the NRC staff. ***(4) Detection of Aging Effects:*** Degradation due to SCC can not occur without crack initiation and growth. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with	Yes, BWRVIP Guideline

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	
B1.1.3	Core Shroud, Shroud Head and Core Plate	Core Plate	SS	288°C, High-Purity Water	Cumulative Fatigue Damage	Fatigue	
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, IASCC	

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
[Supporting documents BWRVIP-03 for reactor pressure vessel internals examination guidelines; BWRVIP-07 and -63 for inspection and evaluation of core shrouds; BWRVIP-76 for ??? ; BWRVIP-14, -59, and -60 for evaluation of crack growth; BWRVIP-44 for weld repair of Ni-alloys; BWRVIP-45 for weldability of irradiated structural components; and BWRVIP-62 for technical basis for inspection relief for internal components with hydrogen injection.]	*(continued from previous page)* applicable, approved BWRVIP guideline is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with applicable, approved BWRVIP guideline. ***(7) Corrective Actions:*** The corrective action proposed by the BWRVIP is under staff review. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Cracking of the core plate has not been reported, but the creviced regions beneath the plate are difficult to inspect. NRC Information Notice (IN) 95-17 discusses cracking in top guides of the U.S. and overseas BWRs. Related experience in other components is reviewed in NRC GL 94-03 and NUREG-1544.	
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. Insert #1.	Yes TLAA
Visual inspection (VT-3) is performed according to ASME Section XI, IWB-2500, category B-N-2. GE Services Information Letter (SIL) 462 Sup. 3 recommends ultrasonic inspection techniques. Implementation of inspection program is plant specific. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-103515 and BWRVIP-29 to minimize the potential of crack initiation and growth. Plant programs also may include water chemistry measures such as strict controls on conductivity, hydrogen addition, and use of noble metal additions such as palladium or platinum to reduce electrochemical potential. [Supporting documents BWRVIP-03 for reactor pressure vessel internals examination guidelines; BWRVIP-14, -59, and -60 for evaluation of crack growth; BWRVIP-44 for weld repair of Ni-alloys; BWRVIP-45 for weldability of irradiated structural components; and BWRVIP-62 for technical basis for inspection relief for internal components with hydrogen injection.]	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC, inservice inspection (ISI) to monitor the effects of SCC on the intended function of the components, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** Maintaining high water purity (many BWRs now operate at <0.15 $\mu S/cm^2$) reduces susceptibility to SCC. Hydrogen additions are effective in reducing electrochemical potentials in the recirculation piping system, but are less effective in the core region. Noble metal additions through a catalytic action ~~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

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	
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	

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with IWB-3520. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWB-3140. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Jet pump boiling water reactors (BWRs) are designed with access holes in the shroud support plate at the bottom of the annulus between the core shroud and the reactor vessel wall. These holes are used for access during construction and are subsequently closed by welding a plate over the hole. Both circumferential (IN 88-03) and radial cracking (IN 92-57) has been observed in the access hole cover.	
Visual inspection (VT-3) is performed according to ASME Section XI, IWB-2500, category B-N-2. GE Services Information Letter (SIL) 462 Sup. 3 recommends ultrasonic inspection techniques. Implementation of inspection program is plant specific. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-103515 and BWRVIP-29 to minimize the potential of crack initiation and growth. Plant programs also may include water chemistry measures such as strict controls on conductivity, hydrogen addition, and use of noble metal additions such as palladium or platinum to reduce electrochemical potential. BWRVIP-38 for shroud support inspection and flaw evaluation guidelines is under staff review. [Supporting documents BWRVIP-03 for reactor pressure vessel internals examination guidelines; BWRVIP-52 for shroud support and vessel bracket repair design criteria; BWRVIP-14, -59, and -60 for evaluation of crack growth; BWRVIP-44 for weld repair of Ni-alloys; BWRVIP-45 for weldability of irradiated structural components; and BWRVIP-62 for technical basis for inspection relief for internal components with hydrogen injection.]	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC, inservice inspection (ISI) to monitor the effects of SCC on the intended function of the components, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** Maintaining high water purity (many BWRs now operate at $<0.15\ \mu S/cm^2$) reduces susceptibility to SCC. Hydrogen additions are effective in reducing electrochemical potentials in the recirculation piping system, but are less effective in the core region. Noble metal additions through a catalytic action ~~appear to~~ increase the effectiveness of hydrogen additions in the core region, ~~but only limited data are available at present to demonstrate their effectiveness~~.Also, the susceptibility of Ni-alloys to SCC is evaluated. ***(3) Parameters Monitored/ Inspected:*** Inspection and flaw evaluation are to be performed in accordance with referenced BWRVIP guideline, as approved by the NRC staff. ***(4) Detection of Aging Effects:*** Degradation due to SCC can not occur without crack initiation and growth. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with applicable, approved BWRVIP guideline is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with applicable, approved BWRVIP guideline. ***(7) Corrective Actions:*** The corrective action proposed by the BWRVIP is under staff review. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Both circumferential (IN 88-03) and radial cracking (IN 92-57) has been observed in the Ni-alloy components.	Yes, BWRVIP Guideline

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	
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, IASCC	
B1.1.8	Core Shroud, Shroud Head and Core Plate	LPCI Coupling	SS	288°C, High-Purity Water	Crack Initiation and Growth	SCC, IASCC	

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Visual inspection (VT-3) is performed according to ASME Section XI, IWB-2500, category B-N-2. Guidance for enhanced VT-1 inspections and UT inspections in plant specific programs and BWRVIP-03. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-103515 and BWRVIP-29 to minimize the potential of crack initiation and growth. Plant programs also may include water chemistry measures such as strict controls on conductivity, hydrogen addition, and use of noble metal additions to reduce electrochemical potential. BWRVIP-27 for standby liquid control system/core plate ΔP inspection and flaw evaluation guidelines is under staff review. [Supporting documents BWRVIP-03 for reactor pressure vessel internals examination guidelines; BWRVIP-53 for standby liquid control line repair design criteria; BWRVIP-14, -59, and -60 for evaluation of crack growth; BWRVIP-44 for weld repair of Ni-alloys; BWRVIP-45 for weldability of irradiated structural components; and BWRVIP-62 for technical basis for inspection relief for internal components with hydrogen injection.]	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC, inservice inspection (ISI) to monitor the effects of SCC on the intended function of the components, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** Maintaining high water purity (many BWRs now operate at <0.15 $\mu S/cm^2$) reduces susceptibility to SCC. Hydrogen additions are effective in reducing electrochemical potentials in the recirculation piping system, but are less effective in the core region. Noble metal additions through a catalytic action ~~appear to~~ increase the effectiveness of hydrogen additions in the core region, ~~but only limited data are available at present to demonstrate their effectiveness~~. ***(3) Parameters Monitored/ Inspected:*** Inspection and flaw evaluation are to be performed in accordance with referenced BWRVIP guideline, as approved by the NRC staff. ***(4) Detection of Aging Effects:*** Degradation due to SCC can not occur without crack initiation and growth. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with applicable, approved BWRVIP guideline is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with applicable, approved BWRVIP guideline. ***(7) Corrective Actions:*** The corrective action proposed by the BWRVIP is under staff review. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Cracking has occurred in a number of vessel internal components. Weld regions are most susceptible, although it is not clear whether this is due to sensitization and/or impurities associated with the welds or the high residual stresses in the weld regions.	Yes, BWRVIP Guideline
Visual inspection (VT-3) is performed according to ASME Section XI, IWB-2500, category B-N-2. Guidance for enhanced VT-1 inspections and UT inspections in plant specific programs. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-103515 and BWRVIP-29 to minimize the potential of crack initiation and growth. Plant	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC, inservice inspection (ISI) to monitor the effects of SCC on the intended function of the components, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** Maintaining high water purity (many BWRs now operate at <0.15 $\mu S/cm^2$) reduces susceptibility to SCC. Hydrogen additions are effective in reducing electrochemical potentials in the recirculation piping system, but are less effective in the core region.	Yes, BWRVIP Guideline

