NUCLEAR ENERGY INSTITUTE

Douglas J. Walters
SR. PROJECT MANAGER, LICENSING
NUCLEAR GENERATION

June 9, 2000

Mr. Christopher I. Grimes
Chief, License Renewal and Standardization Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20006

SUBJECT: Generic Aging Lessons Learned Report Comments

PROJECT NUMBER: 690

Dear Mr. Grimes:

Enclosed are comments on the pressurized water reactor portions of Chapter IV of the Generic Aging Lessons Learned (GALL) Report. Two documents are provided for each section of the chapter. One document is a pen-and-ink mark-up of the existing GALL pages to reflect our comments. Each comment is identified by number. The second document is a table containing our comments, numbered consistent with the marked-up pages.

As we mentioned in previous cover letters, we continue to wrestle with the format incompatibility between that provided by your contractor and our software capabilities. For this reason, we are unable to provide the clean copy we had intended, but continue to explore alternatives to the formatting issue in hopes of providing clean copies in the near future.

We look forward to discussing the enclosed comments with the NRC staff. If you have any questions please call Bob Evans (202.739-8101), or me (202.739-8093).

Sincerely,

FOR

Douglas J. Walters

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

Enclosure

D042

MARK UP 6/8

CHAPTER IV

(12/06/99)

REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

Major Plant Sections

~~A1. Reactor Vessel (Boiling Water Reactor)~~

A2. Reactor Vessel (Pressurized Water Reactor)

~~B1. Reactor Vessel Internals (Boiling Water Reactor)~~

B2. Reactor Vessel Internals (PWR) - Westinghouse

B3. Reactor Vessel Internals (PWR) - Combustion Engineering)

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

~~C1. Reactor Coolant Pressure Boundary (Boiling Water Reactor)~~

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

D1. Steam Generator (Recirculating)

D2. Steam Generator (Once-Through)

MARK - CLN
6/8

A2. Reactor Vessel (Pressurized Water Reactor)

A2.1 Closure Head

A2.1.1 Dome

A2.1.2 Head Flange

A2.1.3 Stud Assembly

A2.1.4 Vessel Flange Leak Detection Line

A2.2 Control Rod Drive Mechanism

A2.2.1 Pressure Housing

A2.3 Nozzles

A2.3.1 Inlet

A2.3.2 Outlet

A2.3.3 Safety Injection (on some)

A2.4 Nozzle Safe Ends

A2.4.1 Inlet

A2.4.2 Outlet

A2.4.3 Safety Injection (on some)

A2.5 Shell

A2.5.1 Upper (Nozzle) Shell

A2.5.2 Intermediate & Lower Shell

A2.5.3 Vessel Flange

A2.6 Core Support Pads

A2.8.1 CRD Mechanism

A2.8.2 Instrumentation

A2.8.3 Leakage Monitoring Tubes

A2.9 Pressure Vessel Support

A2.9.1 Skirt Support

A2.9.2 Cantilever/Column Support

A2.9.3 Neutron Shield Tank

A2. Reactor Vessel (Pressurized Water Reactor)

System, Structures, and Components

The system, structures, and components included in this table comprise the pressurized water reactor (PWR) reactor vessel pressure boundary and consist of vessel shell and flanges, top closure head and bottom head, control rod drive (CRD) mechanism housing, nozzles (including safe ends) for reactor coolant inlet and outlet lines and safety injection, and penetrations through either the closure head or bottom head domes for instrumentation and leakage monitoring tubes. Attachments to the vessel such as closure head lifting lugs and core support pads, as well as pressure vessel support and attachment welds are also included in the table. All systems, structures, and components in the reactor coolant system are classified as Group A Quality Standards.

System Interfaces

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

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

A2. REACTOR VESSEL (Pressurized Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environment	Aging Effect	Aging Mechanism	References
A2.1.1	Closure Head	Dome (Outer Surface)	SA302-Gr B SA533-Gr B with ~~Stainless Steel (SS) Cladding~~ IV A2-1	Air, Leaking chemically aerated borated water or steam up to 340°C	Loss of Material	Corrosion/ Boric Acid Wastage of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 Edition. Plant Technical Specifications. *Operating Experience* NRC IN 86-108 S 3 IV A2-68 WHOLE SECTION

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

A2. REACTOR VESSEL (Pressurized Water Reactor)

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Implementation of NRC Generic Letter 88-05, and based on plant technical specifications, inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, testing category B-P for pressure retaining Class 1 components. SEE IV A2-2 REWRITE TO CORRESPOND WITH EXAMPLE	***(1) Scope of Program:*** The staff guidance of Generic Letter (GL) 88-05 provides assurance that a program has been implemented consisting of systematic measures to ensure that the effects of corrosion caused by leaking coolant containing boric acid does not lead to degradation of the assurance that the reactor coolant pressure boundary will have a extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture. The program includes (a) determination of principal location, (b) examinations requirements and procedures for locating small leaks, and (c) engineering evaluations and corrective actions. ***(2) Preventive Actions:*** Minimizing reactor coolant leakage by frequent monitoring of the locations where potential leakage could occur and timely repair, prevent or mitigate boric acid corrosion. Modifications in the design or operating procedures to reduce the probability of leaks at locations where they may cause corrosion damage and use of suitable corrosion resistant materials or the application of protective coatings. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of boric acid corrosion on the intended function of the component by detection of coolant leakage by implementing the program delineated in GL 88-05 including guidelines for locating small leaks, conducting examinations, and performing engineering evaluations. Inspection requirements of ASME Section XI specify visual VT-2 (IWA-5240) examination during system leakage test (IWB-5221) and system hydrostatic test (IWB-5222) of all pressure retaining Class 1 components, according to Table IWB 2500-1 category B-P. ***(4) Detection of Aging Effects:*** Degradation of the component due to boric acid wastage can not occur without leakage of coolant containing boric acid; extent and schedule of the program delineated in GL 88-05 will assure detection of leakage before the loss of intended function of the component. ***(5) Monitoring and Trending:*** The program delineated in GL 88-05 provides for timely detection of leakage. ***(6) Acceptance Criteria:*** Any leakage is evaluated in accordance with the program proposed by GL 88-05. ***(7) Corrective Actions:*** The leakage source and areas of general corrosion are located and corrective actions are in conformance with the program proposed by GL 88-05, and the requirements of ASME Section XI, IWB-3142 and IWA-5250. Components with local areas of corrosion that reduces the wall thickness by more than 10% require analytical evaluation to demonstrate acceptability. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Boric acid wastage	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

A2. REACTOR VESSEL (Pressurized Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A2.1.1	Closure Head	Dome	SA302-Gr B SA533-Gr B ~~with SS Cladding~~ ASTM A508-Cl 2 CLASS 2 WITH SS CLADDING IV A2-5	Chemically treated water up to 40°C	Crack Initiation and Growth	Stress Corrosion Cracking (SCC), Inter-granular Stress Corrosion Cracking (IGSCC)	ASME Section XI, 1989 Edition Reg. Guide 1.43. Reg. Guide 1.44. EPRI TR-105714. *Operating Experience* NRC IN 84-18. NRC IN 84-89.

SEE IV A2-6 & IV A2-7

SEE IV A2-4

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) implementation of GL 88-05 ensures timely detection of leakage, there may still be a lack of awareness of the conditions that can lead to boric acid wastage.	
Guidelines of Regulatory Guide (RG) 1.44 to avoid sensitization of stainless steels (SSs) and guidelines of RG 1.43 to prevent underclad cracking; and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination categories B-N-1 for vessel interior, B-A for head welds, and B-D for full penetration nozzle-to-head welds. Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 to minimize the potential of crack initiation and growth. SEE IVA2-3 REWRITE TO CORRESPOND WITH EXAMPLE	***(1) Scope of Program:*** The program includes preventive measures to mitigate stress corrosion cracking (SCC) of SS cladding and inservice inspection (ISI) to monitor the effects of SCC and growth of cracks into welds and base metal, on the intended function of component. However, SCC is not anticipated to be an issue for the closure head dome or nozzles because analytical evaluations indicate that cracks in the SS cladding will stop growing in the ferritic base metal. ***(2) Preventive Actions:*** Selection of material in compliance with the guidelines of Regulatory Guide (RG) 1.44 prevent or mitigate IGSCC, and RG 1.43 provides assurance that production cladding complies with ASME Section III and XI requirements to prevent underclad cracking. Control of halogens, sulfates, and oxygen in the primary water to less than 0.05, 0.05, and 0.005 ppm, respectively, during operation, and monitor and control of water chemistry during shut down, mitigate potential of SCC. However, inadvertent introduction of contaminants into the primary coolant system can occur (IN 84-18), and SCC could occur at creviced/cold worked locations (IN 84-89). The AMP must therefore rely upon ISI in accordance with ASME Section XI to detect possible degradation. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of SCC on the intended function of closure head dome by detection and sizing of cracks by ISI. Inspection requirements of Table IWB 2500-1, examination category B-N-1 specifies visual VT-3 of closure head interior surface made accessible for examination during normal refueling outages. Examination category B-A specifies volumetric inspection of all circumferential and meridional welds, and B-D specifies volumetric inspection of all nozzle-to-vessel welds and nozzle inside radius section. ***(4) Detection of Aging Effects:*** Aging degradation of the closure head dome can not occur without crack initiation and growth; extent and schedule of inspection assure detection of cracks before the loss of intended function of the closure head. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI should provide for timely detection of cracks. Closure head dome interior is inspected at 1st refueling outage and subsequent outages at approximately 3 y intervals. Accessible length of all head welds and nozzle-to-head welds are examined each inspection interval. ***(6) Acceptance Criteria:*** Any IGSCC degradation is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and 3520 for visual examination, IWB-3510 for head welds, and IWB-3512 for full penetration nozzle welds. Visual	No

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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
				IVA2-9 CLASS 2			
A2.1.2	Closure Head	Flange (Outer Surface)	SA336, SA508 with SS Cladding	r, Leaking hemically reated orated ater or team up to 40°C	Loss of Material	Corrosion/ Boric Acid Wastage of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 Edition. Plant Technical Specifications.
A2.1.3	Closure Head	Stud Assembly	SA540-B23 & -B24, SA320-L43 Alloy 4340)	ir, Leaking hemically reated orated ater nd/or team up to 40°C	Loss of Material	Corrosion/ Boric Acid Wastage of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 Edition. Plant Technical Specifications.
A2.1.3	Closure Head	Stud Assembly	SA540-B23 & -B24, SA320-L43 (Alloy 4340) SA193-6 IVA2-12	r, Leaking hemically reated rated ater or team up to 40°C	Crack Initiation and Growth IVA2-	SCC, ~~IGSCC~~	ASME Section XI, 1989 Edition. Reg. Guide 1.65 ASME Section III, 1989 Edition. SEE IVA2-11 IVA2-13

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) is in conformance with IWA-4000 and IWB-4000, and reexamination in accordance with requirements of IWA-2200. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** SCC has not been an issue for PWR closure head dome.	
Implementation of NRC Generic Letter 88-05, and based on plant technical specifications, inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, testing category B-P for pressure retaining Class 1 components.	*Same as effect of Corrosion/Boric Acid Wastage of external surfaces of Item A2.1.1 Closure Head Dome.*	No
Implementation of NRC Generic Letter 88-05, and based on plant technical specifications, inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, testing category B-P for pressure retaining Class 1 components.	*Same as effect of Corrosion/Boric Acid Wastage of external surfaces of Item A2.1.1 Closure Head Dome.*	No
Inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination category B-G-1, and testing category B-P for system leakage. Prevention and replacement in accordance with Regulatory Guide 1.65.	***(1) Scope of Program:*** The program is focused on managing the effects of SCC on the intended function of closure head stud assemblies. ***(2) Preventive Actions:*** Design requirements of ASME Section III, Subsection NB, and additional guidance of Regulatory Guide (RG) 1.65 on material selection, preservice inspection, and protection against corrosion prevent or mitigate SCC. High-strength low-alloy steels with controlled tempering procedures are used. Maximum tensile strength is limited to <1172 MPa (<170 ksi) to provide resistance to SCC, and Charpy V energy requirements of Appendix G to 10 CFR Part 50 provide adequate toughness to prevent cracking in the stud threads. Machined stud bolting is avoided to prevent degradation due to corrosion or hydrogen embrittlement. Manufacturer or other acceptable surface treatment, or stable lubricants are permissible. Preservice inspection in conformance with NB-2580 of Section III of the Code requires ultrasonic examination of stud bolting over the entire surface prior to threading. During venting or filling of pressure vessel and while the head is removed, the stud bolts and holes are adequately protected from corrosion and contamination. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of SCC on the intended function of closure stud assembly by detection and sizing of cracks by	No

IV A2-67 WHOLE SECTION

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

A2. REACTOR VESSEL (Pressurized Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A2.1.3	Closure Head	Stud Assembly	SA540-B23 & -B24, SA320-L43	ir, Leaking hemicall reated orated ater or team up to 40°C	~~Attrition~~ LOSS OF MATERIAL IVA2-18	Wear	ASME Section XI, 1989 Edition.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) visual VT-1 examination of surfaces of nuts, washers, and bushings. Requirements for training and qualification of personnel and performance demonstration for procedures and equipment are in conformance with Appendices VII and VIII of ASME Section XI, or any other formal program approved by the NRC. Testing category B-P specify visual VT-2 (IWA-5240) examination of all pressure retaining components extending to and including the second closed valve at the boundary extremity, during system leakage test (IWB-5221) and system hydrostatic test (IWB-5222). ***(4) Detection of Aging Effects:*** Aging degradation of the closure stud assembly can not occur without crack initiation, the extent and schedule of inspection assure detection of cracks before the loss of intended function of closure stud assemblies. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI is effective and adequate for timely detection of cracks. Inspection schedule in accordance with IWB-2400 requires 100% inspection every 10 y. System leakage test is conducted prior to plant startup following each refueling outage, and hydrostatic test at or near the end of each inspection interval. ***(6) Acceptance Criteria:*** Any cracks in closure stud assemblies are evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3515 and 3517. ***(7) Corrective Actions:*** Repair is in conformance with IWB-4000, repair according to IWB-7000, and material and inspection guidance of RG 1.65. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** The AMP based on ASME Section XI inspection requirements and additional guidelines of Reg. Guide 1.65 has been effective in managing the effects of SCC to maintain the intended function of the closure head stud assembly.	
Inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination category B-G-1, and testing category B-P for system leakage. Prevention and replacement in accordance with Regulatory Guide 1.65. SEE IV A2-14 VOLUMETRIC CONCERN	***(1) Scope of Program:*** The program is focused on managing the effects of wear on the intended function of closure head stud assemblies. ***(2) Preventive Actions:*** Design requirements and guidelines of Regulatory Guide (RG) 1.65, frequent performance monitoring, and timely corrective action prevent or mitigate attrition due to wear of closure stud assemblies. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of wear on the intended function of closure stud assembly by performance monitoring and by detection of attrition by inservice inspection (ISI). Inspection requirements of ASME Section XI, Table IWB 2500-1, examination category B-G-1, specify the following for all closure stud	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

A2. REACTOR VESSEL (Pressurized Water Reactor)

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

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

A2. REACTOR VESSEL (Pressurized Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A2.1.3	Closure Head	Stud Assembly	SA540-B23 & -B24, SA320-L43	ir, Leaking hemically reated orated ater or team up to 40°C	Cumulative Fatigue Damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition.
A2.1.5	Closure Head	Vessel Flange Leak Detection Line	Stainless Steel	eaking hemically reated orated ater or team up to 40°C	Crack Initiation and Growth	SCC, IGSCC	Plant Technical Specifications.
A2.2.1	Control Rod Drive (CRD) Mechanism	Pressure Housing	SA182, Types 304 or 316, CF-8	hemically reated orated ater up to	Crack Initiation and Growth	SCC, IGSCC	ASME Section XI, 1989 Edition. Reg. Guide 1.44. EPRI TR-105714.

IVA2-21

IVA2-21

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) and equipment are in conformance with Appendices VII and VIII of ASME Section XI, or any other formal program approved by the NRC. Testing category B-P specify visual VT-2 (IWA-5240) examination of all pressure retaining components [redacted] system leakage test (IWB-5221) and system hydrostatic test (IWB-5222). ***(4) Detection of Aging Effects:*** Degradation due to wear of the closure stud assembly can not occur without attrition, the extent and schedule of inspection assure detection of attrition before the loss of intended function of closure stud assemblies. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI is effective and adequate for timely detection of attrition. Inspection schedule in accordance with IWB-2400 requires 100% inspection every 10 y. System leakage test is conducted prior to plant startup following each refueling outage, and hydrostatic test at or near the end of each inspection interval. ***(6) Acceptance Criteria:*** Any wear damage in closure stud assemblies are evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3515 and 3517. ***(7) Corrective Actions:*** Repair is in conformance with IWB-4000, repair according to IWB-7000, and material and inspection guidance of RG 1.65. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** The present AMP has been effective in managing the effects of wear to maintain the intended function of closure head stud assembly.	
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, [redacted]	Yes TLAA
Plant-specific aging management program: existing programs may not be capable of mitigating or detecting SCC of vessel flange leak detection line.	Plant-specific aging management program is to be evaluated.	Yes, no AMP
Inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB,	***(1) Scope of Program:*** The program includes preventive measures to mitigate stress corrosion cracking (SCC) and inservice inspection (ISI) to manage the effects of SCC on	No

IV A2-15

OR REPLACE

IV A2-19

IV A2-20A
MORE INFORMATION SUGGESTED

IV A2-17

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
(continued from previous page) EPRI guidelines in TR-105714 to minimize the potential of crack initiation and growth. IV A2-21 RE: REG. GUIDE 1.44	*(continued from previous page)* and monitor and control of water chemistry during shut down, mitigate potential of SCC. However, inadvertent introduction of contaminants into the primary coolant system can occur (IN 84–18). The AMP must therefore rely upon ISI in accordance with ASME Section XI to detect possible degradation. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of IGSCC on the intended function of the CRD mechanism housing by detection and sizing of cracks by inservice inspection (ISI). Inspection requirements of Table IWB 2500-1, examination category B-O specifies volumetric or surface examination extending 1/2 in. each side of the CRD housing welds, including weld buttering. Requirements for training and qualification of personnel and performance demonstration for procedures and equipment are in conformance with Appendices VII and VIII of ASME Section XI, or any other formal program approved by the NRC. Testing category B-P specify visual VT-2 (IWA-5240) examination of all pressure retaining components during system leakage test (IWB-5221) and system hydrostatic test (IWB-5222). ***(4) Detection of Aging Effects:*** Aging degradation of the CRD mechanism housing can not occur without crack initiation and growth; the extent and schedule of inspection assure detection of cracks before the loss of intended function of the CRD housing. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2412 should provide timely detection of cracks. 10% peripheral CRD housings are examined each inspection interval. System leakage test is conducted prior to plant startup following each refueling outage, and hydrostatic test at or near the end of each inspection interval. ***(6) Acceptance Criteria:*** Any IGSCC degradation is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3523. Planar and liner flaws are sized according to IWA-3300 and IWA-3400. ***(7) Corrective Actions:*** Repair is in conformance with IWA-4000 and IWB-4000, replacement according to IWB-7000 and IWA-7000, and reexamination with the requirements of IWA-2200. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Although the primary pressure boundary components of PWRs are generally not affected by SCC because of low dissolved oxygen levels and control of water chemistry, the potential of SCC exists from inadvertent introduction of contaminants into the primary coolant system (IN 84–18). Also, SCC could occur at creviced/cold worked locations	

IVA2-23

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

A2. REACTOR VESSEL (Pressurized Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environment	Aging Effect	Aging Mechanism	References
A2.2[illegible]1	~~Control Rod Drive (CRD) Mechanism~~ NOZZLE	Pressure Housing	SB-166 (Alloy 600)	hemically reated orated ater up to 40°C	Crack Initiation and Growth (CRACKING AT OR NEAR HEAT-AFFECTED ZONE)	~~SCC, IGSCC~~ PWSCC	ASME Section XI, 1989 Edition. NRC GL 97-01. EPRI TR-105714. *Operating Experience* NRC IN 90-10. NRC IN 96-11.