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	
B1.2	Top Guide	Top Guide	SS	288°C, High-Purity Water	Crack Initiation and Growth	SCC, IASCC	

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
(continued from previous page) programs also may include water chemistry measures such as strict controls on conductivity, hydrogen addition, and use of noble metal additions to reduce electrochemical potential. BWRVIP-42 for LPCI coupling inspection and flaw evaluation guidelines is under staff review. [Supporting documents BWRVIP-03 for reactor pressure vessel internals examination guidelines; BWRVIP-56 for LPCI coupling repair design criteria; BWRVIP-14, -59, and -60 for evaluation of crack growth; BWRVIP-44 for weld repair of Ni-alloys; BWRVIP-45 for weldability of irradiated structural components; and BWRVIP-62 for technical basis for inspection relief for internal components with hydrogen injection.]	*(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.	
Visual inspection (VT-3) is performed according to ASME Section XI, IWB-2500, category B-N-2. Guidance for enhanced VT-1 inspections and UT inspections in plant specific programs. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-103515 and BWRVIP-29 to minimize the potential of crack initiation and growth. Plant programs also may include water chemistry measures such as strict controls on conductivity, hydrogen addition, and use of noble metal additions such as palladium or platinum to reduce electrochemical potential. BWRVIP-26 for top guide inspection and flaw evaluation guidelines is under staff review. [Supporting documents BWRVIP-03 for reactor pressure vessel internals examination guidelines; BWRVIP-50 for top guide/core plate repair design criteria; BWRVIP-14, -59, and -60 for	***(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/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. Noble metal additions through a catalytic action ~~appear to~~ increase the effectiveness of hydrogen additions in the core region. ~~but only limited data are available at present to demonstrate their effectiveness~~. ***(3) Parameters Monitored/ Inspected:*** Inspection and flaw evaluation are to be performed in accordance with referenced BWRVIP guideline, as approved by the NRC staff. ***(4) Detection of Aging Effects:*** Degradation due to SCC can not occur without crack initiation and growth. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with applicable, approved BWRVIP guideline is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with applicable, approved BWRVIP guideline. ***(7) Corrective Actions:*** The corrective action proposed by the BWRVIP is under staff	Yes, BWRVIP Guideline

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	
B1.2	Top Guide	Top Guide	SS	288°C, High-Purity Water	Cumulative Fatigue Damage	Fatigue	
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, IASCC	

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
(continued from previous page) evaluation of crack growth; BWRVIP-44 for weld repair of Ni-alloys; BWRVIP-45 for weldability of irradiated structural components; and BWRVIP-62 for technical basis for inspection relief for internal components with hydrogen injection.]	*(continued from previous page)* 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. Insert #1.	Yes TLAA
Implementation of the program delineated in NUREG–0619 including inservice inspection (ISI) requirements (ultrasonic, visual and dye penetrant inspections) which depend upon specific plant design and other plant actions (monitoring, etc.). An update to NUREG-0619 with qualified UT inspection methods has been approved by the NRC staff. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-103515 and BWRVIP-29 to minimize the potential of crack initiation and growth. Plant programs also may include water chemistry measures such as strict controls on conductivity, hydrogen addition, and use of noble metal additions such as palladium or platinum to reduce electrochemical potential.	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC, inservice inspection (ISI) to monitor the effects of SCC on the intended function of the components, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** Maintaining high water purity (many BWRs now operate at <0.15 $\mu S/cm^2$) reduces susceptibility to SCC. Hydrogen additions are effective in reducing electrochemical potentials in the recirculation piping system, but are less effective in the core region. Design features aimed at mitigating thermal fatigue cracking, which has been the primary source of degradation for these components, have been implemented as per NUREG–0619. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of SCC on the intended function by detection and sizing of cracks by inservice inspection (ISI). ***(4) Detection of Aging Effects:*** Degradation due to SCC can not occur without crack initiation and growth. An update to NUREG-0619 with qualified UT inspection methods has been approved by the NRC staff. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with NUREG–0619 is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with IWB-3520. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWB-3140. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** NUREG–0619 summarizes work performed by the NRC to resolve Generic Technical Activity A-10, "BWR Nozzle Cracking" and the industry	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

B1. REACTOR VESSEL INTERNALS (Boiling Water Reactor)

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

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.	
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. Insert #1.	Yes TLAA
Implementation of the program delineated in the NRC Inspection and Enforcement Bulletin (IEB) 80-13 including enhanced visual inspection techniques to supplement or replace visual inspection (VT-3) requirement of GE Services Information Letter (SIL) 289. BWRVIP-18 for core spray internals inspection and flaw evaluation guidelines has been approved by the staff. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-103515 and BWRVIP-29 to minimize the potential of crack initiation and growth. Plant programs also may include water chemistry measures such as strict controls on conductivity, hydrogen addition, and use of noble metal additions such as palladium or platinum to reduce electrochemical potential. [Supporting documents BWRVIP-03 for reactor pressure vessel internals examination guidelines; BWRVIP-16 and -19 for internal core spray piping and sparger replacemnet and repair design criteria; BWRVIP-14, -59, and -60 for evaluation of crack growth; BWRVIP-44 for weld repair of Ni-alloys; BWRVIP-45 for weldability of irradiated structural components; and BWRVIP-62 for technical basis for inspection relief for internal components with hydrogen injection.]	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC, inservice inspection (ISI) to monitor the effects of SCC on the intended function of the components, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** Maintaining high water purity (many BWRs now operate at <0.15 $\mu S/cm^2$) reduces susceptibility to SCC. Hydrogen additions are effective in reducing electrochemical potentials in the recirculation piping system, but are less effective in the core region. ***(3) Parameters Monitored/ Inspected:*** Inspection and flaw evaluation are to be performed in accordance with referenced BWRVIP guideline, as approved by the NRC staff. ***(4) Detection of Aging Effects:*** Degradation due to SCC can not occur without crack initiation and growth. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with applicable, approved BWRVIP guideline is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** In the event cracks are identified, an evaluation is performed in accordance with applicable, approved BWRVIP guideline. ***(7) Corrective Actions:*** Corrective actions in accordance with applicable, approved BWRVIP-16 and BWRVIP-19 guidelines are adequate. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** IEB 80-13 reviews instances of cracking in core spray spargers.	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

B1. REACTOR VESSEL INTERNALS (Boiling Water Reactor)

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

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. Insert #1.	Yes TLAA
Implementation of inspection and surveillance programs delineated in NRC Inspection and Enforcement Bulletin (IEB) 80-07 and GE Services Information Letter (SIL) 330 to ensure overall functionality and integrity of jet pump assemblies, and additional recommendations of GE Services Information Letter (SIL) 605 Rev. 1 for jet pump riser pipe. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-103515 and BWRVIP-29 to minimize the potential of crack initiation and growth. Plant programs also may include water chemistry measures such as strict controls on conductivity, and hydrogen addition. BWRVIP-41 for jet pump assembly inspection and flaw evaluation guidelines and BWRVIP-28 for assessment of jet pump riser elbow to thermal sleeve weld cracking are under staff review. [Supporting documents BWRVIP-03 for reactor pressure vessel internals examination guidelines; BWRVIP-51 for jet pump repair design criteria; BWRVIP-14, -59, and -60 for evaluation of crack growth; BWRVIP-44 for weld repair of Ni-alloys; BWRVIP-45 for weldability of irradiated structural components; and BWRVIP-62 for technical basis for inspection relief for internal components with hydrogen injection.]	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC, performance assessment and periodic inservice inspection (ISI) to monitor the effects of SCC on the intended function of the components, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** Maintaining high water purity (many BWRs now operate at <0.15 $\mu S/cm^2$) reduces susceptibility to SCC. Hydrogen additions are effective in reducing electrochemical potentials in the recirculation piping system, but are less effective in the core region. ***(3) Parameters Monitored/Inspected:*** Inspection and flaw evaluation are to be performed in accordance with referenced BWRVIP guideline, as approved by the NRC staff. ***(4) Detection of Aging Effects:*** Degradation due to SCC can not occur without crack initiation and growth, or degradation of jet pump operation. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with applicable, approved BWRVIP guideline is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation in jet pump operation is evaluated in accordance with applicable, approved BWRVIP guideline. ***(7) Corrective Actions:*** The corrective action proposed by the BWRVIP is under staff review. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Instances of cracking have occurred in jet pump assembly (NRC IEB 80-07), hold-down beam (NRC Information Notice (IN) 93-101), and jet pump riser pipe elbows (NRC IN 97-02).	Yes, BWRVIP Guideline

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	
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	
B1.5.4	Jet Pump Assemblies	Castings	CASS	288°C, High-Purity Water	Loss of Fracture Toughness	Thermal Aging and Neutron Irradiation Embrittle-ment	

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. Insert #1.	Yes TLAA
The reactor vessel internals receive a visual inspection (VT-3) according to Category B-N-3 of Subsection IXB, ASME Section XI. This inspection is not sufficient to detect the effects of loss of fracture toughness due to thermal aging and neutron irradiation embrittlement. An acceptable alternative AMP consists of the following: Determination of the susceptibility of CASS components to thermal aging embrittlement based on casting method, Mo content, and percent ferrite. For "potentially susceptible" components, based on the neutron fluence of the component, implement either a supplemental examination of the affected components as part of the applicant's 10-year inservice inspection (ISI) program during the license renewal term or a component-specific evaluation to determine the susceptibility to loss of fracture toughness.	For the acceptable alternative AMP: ***(1) Scope of Program:*** The program includes determination of the susceptibility of CASS components to thermal aging based on casting method, Mo content, and percent ferrite, and for "potentially susceptible" components, to account for the synergistic loss of fracture toughness due to neutron embrittlement and thermal aging embrittlement, implement either a supplemental examination of the affected components as part of a 10-year ISI program during the license renewal term or a component-specific evaluation to determine the susceptibility to loss of fracture toughness. ***(2) Preventive Actions:*** The program provides no guidance on methods to mitigate thermal aging or neutron embrittlement. ***(3) Parameters Monitored/ Inspected:*** The program specifics depend on the neutron fluence and ferrite content of the component. Based on the criteria in NUREG-1705 , the susceptibility to thermal aging embrittlement of CASS piping is determined in terms of casting method, Mo content, and ferrite content. For low-Mo content (0.5 wt.% max.) steels, only static-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement, static-cast steels with ≤20% ferrite and all centrifugal-cast steels 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 Hull's equivalent factors or a method producing an equivalent level of accuracy (±6% deviation between measured and calculated values). Insert #3. ***(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 renewal period including supplemental inspection covering 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	Yes, the existence of a suitable AMP should be evaluated

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	
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	
B1.6.1	Fuel Supports & CRD Assemblies	Orificed Fuel Support	SS, CASS	288°C, High-Purity Water	Cumulative Fatigue Damage	Fatigue	

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) (lighting and surface cleanness) for the ISI bounded by those used to demonstrate the resolution of the inspection technique. Alternatively, perform a component-specific evaluation including a mechanical loading assessment to determine the maximum tensile loading on the component during ASME Code Level A, B, C, and D conditions. If the loading is compressive or low enough (<5 ksi) to preclude fracture, then supplemental inspection of the component is not required. Failure to meet this criteria requires continued use of supplemental inspection program. For all CASS components that have a neutron fluence of less than 10^{17} n/cm^2 (E>1 MeV), implement the supplement inspection program if they are "potentially susceptible" to thermal aging; the existing ASME Section XI inspection requirements are considered adequate if the components are "not susceptible" to thermal aging. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 should provide timely detection of cracks. ***(6) Acceptance Criteria:*** Flaws detected in CASS components are evaluated in accordance with the applicable procedures of IWB-3600. ***(7) Corrective Actions:*** Repair is in conformance with IWA-4000 and IWB-4000, and replacement according to IWA-7000 and IWB-7000. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** The AMP based on susceptibility determination, neutron fluence level, and supplemental inspection is effective in managing the effects of synergistic loss of fracture toughness due to neutron and thermal aging embrittlement on the intended function of CASS components.	
Same as for the effect of Thermal Aging and Neutron Irradiation Embrittlement on Item B1.5.8 jet pump castings.	*Same as for the effect of Thermal Aging and Neutron Irradiation Embrittlement on Item B1.5.8 jet pump castings.*	<u>Yes, the existence of a suitable AMP should be evaluated</u>
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. <u>Insert #1.</u>	Yes TLAA