IVA2-24

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination categories B-O for pressure retaining welds in control rod housing, B-E for pressure retaining partial penetration welds, [redacted] and testing category B-P for system leakage. Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes preventive measures to mitigate stress corrosion cracking (SCC) and inservice inspection (ISI) to detect cracks or leakage. However, the program is inadequate to manage the effects of SCC on the intended function of Ni-alloy components. The applicant should review previous evaluations performed in response to Generic Letter (GL) 97-01, as updated with the most current industry susceptibility model and inspection results. ***(2) Preventive Actions:*** Control of halogens, sulfates, and oxygen in the primary water to less than 0.05, 0.05, and 0.005 ppm, respectively, during operation, and monitor and control of water chemistry during shut down, mitigate potential of SCC. [redacted] ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of SCC on the intended function of components by detection (and sizing) of cracks and coolant leakage by ISI. Requirements of Table IWB 2500-1, examination category B-O specifies volumetric or surface examination extending 1/2 in. each side of the CRD housing welds, including weld buttering. B-E specifies visual VT-2 examination of partial penetration welds during the hydrostatic test. [redacted] Training and qualification of personnel and performance demonstration for procedures and equipment are in conformance with [redacted] any NRC approved program. Testing category B-P specify visual VT-2 (IWA-5240) examination of all pressure retaining components during system leakage test (IWB-5221) and system hydrostatic test (IWB-5222). [redacted] ***(4) Detection of Aging Effects:*** Aging degradation of the CRD mechanism housing can not occur without crack initiation and growth. [redacted] ***(5) Monitoring and***	Yes, Elements 2 to 6 should be further evaluated

PW

IVA2-22

IVA2-25

IVA2-22

IVA2-22

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

A2. REACTOR VESSEL (Pressurized Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A2.2.1	Control Rod Drive (CRD) Mechanism	Pressure Housing	SA182, Types 304 or 316, CF-8, SB-166	hemically reated orated ater up to 40°C	Cumulative Fatigue Damage LOSS OF MATERIAL	Fatigue	ASME Section III, 1989 Edition. GSI-190.
A2.2.1	Control Rod Drive (CRD) Mechanism	Pressure Housing	GF-8	hemically reated orated ater up to 40°C	~~Loss of~~ Fracture Toughness REDUCED	Thermal Aging Embrittle-ment	EPRI TR-106092. ASME Section XI, 1989 Edition.

IVA2-29

IVA2-27

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
IVA2-22 (PAGE 3) IVA2-22 (PAGE 3) IVA2-26	*(continued from previous page)* adequacy of the program or develop an integrated long-term program which includes periodic inspection of the most susceptible locations to detect the occurrence of PWSCC. The frequency of subsequent inspections are based on the finding of the initial inspections and crack growth rate models for Ni alloys. ***(6) Acceptance Criteria:*** Any IGSCC degradation is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3523 or IWB-3522. However, the applicant should provide information on crack initiation and crack growth models and the data used to validate these models to verify adequacy of the inspection program and acceptance criteria (GL 97-01). ***(7) Corrective Actions:*** Repair is in conformance with IWA-4000 and IWB-4000, replacement according to IWB-7000 and IWA-7000, and reexamination with the requirements of IWA-2200. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** SCC of Alloy 600 and austenitic SSs has occurred in domestic and foreign PWRs (IN 90-10). Very little attention has been given to the inspection for SCC in Alloy 600 applications other than that associated with steam generator tubes.	IVA2-2
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-dependent aging mechanism that is to be [illegible] for the period of license renewal and [illegible] fatigue [illegible], [illegible] to be [illegible].~~ THE ORIGINAL LICENSING CRITERIA ENVIRONMENTAL EFFECTS ON FATIGUE ARE TO BE ADDRESSED	es ZAA
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. IVA2-30	***(1) Scope of Program:*** The program includes determination of the susceptibility of CASS components to thermal aging based on casting method, Mo content, and percent ferrite, and for "potentially susceptible" components, to account for the synergistic loss of fracture toughness due to neutron embrittlement and thermal aging embrittlement, implement either a supplemental examination of the affected components as part of a 10-year inservice inspection (ISI) program during the license renewal term or a component-specific evaluation to determine the susceptibility to loss of fracture toughness. ***(2) Preventive Actions:*** The program provides no guidance on methods to mitigate thermal aging. ***(3) Parameters Monitored/ Inspected:*** The program specifics depend on the neutron fluence and ferrite content of the component. The criteria in EPRI TR-106092, with some modifications	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

A2. REACTOR VESSEL (Pressurized Water Reactor)

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) and static-cast steels with ≤20% ferrite are not susceptible. For high-Mo content (2.0 to 3.0 wt.%) steels, static-cast steels with >14% ferrite and centrifugal-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement, static-cast steels with ≤14% ferrite and centrifugal-cast steels with ≤20% ferrite are not susceptible. Ferrite content will be calculated by the Hull's equivalent factors or a method producing an equivalent level of accuracy (±6% deviation between measured and calculated values). ***(4) Detection of Aging Effects:*** For all CASS components that have a neutron fluence greater than 10^{17} n/cm^2 (E>1 MeV), implement a 10-year ISI program during the renewal period that includes supplemental inspection that covers portions of the susceptible components determined to be limiting from the standpoint of thermal aging susceptibility (Ferrite and Mo contents, casting process, and operating temperature), neutron fluence, and cracking susceptibility (applied stress, operating temperature, and environmental conditions). The inspection technique, including the reliability in detecting the features of interest (crack appearance and size) in assuring the integrity of the component, should be specified. For example, enhancement of the visual VT-1 examination to achieve a 1/2-mil (0.0005 in.) resolution, with the conditions (lighting and surface cleanness) for the ISI bounded by those used to demonstrate the resolution of the inspection technique. Alternatively, perform a component-specific evaluation including a mechanical loading assessment to determine the maximum tensile loading on the component during ASME Code Level A, B, C, and D conditions. If the loading is compressive or low enough (<5 ksi) to preclude fracture, then supplemental inspection of the component is not required. Failure to meet this criteria requires continued use of supplemental inspection program. For all CASS components that have a neutron fluence of less than 10^{17} n/cm^2 (E>1 MeV), implement the supplement inspection program if they are "potentially susceptible" to thermal aging, the existing ASME Section XI inspection requirements are considered adequate if the components are "not susceptible" to thermal aging. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 should provide timely detection of cracks. ***(6) Acceptance Criteria:*** Flaws detected in CASS components are evaluated in accordance with the applicable procedures of IWB-3500. ***(7) Corrective Actions:*** Repair is in conformance with IWA-4000 and IWB-4000, and replacement according to IWA-7000 and IWB-7000. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are	

IV A2-40

IV A2-28

IVA2-31

IVA2-32

IVA2-35

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

A2. REACTOR VESSEL (Pressurized Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environment	Aging Effect	Aging Mechanism	References
A2.3.1 thru A2.3.3	Nozzles	Inlet, Outlet, Safety Injection	SA336, SA508 with SS Cladding	hemically reated orated ater up to 40°C	~~Loss of~~ Fracture Toughness REDUCTION IN	Neutron Irradiation Embrittlement	Appendices G & H to 10 CFR Part 50. Reg. Guide 1.99. NRC GL 88-11. NRC GL 92-11, Rev. 1. ASME Section XI, 1989 Edition.
A2.3.1 thru A2.3.3	Nozzles	Inlet, Outlet, Safety Injection	SA336, SA508 with SS Cladding	hemically reated orated ater up to 40°C	Crack Initiation and Growth	~~SCC, IGSCC~~	ASME Section XI, 1989 Edition. Reg. Guide 1.43. Reg. Guide 1.44. EPRI TR-105714.
A2.3.1 thru A2.3.3	Nozzles	Inlet, Outlet, Safety Injection	SA336, SA508 with SS Cladding	hemically reated orated ater up to 40°C	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. GSI-190.
A2.4.1 thru A2.4.3	Nozzle Safe Ends	Inlet, Outlet, Safety Injection	SS, CASS (NiCrFe buttering, and SS or NiCrFe weld)	hemically reated orated ater up to 40°C	Crack Initiation and Growth	SCC, IGSCC	ASME Section XI, 1989 Edition. NRC GL 97-01. EPRI TR-105714.
A2.4.1 thru A2.4.3	Nozzle Safe Ends	Inlet, Outlet, Safety Injection	SS, SB-166 (Alloy 182 butter, and Alloy 82 or 182 weld)	hemically reated orated ater up to 40°C	Cumulative Fatigue Damage CRACK INITIATION AND GROWTH	Fatigue PWSCC	ASME Section III, 1989 Edition ANSI B31.1 GSI-190

IVA2-67 WHOLE SECTION

IVA2-36

IVA2-39

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) toughness due to neutron and thermal aging embrittlement on the intended function of CASS components.	
For a 40 y design life, pressure vessel integrity is assured by fracture toughness and material surveillance program requirements set forth in Appendices G and H to 10 CFR Part 50, and methodology of Regulatory Guide 1.99, Rev. 2, implemented through NRC Generic Letters (GLs) 88-11 and 92-01 Rev. 1 and Supplement 1, to predict effects of neutron irradiation on reactor vessel materials. In addition, inservice inspection of ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, examination categories B-D for nozzle-to-vessel welds, and testing category B-P for system leakage.	Neutron irradiation embrittlement is a time-limited aging analysis (TLAA) to be evaluated for the period of license renewal for all ferritic materials that have a neutron fluence of greater than 10^{17} n/cm^2 (E>1 MeV) at the end of the license renewal term. The TLAA should evaluate the impact of neutron embrittlement on: (a) the adjusted reference temperature, the plant's pressure temperature limits, and the need for inservice inspection of circumferential and axial reactor vessel welds, (b) the Charpy upper shelf energy, and (c) the equivalent margin 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 IVA2-34 AVAILABLE TO UE
Same as for the effect of Stress Corrosion Cracking of Item A2.1.1 closure head dome.	*Same as for the effect of Stress Corrosion Cracking of Item A2.1.1 closure head dome.*	No
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or Section I (Power Boilers) and Section III, Division 1 (Unfired Pressure Vessel).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA THE ORIGINAL LICENSING CRITERIA ER
Same as for the effect of Stress Corrosion Cracking of Item A2.2.1 CRD mechanism pressure housing constructed of Ni-alloy. IVA2-37	*Same as for the effect of Stress Corrosion Cracking of Item A2.2.1 CRD mechanism pressure housing constructed of Ni-alloy.* IVA2-37	Yes Elements 2 to 5 should be further evaluated
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or ANSI B31.1, or other evaluations based	~~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.~~ ENVIRONMENTAL EFFECTS ON FATIGUE ARE TO BE ADDRESSED	Yes TLAA

IVA2-30

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

A2. REACTOR VESSEL (Pressurized Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A2.5.1, A2.5.2	Vessel Shell	Upper Shell, Intermediate and Lower Shell (Including Beltline Welds)	SA302-Gr B SA533-Gr B SA336, SA508-Cl 2 or Cl 3 with Types 308 or 309 Cladding	hemically reated orated ater up to 40°C	~~Loss of~~ Fracture Toughness REDUCED	Neutron irradiation Embrittle-ment	Appendices G & H to 10 CFR Part 50. Reg. Guide 1.99. NRC GL 88-11. NRC GL 92-11, Rev. 1. ASME Section XI, 1989 Edition. NRC IN 98-42.
A2.5.1 thru A2.5.3	Vessel Shell	Upper (Nozzle) Shell, Intermediate and Lower Shell, Vessel Flange	SA302-Gr B SA533-Gr B SA336, SA508 with SS Cladding	hemically reated orated ater up to 40°C	Cumulative Fatigue Damage CRACKING @ WELD JOINTS	Fatigue	ASME Section III, 1989 Edition. GSI-190.
A2.5.3	Vessel Shell	Vessel Flange	SA336, SA508 with SS Cladding	ir, Leaking hemically reated orated ater nd/or team up to 40°C	Loss of Material	Corrosion/ Boric Acid Wastage of External Surfaces AND WEAR	NRC GL 88-05. ASME Section XI, 1989 Edition.
A2.6	Core Support Pads (Lugs)	-	~~SB-166, SB-168~~ Alloy 600	hemically reated orated ater up to 40°C	Crack Initiation and Growth	~~SCC, IGSCC~~ PWSCC	ASME Section XI, 1989 Edition. NRC GL 97-01. EPRI TR-105714.
A2.6	Core Support Pads (Lugs)	-	~~SB-166~~, SB-168	hemically reated orated ater up to 40°C	Attrition	[redacted]	ASME Section XI, 1989 Edition.

IVA2-42

IVA2-43

IVA2-44

IVA2-46

IVA2-48

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
For a 40 y design life, pressure vessel integrity is assured by fracture toughness and material surveillance program requirements set forth in Appendices G and H to 10 CFR Part 50, and methodology of Regulatory Guide 1.99, Rev. 2, implemented through NRC Generic Letters (GLs) 88-11 and 92-01 Rev. 1 and Supplement 1, to predict effects of neutron irradiation on reactor vessel materials. In addition, inservice inspection of ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, examination category B-A of all pressure retaining welds in the vessel and repair welds in beltline region, defined as the region extending for the length of the thermal shield or effective length of reactor fuel elements. Based on NRC Information Notice 98-42, relief from inspection is in accordance with 10 CFR 50.55a (g) (5) (iii). IV A2-41	Neutron irradiation embrittlement is a time-limited aging analysis (TLAA) to be evaluated for the period of license renewal for all ferritic materials that have a neutron fluence of greater than 10^{17} n/cm^2 (E>1 MeV) at the end of the license renewal term. The TLAA should evaluate the impact of neutron embrittlement on: (a) the adjusted reference temperature, the plant's pressure temperature limits, and the need for inservice inspection of circumferential and axial reactor vessel welds, (b) the Charpy upper shelf energy, and (c) the equivalent margins analyses performed in accordance with 10 CFR 50, Appendix G. Reactor surveillance program requires that the existing reactor vessel material surveillance program be evaluated to determine whether there is sufficient material data and dosimetry to monitor irradiation embrittlement at the end of the license renewal term and whether operating restrictions (i.e., inlet temperature, neutron spectrum and flux) are necessary. If surveillance capsules are not removed during the license renewal term, it will be necessary to establish operating restrictions to ensure the plant is operated within the environment of the surveillance capsules.	Yes TLAA
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III, edition specified in 10 CFR 50.55a), Subsection NB, or Section I (Power Boilers) and Section VIII, Division 1 (Unfired Pressure Vessel). IV A2-43 THE ORIGINAL LICENSING CRITERIA OR	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.~~ ENVIRONMENTAL EFFECTS ON FATIGUE ARE TO BE ADDRESSED	Yes TLAA
Implementation of NRC Generic Letter 88-05, and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, testing category B-P for system leakage. IV A2-44	*Same as Corrosion/Boric Acid Wastage of external surfaces of Item A2.1.1 Closure Head Dome.*	No
Same as for the effect of Stress Corrosion Cracking of Item A2.2.1 CRD mechanism pressure housing constructed of Ni-alloy	*Same as for the effect of Stress Corrosion Cracking of Item A2.2.1 CRD mechanism pressure housing constructed of Ni-alloy.* IV A2-47	Yes Elements 2 to 6 should be further evaluated
Inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination category B-N-2 for integrally-welded core support structures.	***(1) Scope of Program:*** The program relies on inservice inspection (ISI) to manage the effects of wear on the intended function of core support pads. ***(2) Preventive Actions:*** Design requirements, frequent inspection, and timely corrective action prevent or mitigate attrition due to wear of core support pads. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of wear on the intended function of core support pads by	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

A2. REACTOR VESSEL (Pressurized Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
A2.7.1	Bottom Head	Dome	SA302-Gr B SA533-Gr B with SS Cladding	hemically reated orated ater up to 40°C	Crack Initiation and Growth	SCC, IGSCC	ASME Section XI, 1989 Edition. Reg. Guide 1.43. Reg. Guide 1.44. EPRI TR-105714.
A2.8.1	Penetrations	CRD Mechanism HEAD ADAPTER	SB-166 Lower Housing Tube with SA182, Types 304 or 316 Upper Housing Flanges	hemically reated orated ater up to 40°C	Crack Initiation and Growth	SCC, IGSCC	ASME Section XI, 1989 Edition. NRC GL 97-01. EPRI TR-105714.
A2.8.2	Penetrations	Instrumentation Tubes	SB-166, SB-167	hemically reated orated ater up to 40°C	Crack Initiation and Growth	SCC, IGSCC	ASME Section XI, 1989 Edition. NRC GL 97-01. EPRI TR-105714.
A2.8.3	Penetrations	Leakage Monitoring Tubes	SB-166, SB-167 with SA312, Type 316 Extensions	hemically reated orated ater up to 40°C	Crack Initiation and Growth	SCC, IGSCC	ASME Section XI, 1989 Edition. NRC GL 97-01. EPRI TR-105714. NRC 90-10

IVA2-49

IVA2-50

IVA2-52

IVA2-54

IVA2-55

IVA2-56

IVA2-57

IVA2-58

IVA2-59

IV A2-60

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

Existing Aging Management Program (AMP)			Evaluation and Technical Basis					Further Evaluation
A2.8.1, A2.8.2	Penetrations	CRD Mechanism, Instrumentation Tubes HEAD ADAPTER	SB-166, SB-167	hemically reated orated ater up to 40°C	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. GSI-190.	
			the loss of intended function of the core support pads. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI should provide for timely detection of attrition. All accessible welds of core support pads are inspected each inspection period. ***(6) Acceptance Criteria:*** Any wear is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and 3520.2 for visual examination. Visual examinations that reveal relevant conditions are supplemented by surface and volumetric examinations (IWB-3200) for flaw characterization, analytical evaluation, corrective measures, and repairs. Acceptance of the component for continued service is based on volumetric and surface examination, repair, replacement, or analytical evaluation (IWB-3132). ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWA-4000 and IWB-4000, and reexamination in accordance with requirements of IWA-2200. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** The present AMP has been effective in managing the effects of wear to maintain the intended function of core support pads.					
Same as for the effect of Stress Corrosion Cracking of Item A2.1.1 closure head dome.			*Same as for the effect of Stress Corrosion Cracking of Item A2.1.1 closure head dome.*					No
Same as for the effect of Stress Corrosion Cracking of Item A2.2.1 CRD mechanism pressure housing constructed of Ni-alloy. IV A2-51			*Same as for the effect of Stress Corrosion Cracking of Item A2.2.1 CRD mechanism pressure housing constructed of Ni-alloy.*					Yes Elements 2 to 6 should be further evaluated
Same as for the effect of Stress Corrosion Cracking of Item A2.2.1 CRD mechanism pressure housing constructed IV A2-54			*Same as for the effect of Stress Corrosion Cracking of Item A2.2.1 CRD mechanism pressure housing constructed of Ni-alloy.*					Yes Elements 2 to 6

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

A2. REACTOR VESSEL (Pressurized Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environment	Aging Effect	Aging Mechanism	References
Same as for the effect of Stress Corrosion Cracking of Item A2.2.1 CRD mechanism pressure housing constructed of Ni-alloy.			*Same as for the effect of Stress Corrosion Cracking of Item A2.2.1 CRD mechanism pressure housing constructed of Ni-alloy.* IV A2-58				Yes Elements 2 to 6 should be further evaluated
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III, Edition specified in 10 CFR 50.55a, Subsection NB, or Section I (Power Boilers) and Section VIII, Division 1 (Unfired Pressure Vessel). IV A2-61 THE ORIGINAL LICENSING CRITERIA OR			~~Fatigue is a time-limited aging analysis that has to be performed for the period of license renewal, and Generic Safety Issue (GSI) 190 is to be addressed.~~ ENVIRONMENTAL EFFECTS ON FATIGUE ARE TO BE ADDRESSED				TLAA
A2.9.1	Pressure Vessel Support	Skirt Support	SA302-Gr B SA533-Gr B SA516-Gr70, SA-36	air	Cumulative Fatigue Damage	Fatigue IV A2-62	Original Licensing Criteria or ASME Section III, 1989 Edition.
A2.9.2, A2.9.3	Pressure Vessel Support	Cantilever/ Column Support, Neutron Shield Tank,	SA302-Gr B SA533-Gr B SA516-Gr70, SA-36	air	~~Loss of~~ Fracture Toughness REDUCED IV A2-64	Neutron Irradiation Embrittlement IV A2-63	Reg. Guide 1.99, Revision 2. NUREG-0933.