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

B1. REACTOR VESSEL INTERNALS (Boiling Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	
B1.7.1 thru B1.7.3	Instrument Housings	Intermediate Range Monitor (IRM) Dry Tubes, Low Power Range Monitor (LPRM) Dry Tubes, Source Range Monitor (SRM) Dry Tubes	SS	288°C, High-Purity Water	Crack Initiation and Growth	SCC, IASCC	
B1.7.1 thru B1.7.3	Instrument Housings	IRM Dry Tubes, LPRM Dry Tubes, SRM Dry Tubes	SS	288°C, High-Purity Water	Cumulative Fatigue Damage	Fatigue	
B1.5.9	Jet Pump Aseemblies	Jet Pump Sensing Line	SS	288°C, High-Purity Water	Crack Initiation and Growth	Unantici-pated Cyclic Loading	

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Implementation of aging management program recommended in GE Services Information Letter (SIL) 409 Rev. 1. BWRVIP-49 for instrument penetration inspection and flaw evaluation guidelines has been approved by the staff. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-103515 and BWRVIP-29 to minimize the potential of crack initiation and growth. Plant programs also may include water chemistry measures such as strict controls on conductivity, and hydrogen addition. [Supporting documents BWRVIP-03 for reactor pressure vessel internals examination guidelines; BWRVIP-57 for instrument penetration repair design criteria; BWRVIP-14, -59, and -60 for evaluation of crack growth; BWRVIP-44 for weld repair of Ni-alloys; BWRVIP-45 for weldability of irradiated structural components; and BWRVIP-62 for technical basis for inspection relief for internal components with hydrogen injection.]	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC and periodic inservice inspection (ISI) to monitor the effects of SCC on the intended function of the components, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** Based on GE SIL 409 Rev. 1 replacement of existing tubes with those fabricated from more IASCC-resistant materials and crevice free design. Maintaining high water purity (many BWRs now operate at <0.15 $\mu S/cm^2$) reduces susceptibility to SCC. Hydrogen additions are effective in reducing electrochemical potentials in the recirculation piping system, but are less effective in the core region. ***(3) Parameters Monitored/Inspected:*** Inspection and flaw evaluation are to be performed in accordance with referenced BWRVIP guideline, as approved by the NRC staff. ***(4) Detection of Aging Effects:*** Degradation due to SCC can not occur without crack initiation and growth. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with applicable, approved BWRVIP guideline is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Crack indications are evaluated in accordance with applicable, approved BWRVIP guideline. ***(7) Corrective Actions:*** Corrective actions in accordance with applicable, approved BWRVIP-57 guidelines are adequate. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Cracking of dry tubes has been observed at 14 or more BWRs. The cracking is intergranular and has been observed in dry tubes without apparent sensitization suggesting that irradiation assisted SCC (IASCC) may also play a role in the cracking.	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. Insert #1.	Yes TLAA
Plant specific aging management program should be implemented.	Plant specific aging management program is to be evaluated.	Yes, no generic AMP

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

<u>C1.1.13 Small Bore Piping</u>

<u>C1.1.14 Jet Pump Sensing Line</u>

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

<u>C1.5 Control Rod Drive (CRD) Hydraulic System</u>

<u>C1.5.1 Piping and Fittings</u>

<u>C1.5.2 Valve Body</u>

<u>C1.5.3 Pump Casing</u>

<u>C1.5.4 Filter</u>

<u>C1.5.5 Accumulator</u>

<u>C1.5.6 Scram Discharge</u>

<u>C1.5.7 CRD Return Line</u>

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	
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)	

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Program delineated in NUREG-1344 and implemented through NRC Generic Letter 89-08; CHECWORKS Code; EPRI guidelines of NSAC-202L-R2 for effective erosion/corrosion program; and water chemistry program based on EPRI guidelines in TR-103515 and BWRVIP--29 for water chemistry in BWRs. [Supporting documents BWRVIP-75 for rtechnical basis for revisions to GL 88-01 inspection schedules.]	***(1) Scope of Program:*** The AMP includes NUMARC program delineated in Appendix A of NUREG-1344 and implemented through NRC Generic Letter (GL) 89-08; CHECWORKS computer Code; and EPRI guidelines of NSAC-202L-R2. The program includes the following recommendations: (a) conduct appropriate analysis and limited baseline inspection, (b) determine the extent of thinning and repair/replace components, and (c) perform follow-up inspections to confirm or quantify and take longer term corrective actions. Technical aspects of the CHECWORKS Code, including the parameters and inputs, are acceptable. However, the EPRI guidance document NSAC-202L-R2 (April 1999) is too general to ensure applicant's flow-accelerated corrosion program will be effective in managing aging in safety-related systems. ***(2) Preventive Actions:*** The rate of E/C is affected by piping material, geometry and hydrodynamic conditions, and operating conditions such as temperature, pH, and dissolved oxygen content. Mitigation is by selecting material considered resistant to E/C, adjusting water chemistry and operating conditions, and improving hydrodynamic conditions through design modifications. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of E/C on the intended function of piping by measuring wall thickness by nondestructive examination and performing analytical evaluations. The inspection program delineated in NUREG-1344 requires ultrasonic or radiographic testing of 10 most susceptible locations and 5 additional locations based on unique operating conditions or special considerations. For each location outside the acceptance guidelines, the inspection sample is expanded based on engineering judgment. Analytical models such as those incorporated into the CHECWORKS code are used to predict E/C in piping systems based on specific plant data including material and hydrodynamic and operating conditions. The inspection data are used to calibrate and benchmark the models and code. ***(4) Detection of Aging Effects:*** Aging degradation of piping and fittings occurs by wall thinning; extent and schedule of inspection assure detection of wall thinning before the loss of intended function of the piping. ***(5) Monitoring and Trending:*** Inspection schedule of NUREG-1344 and EPRI guidelines should provide for timely detection of leakage. Inspections and analytical evaluations are performed during plant outage. If analysis shows unacceptable conditions, inspection of initial sample is performed within 6 months. ***(6) Acceptance Criteria:*** Inspection results are used to calculate number of refueling or operating cycles remaining before the component reaches Code minimum allowable wall thickness. If calculations indicate that an area will reach Code minimum (plus 10% margin), the component must be repaired or replaced. However, NRC staff has identified the problems in implementing E/C program that pertain to weakness or errors in (a) using predictive models, (b) calculating minimum wall thickness acceptance criteria, (c) analyzing the results of UT examinations, and (d) assessment of E/C program activities (NRC	Yes; Element 1 should be further evaluated

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	
C1.1.1	Piping & Fittings	Main Steam	CS SA106-Gr B, SA333-Gr 6, SA155-Gr KCF70	288°C Steam	Cumulative Fatigue Damage	Fatigue	
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	
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	
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	

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) Information Notice IN 93-21). ***(7) Corrective Actions:*** Prior to service, repair or replace to meet the requirements of NUREG-1344. Follow-up inspections are performed to confirm or quantify thinning and take longer term corrective actions such as adjustment of chemistry and operating parameters, or selection of materials resistant to E/C. However, NRC staff has identified weakness or errors in (a) dispositioning components after reviewing the results of the inspection analysis, and (b) repairing or replacing components that failed to meet the acceptance criteria (IN 93-21). ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Wall-thinning problems in single-phase systems have occurred in feedwater and condensate systems (NRC Bulletin No. 87-01, INs 81-28, 92-35, 95-11), and in two-phase piping in extraction steam lines (INs 89-53, 97-84) and moisture separation reheater and feedwater heater drains (INs 89-53, 91-18, 93-21, 97-84). The AMP outlined in NUREG-1344 and EPRI report and implemented through GL 89-08 has provided effective means of ensuring the structural integrity of all high-energy carbon steel systems.	
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or ANSI B31.1, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed. <u>Insert #1.</u>	Yes TLAA
Same as for the effect of Erosion/Corrosion on Item C1.1.1 Main Steam Line Piping and Fittings.	*Same as for the effect of Erosion/Corrosion on Item C1.1.1 Main Steam Line Piping and Fittings.*	<u>Yes. Element 1 should be further evaluated</u>
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. <u>Insert #1.</u>	Yes TLAA
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or ANSI B31.1, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed. <u>Insert #1.</u>	Yes TLAA

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Program delineated in NUREG-0313, Rev. 2 and NRC Generic letter (GL) 88-01 and its Supplement 1, and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination categories B-J for pressure retaining welds in piping and B-F for pressure retaining dissimilar metal welds, and testing category B-P for system leakage. BWRVIP-75 technical basis for revisions to GL 88-01 inspection schedule, BWRVIP-27 for standby liquid control/core plate ΔP inspection and flaw evaluation guidelines, and BWRVIP-42 for LPCI coupling inspection and flaw evaluation guidelines are 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. [Supporting documents BWRVIP-03 for reactor pressure vessel internals examination guidelines; BWRVIP-14, -59, and -60 for evaluation of crack growth; BWRVIP-53 [standby liquid control line repair design criteria; BWRVIP-61 for BWR vessel and internals induction heating stress improvement effectiveness on crack growth in operating plants; and BWRVIP-62 for technical basis for inspection relief for internal components with hydrogen injection.]	***(1) Scope of Program:*** The program focuses on managing and implementing countermeasures to mitigate IGSCC and inservice inspection (ISI) to monitor IGSCC and its effects on the intended function of austenitic stainless steel (SS) piping 4 in. or larger in diameter, and reactor vessel attachments and appurtenances. NUREG-0313 and GL 88-01, respectively, describe the technical basis and staff guidance regarding mitigating IGSCC in BWRs. ***(2) Preventive Actions:*** 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 according to EPRI guidelines in TR-103515 and BWRVIP-29 to minimize the potential of crack initiation and growth. Also, hydrogen water chemistry and stringent control of conductivity is used to inhibit IGSCC. ***(3) Parameters Monitored/Inspected:*** Inspection and flaw evaluation are to be performed in accordance with referenced BWRVIP guideline, as approved by the NRC staff. ***(4) Detection of Aging Effects:*** Aging degradation of the piping can not occur without crack initiation and growth; extent, method, and schedule of inspection as delineated in GL 88-01 and updated in BWRVIP-75 is adequate and will assure timely detection of cracks before the loss of intended function of austenitic SS piping and fittings. Insert #5. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with GL 88-01 or applicable approved BWRVIP guideline. ***(6) Acceptance Criteria:*** Any IGSCC degradation is evaluated according to applicable approved BWRVIP guideline. ***(7) Corrective Actions:*** Insert #6. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** IGSCC has occurred in small- and large-diameter BWR piping made of austenitic SSs and Nickel-base alloys. Significant cracking has occurred in recirculation, core spray, and RHR systems and reactor water cleanup system piping welds.	Yes, BWRVIP Guideline
The reactor coolant system components are inspected in accordance with ASME Section XI, Subsection IWB. This inspection is not sufficient to detect the effects of loss of fracture toughness due to thermal aging embrittlement. An acceptable alternative AMP consists of the following: Determination of the susceptibility of CASS piping to thermal aging embrittlement based on casting method, Mo content, and percent ferrite. For "potentially susceptible" piping, aging	For the acceptable alternative AMP: ***(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 NUREG-1705 , the susceptibility to thermal aging embrittlement of CASS piping is determined in terms of casting method, Mo content, and ferrite content.	Yes, the existence of a suitable AMP should be evaluated

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
(continued from previous page) management is accomplished either through enhanced volumetric examination or plant/component-specific flaw tolerance evaluation. Additional inspection or evaluations are not required for "not susceptible" piping to demonstrate that the material has adequate fracture toughness. For pump casings and valve bodies, screening for susceptibility to thermal aging is not required. Also, the existing ASME Section XI inspection requirements, including the alternative requirements of ASME Code Case N-481 for pump casings, are considered adequate for all pump casings and valve bodies.	*(continued from previous page)* For low-Mo content (0.5 wt.% max) steels, only static-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement, static-cast steels with ≤20% ferrite and all centrifugal-cast steels 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). Insert #3. For pump casings and valve bodies, screening for susceptibility to thermal aging is not required. ***(4) Detection of Aging Effects:*** For "not susceptible" piping, no additional inspection or evaluations are required to demonstrate that the material has adequate fracture toughness. For "potentially susceptible" piping, because the base metal does not receive periodic inspection per ASME Section XI, volumetric examination should be performed on the base metal, with the scope of the inspection covering the portions determined to be limiting from the standpoint of applied stress, operating time, and environmental considerations. Alternatively, a plant/component- specific flaw tolerance evaluation, using specific geometry and stress information, can be used to demonstrate that the thermally-embrittled material has adequate toughness. Current volumetric examination methods are inadequate for reliable detection of cracks in CASS components; the performance of the equipment and techniques when developed, should be demonstrated through the program consistent with the ASME Section XI, Appendix VIII. For all pump casings and valve bodies, the existing ASME Section XI inspection requirements, including the alternative requirements of ASME Code Case N-481 for pump casings, are considered adequate. For valve bodies less than NPS 4, the adequacy of inservice inspection according to ASME Section XI has been demonstrated by a NRC performed bounding fracture analysis. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 and reliable examination methods 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***	