LOSS OF MATERIAL BY BORIC ACID WASTAGE IV A2-65

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

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

IV A2-6

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IV

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV A2 - 1	Reactor Vessel (PWR)	A2.1.1, A2.1.2	IV A2-4, 6, 8		The "Region of Interest" is the outer surface of the vessel dome/flange. The stainless steel cladding reference in the "Material" column should be deleted.
IV A2 - 2	Reactor Vessel (PWR)	1.1	IV A2-4		For the Aging Mechanism Corrosion/Boric Acid Wastage, the "Existing AMP" entry lists both a Boric Acid Corrosion Inspection Program (meeting the requirements of GL 88-05) and ISI. The Boric Acid Corrosion Inspection Program itself was deemed adequate for aging management in NUREG-1705. The References, Existing AMP, Evaluation and Technical Basis, and Further Evaluation entries should be rewritten to correspond to the provided example.
IV A2 - 3	Reactor Vessel (PWR)	1.1, 1.3, 1.5, 2.1, 3.1, 3.3, 4.1, 4.3, and 7.1	IV A2-4, 6, 8, 12, 14, 16, 22, 26		SCC/IGSCC is listed as an Aging Mechanism. Stainless steel subcomponents which have been stress relieved are not susceptible to SCC because of the low total stress and the low oxygen content of the primary coolant. Stainless steel subcomponents with fluences less than 5E20 n/cm2 are not susceptible to IASCC. Those RPV subcomponents fabricated from alloy steel not in contact with reactor coolant fluid are not susceptible to SCC or IGSCC. All low alloy ferritic steels are fully stress relieved and experience relatively low total stress. This combined with water chemistry control also help make these ARDMs unlikely for the RPV and CEDM subcomponents. Low alloy steel bolting material with a specified maximum hardness of 42 Rc or less is not susceptible to SCC. Therefore, for SCC, chemistry programs in accordance with industry guidelines alone should be credited. A similar position was accepted in NUREG-1705. The References, Existing AMP, Evaluation and Technical Basis, and Further Evaluation entries should be rewritten to correspond to the provided example.
IV A2 - 4	Reactor Vessel (PWR)	1.1	IV A2-4		SCC/IGSCC is listed as an Aging Mechanism for the Closure Head Dome. (See previous, IV A2 -3, global comments on programs for SCC.) Reg. Guide 1.43 and 1.44 Guidelines are not an AMP, they

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
					are a design feature and as such should be listed in the Material entry. In addition, many vessels were fabricated before the issuance of the Reg. Guides so this information should be qualified. Evaluation and Technical Basis entry Item (1) states "SCC is not anticipated to be an issue for the Closure Head Dome or Nozzles". If SCC is not an issue this entry should be removed.
IV A2 - 5	Reactor Vessel (PWR)	A2.1.1	IV A2-4		ASTM A508-64 Class 2 with stainless steel cladding should be added as a material
IV A2 - 6	Reactor Vessel (PWR)	A2.1.1	IV A2-6		For Aging Effect = Crack Initiation and Growth (SCC and IGSCC), this effect should be deleted. Conditions do not exist to support this mechanism for the RV materials. Refer to item (1) under Technical Basis, "However, SCC is not anticipated to be an issue for the closure head dome or nozzles because analytical evaluations indicate that cracks in the SS cladding will stop growing in the ferritic base metal."
IV A2 - 7	Reactor Vessel (PWR)	A2.1.1	IV A2-6		Same comment as IV A2-5 with regard to material. The aging effect should be cracking at welded joints. Neither SCC nor IGSCC are credible aging mechanisms for ferritic pressure boundary metals with the possible exception of high-strength bolting. To understand the correct mechanisms, a review of the basis for the ASME Section XI inspections must be performed--see section B-1of the EPRI document entitled Nondestuctive Examination Acceptance Standards (EPRI NP-1406-SR, May 1980). In section B, page B-1, the third paragraph states "Components fabricated in accordance with these construction codes were presumed to contain material and fabrication flaws whose sizes and character are below the detection threshold of the examination method employed, or less that the acceptance standards." "The presence of such flaws, although not detrimental to the structural integrity of the newly constructed component, lead to the recognition that these flaws might grow in size as a consequence of

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
					loadings imposed on the component during the service lifetime. The inservice examinations of components are intended to detect significant flaw growth and development." The acceptance standards in Section XI are based on this foundation and not on SCC and IGSCC. Cracking of the cladding by SCC and IGSCC is unlikely owing to primary water chemistry requirements but should be called out separately if the staff requires that an AMP be listed to manage this effect. Cracking by fatigue is not listed, is applicable to all RV pressure boundary items, and is a TLAA. For some reason cumulative fatigue damage is called out only for selected RV items.
IV A2 - 8	Reactor Vessel (PWR)	A2.1.2	IVA2-7		It is not clear why Reg. Guide 1.43 and 1.44 are listed as AMPs. These guidelines are followed during fabrication, but for all B&W-designed vessels these Reg. Guides were not in place when the vessels were fabricated. Is the staff referring to replacement items versus originally fabricated items? Underclad cracking of base metal (508 Class 2 forgings made using a coarse grain practice and clad using a high heat-input cladding process) may contain underclad cracks that need to be addressed for the period of extended operation. Analytical evaluation of the cracks is a TLAA for all of the B&W operating plants. See BAW-2251A, Appendix C, for a complete description of the treatment of underclad cracking for the B&W operating plants for license renewal.
IV A2 - 9	Reactor Vessel (PWR)	A2.1.2	IV A2-8		Clarify A508 Class 2. Believe Class 1 is carbon steel and not low-alloy steel.
IV A2 - 10	Reactor Vessel (PWR)	A2.1.3	IV A2-8, 9		This item identifies both SCC and IGSCC as aging mechanisms (effects) for vessel closure studs. The reference to IGSCC in the

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
					"Aging Mechanism" and "Evaluation and Technical Basis" columns should be deleted. The aging mechanism (effect) of concern is only SCC for this class of materials. This deletion will not affect the technical content of the item.
IV A2 - 11	Reactor Vessel (PWR)	A2.1.3	IV A2-8		SCC/IGSCC is listed as an Aging Mechanism for the Closure Head Studs. (See previous, IV A2 -3, global comments on programs for SCC.) Reg. Guide 1.65 Guidelines are not an AMP; they are a design feature and as such should be listed in the Material entry. In addition, many vessels were fabricated before the issuance of the Reg. Guide so this information should be qualified. Preservice inspection identified in Evaluation and Technical Basis entry Item (2) could not detect aging because the components have not yet been placed in service.
IV A2 - 12	Reactor Vessel (PWR)	A2.1.3	IV A2-8, 10, 12		Add material **SA193-6.**
IV A2 - 13	Reactor Vessel (PWR)	A2.1.3	IV A2-8		Reg. Guide 1.65 is only used if the studs are replaced. Otherwise Section XI is used to manage applicable aging effects. See BAW-2251A, Appendix E, response to RAI 14. This comment is similar to comment, IVA2-8, regarding Reg. Guide 1.43 and 1.44.
IV A2 - 14	Reactor Vessel (PWR)	A2.1.3	IV A2-10		Wear is listed as an Aging Mechanism for the Closure Head Studs. The ISI Program credited includes both volumetric and visual techniques for Aging Management. Visual techniques alone were considered adequate in NUREG-1705. The requirement for the volumetric examination should be removed.
IV A2 - 15	Reactor Vessel (PWR)	A2.1.3	IV A2-11		Under Technical Basis (3), last sentence. "Testing category B-P specifies visual VT-2 (IWA-5240) examination of all pressure retaining components ~~extending to and including the second closed valve at the boundary extremity.~~ during system leakage test (IWB-5221) and system hydrostatic test(IWB-5222)

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
					Under Technical Basis (7) Include **Replacement** as an available Corrective Action
IV A2 - 16	Reactor Vessel (PWR)	A2.1.3	IV A2-13		For Aging Mechanism = WEAR, Under Technical Basis (3), last sentence. "Testing category B-P specifies visual VT-2 (IWA-5240) examination of all pressure retaining components ~~extending to and including the second closed valve at the boundary extremity.~~ during system leakage test (IWB-5221) and system hydrostatic test(IWB-5222) Under Technical Basis (7) Include **Replacement** as an available Corrective Action
IV A2 - 17	Reactor Vessel (PWR)	A2.1.3	IV A2-13		For Stud Assembly , Aging Mechanism = Fatigue, under Technical Basis, delete the text "~~and Generic Safety Issue (GSI) 190 is to be addressed~~" The studs are not in a reactor water environment.
IV A2 - 18	Reactor Vessel (PWR)	A2.1.3	IV A2-10		Aging Effect--Suggest replacing attrition with loss of material. Attrition is grinding down by friction, which is loss of material.
IV A2 - 19	Reactor Vessel (PWR)	A2.1.5	IV A2-12		For most WOG plants, this component does not require an aging management review since it does not perform any "intended" function.
IV A2 - 20	Reactor Vessel (PWR)	2.1	IV A2-12		SCC/IGSCC is listed as an Aging Mechanism for the CRDM Pressure Housing. (See previous global comments on programs for SCC.) RG 1.44 is described as an AMP under the Evaluation and Technical Basis entry Item (2), despite having not been previously listed in the Existing AMP entry. Irrespective of this inconsistency, Reg. Guide 1.44 Guidelines are not an AMP; they are a design feature and as such should be listed in the Material entry. In addition, many vessels were fabricated before the issuance of the Reg. Guide so this information should be qualified.

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV A2 - 20A	Reactor Vessel (PWR)	2.1	IV A2-12		The "Existing AMP" entry states that components have been designed for fatigue for a 40 year design life in accordance with ASME III. This is not strictly true; it has been assumed that the components in question will experience certain operational stresses during the plant lifetime, for example, less than or equal to 500 thermal cycles caused by plant startups and shutdowns. This number of transients will have originally been estimated with some consideration of a 40-year plant life, but there is no explicit correlation to a time period in this evaluation. Also, simply identifying Fatigue as a TLAA does not identify an AMP per se. More information should be provided, for example, relevant to Fatigue Monitoring Programs that calculate Cumulative Usage Factors based on partial cycles.
IV A2 - 21	Reactor Vessel (PWR)	A2.2.1	IV A2-12		I assume that the item referred to here is the CRDM motor tube housing for the B&W-design. The motor tube is bolted to a stainless steel flange that is connected to the Alloy 600 CRDM nozzle. The CRDM nozzles are attached to the RV closure head using partial penetration welds. Material—B&W designs use Type 403 SS in the center section of the Type C drive, and low-alloy steel (508 Class 2) clad with Alloy 82/182 is used for the center section of the Type A and B drives. Aging Effect-Cracking anywhere or cracking at welded joints. If the mechanism is SCC or IGSCC, cracking most likely to occur at welded joints for stainless steel items. Growth of fabrication flaws most likely at welded joints. References and AMP--Same comment, IVA2-8, as previous with regard to Reg. Guide 1.44.
IV A2 - 22	Reactor Vessel (PWR)	A2.2.1	IV A2-17 through IV		This is one of the 17 open license renewal technical issues identified and described in EPRI TR-107521 (NEI/NRC 98-0033; PWSCC of

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
			A2-19		High-Nickel Alloys). Elements 2 through 6 are identified as requiring further evaluation in the "Evaluation and Technical Basis" column for PWSCC of Ni-Fe-Cr alloys used for CRDM pressure housings. The text implies that requirements are being placed on license renewal applicants that cannot be justified for PWSCC in current operating plants. A number of changes to this text would place the appropriate requirements in context. For example, Element 1 already states that "The applicant should review previous evaluations performed in response to Generic Letter (GL) 97-01, as updated with the most current industry susceptibility model and inspection results." In Element 2, the last sentence should read "Based upon industry actions, as represented by the responses to NRC Information Notices (IN) 90-10 and 96-11, and Generic Letter (GL) 97-01, the susceptibility of Ni-alloys and welds to primary water SCC is currently being addressed, even for cases when demineralizer resins contaminate the reactor coolant system." To state otherwise is to imply that an unresolved safety question exists for current operating plants. Element 3 contains inservice examination program elements that do not apply to the CRD pressure housings. IGSCC should be changed to PWSCC. Examination Category B-O and Examination Category B-P are the only program elements that apply. The others apply for other Ni-alloy components, but not the CRD pressure housings. Furthermore, the reference to ASME Section XI Appendices VII and VIII should be removed, since there are no supplements to Appendix VIII that apply for the welds to be inspected. Finally, the last sentence of Element 3 should read "The applicant should continue the commitments made in response to Generic Letter 97-01 through the license renewal term, including susceptibility findings leading to augmented surface or volumetric inspections." The second sentence of Element 4 should read "The applicant

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
					should continue the commitments made in response to Generic Letter 97-01, in particular any visual (leak detection), surface, or volumetric inspections needed to assure detection of cracks before the loss of intended function of the components. The third sentence of Element 5 should read "The applicant should review the technical basis used in the response to Generic Letter 97-01 for determining the adequacy of the integrated long-term program which includes periodic inspection of the most susceptible locations to detect the occurrence of PWSCC." The last sentence of Element 6 should read "The applicant should provide relevant information from the response to Generic Letter 97-01 on crack initiation and crack growth models that are intended to verify the adequacy of the inspection program and acceptance criteria."
IV A2 - 23	Reactor Vessel (PWR)	A2.2.1	IV A2-16		Material--The material is SB-166 and this should be classified as a nozzle since it is attached to the RV using partial penetration welds. Aging Effect—Suggest cracking at or near the heat-affected zone Aging Mechanism--The aging mechanism should be primary water stress corrosion cracking (PWSCC).
IV A2 - 24	Reactor Vessel (PWR)	A2.2.1,	IV A2-16		Under Material, add **Alloy 600**, as the more common name
IV A2 - 25	Reactor Vessel (PWR)	A2.2.1	IV A2-17		Under Existing AMP, first sentence, "……examination categories B-O for pressure retaining welds in control rod housing, B-E for pressure retaining partial penetration welds, ~~B-F for pressure retaining dissimilar metal welds, B-N-2 for integrally-welded core support structure~~, and testing category B-P for system leakage." Under Technical Basis, (3) Parameters Monitored / Inspected, "……~~B-F specifies for all nozzle-to-safe end butt welds NPS 4 or~~

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
					~~larger, volumetric and surface examination of ID region and surface examination of OD surface; only surface examination is conducted for all butt welds less than NPS 4 and for al nozzle-to-safe end socket welds. Category B-N-2 specifies visual VT-3 of accessible welds of core support pads.~~ Under "Further Evaluation" clarify was is expected for "Yes, Elements 2 to 6 should be further evaluated."
IV A2 - 26	Reactor Vessel (PWR)	A2.2.1	IV A2-19		For Aging Mechanism = Fatigue, under Existing AMP, "Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of **the original licensing criteria or** ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or Section I (Power Boilers) and Section VIII, Division 1 (Unfired Pressure Vessel). Under "Technical Basis" recognize that GSI-190 has been closed with no additional regulatory requirements. Change the wording to say "**environmental effects on fatigue are to be addressed.**"
IV A2 - 27	Reactor Vessel (PWR)	A2.2.1	IV A2-18		For Material = CF8 and Aging Effect = ~~Loss of~~ **Reduced** Fracture Toughness
IV A2 - 28	Reactor Vessel (PWR)	A2.2.1	IV A2-21		For Material = CF8 and Aging Mechanism = Thermal Aging Embrittlement, the section on Existing AMP implies a neutron fluence threshold to initiate action. In the subsequent Technical Basis, a threshold value of 10^{17} n/cm^2 is mentioned. WOG does not agree with this value and feels it should be 10^{21}. Last sentence in (4) "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."~~

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV A2 - 29	Reactor Vessel (PWR)	A2.2.1	IV A2-22		ADD NEW AGING EFFECT = Loss of Material, Aging Mechanism = Corrosion /Boric Acid Wastage. Existing AMP uses implementation of GL 88-05, Section XI examination category B-E. For Technical Basis, use information for aging effects = SCC, PWSCC
IV A2 - 30	Reactor Vessel (PWR)	A2.2.1	IV A2-21		For CF-8 material. AMP--The ferrite screening criteria is not consistent with the NRC-specified screening criteria contained in the Oconee license renewal application. Reduction of fracture toughness by thermal embrittlement is an applicable aging effect for CASS items under the following circumstances. **High-Mo Content** Statically Cast and ferrite > 10% Centrifugally Cast and Ferrite > 20% **Low-Mo Content** Statically Cast and Ferrite > 20% Low Moly Centrifugally Cast Not Susceptible to Thermal Embrittlement • Flaws in CASS with ferrite levels less than 25% and no niobium may be evaluated using ASME Section XI, IWB-3640 procedures • Flaws in CASS with ferrite levels exceeding 25% must be evaluated using IWB-3640 procedures. If such flaws occur, fracture toughness data will be presented on a case-by-case basis

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV A2 - 31	Reactor Vessel (PWR)	A2.3.1 through A2.3.3	IV A2-23		This is one of the 17 open license renewal technical issues identified and described in EPRI TR-107521 (NEI/NRC 98-0036; Neutron Irradiation Embrittlement – Definition of Reactor Vessel Beltline). See the comment on A1.2.4. The same text change is recommended here.
IV A2 - 32	Reactor Vessel (PWR)	A 2.3.1 through A2.3.3	IV A 2-22		Aging Effect—Suggest Reduction of Fracture Toughness if within the beltline region for the period of extended operation versus Loss of Fracture Toughness Reduction of fracture toughness is applicable to the nozzles only if they are within the beltline region for license renewal. The 1.0E17 criteria is from 10 CFR 50, Appendix H, and is the criteria for which a surveillance program is needed if end-of-life fluence exceeds 1.0E17 n/cm2 (E>1.0MeV). This criteria should not be used to determine if a ferritic item is or is not susceptible to reduction of fracture toughness. ***Beltline, as defined in 10 CFR 50.61(a)(3)***, is the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection for the most limiting material with regard to radiation damage. The fluence at the inlet/exit nozzles for the B&W-designed plants exceeds 1.0E17 at 48 EFPY but is not the most limiting material and is not within the beltline region for license renewal—see BAW-2251A. The B&WOG performed a fracture mechanics evaluation to justify this position and other license renewal applicatnts should have this option as well since the fluence at some portion of the nozzles will most likely exceed 1.0E17 at 48 EFPY.

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV A2 - 33	Reactor Vessel (PWR)	3.1, 3.2, and 3.3	IV A2-22		Neutron Irradiation Embrittlement is listed as an Aging Mechanism for the Nozzles. The nozzles are not located near the beltline region of the reactor vessel, and as such do not experience the conditions necessary for this Aging Mechanism. This position was accepted in NUREG-1705. This entry should be removed.
IV A2 - 34	Reactor Vessel (PWR)	3.1, 3.2, and 3.3	IV A2-22		Neutron Irradiation Embrittlement is listed as an Aging Mechanism for the Nozzles. (This mechanism was considered Not Plausible for CCNPP; see separate comments on the Aging Mechanism elsewhere. Comments to follow assume that the Aging Mechanism is retained in Report despite earlier comment.) Recommend next to last sentence under Evaluation and Technical Basis entry to read "material data and/or dosimetry". Recommend revising last sentence under Evaluation and Technical Basis entry to read, "are not available to be removed".
IV A2 - 35	Reactor Vessel (PWR)	A2.3.1 thru A2.3.3,	IV A2-22		For Aging Mechanism = Neutron Irradiation Embrittlement, change Aging Effect = ~~Loss of~~ **Reduced** Fracture Toughness
IV A2 - 36	Reactor Vessel (PWR)	A2.3.1 thru A2.3.3,	IV A2-22		For Aging Effect = Crack Initiation and Growth ; Aging Mechanism = SCC, IGSCC, this effect should be deleted. Conditions do not exist to support this mechanism for the RV materials. Refer to item (1) under Technical Basis for A2.1.1, "However, SCC is not anticipated to be an issue for the closure head dome or nozzles because analytical evaluations indicate that cracks in the SS cladding will stop growing in the ferritic base metal."
IV A2 - 37	Reactor Vessel (PWR)	A2.4.1 thru A2.4.3,	IV A2-23		For Aging Mechanism = SCC, IGSCC, the Existing AMP entry refers to the item for A2.2.1 CRDM Pressure Housing. The following changes apply: First sentence, ".....examination categories ~~B-O for pressure retaining welds in control rod housing~~, ~~B-E for pressure retaining partial penetration welds~~, B-F for pressure retaining dissimilar metal welds, ~~B-N-2 for integrally-welded core support structure~~, and testing category B-P for system leakage."