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	
C1.1.5, C1. 1.11	Piping & Fittings	Recirculation, Lines to RWC and SLC Systems	SS	288°C, Oxygenated Water	Cumulative Fatigue Damage	Fatigue	
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	
C1.2.1 thru C1.2.3	Recirculation Pump	Bowl/Casing, Cover, Seal Flange	CASS, SS	288°C, Oxygenated Water	Cumulative Fatigue Damage	Fatigue	
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	
C1.2.1	Recirculation Pump	Bowl/Casing	CASS, SS	288°C, Oxygenated Water	Crack Initiation and Growth	SCC, IGSCC	

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	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 proposed AMP is effective in managing the effects of thermal aging on the intended function of CASS components.	
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or ANSI B31.1, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed. Insert #1.	Yes TLAA
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or ANSI B31.1, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed. Insert #1.	Yes TLAA
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or ANSI B31.1, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed. Insert #1.	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.*	Yes, existence of a suitable AMP should be evaluated
Guidelines of NUREG-0313, Rev. 2 and NRC Generic letter (GL) 88-01 and its Supplement 1; inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination categories B-L-1 for pump casing welds and B-L-2 for pump casing, and testing category B-P for system leakage. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-103515 and BWRVIP-29 to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC and inservice inspection (ISI) to monitor the effects of SCC on intended function of the pump. NUREG-0313 and GL 88-01, respectively, describe the technical basis and staff guidance regarding the problem of IGSCC in BWRs. ***(2) Preventive Actions:*** Mitigation of IGSCC is by selection of material considered resistant to sensitization and IGSCC, e.g., low-carbon grades of cast SSs and weld metal, with a maximum carbon of 0.035% and minimum 7.5% ferrite. Also, hydrogen water chemistry and stringent control of conductivity is used to inhibit IGSCC. ~~However~~, High-carbon grades of cast SS, e.g., CF-8 and CF-8M ~~may be~~ are susceptible to SCC. The aging management program must therefore rely upon ISI in accordance with GL 88-01 to detect possible degradation. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of SCC on 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	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	
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~~ Loss of Material	Wear	

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) to IWA-5000 specify visual VT-2 (IWA-5240) examination of all pressure retaining boundary of the pump during system leakage test (IWB-5221) and system hydrostatic test (IWB-5222). Also, coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-103515 and BWRVIP-29 to minimize the potential of crack initiation and growth. ***(4) Detection of Aging Effects:*** Degradation of the pump due to SCC can not occur without crack initiation and growth; extent and schedule of inspection as delineated in GL 88-01 will assure detection of cracks before the loss of intended function of the pump. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with GL 88-01 should provide timely detection of cracks. All welds are inspected each inspection period from at least one pump in each group performing similar functions in the system. Visual examination is required only when the pump is disassembled for maintenance, repair, or volumetric examination, but at least once during the period. System leakage test is conducted prior to plant startup following each refueling outage, and hydrostatic test is conducted at or near the end of each inspection interval. ***(6) Acceptance Criteria:*** Any SCC degradation is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400; IWB-3518 for volumetric examination of welds and IWB-3519 for visual examination of pump internal surfaces. Supplementary surface examination may be performed on interior and/or exterior surfaces when flaws are detected in volumetric examination. ***(7) Corrective Actions:*** Repair is in conformance with IWA-4000 and IWB-4000 or GL 88-01. Continued operation without repair require that crack growth calculations be performed according to the guidance of GL 88-01 or other approved procedure. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** The comprehensive AMP outlined in NUREG-0313 and GL 88-01 addresses improvements in all elements that cause IGSCC and has provided effective means of ensuring structural integrity of the primary coolant pressure boundary.	
Recommendations for a comprehensive bolting integrity program delineated in NUREG–1339 on resolution of Generic Safety Issue 29 and implemented through NRC Generic Letter 91-17; additional details on bolting integrity outlined in EPRI NP-5769; and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination categories B-G–1 or B-G-2 for pressure retaining bolting, and category B-P for system leakage.	***(1) Scope of Program:*** The staff guidance of NRC Generic Letter (GL) 91-17 provides assurance that plant specific comprehensive bolting integrity programs have been implemented to ensure bolting reliability. The NRC staff recommendations and guidelines for a comprehensive bolting integrity program is delineated in NUREG-1339, and the industry's technical basis for the program is outlined in EPRI NP-5769. ***(2) Preventive Actions:*** Selection of bolting material and the use of lubricants and sealants in accordance with guidelines of EPRI NP-5769 and additional requirements of NUREG 1339, prevent or mitigate degradation and failure of all safety-related closure bolting. ***(3) Parameter Monitored/ Inspected:***	No

Item	Structure and Component	Region of Interest	Material	Environment	Aging Effect	Aging Mechanism	
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	
C1.2.4	Recirculation Pump	Closure Bolting	HSLAS SA193 GrB7	Air, Leaking Oxygenated Water and/or Steam at 288°C	Cumulative Fatigue Damage	Fatigue	

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) The AMP monitors the effects of aging degradation on the intended function of closure bolting by detection of coolant leakage, and by detection and sizing of cracks by inservice inspection (ISI). Inspection requirements of ASME Section XI, Table IWB 2500-1, examination category B-G-1 for bolting greater than 2 in. in diameter specify volumetric examination of studs and bolts, and visual VT-1 examination of surfaces of nuts, washers, bushings, and flanges. Examination category B-G-2 for bolting 2 in. or smaller specifies only visual VT-1 examination of surfaces of bolts, studs, and nuts. However, because most failures have occurred in fasteners 2 in. or smaller, based on IE Bulletin 82-02, enhanced inspection and improved techniques are recommended. Inspection requirements of ASME Section XI testing category B-P specify visual VT-2 (IWA-5240) examination of all pressure retaining components during system leakage test (IWB-5221) and system hydrostatic test (IWB-5222). ***(4) Detection of Aging Effects:*** Degradation of the closure bolting due to crack initiation, loss of prestress, or attrition of the closure bolting would result in leakage. The extent and schedule of inspection assure detection of aging degradation before the loss of intended function of closure bolting. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI are effective and adequate for timely detection of cracks and leakage. ***(6) Acceptance Criteria:*** Any cracks in closure bolting are evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3515 and 3517. ***(7) Corrective Actions:*** Repair and replacement is in conformance with IWB-4000 and guidelines and recommendations of EPRI NP-5769. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** The bolting integrity programs developed and implemented in accordance with commitments made in response to NRC communications on bolting events have provided effective means of ensuring bolting reliability.	
Same as for the effect of wear on Item C1.2.4 Closure Bolting for Recirculation Pump.	*Same as for the effect of wear on Item C1.2.4 Closure Bolting for Recirculation Pump.*	No
Components have been designed or evaluated for fatigue for a 40 y design life, 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. <u>Insert #1.</u>	Yes TLAA