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
					Under Technical Basis, (3) Parameters Monitored / Inspected, "……., examination category ~~B-O specifies volumetric or surface examination extending ½ in. each side of the CRD housing welds, including weld buttering~~. ~~B-E specifies visual VT-2 examination of partial penetration welds during the hydrostatic test~~ B-F specifies for all nozzle-to-safe end butt welds NPS 4 or larger, volumetric and surface examination of ID region and surface examination of OD surface; only surface examination is conducted for all butt welds less than NPS 4 and for all nozzle-to-safe end socket welds. ~~Category B-N-2 specifies visual VT-3 of accessible welds of core support pads.~~ Under "Further Evaluation" clarify what is expected for "Yes, Elements 2 to 6 should be further evaluated."
IV A2 - 38	Reactor Vessel (PWR)	A2.4.1 through A2.4.3	IV A2-22		For Aging Mechanism = Fatigue, Material should be the same as for Aging Mechanism = SCC, IGSCC For Aging Mechanism = Fatigue, Existing AMP = "Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of **the original licensing criteria or** ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or ANSI B31.1, or other evaluations based on cumulative usage factor (CUF) Under "Technical Basis" recognize that GSI-190 has been closed with no additional regulatory requirements. Change the wording to say "**environmental effects on fatigue are to be addressed.**"
IV A2 - 39	Reactor Vessel (PWR)	A2.4.1 through A2.4.3	IV A2-22		Material--SB-166 Aging Effect—Cracking initiation and growth was omitted Mechanism—PWSCC should be listed. For Alloy 600 items other than the CRDM nozzles, a separate Alloy 600 program may be required.

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV A2 - 40	Reactor Vessel (PWR)	A2.2.1	IV A2-21		This is one of the 17 open license renewal technical issues identified and described in EPRI TR-107521 (NEI/NRC 98-0030; Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components). The text in the columns "Aging Management Program" and "Evaluation and Technical Basis" represents an excellent summary of technical agreements reached between the NRC staff and the industry on this issue. In particular, the alternative of mechanical loading assessment in Element 4 and the associated acceptance criteria are to be commended. One change to the discussion is Element 4 is suggested. The third sentence should read "For example, the applicant may wish to determine the maturity of any cracking on the structural integrity of the component through flaw tolerance evaluation and enhance the visual examination requirements accordingly (visual acuity and lighting)."
IV A2 - 41	Reactor Vessel (PWR)	A2.5.1, A2.5.2	IV A2-25		This is one of the 17 open license renewal technical issues identified and described in EPRI TR-107521 (NEI/NRC 98-0036; Neutron Irradiation Embrittlement – Definition of the Reactor Vessel Beltline). The redefinition of the "beltline" region in the "Aging Management Program" column is not needed. The beltline region is already defined by the regulations. The next-to-last sentence should read "In addition, inservice inspection of ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Examination Category B-A of all pressure retaining welds in the vessel and repair welds in the beltline, as defined in 10 CFR 50, Appendix H."
IV A2 - 42	Reactor Vessel (PWR)	A2.5.1, A2.5.2,	IV A2-24		Materials listed for this item and for the following item (A2.5.1 thru A2.5.3) should be the same. For Neutron Irradiation Embrittlement, Aging Effect = ~~Loss of~~ **Reduced** Fracture Toughness

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV A2 - 43	Reactor Vessel (PWR)	A2.5.1 thru A2.5.3,	IV A2-24		For Aging Mechanism = Fatigue, under Existing AMP, "Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of **the original licensing criteria or** ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or Section I (Power Boilers) and Section VIII, Division 1 (Unfired Pressure Vessel). Under "Technical Basis" recognize that GSI-190 has been closed with no additional regulatory requirements. Change the wording to say "**environmental effects on fatigue are to be addressed.**"
IV A2 - 44	Reactor Vessel (PWR)	A2.5.3	IV A2-24		Add **Aging Effect = Loss of Material, Aging Mechanism=WEAR** *To reflect mechanical wear on threaded stud holes and vessel flange surface.* Existing Aging Management Program is ASME Section XI Examination Category B-G-1 (Visual VT-1 inspection of exposed surfaces of closure assemblies) and B-P (Visual VT-2 inspection of pressure retaining boundary during pressure boundary test) Acceptance criteria per ASME Section XI visual examination acceptance criteria and ASME Section III sizing and stress intensity limits. Corrective actions include replacing studs, repairing surface damage by blending to maintain minimum thickness and stress intensity limits.
IV A2 - 45	Reactor Vessel (PWR)	A2.5.1 through A2.5.3	IV A2-24		Aging Effect—Cracking initiation and growth is missing. Suggest cracking at welded joints be listed (see response to item A2.1-1 above). If not an applicable aging effect then Examination Category B-A volumetric inspections of the beltline welds are not required for license renewal. In addition, growth of intergranular separations by fatigue should be listed as an aging mechanism. A few of the B&W-designed vessels mat contain the subject fabrication flaws in the beltline region. See BAW-2251A, Appendix C, for a suggested resolution to this issue.

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV A2 - 46	Reactor Vessel (PWR)	A2.6	IV A2-24		Material is Alloy 600. Aging Mechanism-PWSCC should be listed for Alloy 600 items.
IV A2 - 47	Reactor Vessel (PWR)	A2.6	IV A2-25		For Aging Mechanism = SCC, IGSCC, the Existing AMP entry refers to the item for A2.2.1 CRDM Pressure Housing. The following changes apply: First sentence, ".....examination category ~~es B-O for pressure retaining welds in control rod housing, B-E for pressure retaining partial penetration welds, B-F for pressure retaining dissimilar metal welds~~, B-N-2 for integrally-welded core support structure~~, and testing category B-P for system leakage~~." Under Technical Basis, (3) Parameters Monitored / Inspected, ".....Requirements of Table IWB 2500-1, ~~examination category B-O specifies volumetric or surface examination extending ½ in. each side of the CRD housing welds, including weld buttering. B-E specifies visual VT-2 examination of partial penetration welds during the hydrostatic test. B-F specifies for all nozzle-to-safe end butt welds NPS 4 or larger, volumetric and surface examination of ID region and surface examination of OD surface; only surface examination is conducted for all butt welds less than NPS 4 and for all nozzle-to-safe end socket welds.~~ Category B-N-2 specifies visual VT-3 of accessible welds of core support-pads. Under "Further Evaluation" clarify what is expected for "Yes, Elements 2 to 6 should be further evaluated."
IV A2 - 48	Reactor Vessel (PWR)	A2.6	IV A2-24		For Aging Effect = Attrition and Aging Mechanism = Wear, this mechanism should be deleted. There is insufficient relative motion between the pad and adjacent parts to generate significant degradation.

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV A2 - 49	Reactor Vessel (PWR)	A2.7.1	IV A2-27		For Aging Effect = Crack Initiation and Growth (SCC and IGSCC), this effect should be deleted. Conditions do not exist to support this mechanism for the RV materials. Refer to item (1) under Technical Basis, "However, SCC is not anticipated to be an issue for the closure head dome or nozzles because analytical evaluations indicate that cracks in the SS cladding will stop growing in the ferritic base metal."
IV A2 - 50	Reactor Vessel (PWR)	A2.8.1	IV A2-26		Add the terms (**head adapter**) to the "region of interest" column
IV A2 - 51	Reactor Vessel (PWR)	A2.8.1	IV A2-26		For Aging Mechanism = SCC, IGSCC, the Existing AMP entry refers to the item for A2.2.1 CRDM Pressure Housing. The following changes apply: First sentence, ".....examination categories ~~B-O for pressure retaining welds in control rod housing,~~ B-E for pressure retaining partial penetration welds, B-F for pressure retaining dissimilar metal welds, ~~B-N-2 for integrally-welded core support structure~~, and testing category B-P for system leakage." Under Technical Basis, (3) Parameters Monitored / Inspected, ".....Requirements of Table IWB 2500-1, examination category ~~B-O specifies volumetric or surface examination extending ½ in. each side of the CRD housing welds, including weld buttering.~~ B-E specifies visual VT-2 examination of partial penetration welds during the hydrostatic test. B-F specifies for all nozzle-to-safe end butt welds NPS 4 or larger, volumetric and surface examination of ID region and surface examination of OD surface; only surface examination is conducted for all butt welds less than NPS 4 and for all nozzle-to-safe end socket welds. ~~Category B-N-2 specifies visual VT-3 of accessible welds of core support pads.~~ Under "Further Evaluation" clarify what is expected for "Yes, Elements 2 to 6 should be further evaluated."

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV A2 - 52	Reactor Vessel (PWR)	A2.8.1	IV A2-26		For Penetration – CRD Mechanism, need to add Aging Effect = **Loss of Material**, Aging Mechanism = **Corrosion / Boric Acid Wastage of External Surfaces**. Existing Program is **Implementation of NRC Generic Letter 88-05, and based on plant technical specifications, inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Examination Category B-E (visual VT-2 inspection of partial penetration welds during pressure test), testing category B-P for pressure retaining Class 1 components.** ***The intent of this addition is reflect the application of the 88-05 program in searching for sources of boric acid leakage.***
IV A2 - 53	Reactor Vessel (PWR)	A2.8.2	IV A2-26		Under "References" add **NRC IN 90-10**
IV A2 - 54	Reactor Vessel (PWR)	A2.8.2	IV A2-26		For Aging Mechanism = SCC, IGSCC, the Existing AMP entry refers to the item for A2.2.1 CRDM Pressure Housing. The following changes apply: First sentence, ".....examination categories ~~B-O for pressure retaining welds in control rod housing,~~ B-E for pressure retaining partial penetration welds, B-F for pressure retaining dissimilar metal welds, ~~B-N-2 for integrally-welded core support structure~~, and testing category B-P for system leakage," Under Technical Basis, (3) Parameters Monitored / Inspected, ".....Requirements of Table IWB 2500-1, examination category ~~B-O specifies volumetric or surface examination extending ½ in. each side of the CRD housing welds, including weld buttering.~~ B-E specifies visual VT-2 examination of partial penetration welds during the hydrostatic test. B-F specifies for all nozzle-to-safe end butt welds NPS 4 or larger, volumetric and surface examination of ID region and surface examination of OD surface; only surface examination is conducted for all butt welds less than NPS 4 and for

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
					all nozzle-to-safe end socket welds. ~~Category B-N-2 specifies visual VT-3 of accessible welds of core support pads.~~ Under "Further Evaluation" clarify what is expected for "Yes. Elements 2 to 6 should be further evaluated."
IV A2 - 55	Reactor Vessel (PWR)	A2.8.1	IV A2-26		What is the difference between a penetration and a nozzle. If this is the nozzle that is attached to vessel then what is item A2.2.1. Aging Effect—Suggest cracking at or near the heat-affected zone Aging Mechanism—PWSCC was omitted
IV A2 - 56	Reactor Vessel (PWR)	A2.8.2	IV A2-26		These penetrations are at the bottom of the vessel and are ¾-inch Schedule 160 tubes for the B&W-designed plants Aging Mechanism—PWSCC was omitted AMP—Generic Letter 97-01 does not apply to these penetrations since they are attached to the bottom head of the vessel.
IV A2 - 57	Reactor Vessel (PWR)	A2.8.3	IV A2-26		Under "Region of Interest" add the clarifying term **head vent pipe** to distinguish from vessel flange leakage monitoring tubes.
IV A2 - 58	Reactor Vessel (PWR)	A2.8.3	IV A2-26		For Aging Mechanism = SCC, IGSCC, the Existing AMP entry refers to the item for A2.2.1 CRDM Pressure Housing. The following changes apply: First sentence, ".....examination categories ~~B-O for pressure retaining welds in control rod housing,~~ B-E for pressure retaining partial penetration welds, B-F for pressure retaining dissimilar metal welds, ~~B-N-2 for integrally welded core support structure~~, and testing category B-P for system leakage." Under Technical Basis, (3) Parameters Monitored / Inspected, ".....Requirements of Table IWB 2500-1, examination category ~~B-O specifies volumetric or surface examination extending ½ in. each side of the CRD housing welds, including weld buttering.~~ B-E specifies visual VT-2 examination of partial penetration welds

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
					during the hydrostatic test, B-F specifies for all nozzle-to-safe end butt welds NPS 4 or larger, volumetric and surface examination of ID region and surface examination of OD surface; only surface examination is conducted for all butt welds less than NPS 4 and for all nozzle-to-safe end socket welds. ~~Category B-N-2 specifies visual VT-3 of accessible welds of core support pads.~~ Under "Further Evaluation" clarify what is expected for "Yes, Elements 2 to 6 should be further evaluated."
IV A2 - 59	Reactor Vessel (PWR)	A2.8.3	IV A2-26		For Penetration, Leakage Monitoring Tubes, need to add Aging Effect = **Loss of Material**, Aging Mechanism = **Corrosion / Boric Acid Wastage of External Surfaces**. Existing Program is **Implementation of NRC Generic Letter 88-05, and based on plant technical specifications, inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Examination Category B-E (visual VT-2 inspection of partial penetration welds during pressure test), testing category B-P for pressure retaining Class 1 components.** ***The intent of this addition is reflect the application of the 88-05 program in searching for sources of boric acid leakage.***
IV A2 - 60	Reactor Vessel (PWR)	A2.8.1 A2.8.2,	IV A2-26		Under "Region of Interest" add the clarifying term CRD Mechanism **head adapter** .
IV A2 - 61	Reactor Vessel (PWR)	A2.8.1 A2.8.2,	IV A2-27		For Aging Mechanism = Fatigue, under Existing AMP, "Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of **the original licensing criteria or** ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or Section I (Power Boilers) and Section VIII, Division 1 (Unfired Pressure Vessel). Under "Technical Basis" recognize that GSI-190 has been closed with no additional regulatory requirements. Change the wording to say "**environmental effects on fatigue are to be addressed.**"

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IV

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV A2 - 62	Reactor Vessel (PWR)	9.1	IV A2-28		Fatigue is listed as an Aging Mechanism for the Vessel Support. Low cycle fatigue is not plausible for these components because components are maintained at a relatively constant temperature and do not experience significant thermal changes. This position was accepted in NUREG-1705. This entry should be removed.
IV A2 - 63	Reactor Vessel (PWR)	9.2 and 9.3	IV A2-28		Items list Neutron Irradiation Embrittlement as an Aging Mechanism for Pressure Vessel Support components (Cantilever/Column Support, Neutron Shield Tank) but then indicates that based on NUREG-0933 Neutron Irradiation Embrittlement is not a concern. This entry should be deleted since the Aging Mechanism is not of concern.
IV A2 - 64	Reactor Vessel (PWR)	A2.9.2 A2.9.3	IV A2-28		For Neutron Irradiation Embrittlement, Aging Effect = ~~Loss of~~ **Reduced** Fracture Toughness Since Neutron Irradiation Embrittlement is not a concern for the extended period of license renewal, this aging effect should be deleted completely.
IV A2 - 65	Reactor Vessel (PWR)	A2.9.1	IV A2-28		Aging Effect—Loss of material by boric acid wastage was not listed AMP-Why is ASME Section XI, Subsection IWF, not listed as an aging management program for the skirt.
IV A2-66	Reactor Vessel (PWR)	Various	IV A2-22 and 24		Replace "thru" with "through" or use dashes, (e.g., A1.1.1 – A1.1.3)
IV A2-67	Reactor Vessel (PWR)	A2.1.1 A2.1.3 A2.1.5 A2.2.1 A2.3.1-A2.3.3 A2.4.1-A2.4.3 A2.6, A2.7.1, A2.8.1-A2.8.3	IV A2-6, 8, 12, 22, 24, and 26.		Replace "SCC, IGSCC" or "Stress Corrosion Cracking (SCC), Intergranular Stress Corrosion Cracking (IGSCC)" with "SCC".

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IV

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV A2-68	Reactor Vessel (PWR)	A2.1.1 A2.2.1 A2.3.1-A2.3.3 A2.4.1-A2.4.3 A2.6 A2.7.1 A2.8.1-A2.8.3	IV A2-6, 7, 8, 9, 12, 13, 14, 15, 16, 17, 22, 24, and 26.		Replace "EPRI TR-105714" with "the latest version of EPRI TR-105714" or "EPRI PWR Primary Water Chemistry Guidelines, Revision 4, March 1999, EPRI TR-105714".

B2. Reactor Vessel Internals (PWR) - Westinghouse

B2.1 Upper Internals Assembly

- B2.1.1 Upper Support Plate
- B2.1.2 Upper Support Column
- B2.1.3 Upper Support Column Bolts
- B2.1.4 Upper Core Plate
- B2.1.5 Upper Core Plate Alignment Pins
- B2.1.6 Fuel Pins
- B2.1.7 Hold-Down Spring

B2.2 RCCA Guide Tube Assemblies

- B2.2.1 RCCA Guide Tubes
- B2.2.2 RCCA Guide Tube Bolts
- B2.2.3 RCCA Guide Tube Support Pins

B2.3 Core Barrel

- B2.3.1 Core Barrel
- B2.3.2 Upper Core Barrel Flange
- B2.3.3 Core Barrel Nozzles
- B2.3.4 Thermal Shield

B2.4 Baffle/Former Assembly

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

B2.5.3 Lower Support Plate

B2.5.4 Lower Support Plate Column

B2.5.5 Lower Support Plate Column Bolts

B2.5.6 Radial Keys and Clevis Inserts

B2.5.7 Clevis Insert Bolts

B2.6 Instrumentation Support Structure

B2.6.1 Flux Thimble Guide Tubes

B2.6.2 Flux Thimbles

B2. Reactor Vessel Internals (PWR) - Westinghouse

System, Structures, and Components

The system, structures, and components included in this table comprise the Westinghouse pressurized water reactor (PWR) reactor vessel internals and consist of the Upper Internals Assembly, Rod Control Cluster Assemblies (RCCA) Guide Tube Assemblies, Core Barrel, Baffle/Former Assembly, and Lower Internal Assembly. All structures and components in the reactor vessel are classified as Group A or B Quality Standards.

System Interfaces

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

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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B2.1.1, B2.1.4, B2.1.7	Upper Internals Assembly	Upper Support Plate, Upper Core Plate, Hold-Down Spring	Stainless Steel (SS)	hemically reated orated ater up to 40°C	Crack Initiation and Growth	Stress Corrosion Cracking (SCC) IVB2-8	ASME Section XI, 1989 Edition. EPRI TR-105714. *Operating Experience* NRC IN 84-18. NRC IN 98-11. IVB2-7 IVB2-36
	IVB2-1 IVB2-2 IVB2-3 IVB2-4 IVB2-5 IVB2-6						
B2.1.1, B2.1.4, B2.1.7	Upper Internals Assembly	Upper Support Plate, Upper Core Plate,	Stainless Steel (SS)	hemically reated orated ater up to	Crack Initiation and Growth	Irradiation Assisted Stress Corrosion	ASME Section XI, 1989 Edition. EPRI TR-105714.