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

C1. REACTOR COOLANT PRESSURE BOUNDARY (Boiling Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	
C1.3.1	Valves (Check, Control, Hand, Motor-Operated, and Relief Valves)	Body	CS	288°C, Oxygenated Water	Wall Thinning	Erosion/ Corrosion	
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	
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	

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.*	Yes, existence of a suitable AMP should be evaluated
Guidelines of NUREG-0313, Rev. 2 and NRC Generic letter (GL) 88-01 and its Supplement 1; inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination categories B-M-1 for valve body welds and B-M-2 for valve body, and testing category B-P for system leakage. Coolant water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-103515 and BWRVIP-29 to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes preventive measures to mitigate stress corrosion cracking (SCC) and inservice inspection (ISI) to monitor the effects of SCC on intended function of the valves. NUREG-0313 and GL 88-01, respectively, describe the technical basis and staff guidance regarding the problem of IGSCC in BWRs. ***(2) Preventive Actions:*** Mitigation of IGSCC is by selection of material considered resistant to sensitization and IGSCC, e.g., low-carbon grades of cast SSs and weld metal, with a maximum carbon of 0.035% and minimum 7.5% ferrite. Also, hydrogen water chemistry and stringent control of conductivity is used to inhibit IGSCC. ~~However~~, High-carbon grades of cast SS, e.g., CF-8 and CF-8M ~~may be~~ <u>are</u> 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	
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~~ Loss of Material	Wear	
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	
C1.3.4	Valves	Closure Bolting	HSLAS SA193 GrB7	Air, Leaking Oxygenated Water and/or Steam at 288°C	Loss of Preload	Stress Relaxation	
C1.3.4	Valves	Closure Bolting	HSLAS SA193 GrB7	Air, Leaking Oxygenated Water and/or Steam at 288°C	Cumulative Fatigue Damage	Fatigue	
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	Crack Initiation and Growth	SCC, Unanticipated Cyclic Loading	

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400; IWB-3518 for volumetric examination of welds and 3519 for visual examination of valve internal surfaces. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWA-4000 and IWB-4000 or GL 88-01, and reexamination in accordance with requirements of IWA-2200. Continued operation without repair require that crack growth calculations be performed according to the guidance of GL 88-01 or other approved procedure. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** The comprehensive AMP outlined in NUREG-0313 and GL 88-01 has provided effective means of ensuring structural integrity of the primary coolant pressure boundary.	
Same as for the effect of wear on Item C1.2.4 Closure Bolting for Recirculation Pump.	*Same as for the effect of wear on Item C1.2.4 Closure Bolting for Recirculation Pump.*	No
Components have been designed or evaluated for fatigue for a 40 y design life, 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. Insert #1.	Yes TLAA
Same as for the effect of wear on Item C1.2.4 Closure Bolting for Recirculation Pump.	*Same as for the effect of wear on Item C1.2.4 Closure Bolting for Recirculation Pump.*	No
Components have been designed or evaluated for fatigue for a 40 y design life, 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. Insert #1.	Yes TLAA
ASME Section XI (edition specified in 10 CFR 50.55a or CLB), Table IWC 2500-1, examination category C-H for pressure retaining Class 2 components should be augmented by a program of temperature and radioactivity monitoring of the shell side water, and eddy current testing of tubes	***(1) Scope of Program:*** The program includes inservice inspection in accordance with ASME Section XI, and should be augmented with temperature and radioactivity monitoring of the shell side water, and eddy current testing of the tubes. ***(2) Preventive Actions:*** Monitor isolation condenser system performance based on the plant technical specifications and measurements of temperature and radioactivity in the shell side water. Perform ASME Section XI inspections and eddy current	Yes, plant specific augmentation program

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

C1. REACTOR COOLANT PRESSURE BOUNDARY (Boiling Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environment	Aging Effect	Aging Mechanism	
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	General, Pitting, and Crevice Corrosion	

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) testing. ***(3) Parameters Monitored/ Inspected:*** The temperature monitoring is directly related to detecting leakage of the condensate return valves, the radioactivity measurement, ASME Section XI inspections, and eddy current testing to detect tube cracking. ***(4) Detection of Aging Effects:*** Cumulative fatigue damage to condenser tubes would result in degradation of component performance. Monitoring of temperature would detect valve leakage; monitoring of radioactivity in shell side water and ASME inspection and eddy current testing assure detection of cumulative fatigue damage to condenser tubes before the loss of intended function of the component. ***(5) Monitoring and Trending:*** The results of temperature and radioactivity monitoring are monitored and trended. ***(6) Acceptance Criteria:*** The monitoring, testing and inspection results are related to cumulative fatigue damage to condenser tubes and 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 is 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 cumulative fatigue damage to condenser tubes.	
Same as for the effect of SCC andUnantici-pated Cyclic Loading *on Items C1.4.1 - C1.4.4 isolation condenser components.*	*Same as for the effect of* SCC andUnantici-pated Cyclic Loading *on Items C1.4.1 - C1.4.4 isolation condenser components.*	Yes, plant specific augmenta-tion program

Item	Structure and Component	Region of Interest	Material	Environment	Aging Effect	Aging Mechanism	
C1. 1.13	Piping & Fittings	Small-Bore Piping	SS, CS	288°C, Oxygenated Water	Crack Initiation and Growth	Unanticipated Thermal and Mechanical Loading	

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 category B-J for pressure retaining welds in piping and testing category B-P for system leakage, and primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-103515 and BWRVIP-29 to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes preventive measures to inhibit cracking and inservice inspection (ISI) to monitor the effects of cracking on the intended function of small-bore piping of reactor coolant system and connected lines. ***(2) Preventive Actions:*** Coolant water chemistry is monitored and maintained according to 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:*** The AMP monitors the effects of cracking on the intended function of piping and fittings by detection cracks and leakage by ISI. Inspection requirements of Table IWB 2500-1, examination category B-J specifies surface examination for circumferential and longitudinal welds in each pipe or branch run less than 4 inches nominal pipe size (NPS), and category B-P specifies visual VT-2 (IWA-5240) examination of all pressure retaining components during system leakage test (IWB-5221) and system hydrostatic test (IWB-5222). However, inspection in accordance with ASME Section XI does not require volumetric examination of pipes less than NPS 4. A plant-specific destructive examination or a nondestructive examination (NDE) that permits inspection of the inside surfaces of the piping should be conducted to ensure that cracking has not occurred and the component intended function will be maintained during the extended period. ***(4) Detection of Aging Effects:*** Degradation of the piping due to cracking would result in leakage of coolant. A one-time inspection of a sample of locations most susceptible to cracking should be conducted to verify that service-induced weld cracking is not occurring in the small-bore piping less than NPS 4, including pipe, fittings, and branch connections. Actual inspection locations should be based on risk-informed approaches and physical accessibility, exposure levels, and NDE examinations techniques, and locations identified in NRC Information Notice (IN) 97-46. ***(5) Monitoring and Trending:*** System leakage test is conducted prior to plant startup following each refueling outage, and hydrostatic test at or near the end of each inspection interval. The results of one-time inspection will be used to dictate the frequency of future inspections. ***(6) Acceptance Criteria:*** Any relevant conditions that may be detected during the leakage tests are evaluated in accordance with IWC-3516. ***(7) Corrective Actions:*** Repair is in conformance with IWA-4000 and IWB-4000, replacement according to IWA-7000 and IWB-7000. If destructive examination is employed, repair and replacement are in accordance with ASME Section XI rules. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Cracking has occurred in HPCI piping (IN 89-80) and instrument lines (LER 50-249/99-003-1) due to thermal and mechanical loading.	Yes Elements 3 and 4 should be further evaluated

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	
C1.5.1	Control Rod Drive (CRD) Hydraulic System	Pipings and Fittings (Outside Surface)	SS	Air, Leaking Oxygenated Water up to 288°C	Crack Initiation and Growth	Stress Corrosion Cracking	
C1.5.1, C1.5.4, C1.5.7	Control Rod Drive (CRD) Hydraulic System	Pipings and Fittings, Filter, CRD Return Line	SS	Oxygenated Water up to 288°C	Crack Initiation and Growth	Stress Corrosion Cracking	
C1.5.1, C1.5.4, C1.5.7	Control Rod Drive (CRD) Hydraulic System	Pipings and Fittings, Filter, CRD Return Line	Carbon Steel, SS	Oxygenated Water up to 288°C	Cumulative Fatigue Damage	Fatigue	
C1.5.2	Control Rod Drive (CRD) Hydraulic System	Valve Body	SS	Oxygenated Water up to 288°C	Crack Initiation and Growth	Stress Corrosion Cracking	
C1.5.3	Control Rod Drive (CRD) Hydraulic System	Pump Casing	SS	Oxygenated Water up to 288°C	Crack Initiation and Growth	Stress Corrosion Cracking	

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Leaching of chlorides from insulation jackets and other sourses can cause externally-initiated transgranular stress corrosion cracking (TGSCC) in the stainless steel heat-traced lines. Plant specific aging management program should be implemented.	Plant specific aging management program is to be evaluated.	Yes, no generic AMP
Same as for the effect of SCC/IGSCC on piping and fittings in Items C1.1.1 thru C1.1.11.	*Same as for the effect of SCC/IGSCC on piping and fittings in Items C1.1.1 thru C1.1.11.*	Yes, BWRVIP Guideline
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. Insert #1.	Yes TLAA
Same as for the effect of SCC/IGSCC on Item C1.3.1 valve body.	*Same as for the effect of SCC/IGSCC on Item C1.3.1 valve body.*	No
Same as for the effect of SCC/IGSCC on Item C1.2.1 recirculation pump bowl/casing.	*Same as for the effect of SCC/IGSCC on Item C1.2.1 recirculation pump bowl/casing.*	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	
C1.5.5, C1.5.6	Control Rod Drive (CRD) Hydraulic System	Accumulator, Scram Discharge Volume	Carbon Steel	Oxygenated Water up to 288°C	Loss of Material	General, Pitting, and Crevice Corrosion	
C1.5.2	CRD Hydraulic System	Valve body	Carbon Steel	Oxygenated Water up to 288°C	Wall Thinning	Erosion/ Corrosion (E/C)	

Inserts

Insert #1

The staff recommendation for the closure of GSI-190 is contained in a December 26, 1999, memorandum from Ashok Thadani to William Travers. The staff recommended that licensees address the effects of the coolant environment on component fatigue life as aging management programs are formulated in support of license renewal. One method acceptable to the staff of satisfying this recommendation is to assess the impact of the reactor coolant environment on a sample of critical components. These critical components should include, as a minimum, those components selected in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components." The sample of critical components can be evaluated by applying environmental correction factors to the existing code fatigue analyses. Formulas for calculating the environmental life corrections factors for carbon and low-alloy steels are contained in NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves for Carbon and Low-Alloy Steels." The formula for calculating the environmental life corrections factor for stainless steels is contained in NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels."

Insert #2

The reactor vessel internals receive a visual inspection (VT-3) according to Category B-N-3 of Subsection IXB, ASME Section XI. This inspection is not sufficient to detect the effects of changes in dimension due to void swelling.

An acceptable alternative AMP consists of the following:

1. Participation in industry programs to address the significance of change in dimensions due to void swelling.

2. Implementation of an inspection program should the results of the industry programs indicate the need for such inspections.

Insert #3

Components containing Nb are considered susceptible and require evaluation on a case-by-case basis.

Insert #4

(1) Scope of Program: The program includes inservice inspection (ISI) to monitor the condition of components that depend on preload, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** No practical preventative actions are possible. ***(3) Parameters Monitored/Inspected:*** The AMP utilizes ISI to monitor the effects of stress relaxation on the intended function of the component by detection and sizing of cracks that could be formed by excessive vibration etc. that may occur if the preload is lost. Table IWB-2500, category B-N-3 specifies visual VT-3 examination of all accessible surfaces of reactor internals. Because VT-3 inspection can only detect degradation that occurs after the loss of preload, it may be adequate if there is sufficient redundancy that loss of some bolting between inspections is accepatable. In some cases additional inspection may be required. ***(4) Detection of Aging Effects:*** As part of the AMP it may be possible to identify acceptable levels of preload and demonstrate whether under the fluence of interest whether loss of acceptable preload is likely. VT-3 may not be adequate to detect tight cracks. Also, creviced regions are difficult to inspect visually. Supplementary inspections by techniques such as ultrasonic testing (UT) or other nondestructive methods may be needed to detect cracking in inaccessible regions. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with IWB-3520. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWB-3140. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** There are no reports of stress relaxation producing damage in reactor vessel internals.

<u>Insert #5</u>

The inspection guidance in BWRVIP-75 is under staff review. The topical (BWRVIP-75) when approved by the staff may serve to replace the inspection extent and schedule in GL 88-01.

<u>Insert #6</u>

The guidance for weld overlay repair, stress improvement or replacement is provided in GL 88-01, Code Case N 504-1, or ASME Section XI.

<u>Insert #7</u>

The extent and schedule of the inspections and test techniques prescribed by the program are designed to ensure continued tube integrity and that aging effects will be discovered an repaired before there is a loss of intended function.

Insert #8 (originally defined as OKC-steam)

The staff recommendation for the closure of GSI-190 is contained in a December 26, 1999, memorandum from Ashok Thadani to William Travers. The staff recommended that licensees address the effects of the coolant environment on component fatigue life as aging management programs are formulated in support of license renewal. An acceptable method of satisfying this recommendation is to use the high-temperature water data to assess the environmental effects on fatigue life.