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI (edition specified in 10 CFR 50.55a). Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 to minimize the potential of crack initiation and growth. IVB2-7	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC of SS components, inservice inspection (ISI) to monitor the effects of SCC on the intended function of the components, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** PWR operating chemistry limits the halogens, sulfates, and oxygen in the primary water to less than 0.05, 0.05, and 0.005 ppm, respectively, during operation. However, introduction of oxygen can occur during shutdown and potential exists for the formation of more aggressive chemistry conditions by radiolysis in creviced regions or in low-flow stagnant regions. The AMP must rely upon inservice inspection (ISI) in accordance with ASME Section XI to detect possible degradation. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of SCC on the intended function by detection and sizing of cracks by ISI. Table IWB-2500, category B-N-3 specifies visual VT-3 examination of all accessible surfaces of reactor internals. ***(4) Detection of Aging Effects:*** Degradation due to SCC can not occur without crack initiation and growth. VT-3 may not be adequate to detect tight cracks. Creviced and other inaccessible regions are difficult to inspect visually and supplementary UT or other nondestructive examinations may be needed. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with IWB-3520. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWB-3140. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Although stainless steel components in PWRs have generally not been found to be affected by SCC because of low dissolved oxygen levels and control of primary water chemistry, potential for SCC exists from inadvertent introduction of contaminants into the primary coolant system (IN 84-18), from the introduction of relatively high levels of dissolved oxygen during shutdown, or from aggressive chemistries that may develop in creviced regions. Cracking has occurred in SS baffle former bolts in a number of foreign plants (IN 98-11) and has now been observed in US plants. The mechanism of this particular cracking has not yet been resolved. IVB2-9	Yes Element 4 should be further evaluated IVB2-7
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC of SS components, inservice inspection (ISI) to monitor the effects of SCC on the	Yes Element 4 should be

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

B2. REACTOR VESSEL INTERNALS (PWR) - Westinghouse

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) greatly reduces susceptibility to IASCC. However, introduction of oxygen can occur during shutdown and potential exists for the formation of more aggressive chemistry conditions by radiolysis in creviced regions or in low–flow stagnant regions. Also for sufficiently high fluence levels, IASCC can occur even in low oxygen environments. The AMP must rely upon ISI in accordance with ASME Section XI to detect possible degradation. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of SCC on the intended function by detection and sizing of cracks by inservice inspection (ISI). Table IWB-2500, category B-N-3 specifies visual VT-3 examination of all accessible surfaces of reactor internals. ***(4) Detection of Aging Effects:*** Degradation due to IASCC can not occur without crack initiation and growth. VT–3 may not be adequate to detect tight cracks. The inspection technique, including the reliability in detecting the features of interest (crack appearance and size) in assuring the integrity of the component, should be specified. For example, enhancement of the visual VT-1 examination to achieve a 1/2-mil (0.0005 in.) resolution, with the conditions (lighting and surface cleanness) for the ISI bounded by those used to demonstrate the resolution of the inspection technique. Creviced regions are difficult to inspect visually and supplementary UT or other nondestructive examinations may be needed. As an alternate to enhanced inspection, perform a component-specific evaluation including a mechanical loading assessment to determine the maximum tensile loading on the component during ASME Code Level A, B, C, and D conditions. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with IWB-3520. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWB-3140. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Although the only cracking presently directly attributable to IASCC in PWRs occurred in SS fuel cladding used in early reactors, susceptibility to this degradation will increase as plant operating time and hence accumulated fluence levels increase. The role that IASCC has played in the more recent cracking that has occurred in SS baffle/former bolts in U.S. plants as well as foreign plants (IN 98-11) in not yet universally	

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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.			Plant specific aging management program is to be evaluated. The applicant must provide the basis for concluding that void swelling is not an issue for the component or must provide an AMP.				Yes TLAA
B2.1.1, B2.1.4, B2.1.7	Upper Internals Assembly	Upper Support Plate, Upper Core Plate, Hold-Down Spring	SS	hemically reated orated ater up to 40°C	Cumula-tive fatigue damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B2.1.7	Upper Internals Assembly	Hold-Down Spring	SS	hemically reated orated ater up to 40°C	Loss of Preload	Stress Relaxation	ASME Section XI, 1989 Edition.

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

Existing Aging Management Program (AMP)			Evaluation and Technical Basis					Further Evaluation
B2.1.2	Upper Internals Assembly	Upper Support Column	SS, CASS	Chemically treated borated water up to 40°C	Crack Initiation and Growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.	

IV B2-10

IV B2-13

Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	***(1) Scope of Program:*** The program includes inservice inspection (ISI) to monitor the condition of components that depend on preload, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** No practical preventative actions are possible. ***(3) Parameters Monitored/Inspected:*** The AMP utilizes ISI to monitor the effects of stress relaxation on the intended function of the component by detection and sizing of cracks that could be formed by excessive vibration that may occur if the preload is lost. Table IWB-2500, category B-N-3 specifies visual VT-3 examination of all accessible surfaces of reactor internals. Because VT-3 inspection can only detect degradation that occurs after the loss of preload, it may be adequate if there is sufficient redundancy that loss of some bolting between inspections is acceptable. In some cases additional inspection may be required. ***(4) Detection of Aging Effects:*** As part of the AMP it may be possible to identify acceptable levels of preload and demonstrate whether under the fluence of interest whether loss of acceptable preload is likely. VT-3 may not be adequate to detect tight cracks. Also, creviced regions are difficult to inspect visually. Supplementary inspections by techniques such as ultrasonic testing (UT) or other nondestructive methods may be needed to detect cracking in inaccessible regions. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with IWB-3520. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWB-3140. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** There are no reports of	Yes Elements 3 and 4 should be further evaluated

IV B2-12

IV B2-11

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

Item	Structure and Component	Region of Interest	Material	Environment	Aging Effect	Aging Mechanism	References
Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Items B2.1.1, upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.			*Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Items B2.1.1, upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.* IV B2-14				Yes Element 4 should be further evaluated
B2.1.2	Upper Internals Assembly	Upper Support Column	SS, CASS	hemically reated orated ater up to 40°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.

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

Existing Aging Management Program (AMP)			Evaluation and Technical Basis					Further Evaluation
B2.1.2	Upper Internals Assembly	Upper Support Column	CASS	hemically reated orated ater up to 40°C	Loss of fracture toughness	Thermal Aging and Neutron Irradiation Embrittle-ment	EPRI TR-106092. ASME Section XI, 1989 Edition.	

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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
Determination of the susceptibility of CASS components to thermal aging embrittlement based on casting method, Mo content, and percent ferrite. For "potentially susceptible" components, based on the neutron fluence of the component, implement either a supplemental examination of the affected components as part of the applicant's 10-year inservice inspection (ISI) program during the license renewal term or a component-specific evaluation to determine the susceptibility to loss of fracture toughness.			***(1) Scope of Program:*** The program includes determination of the susceptibility of CASS components to thermal aging based on casting method, Mo content, and percent ferrite, and for "potentially susceptible" components, to account for the synergistic loss of fracture toughness due to neutron embrittlement and thermal aging embrittlement, implement either a supplemental examination of the affected components as part of a 10-year inservice inspection (ISI) program during the license renewal term or a component-specific evaluation to determine the susceptibility to loss of fracture toughness. ***(2) Preventive Actions:*** The program provides no guidance on methods to mitigate thermal aging or neutron embrittlement. ***(3) Parameters Monitored/ Inspected:*** The program specifics depend on the neutron fluence and ferrite content of the component. The criteria in EPRI TR-106092, with some modifications approved by the NRC staff, can be used to determine whether CASS components are potentially susceptibility to thermal aging embrittlement based on the casting method, Mo content, and ferrite content. For low-Mo content (0.5 wt.% max.) steels, only static-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement, all centrifugal-cast and static-cast steels with ≤20% ferrite are not susceptible. For high-Mo content (2.0 to 3.0 wt.%) steels, static-cast steels with >14% ferrite and centrifugal-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement, static-cast steels with ≤14% ferrite and centrifugal-cast steels with ≤20% ferrite are not susceptible. Ferrite content will be calculated by the Hull's equivalent factors or a method producing an equivalent level of accuracy (±6% deviation between measured and calculated values. ***(4) Detection of Aging Effects:*** For all CASS components that have a neutron fluence of greater than 10^{17} n/cm^2 (E>1 MeV), implement a 10-year ISI program during the renewal period that includes supplemental inspection that covers portions of the susceptible components determined to be limiting from the standpoint of thermal aging susceptibility (Ferrite and Mo contents, casting process, and operating temperature), neutron fluence, and cracking susceptibility (applied stress, operating temperature, and environmental conditions). The inspection technique, including the reliability in detecting the features of interest (crack appearance and size) in assuring the integrity of the component, should be specified. For example, enhancement of the visual VT-1 examination to achieve a 1/2-mil (0.0005 in.) resolution, with the conditions (lighting and surface cleanness) for the ISI bounded by those used to demonstrate the resolution of the inspection technique. As an alternate to enhanced inspection,				No

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

Existing Aging Management Program (AMP)			Evaluation and Technical Basis				Further Evaluation
B2.1.2	Upper Internals Assembly	Upper Support Column	SS, CASS	hemically reated orated ater up to 40°C	Cumula-tive fatigue damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B2.1.3, B2.1.5, B2.1.6	Upper Internals Assembly	Upper Support Column Bolts, Upper Core Plate Alignment Pins, Fuel Pins	SS, Ni Alloy	hemically reated orated ater at emperature up to 40°C	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714. *Operating Experience* NRC IN 82-29. NRC IN 84-18. NRC IN 98-11.

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
			(continued from previous page) perform a component-specific evaluation including a mechanical loading assessment to determine the maximum tensile loading on the component during ASME Code Level A, B, C, and D conditions. If the loading is compressive or low enough (<5 ksi) to preclude fracture, then supplemental inspection of the component is not required. Failure to meet this criteria requires continued use of supplemental inspection program. For all CASS components that have a neutron fluence of less than 10^{17} n/cm^2 (E>1 MeV), implement the supplement inspection program if they are "potentially susceptible" to thermal aging; the existing ASME Section XI inspection requirements are considered adequate if the components are "not susceptible" to thermal aging. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 should provide timely detection of cracks. ***(6) Acceptance Criteria:*** Flaws detected in CASS components are evaluated in accordance with the applicable procedures of IWB-3600. ***(7) Corrective Actions:*** Repair is in conformance with IWA-4000 and IWB-4000, and replacement according to IWA-7000 and IWB-7000. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** The AMP based on susceptibility determination, neutron fluence level, and supplemental inspection is effective in managing the effects of synergistic loss of fracture toughness due to neutron and thermal aging embrittlement on the intended function of CASS components.				
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.			Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.				Yes TLAA
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI (edition specified in 10 CFR 50.55a). Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 to minimize the potential of crack initiation and growth.			***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC of components constructed from SS and Ni alloys, inservice inspection (ISI) to monitor the effects of SCC on the intended function of the components, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** PWR operating chemistry limits the halogens, sulfates, and oxygen in the primary water to less than 0.05, 0.05, and 0.005 ppm, respectively, during operation. However, preventive actions are considered inadequate because of possible effects of introduction of oxygen during shutdown and the potential formation of				Yes Element 4 should be further evaluated

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

Existing Aging Management Program (AMP)			Evaluation and Technical Basis				Further Evaluation
B2.1.3, B2.1.5, B2.1.6	Upper Internals Assembly	Upper Support Column Bolts, Upper Core Plate Alignment Pins, Fuel Pins	SS, Ni Alloy	hemically reated orated ater up to 40°C	Crack Initiation and Growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B2.1.3, B2.1.5, B2.1.6	Upper Internals Assembly	Upper Support Column Bolts, Upper Core Plate Alignment Pins, Fuel Pins	SS, Ni Alloy	hemically reated orated ater up to 40°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
			(continued from previous page) internals. ***(4) Detection of Aging Effects:*** Degradation due to SCC can not occur without crack initiation and growth. Historically the VT-3 visual examinations have not identified bolt cracking because cracking occurs at the juncture of the bolt head and shank, which is not accessible for visual inspection. Creviced and other inaccessible regions are difficult to inspect visually. Supplementary UT examinations may be needed. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with IWB-3520. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWB-3140. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Although stainless steel components in PWRs have generally not been found to be affected by SCC because of low dissolved oxygen levels and control of primary water chemistry, potential for SCC exists from inadvertent introduction of contaminants into the primary coolant system (IN 84–18), from the introduction of relatively high levels of dissolved oxygen during shutdown, or from aggressive chemistries that may develop in creviced regions. Cracking has occurred in SS baffle former bolts in a number of foreign plants (IN 98-11) and has now been observed in US plants. The mechanism of this particular cracking has not yet been resolved. SCC has also been observed in Ni alloy (Alloy X–750) control rod drive guide tube support pins (IN 82-29). Replacement components with a different heat treatment appear to be less susceptible to cracking.	
Same as for the effect of Stress Corrosion Cracking on Items B2.1.3 upper support column bolts, B2.1.5 upper core plate alignment pins, and B2.1.6 fuel pins.			*Same as for the effect of Stress Corrosion Cracking on Items B2.1.3 upper support column bolts, B2.1.5 upper core plate alignment pins, and B2.1.6 fuel pins.*				Yes Element 4 should be further evaluated
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the			Plant specific aging management program is to be evaluated. The applicant must provide the basis for concluding that void swelling is not an issue for the component or must provide an AMP.				Yes TLAA

IV B2-15

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

Existing Aging Management Program (AMP)			Evaluation and Technical Basis				Further Evaluation
B2.1.4	Upper Internals Assembly	Upper Core Plate	SS	hemically reated orated ater up to 40°C	Cumula-tive fatigue damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B2.1.5, B2.1.6	Upper Internals Assembly	Upper Core Plate Alignment Pins, Fuel Pins	SS, Ni Alloy	hemically reated orated ater up to 40°C	Attrition	Wear	ASME Section XI, 1989 Edition.
B2.1.6	Upper Internals Assembly	Fuel Pins	SS, Ni Alloy	hemically reated orated ater up to 40°C	Cumula-tive fatigue damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B2.2.1	RCCA Guide Tube Assemblies	RCCA Guide Tubes	SS	hemically reated orated ater up to 40°C	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B2.2.1	RCCA Guide Tube Assemblies	RCCA Guide Tubes	SS	hemically reated orated ater up to 40°C	Crack Initiation and Growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.

IV B2-17

IV B2-16

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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.				Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.			Yes TLAA
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.				***(1) Scope of Program:*** The program includes inservice inspection (ISI) to monitor the condition of components and assure that they can perform their intended function. ***(2) Preventive Actions:*** No practical preventative actions are possible. ***(3) Parameters Monitored/Inspected:*** The AMP program monitors the effects of wear on the intended function by inservice inspection (ISI) to determine the general mechanical and structural condition of components. ***(4) Detection of Aging Effects:*** Inspection can reveal indications of degradation due to wear such as the verification of clearances, settings, physical displacements, loose of missing parts, debris, corrosion, wear, erosion, or the loss of integrity at bolted or welded connections. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 is adequate for timely detection of wear. ***(6) Acceptance Criteria:*** Acceptance standard for the inspection is IWB-3520. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWB-3140. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** To date aging management program has been effective in managing the effects of wear on reactor internals.			No
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.				Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.			Yes TLAA
Same as for the effect of Stress Corrosion Cracking on Items B2.1.1, upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.				*Same as for the effect of Stress Corrosion Cracking on Items B2.1.1, upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.*			Yes Element 4 should be further evaluated
Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Items B2.1.1, upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.				*Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Items B2.1.1, upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.*			Yes Element 4 should be further evaluated
Plant specific aging management				Plant specific aging management program is to be			Yes

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

Existing Aging Management Program (AMP)			Evaluation and Technical Basis				Further Evaluation
B2.2.1	RCCA Guide Tube Assemblies	RCCA Guide Tubes IV B2-18	SS	hemically reated orated ater up to 40°C	Attrition	Wear	ASME Section XI, 1989 Edition.
B2.2.1	RCCA Guide Tube Assemblies	RCCA Guide Tubes	SS	hemically reated orated ater up to 40°C	Cumula-tive fatigue damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B2.2.2, B2.2.3	RCCA Guide Tube Assemblies	RCCA Guide Tube Bolts, RCCA Guide Tube Support Pins	SS, Ni Alloy	hemically reated orated ater up to 40°C	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B2.2.2, B2.2.3	RCCA Guide Tube Assemblies	RCCA Guide Tube Bolts, RCCA Guide Tube Support Pins	SS, Ni Alloy	hemically reated orated ater up to 40°C	Crack Initiation and Growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B2.2.2, B2.2.3	RCCA Guide Tube Assemblies	RCCA Guide Tube Bolts, RCCA Guide Tube Support Pins	SS, Ni Alloy	hemically reated orated ater up to 40°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.
B2.2.2	RCCA Guide Tube Assemblies	RCCA Guide Tube Bolts IV B2-19	SS, Ni Alloy	hemically reated orated ater up to 40°C	Loss of Preload	Stress Relaxation	ASME Section XI, 1989 Edition.
B2.2.2, B2.2.3	RCCA Guide Tube Assemblies	RCCA Guide Tube Bolts, RCCA Guide Tube Support Pins	SS, Ni Alloy	hemically reated orated ater up to 40°C	Cumula-tive fatigue damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B2.3.1 thru B2.3.4	Core Barrel	Core Barrel, Upper Core Barrel Flange, Core Barrel Nozzles, Thermal Shield	SS	hemically reated orated ater up to 40°C	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B2.3.1 thru B2.3.4	Core Barrel	Core Barrel, Upper Core Barrel Flange,	SS	hemically reated orated	Crack Initiation and Growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*Same as the effect of Wear on Items B2.1.5 and B2.1.6, upper core plate alignment pins, fuel pins.*						No
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.						Yes TLAA
Same as for the effect of Stress Corrosion Cracking on Items B2.1.3 upper support column bolts, B2.1.5 upper core plate alignment pins, and B2.1.6 fuel pins.	*Same as for the effect of Stress Corrosion Cracking on Items B2.1.3 upper support column bolts, B2.1.5 upper core plate alignment pins, and B2.1.6 fuel pins.*						Yes Element 4 should be further evaluated
Same as for the effect of Stress Corrosion Cracking on Items B2.1.3 upper support column bolts, B2.1.5 upper core plate alignment pins, and B2.1.6 fuel pins.	*Same as for the effect of Stress Corrosion Cracking on Items B2.1.3 upper support column bolts, B2.1.5 upper core plate alignment pins, and B2.1.6 fuel pins.*						Yes Element 4 should be further evaluated
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.	Plant specific aging management program is to be evaluated. The applicant must provide the basis for concluding that void swelling is not an issue for the component or must provide an AMP.						Yes TLAA
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*Same as the effect of Stress Relaxation on Item B2.1.7, hold-down spring*						Yes Elements 3 and 4 should be further evaluated
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.						Yes TLAA
Same as for the effect of Stress Corrosion Cracking on Items B2.1.1, upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.	*Same as for the effect of Stress Corrosion Cracking on Items B2.1.1, upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.*						Yes Element 4 should be further evaluated
Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Items B2.1.1, upper support plate,	*Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Items B2.1.1, upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.*						Yes Element 4 should be

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

Existing Aging Management Program (AMP)			Evaluation and Technical Basis				Further Evaluation
B2.3.1 thru B2.3.4	Core Barrel	Core Barrel, Upper Core Barrel Flange, Core Barrel Nozzles, Thermal Shield	SS	hemically reated orated ater up to 40°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.
B2.3.1 thru B2.3.4	Core Barrel	Core Barrel, Upper Core Barrel Flange, Core Barrel Nozzles, Thermal Shield	SS	hemically reated orated ater up to 40°C	~~Loss of~~ Fracture Toughness	Neutron Irradiation Embrittle-ment	ASME Section XI, 1989 Edition.
B2.3.1 thru B2.3.4	Core Barrel	Core Barrel, Upper Core Barrel Flange, Core Barrel Nozzles, Thermal	SS	hemically reated orated ater up to 40°C	Cumula-tive fatigue damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.

IVB2-20

REDUCED

IVB2-21

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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.				Plant specific aging management program is to be evaluated. The applicant must provide the basis for concluding that void swelling is not an issue for the component or must provide an AMP.			Yes TLAA
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI. IVB2-21				***(1) Scope of Program:*** The program includes inservice inspection (ISI) to detect cracking and/or failure, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** No practical preventative actions are possible. Stainless steels are susceptible to embrittlement under neutron irradiation. Fracture toughness will depend strongly on the fluence on a particular component. Components can be screened out if the maximum tensile loading on the component under ASME Code Level A, B, C and D conditions is sufficiently low. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of neutron irradiation embrittlement on the intended function of the component by detection and sizing of cracks by inservice inspection (ISI). Table IWB-2500, category B-N-3 specifies visual VT-3 examination of all accessible surfaces of reactor internals. ***(4) Detection of Aging Effects:*** Loss of fracture toughness is of consequence only if cracks exist. Cracking is expected to initiate at the surface and should be detectable by ISI except for some crevice regions. VT-3 may not be adequate to detect tight cracks. Also, creviced regions are difficult to inspect visually and supplementary UT or other nondestructive examinations may be needed. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with IWB-3520. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWB-3140. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** No instances of internals degradation have been recorded that have been definitely attributed to irradiation embrittlement. IVB2-21			Yes Elements 3 and 4 should be further evaluated
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.				Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.			Yes TLAA

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

Existing Aging Management Program (AMP)			Evaluation and Technical Basis				Further Evaluation
B2.4.1	Baffle/Former Assembly	Baffle and Former Plates	SS	hemically reated orated ater up to 40°C	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B2.4.1	Baffle/Former Assembly	Baffle and Former Plates	SS	hemically reated orated ater up to 40°C	Crack Initiation and Growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B2.4.1	Baffle/Former Assembly	Baffle and Former Plates	SS	hemically reated orated ater up to 40°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.
B2.4.2	Baffle/Former Assembly	Baffle/Former Bolts	SS (Type 347 & cold-worked Type 316)	hemically reated orated ater up to 40°C and igh fluence >10 dpa or x 10^{21} /cm^2 >1 MeV)	Crack Initiation and Growth	SCC	Plant Technical Specification
B2.4.2	Baffle/Former Assembly	Baffle/Former Bolts	SS (Type 347 & cold-worked Type 316)	hemically reated orated ater up to 40°C and igh fluence >10 dpa or x 10^{21} /cm^2 >1 MeV)	Crack Initiation and Growth	IASCC	Plant Technical Specification
B2.4.2	Baffle/Former Assembly	Baffle/Former Bolts	SS (Type 347 & cold-worked Type 316)	hemically reated orated ater up to 40°C and	Changes in Dimensions	Void Swelling	EPRI TR-107521.

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

Item	Structure and Component	Region of Interest	Material	Environment	Aging Effect	Aging Mechanism	References
Same as for the effect of Stress Corrosion Cracking on Items B2.1.1, upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.			*Same as for the effect of Stress Corrosion Cracking on Items B2.1.1, upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.* IV B2-23				Yes Element 4 should be further evaluated
Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Items B2.1.1, upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.			*Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Items B2.1.1, upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.*				Yes Element 4 should be further evaluated
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.			Plant specific aging management program is to be evaluated. The applicant must provide the basis for concluding that void swelling is not an issue for the component or must provide an AMP.				Yes TLAA
Plant specific aging management program. Historically the VT-3 visual examinations have not identified baffle/former bolt cracking because cracking occurs at the junction of the bolt head and shank, which is not accessible for visual inspection. However, recent UT examinations of the baffle/former bolt have identified cracking in several plants. The industry is currently addressing the issue of baffle bolt cracking in the PWR Materials Reliability Project, Issues Task Group (ITG) activities to determine, develop, and implement the necessary steps and plans to manage the applicable aging effects on a plant specific basis. IV B2-24			Plant-specific aging management program is to be evaluated.				Yes No AMP
Same as for the effect of Stress Corrosion Cracking on Item B2.4.2 baffle/former bolts.			*Same as for the effect of Stress Corrosion Cracking on Item B2.4.2 baffle/former bolts.*				Yes No AMP
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the			Plant specific aging management program is to be evaluated. The applicant must provide the basis for concluding that void swelling is not an issue for the component or must provide an AMP.				Yes TLAA

IVB2-25

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

Existing Aging Management Program (AMP)			Evaluation and Technical Basis				Further Evaluation
B2.4.1, B2.4.2	Baffle/Former Assembly	Baffle and Former Plates, Baffle/Former Bolts	SS, Ni Alloy (bolts)	hemically reated orated ater up to 40°C	Loss of Fracture Toughness REDUCED	Neutron Irradiation Embrittle-ment	ASME Section XI, 1989 Edition.
B2.4.1, B2.4.2	Baffle/Former Assembly	Baffle and Former Plates, Baffle/Former Bolts	SS, Ni Alloy (bolts)	hemically reated orated ater up to 40°C	Cumula-tive fatigue damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B2.4.2	Baffle/Former Assembly	Baffle/Former Bolts	SS, Ni Alloy	hemically reated orated ater up to 40°C	Loss of Preload	Stress Relaxation	ASME Section XI, 1989 Edition.
B2.5.1, B2.5.6, B2.5.7	Lower Internal Assembly	Lower Core Plate, Radial Keys and Clevis Inserts, Clevis Insert Bolts	SS	hemically reated orated ater up to 40°C	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B2.5.1, B2.5.6, B2.5.7	Lower Internal Assembly	Lower Core Plate, Radial Keys and Clevis Inserts, Clevis Insert Bolts	SS	hemically reated orated ater up to 40°C	Crack Initiation and Growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B2.5.1, B2.5.6, B2.5.7	Lower Internal Assembly	Lower Core Plate, Radial Keys and Clevis Inserts, Clevis Insert Bolts	SS	hemically reated orated ater up to 40°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.
B2.5.1	Lower Internal Assembly	Lower Core Plate	SS	hemically reated orated ater up to 40°C	Loss of Fracture Toughness REDUCES	Neutron Irradiation Embrittle-ment	ASME Section XI, 1989 Edition.
B2.5.1, B2.5.4	Lower Internal Assembly	Lower Core Plate, Lower Support Plate Columns	SS	hemically reated orated ater up to 40°C	Cumula-tive fatigue damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.

IVB2-28

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

Item	Structure and Component	Region of Interest	Material	Environment	Aging Effect	Aging Mechanism	References
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI. IVB2-29			*Same as for the effect of Neutron Irradiation Embrittlement on Items B2.3.1 thru B2.3.3 core barrel, upper core barrel flange, core barrel nozzles.*				Yes Elements 3 and 4 should be further evaluated
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.			Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.				Yes TLAA
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI. IVB2-26			*Same as for the effect of Stress Relaxation on Item B2.1.7 hold-down spring.*				Yes Elements 3 and 4 should be further evaluated
Same as for the effect of Stress Corrosion Cracking on Items B2.1.1, upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.			*Same as for the effect of Stress Corrosion Cracking on Items B2.1.1, upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.*				Yes Element 4 should be further evaluated
Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Items B2.1.1, upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring. IVB2-27			*Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Items B2.1.1, upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.*				Yes Element 4 should be further evaluated
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.			Plant specific aging management program is to be evaluated. The applicant must provide the basis for concluding that void swelling is not an issue for the component or must provide an AMP.				Yes TLAA
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI. IVB2-28			*Same as for the effect of Neutron Irradiation Embrittlement on Items B2.3.1 thru B2.3.3 core barrel, upper core barrel flange, core barrel nozzles.*				Yes Elements 3 and 4 should be further evaluated
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.			Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.				Yes TLAA

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

Existing Aging Management Program (AMP)			Evaluation and Technical Basis				Further Evaluation
B2.5.2, B2.5.5	Lower Internal Assembly	Fuel Pins, Lower Support Plate Column Bolts	SS, Ni Alloy	hemically reated orated ater up to 40°C	Crack Initiation and Growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B2.5.2, B2.5.5	Lower Internal Assembly	Fuel Pins, Lower Support Plate Column Bolts	SS, Ni Alloy	hemically reated orated ater up to 40°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.
B2.5.2, B2.5.5	Lower Internal Assembly	Fuel Pins, Lower Support Plate Column Bolts	SS, Ni Alloy	hemically reated orated ater up to 40°C	~~Loss of~~ Fracture Toughness REDUCES	Neutron Irradiation Embrittle-ment	ASME Section XI, 1989 Edition.
B2.5.2, B2.5.5, B2.5.7	Lower Internal Assembly	~~Fuel Pins~~, Lower Support Plate Column Bolts, Clevis Insert Bolts	SS, Ni Alloy	hemically reated orated ater up to 40°C	~~Loss of~~ Preload	Stress Relaxation	ASME Section XI, 1989 Edition.
B2.5.2	Lower Internal Assembly	~~Fuel Pins~~	SS, Ni Alloy	hemically reated orated ater up to 40°C	Attrition	Wear	ASME Section XI, 1989 Edition.
B2.5.2, B2.5.5	Lower Internal Assembly	Fuel Pins, Lower Support Plate Column Bolts	SS, Ni Alloy	hemically reated orated ater up to 40°C	Cumula-tive fatigue damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B2.5.3, B2.5.4	Lower Internal Assembly	Lower Support Plate, Lower Support Plate Columns	SS, CASS	hemically reated orated ater up to 40°C	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B2.5.3, B2.5.4	Lower Internal Assembly	Lower Support Plate, Lower Support Plate Columns	SS, CASS	hemically reated orated ater up to 40°C	Crack Initiation and Growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B2.5.3, B2.5.4	Lower Internal Assembly	Lower Support Plate, Lower Support Plate Columns	SS, CASS	hemically reated orated ater up to 40°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.

IV B2-30

IV B2-31

IV B2-32

IV B2-33

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

Item	Structure and Component	Region of Interest	Material	Environment	Aging Effect	Aging Mechanism	References
Same as for the effect of Stress Corrosion Cracking on Items B2.1.3 upper support column bolts, B2.1.5 upper core plate alignment pins, and B2.1.6 fuel pins.			*Same as for the effect of Stress Corrosion Cracking on Items B2.1.3 upper support column bolts, B2.1.5 upper core plate alignment pins, and B2.1.6 fuel pins.*				Yes Element 4 should be further evaluated
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.			Plant specific aging management program is to be evaluated. The applicant must provide the basis for concluding that void swelling is not an issue for the component or must provide an AMP.				Yes TLAA
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.			*Same as for the effect of Neutron Irradiation Embrittlement on Items B2.3.1 thru B2.3.3 core barrel, upper core barrel flange, core barrel nozzles.*				Yes Elements 3 and 4 should be further evaluated
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.			*Same as for the effect of Stress Relaxation on Item B2.1.7 hold-down spring.*				Yes Elements 3 and 4 should be further evaluated
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.			*Same as for the effect of Wear on Items B2.1.5 upper core plate alignment pins and B2.1.6 fuel pins.*				No
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.			Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.				Yes TLAA
Same as for the effect of Stress Corrosion Cracking on Items B2.1.1, upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.			*Same as for the effect of Stress Corrosion Cracking on Items B2.1.1, upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.*				Yes Element 4 should be further evaluated
Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Items B2.1.1, upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.			*Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Items B2.1.1, upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.*				Yes Element 4 should be further evaluated
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the			Plant specific aging management program is to be evaluated. The applicant must provide the basis for concluding that void swelling is not an issue for the component or must provide an AMP.				Yes TLAA

IV B2-29

IV B2-30

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

Existing Aging Management Program (AMP)			Evaluation and Technical Basis				Further Evaluation
B2.5.3, B2.5.4	Lower Internal Assembly	Lower Support Plate, Lower Support Plate Columns	SS, CASS	hemically reated orated ater up to 40°C	~~Loss of~~ fracture toughness REDUCED	Thermal Aging and Neutron Irradiation Embrittle-ment	EPRI TR-106092. ASME Section XI 1989 Edition.
B2.5.6	Lower Internal Assembly	Radial Keys and Clevis Inserts	SS	hemically reated orated ater up to 40°C	Attrition	Wear	ASME Section XI, 1989 Edition.
B2.5.6, B2.5.7	Lower Internal Assembly	Radial Keys and Clevis Inserts, Clevis Insert Bolts	SS	hemically reated orated ater up to 40°C	Cumula-tive fatigue damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B2.6.1	Instru-mentation Support Structure	Flux Thimble Guide Tubes	SS	hemically reated orated ater up to 40°C	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B2.6.1	Instru-mentation Support Structure	Flux Thimble Guide Tubes	SS	hemically reated orated ater up to 40°C	Crack Initiation and Growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B2.6.1	Instru-mentation Support Structure	Flux Thimble Guide Tubes	SS	hemically reated orated ater up to 40°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.
B2.6.2	Instru-mentation Support Structure	Flux Thimbles	SS	hemically reated orated ater up to 40°C	Attrition	Wear	ASME Section XI, 1989 Edition.

IV B2-34

IV B2-35

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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
Same as for the effect of Thermal Aging and Neutron Irradiation Embrittlement on Item B2.1.2, upper support column constructed of CASS.			*Same as for the effect of Thermal Aging and Neutron Irradiation Embrittlement on Item B2.1.2, upper support column constructed of CASS.*				No
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.			*Same as for the effect of Wear on Items B2.1.5 upper core plate alignment pins and B2.1.6 fuel pins.*				No
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.			Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.				Yes TLAA
Same as for the effect of Stress Corrosion Cracking on Items B2.1.1, upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.			*Same as for the effect of Stress Corrosion Cracking on Items B2.1.1, upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.*				Yes Element 4 should be further evaluated
Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Items B2.1.1, upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.			*Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Items B2.1.1, upper support plate, B2.1.4 upper core plate, and B2.1.7 hold-down spring.*				Yes Element 4 should be further evaluated
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.			Plant specific aging management program is to be evaluated. The applicant must provide the basis for concluding that void swelling is not an issue for the component or must provide an AMP.				Yes TLAA
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.			*Same as for the effect of Wear on Items B2.1.5 upper core plate alignment pins and B2.1.6 fuel pins.*				No

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IV

GALL SECTION	TITLE	ITEM NO	PAGE	REVIEWER	COMMENT/CHANGES
IV B2-1	Reactor Vessel Internals (PWR-W)	General	Various		Void Swelling should not be included in the GALL report. Void Swelling is a phenomenon originally referred to as a "change in dimension" initially directed towards the baffle/former assembly components for PWRs. The reference (EPRI TR-107521) in the GALL report for Void Swelling explicitly states that Void Swelling is insignificant and that the only components susceptible are those subjected to high radiation fields. The only component specifically noted in this reference as remotely susceptible to Void Swelling is the baffle/former assemblies. Until sufficient evidence exists to warrant the inclusion of Void Swelling in the GALL report, the burden of proof should not reside with the individual utilities to conclude that Void Swelling is not an issue for each component. If Void Swelling is to remain in the GALL report, then the degradation mechanism should only be applicable to the baffle/former assemblies and listed as a concern under Irradiation Creep and/or Irradiation Assisted Stress Corrosion Cracking.
IV B2-2	Reactor Vessel Internals (PWR-W)	General	Various		The descriptions/names of the reactor vessel internal components should follow the "official" vendor names for the component. If the component names differ between vendors/reactor types, then all names for the component should be listed.
IV B2-3	Reactor Vessel Internals (PWR-W)	General	Various		The GALL report should incorporate the accepted conclusions from WCAP-14573 "Aging Management Evaluation for Reactor Internals as well as any other report concerning issues related to the RVI. This will prevent unnecessary questions based on the GALL report which have been addressed in previously reviewed and accepted reports.
IV B2-4	Reactor Vessel Internals (PWR-W)	General	Various		Remove all references to GSI-190 in the GALL Report. GSI-190 has been closed by the NRC. The issue of environmental factors impacting fatigue should not be addressed in the GALL Report. Recommend NEI Task Force address with the NRC.

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IV

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT/CHANGES
IV B2-5	Reactor Vessel Internals (PWR-W)	General	Various		Many subcomponents are assumed subject to IASCC, although most subcomponents do not accumulate sufficient fluence to initiate this mechanism. Item 10 of the technical basis supports this "Although the only cracking presently directly attributable to IASCC in PWRs occurred in SS fuel cladding used in early reactors, susceptibility to this degradation will increase…." In addition, the technical basis does not provide a quantified level for fluence, which is necessary to determine if the mechanism is applicable. WOG report WCAP-14577 justifies a fluence threshold of 10^{21}, which these components will not realize over 60 years of operation. This category should be deleted or at least provide threshold for applicability of IASCC
IV B2-6	Reactor Vessel Internals (PWR-W)	General	Various		Similarly, many subcomponents are assumed to be subject to Reduction in Fracture Toughness due to irradiation embrittlement. This effect is effectively limited to the beltline region of the reactor pressure vessel (alloy steel) and internals (SS).
IV.B2-7	Reactor Vessel Internals (PWR-W)	B2.1.1, B2.1.4, B2.1.7, various	IV B2-4, various		Reference to ASME Section XI should be deleted from the References column, the Existing AMP column, and the Evaluation and Technical Basis column. The AMA should only be the "*Chemistry Program (Primary and Secondary Water Systems)*". Refer to the proposed program description. This is one of the 17 technical issues identified and described in EPRI TR-107521 ("Stress Corrosion Cracking in PWR System Components"). The effects of stress corrosion cracking (SCC) of PWR austenitic stainless steel are precluded by material selection (e.g., Reg. Guide 1.43) and control of chemistry (oxygen and other debilitating constituents) in the reactor coolant. Hence no further evaluation is required. Note: License Renewal applicants who are unable to demonstrate the satisfaction of appropriate material chemistry requirements, can manage SCC effects through existing ASME Code Section XI in-service inspection requirements.

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IV

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT/CHANGES
IV B2-8	Reactor Vessel Internals (PWR-W)	B2.1.1, B2.1.4, and B2.1.7	IV B2-5		This is one of the 17 technical issues identified and described in EPRI TR-107521 (NEI/NRC 98-0038, Visual Examination of Components and Structures) It should be the responsibility of the applicant (or the industry, on a generic basis) to demonstrate, by flaw tolerance or other means, whether the structural integrity of internals components is threatened by a mature stage of crack growth. This same comment applies to all other items in B2 (W), B3 (CE), and B4 (B&W) that deal with SCC.
IV B2-9	Reactor Vessel Internals (PWR-W)	B2.1.1, B2.1.4, and B2.1.7	IV B2-7		This is one of the 17 technical issues identified and described in EPRI TR-107521 (NEI/NRC 98-0038, Visual Examination of Components and Structures). The fourth sentence of Element 4 in the "Evaluation and Technical Basis" column should read "For example, the applicant may wish to determine the maturity of any cracking on the structural integrity of the component through flaw tolerance evaluation and enhance the visual examination requirements accordingly (visual acuity and lighting)." The last sentence would be more helpful if specific criteria (e.g., compressive loads or tensile loads with stresses < 5 ksi) were added. This same comment applies to all other items in B2 (W), B3 (CE), and B4 (B&W) that deal with IASCC.
IV.B2-10	Reactor Vessel Internals (PWR-W)	B2.1.2	IV B2-8, various		Not all plants utilize CASS for these parts.
IV.B2-11	Reactor Vessel Internals (PWR-W)	B2.1.7	IV B2-9, various		Evaluation and technical basis section: Item (4) "Detection of Aging Effects" should be revised to provide some standard wording such as "Based on inspection results from utilities as documented by operating experience feedback, etc., there may be a need for more focused examinations, both in method and location."
IV B2-12	Reactor Vessel Internals (PWR-W)	B2.1.7	IV B2-9		This is one of the 17 technical issues identified and described in EPRI TR-107521 (NEI/NRC 98-0032, Stress Relaxation of PWR Internals).

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT/CHANGES
					Element 3 in the "Evaluation and Technical Basis" column. It should be the responsibility of the applicant (or the industry, on a generic basis) to demonstrate, by flaw tolerance or other means, whether the structural integrity of internals components is threatened by a mature stage of crack growth. This same comment applies to all other items in B2 (W), B3 (CE), and B4 (B&W) that deal with stress relaxation and loss of preload.
IV B2-13	Reactor Vessel Internals (PWR-W)	B2.1.2	IV B2-10		Change Aging Effect = Reduced fracture toughness For Material = CASS and Aging Mechanism = Thermal Aging and Neutron Irradiation Embrittlement. WOG plants do not have CASS in the upper support columns. Some plants do have mixing vane devices made of CASS, however these do not perform any intended function. Delete this Category
IV B2-14	Reactor Vessel Internals (PWR-W)	B2.1.2	IV B2-11		This is one of the 17 technical issues identified and described in EPRI TR-107521 (NEI/NRC 98-0030, Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components). The third sentence of Element 4 should be changed to read "For example, the applicant may wish to determine the maturity of any cracking on the structural integrity of the component through flaw tolerance evaluation and enhance the visual examination requirements accordingly (visual acuity and lighting. This same comment applies to all other items in B2 (W), B3 (CE), and B4 (B&W) that deal with thermal aging embrittlement and loss of fracture toughness.
IV B2-15	Reactor Vessel Internals (PWR-W)	B2.1.3, B2.1.5, B2.1.6	IV B2-12		Technical Basis Item 10 : References to baffle bolts and guide tube alignment pins are not appropriate for this item
IV B2-16	Reactor Vessel Internals (PWR-W)	B2.1.5 B2.1.6	IV B2-16		For Aging Mechanism = Wear, the item "Fuel Pins – B2.1.6" should be deleted. Wear is not a significant aging mechanism for fuel pins.
IV B2-17	Reactor Vessel Internals (PWR-W)	B2.1.4	IV B2-16		Upper Core Plate specifying fatigue as aging mechanism is a duplicate. It is also listed on page IV B2-8

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IV

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT/CHANGES
IV B2-18	Reactor Vessel Internals (PWR-W)	B2.2.1	IV B2-18		For Aging Mechanism = Wear, the only components subject to wear are the guide tube cards, and measurements have shown the effect to be not significant. Rod drop time testing which is done at the beginning of each fuel cycle will provide adequate indications of wear. This category should be deleted.
IV B2-19	Reactor Vessel Internals (PWR-W)	B2.2.2	IV B2-18		For Aging Effect = Loss of Preload and Aging Mechanism – Stress Relaxation, this effect should be deleted. The GT Bolts do not require significant preload in order to perform their intended function.
IV-B2-20	Reactor Vessel Internals (PWR) Westinghouse	B2.3.1 thru B.3.4	IV B2-18 IV B2-20		The Thermal Shield (item B2.3.4) is included in the aging management programs. The thermal shield does not perform or support any intended function, so does not require aging management. Delete thermal shield from all consideration in the reactor internals section.
IV B2-21	Reactor Vessel Internals (PWR-W)	B2.3.1 Through B2.3.4	IV B2-20		Change Aging Effect – Reduced fracture toughness This effect is only applicable to item B2.3.1, Core Barrel, in the active core region. Delete the other items for this effect. Change Existing AMP to be examination category B-N-2/B-N-3. Under "Technical Basis," Item 3, Change inspection category to B-N-2/B-N-3
IV B2-22	Reactor Vessel Internals (PWR-W)	B2.3.2	IV B2-20		For Aging Mechanism – Wear. This component does not experience significant relative motion to experience this mechanism. This aging effect should be deleted.
IV B2-23	Reactor Vessel Internals (PWR-W)	B2.4.1	IV B-22		For Aging Effect = Crack Initiation and Growth, Aging Mechanism = IASCC, change Existing AMP to be examination category B-N-2/B-N-3 Under "Technical Basis," Item 3, Change inspection category to B-N-2/B-N-3
IV B2-24	Reactor Vessel Internals (PWR-W)	B2.4.2	IV B2-22		For Aging Effect = Crack Initiation and Growth, Aging Mechanism = SCC, and also For Aging Effect = Crack Initiation and Growth,

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT/CHANGES
					Aging Mechanism = IASCC Existing AMP references the MRP ITG to develop this program. NO AMP IS ACKNOWLEDGED; PLANT-SPECIFIC APPROACH REQUIRED
IV B2-25	Reactor Vessel Internals (PWR-W)	B2.4.1 B2.4.2	IV B2-24		Change Aging Effect =Reduced fracture toughness Change Existing AMP to be examination category B-N-2/B-N-3 Under "Technical Basis," Item 3, Change inspection category to B-N-2/B-N-3 Bolts (B2.4.2) should not be included in this program. Any program resulting from MRP ITG will cover this mechanism (irradiation embrittlement) as well as SCC and IASCC.
IV B2-26	Reactor Vessel Internals (PWR-W)	B2.4.2	IV B2-24		For Aging Effect = Loss of Preload and Aging Mechanism = Stress Relaxation. An EPRI-sponsored MRP will identify any needed AMP for the baffle/former bolts considering all active degradation effects.
IV B2-27	Reactor Vessel Internals (PWR-W)	B2.5.1 B2.5.6 B2.5.7	IV B2-24		For Aging Effect = Crack Initiation and Growth, Aging Mechanism = IASCC, sufficient irradiation exposure will exist to make this a likely mechanism for only the Lower Core Plate. For the radial keys, clevis inserts, and clevis insert bolts, the effect of IASCC is not significant. Make this effect applicable only to the Lower Core Plate (B2.5.1); change Existing AMP to be examination category B-N-2/B-N-3 Under "Technical Basis," Item 3, Change inspection category to B-N-2/B-N-3
IV B2-28	Reactor Vessel Internals (PWR-W)	B2.5.1	IV B2-24		Change Aging Effect Reduced fracture toughness Change Existing AMP to be examination category B-N-2/B-N-3 Under "Technical Basis," Item 3, Change inspection category to B-N-2/B-N-3
IV B2-29	Reactor Vessel Internals (PWR-W)	B2.5.2 B2.5.5	IV B2-26		For Aging Effect = Crack Initiation and Growth, Aging Mechanism = IASCC, sufficient irradiation exposure will exist to make this a

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT/CHANGES
					likely mechanism for only the fuel pins. For the lower SP column bolts, the effect of IASCC is not significant. Make this effect applicable only to the Fuel Pins (B2.5.2); change Existing AMP to be examination category B-N-2/B-N-3 Under "Technical Basis," Item 3, Change inspection category to B-N-2/B-N-3
IV B2-30	Reactor Vessel Internals (PWR-W)	B2.5.2 B2.5.5	IV B2-26		Change Aging Effect – Reduced fracture toughness Sufficient irradiation exposure will exist to make this a likely mechanism for only the fuel pins. For the lower SP column bolts, the effect of IASCC is not significant. Make this effect applicable only to the Fuel Pins (B2.5.2); Change Existing AMP to be examination category B-N-2/B-N-3 Under "Technical Basis," Item 3, Change inspection category to B-N-2/B-N-3
IV B2-31	Reactor Vessel Internals (PWR-W)	B2.5.2 B2.5.5 B2.5.7	IV B2-26		Aging Effect "Loss of Preload" due to Stress Relaxation is not applicable to Fuel Pins (B2.5.2). They should be deleted from this category.
IV B2-32	Reactor Vessel Internals (PWR-W)	B2.5.2	IV B2-26		Attrition due to Wear is not applicable to fuel pins. There is not sufficient relative motion of adjacent components to produce wear. Delete this category.
IV B2-33	Reactor Vessel Internals (PWR-W)	B2.5.3 B2.5.4	IV B2-26		For Aging Effect = Crack Initiation and Growth, Aging Mechanism IASCC, there is generally not sufficient irradiation exposure to make this a likely mechanism. Only 2 WOG plants (the XL core design) are affected.
IV B2-34	Reactor Vessel Internals (PWR-W)	B2.5.3 B2.5.4	IV B2-28		Change Aging Effect –Reduced fracture toughness Sufficient irradiation exposure will not exist to make this a likely mechanism for these items. Delete this category

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IV

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT/CHANGES
IV B2-35	Reactor Vessel Internals (PWR-W)	B2.6.2	IV B2-28		For Aging effect = Attrition and Aging Mechanism = WEAR, the Reference column should include I&E Bulletin 88-09. Existing program material should be replaced with "utility response to Bulletin 88-09" Technical Basis should reflect Bulletin 88-09 requirements.
IV B2-36	Reactor Vessel Internals (PWR-W)	B2.1.1 and various	IV B2-4 and various		Replace "EPRI TR-105714" with "the latest version of EPRI TR-105714" or "EPRI PWR Primary Water Chemistry Guidelines Rev. 4 March 1999. EPRI TR-105714"

B3. Reactor Vessel Internals (PWR) - Combustion Engineering

B3.1 Upper Internals Assembly

- B3.1.1 Upper Guide Structure Support Plate
- B3.1.2 Fuel Alignment Plate
- B3.1.3 Fuel Alignment Plate Guide Lugs
- B3.1.4 Hold-Down Rings

B3.2 CEA Shroud Assemblies

- B3.2.1 CEA Shrouds
- B3.2.2 CEA Shrouds Bolts
- B3.2.3 CEA Shrouds Extension Shaft Guides

B3.3 Core Support Barrel

- B3.3.1 Core Support Barrel
- B3.3.2 Core Support Barrel Upper Flange
- B3.3.3 Core Support Barrel Alignment Keys

B3.4 Core Shroud Assembly

- B3.4.1 Core Shroud Assembly
- B3.4.2 Core Shroud Assembly Bolts
- B3.4.3 Core Shroud Tie Rods

B3.5 Lower Internal Assembly

- B3.5.1 Core Support Plate
- B3.5.2 Fuel Alignment Pins

B3.5.6 Core Support Barrel Snubber Assemblies

B3. Reactor Vessel Internals (PWR) - Combustion Engineering

System, Structures, and Components

The system, structures, and components included in this table comprise the ABB/Combustion Engineering pressurized water reactor (PWR) reactor vessel internals and consist of the Upper Internals Assembly, CEA Shroud Assemblies, Core Support Barrel, Core Shroud Assembly, and Lower Internal Assembly. All structures and components in the reactor vessel are classified as Group A or B Quality Standards.

System Interfaces

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

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B3. REACTOR VESSEL INTERNALS (PWR) - ABB/Combustion Engineering

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B3.1.1 thru B3.1.3	Upper Internals Assembly	Upper Guide Structure Support Plate, Fuel Alignment Plate, Fuel Alignment Plate Guide Lugs	Stainless Steel (SS)	hemically reated orated ater up to 40°C	Crack Initiation & Growth	Stress Corrosion Cracking (SCC)	ASME Section XI, 1989 Edition. EPRI TR-105714. *Operating Experience* NRC IN 84-18. NRC IN 98-11.
B3.1.1 thru B3.1.3	Upper Internals Assembly	Upper Guide Structure Support Plate, Fuel	SS	hemically reated orated ater up to	Crack Initiation & Growth	Irradiation Assisted Stress Corrosion	ASME Section XI, 1989 Edition. EPRI TR-105714.

IVB3-1 WHOLE SECTION

IVB3-2 WHOLE SECTION

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B3. REACTOR VESSEL INTERNALS (PWR) - ABB/Combustion Engineering

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

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B3. REACTOR VESSEL INTERNALS (PWR) - ABB/Combustion Engineering

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B3.1.1 thru B3.1.3	Upper Internals Assembly	Upper Guide Structure Support Plate, Fuel	SS	hemically reated orated ater up to	Changes in Dimensions	Void Swelling	EPRI TR-107521.

IVB3-3
IVB3-4
WHOLE SECTION

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) potential exists for the formation of more aggressive chemistry conditions by radiolysis in creviced regions or in low-flow stagnant regions. Also for sufficiently high fluence levels, IASCC can occur even in low oxygen environments. The AMP must rely upon ISI in accordance with ASME Section XI to detect possible degradation. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of SCC on the intended function by detection and sizing of cracks by inservice inspection (ISI). Table IWB-2500, category B-N-3 specifies visual VT-3 examination of all accessible surfaces of reactor internals. ***(4) Detection of Aging Effects:*** Degradation due to IASCC can not occur without crack initiation and growth. VT-3 may not be adequate to detect tight cracks. The inspection technique, including the reliability in detecting the features of interest (crack appearance and size) in assuring the integrity of the component, should be specified. For example, enhancement of the visual VT-1 examination to achieve a 1/2-mil (0.0005 in.) resolution, with the conditions (lighting and surface cleanness) for the ISI bounded by those used to demonstrate the resolution of the inspection technique. Creviced regions are difficult to inspect visually and supplementary UT or other nondestructive examinations may be needed. As an alternate to enhanced inspection, perform a component-specific evaluation including a mechanical loading assessment to determine the maximum tensile loading on the component during ASME Code Level A, B, C, and D conditions. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with IWB-3520. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWB-3140. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Although the only cracking presently directly attributable to IASCC in PWRs occurred in SS fuel cladding used in early reactors, susceptibility to this degradation will increase as plant operating time and hence accumulated fluence levels increase. The role of IASCC in the more recent cracking that has occurred in SS baffle/former bolts in U.S. plants as well as foreign plants (IN 98-11) has not yet been universally agreed upon.	
Plant specific aging management program. Based on EPRI TR-107521,	Plant specific aging management program is to be evaluated. The applicant must provide the basis for	Yes TLAA

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B3. REACTOR VESSEL INTERNALS (PWR) - ABB/Combustion Engineering

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B3.1.1, B3.1.3, B3.1.4	Upper Internals Assembly	Upper Guide Structure Support Plate, Fuel Alignment Plate Guide Lugs, Hold-Down Ring	SS	hemically reated orated ater up to 40°C	Attrition	Wear	ASME Section XI, 1989 Edition.
B3.2.1	CEA Shroud Assemblies	CEA Shroud	SS, Cast Austenitic SS (CASS)	hemically reated orated ater up to 40°C	Crack Initiation & Growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B3.2.2	CEA Shroud Assemblies	CEA Shroud Bolts	SS, Ni alloy	hemically reated orated ater up to 40°C	Crack Initiation & Growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714. *Operating Experience* NRC IN 82-29. NRC IN 84-18. NRC IN 98-11.

IV B3-5

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	***(1) Scope of Program:*** The program includes inservice inspection (ISI) to monitor the condition of components and assure that they can perform their intended function. ***(2) Preventive Actions:*** No practical preventative actions are possible. ***(3) Parameters Monitored/Inspected:*** The AMP program monitors the effects of wear on the intended function by inservice inspection (ISI) to determine the general mechanical and structural condition of components. ***(4) Detection of Aging Effects:*** Inspection can reveal indications of degradation due to wear such as the verification of clearances, settings, physical displacements, loose or missing parts, debris, corrosion, wear, erosion, or the loss of integrity at bolted or welded connections. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 is adequate for timely detection of wear. ***(6) Acceptance Criteria:*** Acceptance standard for the inspection is IWB-3520. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWB-3140. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** To date aging management program has been effective in managing the effects of wear on reactor internals.	No
Same as for the effect of Stress Corrosion Cracking on Items B3.1.1 upper guide structure support plate, B3.1.2 fuel alignment plate, and B3.1.3 fuel alignment plate guide lugs.	*Same as for the effect of Stress Corrosion Cracking on Items B3.1.1 upper guide structure support plate, B3.1.2 fuel alignment plate, and B3.1.3 fuel alignment plate guide lugs.*	Yes Element 4 should be further evaluated
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI (edition specified in 10 CFR 50.55a). Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 to minimize the potential of crack initiation and growth.	***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC of components constructed from SS and Ni alloys, inservice inspection (ISI) to monitor the effects of SCC on the intended function of the components, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** PWR operating chemistry limits the halogens, sulfates, and oxygen in the primary water to less than 0.05, 0.05, and 0.005 ppm, respectively, during operation. However, preventive actions are considered inadequate because of possible effects of introduction of oxygen during shutdown and the potential formation of more aggressive chemistry conditions by radiolysis in creviced regions or in low-flow stagnant regions. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of SCC on the intended function by detection and sizing of cracks by inservice inspection (ISI). Table IWB-2500, category B-N-3 specifies visual VT-3 examination of all accessible surfaces of reactor internals. ***(4) Detection of Aging Effects:*** Degradation due to SCC	Yes Element 4 should be further evaluated

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B3. REACTOR VESSEL INTERNALS (PWR) - ABB/Combustion Engineering

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B3.2.1	CEA Shroud Assemblies	CEA Shroud	SS, CASS	hemically reated orated ater up to 40°C	Crack Initiation & Growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B3.2.2	CEA Shroud Assemblies	CEA Shroud Bolts	SS, Ni alloy	hemically reated orated ater up to 40°C	Crack Initiation & Growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B3.2.1, B3.2.2	CEA Shroud Assemblies	CEA Shroud, CEA Shroud Bolts	SS, CASS, Ni alloy	hemically reated orated ater up to 40°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.
B3.2.3	CEA Shroud Assemblies	CEA Shroud Extension Shaft Guides	SS	hemically reated orated ater up to	Attrition	Wear	ASME Section XI, 1989 Edition.

IVB3-3
IVB3-4

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) inaccessible regions are difficult to inspect visually. Supplementary UT examinations may be needed. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with IWB-3520. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWB-3140. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Although stainless steel components in PWRs have generally not been found to be affected by SCC because of low dissolved oxygen levels and control of primary water chemistry, potential for SCC exists from inadvertent introduction of contaminants into the primary coolant system (IN 84–18), from the introduction of relatively high levels of dissolved oxygen during shutdown, or from aggressive chemistries that may develop in creviced regions. Cracking has occurred in SS baffle former bolts in a number of foreign plants (IN 98-11) and has now been observed in US plants. The mechanism of this particular cracking has not yet been resolved. SCC has also been observed in Ni alloy (Alloy X–750) control rod drive guide tube support pins (IN 82-29). Replacement components with a different heat treatment appear to be less susceptible to cracking.	
Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Items B3.1.1 upper guide structure support plate, B3.1.2 fuel alignment plate, and B3.1.3 fuel alignment plate guide lugs.	*Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Items B3.1.1 upper guide structure support plate, B3.1.2 fuel alignment plate, and B3.1.3 fuel alignment plate guide lugs.*	Yes Element 4 should be further evaluated
Same as for the effect of Stress Corrosion Cracking on Item B3.2.2 CEA shroud bolts.	*Same as for the effect of Stress Corrosion Cracking on Item B3.2.2 CEA shroud bolts.*	Yes Element 4 should be further evaluated
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.	Plant specific aging management program is to be evaluated. The applicant must provide the basis for concluding that void swelling is not an issue for the component or must provide an AMP.	Yes TLAA
Visual inspection (VT–3) is performed according to Category B–N–3 of Subsection IWB, ASME Section XI.	*Same as for the effect of Wear on Items B3.1.1 upper guide structure support plate, B3.1.3 fuel alignment plate guide lugs., and B3.1.4 hold-down ring.*	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

B3. REACTOR VESSEL INTERNALS (PWR) - ABB/Combustion Engineering

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B3.2.1	CEA Shroud Assemblies	CEA Shroud	CASS	hemically reated orated ater up to 40°C	Loss of Fracture Toughness	Thermal Aging and Neutron Irradiation Embrittle-ment	EPRI TR-106092. ASME Section XI, 1989 Edition.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B3. REACTOR VESSEL INTERNALS (PWR) - ABB/Combustion Engineering

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) components that have a neutron fluence of less than 10^{17} n/cm^2 (E>1 MeV), implement the supplement inspection program if they are "potentially susceptible" to thermal aging; the existing ASME Section XI inspection requirements are considered adequate if the components are "not susceptible" to thermal aging. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 should provide timely detection of cracks. ***(6) Acceptance Criteria:*** Flaws detected in CASS components are evaluated in accordance with the applicable procedures of IWB-3600. ***(7) Corrective Actions:*** Repair is in conformance with IWA-4000 and IWB-4000, and replacement according to IWA-7000 and IWB-7000. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** The AMP based on susceptibility determination, neutron fluence level, and supplemental inspection is effective in managing the effects of synergistic loss of fracture toughness due to neutron and thermal aging embrittlement on the intended function of CASS components.	
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	***(1) Scope of Program:*** The program includes inservice inspection (ISI) to monitor the condition of the components that depend on preload, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** No practical preventative actions are possible. ***(3) Parameters Monitored/Inspected:*** The AMP utilizes inservice inspection (ISI) to monitor the effects of stress relaxation on the intended function of the component by detection and sizing of cracks that could be formed by excessive vibration etc. if the preload is lost. Table IWB-2500, category B-N-3 specifies visual VT-3 examination of all accessible surfaces of reactor internals. Because VT-3 inspection can only detect degradation that occurs after the loss of preload, it may be adequate if there is sufficient redundancy that loss of some bolting between inspections is acceptable. In some cases additional inspection may be required. ***(4) Detection of Aging Effects:*** As part of the AMP it may be possible to identify acceptable levels of preload and demonstrate under the	Yes Elements 3 and 4 should be further evaluated

IV B3-7

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B3. REACTOR VESSEL INTERNALS (PWR) - ABB/Combustion Engineering

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B3.2.1, B3.2.2	CEA Shroud Assemblies	CEA Shroud, CEA Shroud Bolts	SS, CASS, Ni Alloy	hemically reated orated ater up to 40°C	Cumulative Fatigue Damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B3.2.2	CEA Shroud Assemblies	CEA Shroud Bolts	SS, Ni Alloy	hemically reated orated ater up to 40°C	Loss of Preload	Stress Relaxation	ASME Section XI, 1989 Edition.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B3. REACTOR VESSEL INTERNALS (PWR) - ABB/Combustion Engineering

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B3.3.1, B3.3.2	Core Support Barrel	Core Support Barrel, Core Support Barrel Upper Flange	SS	hemically reated orated ater up to 40°C	Crack Initiation & Growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B3.3.1, B3.3.2	Core Support Barrel	Core Support Barrel, Core Support Barrel Upper Flange	SS	hemically reated orated ater up to 40°C	Crack Initiation & Growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B3.3.1, B3.3.2	Core Support Barrel	Core Support Barrel, Core Support Barrel Upper Flange	SS	hemically reated orated ater up to 40°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.
B3.3.1, B3.3.2	Core Support Barrel	Core Support Barrel, Core Support Barrel Upper Flange	SS	hemically reated orated ater up to 40°C	Loss of Fracture Toughness	Neutron Irradiation Embrittle-ment IVB3-8 IVB3-9	ASME Section XI, 1989 Edition.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) Inspection schedule in accordance with IWB-2400 is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with IWB-3520. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWB-3140. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** There are no reports of stress relaxation producing damage in reactor vessel internals.	
Same as for the effect of Stress Corrosion Cracking on Items B3.1.1 upper guide structure support plate, B3.1.2 fuel alignment plate, and B3.1.3 fuel alignment plate guide lugs.	*Same as for the effect of Stress Corrosion Cracking on Items B3.1.1 upper guide structure support plate, B3.1.2 fuel alignment plate, and B3.1.3 fuel alignment plate guide lugs.*	Yes Element 4 should be further evaluated
Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Items B3.1.1 upper guide structure support plate, B3.1.2 fuel alignment plate, and B3.1.3 fuel alignment plate guide lugs.	*Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Items B3.1.1 upper guide structure support plate, B3.1.2 fuel alignment plate, and B3.1.3 fuel alignment plate guide lugs.*	Yes Element 4 should be further evaluated
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.	Plant specific aging management program is to be evaluated. The applicant must provide the basis for concluding that void swelling is not an issue for the component or must provide an AMP.	Yes TLAA

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B3 REACTOR VESSEL INTERNALS (PWR) - ABB/Combustion Engineering

Item	Structure and Component	Region of Interest	Material	Environment	Aging Effect	Aging Mechanism	References
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI. IVB3-8			***(1) Scope of Program:*** The program includes inservice inspection (ISI) to detect cracking and/or failure, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** No practical preventative actions are possible. Stainless steels are susceptible to embrittlement under neutron irradiation. Fracture toughness will depend strongly on the fluence on a particular component. Components can be screened out if the maximum tensile loading on the component under ASME Code Level A, B, C, and D conditions is sufficiently low. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of neutron irradiation embrittlement on the intended function of the component by detection and sizing of cracks by inservice inspection (ISI). Table IWB-2500, category B-N-3 specifies visual VT-3 examination of all accessible surfaces of reactor internals. IVB3-9 ***(4) Detection of Aging Effects:*** Loss of fracture toughness is of consequence only if cracks exist. Cracking is expected to initiate at the surface and should be detectable by ISI except for some crevice regions. VT-3 may not be adequate to detect tight cracks. Also, creviced regions are difficult to inspect visually and supplementary UT or other nondestructive examinations may be needed. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is				Yes Elements 3 and 4 should be further evaluated
B3.3.1, B3.3.2	Core Support Barrel	Core Support Barrel, Core Support Barrel Upper Flange	SS	hemically reated orated ater up to 40°C	Cumulative Fatigue Damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B3.3.2, B3.3.3	Core Support Barrel	Core Support Barrel Upper Flange, Core Support Barrel Alignment	SS	hemically reated orated ater up to 40°C	Attrition	Wear	ASME Section XI, 1989 Edition.

Existing Aging Management Program (AMP)			Evaluation and Technical Basis					Further Evaluation
B3.4.1, B3.4.3	Core Shroud Assembly	Core Shroud Assembly, Core Shroud Tie Rods (core support plate attached by welds in later plants)	SS	hemically reated orated ater up to 40°C	Crack Initiation & Growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.	
B3.4.1, B3.4.3	Core Shroud Assembly	Core Shroud Assembly, Core Shroud Tie Rods (core support plate attached by welds in later plants)	SS	hemically reated orated ater up to 40°C	Crack Initiation & Growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.	
B3.4.1, B3.4.3	Core Shroud Assembly	Core Shroud Assembly, Core Shroud Tie Rods (core support plate attached by welds in later plants)	SS, CASS, Ni alloy	hemically reated orated ater up to 40°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.	
B3.4.1, B3.4.3	Core Shroud Assembly	Core Shroud Assembly, Core Shroud Tie Rods (core support plate attached by welds in later plants)	SS	hemically reated orated ater up to 40°C	Loss of Fracture Toughness	Neutron Irradiation Embrittle-ment	ASME Section XI, 1989 Edition.	

	(continued from previous page) evaluated in accordance with IWB-3520. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWB-3140. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** No instances of internals degradation recorded have been definitely attributed to irradiation embrittlement.	
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B3. REACTOR VESSEL INTERNALS (PWR) - ABB/Combustion Engineering

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
Same as for the effect of Stress Corrosion Cracking on Items B3.1.1 upper guide structure support plate, B3.1.2 fuel alignment plate, and B3.1.3 fuel alignment plate guide lugs.			*Same as for the effect of Stress Corrosion Cracking on Items B3.1.1 upper guide structure support plate, B3.1.2 fuel alignment plate, and B3.1.3 fuel alignment plate guide lugs.*				Yes Element 4 should be further evaluated
Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Items B3.1.1 upper guide structure support plate, B3.1.2 fuel alignment plate, and B3.1.3 fuel alignment plate guide lugs.			*Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Items B3.1.1 upper guide structure support plate, B3.1.2 fuel alignment plate, and B3.1.3 fuel alignment plate guide lugs.*				Yes Element 4 should be further evaluated
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.			Plant specific aging management program is to be evaluated. The applicant must provide the basis for concluding that void swelling is not an issue for the component or must provide an AMP.				Yes TLAA
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.			*Same as for the effect of Neutron Irradiation Embrittlement on Items B3.3.1 core support barrel and B3.3.2 core support barrel upper flange.*				Yes Elements 3 and 4 should be further evaluated
B3.4.1 thru B3.4.3	Core Shroud Assembly	Core Shroud Assembly, Core Shroud Assembly Bolts, Core Shroud Tie Rods	SS, Ni Alloy (bolts)	hemically reated orated ater up to 40°C	Cumulative Fatigue Damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B3.4.2	Core Shroud Assembly	Core Shroud Assembly Bolts (later plants are welded)	SS, Ni Alloy	hemically reated orated ater up to 40°C	Crack Initiation & Growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B3.4.2	Core Shroud Assembly	Core Shroud Assembly Bolts (later plants are welded)	SS, Ni Alloy	hemically reated orated ater up to 40°C	Crack Initiation & Growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B3.4.2	Core Shroud	Core Shroud	SS,	hemically	Changes in	Void	EPRI TR-107521.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B3. REACTOR VESSEL INTERNALS (PWR) - ABB/Combustion Engineering

Existing Aging Management Program (AMP)			Evaluation and Technical Basis					Further Evaluation
B3.4.2	Core Shroud Assembly	Core Shroud Assembly Bolts (later plants are welded)	SS, Ni Alloy	hemically reated orated ater up to 40°C	Loss of Fracture Toughness	Neutron Irradiation Embrittle-ment	ASME Section XI, 1989 Edition.	
B3.4.2, B3.4.3	Core Shroud Assembly	Core Shroud Assembly Bolts, Core Shroud Tie Rods	SS, Ni Alloy	hemically reated orated ater up to 40°C	Loss of Preload	Stress Relaxation	ASME Section XI, 1989 Edition.	
B3.5.1, B3.5.3, B3.5.4, B3.5.6	Lower Internal Assembly	Core Support Plate, Lower Support Structure Beam Assemblies, Core Support Column, Core Support Barrel Snubber Assemblies	SS	hemically reated orated ater up to 40°C	Crack Initiation & Growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.	
B3.5.2, B3.5.5	Lower Internal Assembly	Fuel Alignment Pins, Core Support Column Bolts,	SS, Ni Alloy	hemically reated orated ater up to 40°C	Crack Initiation & Growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.	

Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Same as for the effect of Stress Corrosion Cracking on Item B3.2.2 CEA shroud bolts.	*Same as for the effect of Stress Corrosion Cracking on Item B3.2.2 CEA shroud bolts.*	Yes Element 4 should be further evaluated
Same as for the effect of Stress Corrosion Cracking on Item B3.2.2 CEA shroud bolts.	*Same as for the effect of Stress Corrosion Cracking on Item B3.2.2 CEA shroud bolts.*	Yes Element 4 should be further evaluated

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B3. REACTOR VESSEL INTERNALS (PWR) - ABB/Combustion Engineering

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.			*Same as for the effect of Neutron Irradiation Embrittlement on Items B3.3.1 core support barrel and B3.3.2 core support barrel upper flange.*				Yes Elements 3 and 4 should be further evaluated
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.			*Same as for the effect of Stress Relaxation on Item B3.2.2 CEA shroud bolts.*				Yes Elements 3 and 4 should be further evaluated
Same as for the effect of Stress Corrosion Cracking on Items B3.1.1 upper guide structure support plate, B3.1.2 fuel alignment plate, and B3.1.3 fuel alignment plate guide lugs.			*Same as for the effect of Stress Corrosion Cracking on Items B3.1.1 upper guide structure support plate, B3.1.2 fuel alignment plate, and B3.1.3 fuel alignment plate guide lugs.*				Yes Element 4 should be further evaluated
Same as for the effect of Stress Corrosion Cracking on Item B3.2.2 CEA shroud bolts.			*Same as for the effect of Stress Corrosion Cracking on Item B3.2.2 CEA shroud bolts.*				Yes Element 4 should be further evaluated
B3.5.1, B3.5.3, B3.5.4, B3.5.6	Lower Internal Assembly	Core Support Plate, Lower Support Structure Beam Assemblies, Core Support Column, Core Support Barrel Snubber Assemblies	SS	hemically reated orated ater up to 40°C	Crack Initiation & Growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B3.5.2, B3.5.5	Lower Internal Assembly	Fuel Alignment Pins, Core Support Column Bolts,	SS, Ni Alloy	hemically reated orated ater up to 40°C	Crack Initiation & Growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B3. REACTOR VESSEL INTERNALS (PWR) - ABB/Combustion Engineering

Existing Aging Management Program (AMP)			Evaluation and Technical Basis					Further Evaluation
B3.5.1 thru B3.5.6	Lower Internal Assembly	Core Support Plate, Fuel Alignment Pins, Lower Support Structure Beam Assemblies, Core Support Column, Core Support Column Bolts Core Support Barrel Snubber Assemblies	SS, Ni Alloy (pins/bolts), CASS (support column)	hemically reated orated ater up to 40°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.	
B3.5.1 thru B3.5.6	Lower Internal Assembly	Core Support Plate, Fuel Alignment Pins, Lower Support Structure Beam Assemblies, Core Support Column, Core Support Column Bolts Core Support Barrel Snubber Assemblies	SS, Ni Alloy (pins/bolts), CASS (support column)	hemically reated orated ater up to 40°C	Loss of Fracture Toughness	Neutron Irradiation Embrittle-ment	ASME Section XI, 1989 Edition.	
B3.5.2, B3.5.5	Lower Internal Assembly	Fuel Alignment Pins, Core Support Column Bolts	SS, Ni Alloy	hemically reated orated ater up to 40°C	Loss of Preload	Stress Relaxation	ASME Section XI, 1989 Edition.	

IV B3-11

Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Items B3.1.1 upper guide structure support plate, B3.1.2 fuel alignment plate, and B3.1.3 fuel alignment plate guide lugs.	*Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Items B3.1.1 upper guide structure support plate, B3.1.2 fuel alignment plate, and B3.1.3 fuel alignment plate guide lugs.*	Yes Element 4 should be further evaluated

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B3. REACTOR VESSEL INTERNALS (PWR) - ABB/Combustion Engineering

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.			Plant specific aging management program is to be evaluated. The applicant must provide the basis for concluding that void swelling is not an issue for the component or must provide an AMP.				Yes TLAA
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.			*Same as for the effect of Neutron Irradiation Embrittlement on Items B3.3.1 core support barrel and B3.3.2 core support barrel upper flange.*				Yes Elements 3 and 4 should be further evaluated
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.			*Same as for the effect of Stress Relaxation on Item B3.2.2 CEA shroud bolts.*				Yes Elements 3 and 4 should be further evaluated
B3.5.2, B3.5.6	Lower Internal Assembly	Fuel Alignment Pins, Core Support Barrel Snubber Assemblies	SS, Ni Alloy (pins)	hemically reated orated ater up to 40°C	Attrition	Wear	ASME Section XI, 1989 Edition.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B3. REACTOR VESSEL INTERNALS (PWR) - ABB/Combustion Engineering

Existing Aging Management Program (AMP)			Evaluation and Technical Basis				Further Evaluation
B3.5.1 thru B3.5.6	Lower Internal Assembly	Core Support Plate, Fuel Alignment Pins, Lower Support Structure Beam Assemblies, Core Support Column, Core Support Column Bolts Core Support Barrel Snubber Assemblies	SS, Ni Alloy (pins/bolts), CASS (support column)	hemically reated orated ater up to 40°C	Cumulative Fatigue Damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B3.5.4	Lower Internal Assembly	Core Support Column	CASS	hemically reated orated ater up to 40°C	Loss of Fracture Toughness	Thermal Aging and Neutron Irradiation Embrittle-ment	EPRI TR-106092. ASME Section XI, 1989 Edition.

Visual inspection (VT-3) is performed	*Same as for the effect of Wear on Items B3.1.1 upper guide*	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B3. REACTOR VESSEL INTERNALS (PWR) - ABB/Combustion Engineering

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.			Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.				Yes TLAA
Same as for the effect of Thermal Aging and Neutron Irradiation Embrittlement on Item B3.2.1 CEA shroud.			*Same as for the effect of Thermal Aging and Neutron Irradiation Embrittlement on Item B3.2.1 CEA shroud.*				No

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IV

GALL SECTION	TITLE	ITEM NO	PAGE	REVIEWER	COMMENT
IV B3-1	Reactor Vessel Internals (PWR) – (Combustion Engineering)	B3.1.1, B3.1.2, B3.1.3, B3.2.1, B3.2.2, B3.3.1, B3.3.2, B3.4.1, B3.4.2, B3.4.3, B3.5.2 & B3.5.5 (All IASCC)	IV B3-4, 5, 10, 11, 16, 18, 19, 20, 21, 22 & 23		IASCC is listed as an Aging Mechanism for the Upper Internals Assembly, CEA Shroud Assembly, Core Shroud Bolts, Core Support Barrel, Core Shroud/Tie Rod, and Lower Internals Assembly. The low levels of dissolved oxygen in a PWR environment and the low applied strain of the RV Internals components cause IASCC to be an unlikely Aging Mechanism for this device type. This position was accepted in NUREG-1705. This mechanism has been observed in BWRs where oxygen levels are considerably higher than in PWRs. A similar Aging Mechanism has also been observed in PWR CEDM tips where very high strain is applied at very low strain rate in a high fluence field. However, there is not conclusive evidence of IASCC for device types with the temperature, oxygen and radiation levels present for the RV Internals either in operating plants or in laboratory tests. Since there is not clear agreement on this Aging Mechanism, and since new information is emerging, such as from the EPRI Materials Reliability Project, applicants could agree to participate in industry efforts to address the significance of IASCC, such as with a program similar to that accepted in NUREG-1705 for Void swelling, in lieu of the listed Existing AMP. Prior to year 40, if it is determined that IASCC is a significant issue in the renewal term, they would agree to develop a sufficient inspection program (including the basis, methods, locations to be examined, timing frequency and acceptance criteria) for management of the issue based upon the results of the industry information. This agreement would not constitute consideration of this Aging Mechanism as requiring management and the agreement would not constitute a "credited program" at this time.

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV.B3-2	Reactor Vessel Internals (PWR) – (Combustion Engineering)	B3.1.1, B3.1.2, B3.1.3, B3.2.1, B3.2.2, B3.3.1, B3.3.2, B3.4.1, B3.4.2, B3.4.3 & B3.5.1 Through B3.5.6 (All SSC)	IV.B3-4, 5, 8, 9, 11 & 16 through 21		SCC is listed as an Aging Mechanism for the Upper Internals Assembly, CEA Shroud Assembly, Core Shroud Bolts, Core Support Barrel, Core Shroud/Tie Rods, and Lower Internals Assembly. SCC/IGSCC is not plausible for this device type due to non-susceptible material (Alloy Steel, Stainless Steel and/or Nickel Base Stainless Steel), lack of high tensile stresses and control of water chemistry. SCC is not a concern for SS components in treated borated water where chemistry controls maintain halides < 150 ppb or sulfates < 100 ppb (BAW-2270). Chemistry controls in accordance with industry guidelines assure this requirement is met. Therefore, for SCC, chemistry programs in accordance with industry guidelines alone should be credited. A similar position was accepted in NUREG-1705. The References, Existing AMP, Evaluation and Technical Basis, and Further Evaluation entries should be rewritten to correspond to the provided example.

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IV

GALL SECTION	TITLE	ITEM NO	PAGE	REVIEWER	COMMENT
IV B3-3	Reactor Vessel Internals (PWR) (Combustion Engineering)	B3.1.1, B3.1.2, B3.1.3, B3.2.1, B3.2.2, B3.3.1, B3.3.2, B3.4.1, B3.4.2, B3.4.3 & B3.5.1 Through B3.5.6 (All Void Swelling)	IV B3-6, 7, 10, 11 & 16 through 23		Void Swelling is listed as an Aging Mechanism. BGE did not identify void swelling as a plausible degradation mechanism in its license renewal application for Calvert Cliffs nor did the NRC staff identify this potential aging effect in its initial review of the Calvert Cliff application. Subsequently, void swelling was identified by the NRC staff as an applicable aging effect for reactor vessel internals during the review of the B&W topical report and the B&WOG vessel internals program. BGE maintains that void swelling is not plausible but agreed to participate in industry programs to address the significance of void swelling. Prior to year 40, if BGE determines that void swelling is a significant issue in the renewal term, BGE agrees to develop a sufficient inspection program (including the basis, methods, locations to be examined, timing frequency and acceptance criteria) for management of the issue based upon the results of the industry programs, and performed in conjunction with the 10-year inservice inspection program. If BGE has made its determination far enough in advance of the end of the current license period, BGE will implement the inspection program prior to the end of that period. Otherwise, the program will be implemented as soon as practicable thereafter. This agreement does not constitute consideration of this Aging Mechanism as "Plausible" and the agreement does not constitute a "credited program". This position was accepted in NUREG-1705.
IV.B3-4	Reactor Vessel Internals (PWR) – (Combustion Engineering)	All sections with Void Swelling See previous comment	IV B3-7, 11, 17, 19, 21 & 23		Void Swelling is listed as an Aging Mechanism. (This mechanism was considered Not Plausible for CCNPP; see separate comments on the Aging Mechanism elsewhere. Comments to follow assume that the Aging Mechanism is retained in Report despite earlier comment.) Since information on this Aging Mechanism and its applicability to these components is only now emerging, relative to commercial PWRs, and no consideration has to this point been afforded it in plant design or analysis calculations for US PWRs, it is impossible that a TLAA would exist on this issue.
IV B3-5	Reactor Vessel Internals (PWR) Combustion	B3.1.1, B3.1.2 & B3.1.4	IV B3-8		Wear is listed as an Aging Mechanism for the Upper Internals Assembly. The applicability to the subcomponents does not agree with EPRI TR 104305. The Upper Guide Structure Support (B3.1.1) should not be listed and the Fuel

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
	Engineering				Alignment Plate (B3.1.2) should be listed.
IV.B3-6	Reactor Vessel Internals (PWR) – (Combustion Engineering)	B3.2.2	IV B3-10 & 11		The CEA Shroud Bolts have been determined to be not required for the CEA Shroud and Fuel Alignment Plate functions to be performed during normal operation and design basis events for CCNPP, pending additional analysis. Inclusion of this item in the Report should be qualified with this statement and the fact that they may be excluded for plants with the same configuration as CCNPP. This position was accepted in NUREG-1705.
IV.B3-7	Reactor Vessel Internals (PWR) (Combustion Engineering)	B3.2.1 & B3.2.2	IV B3-14 & 15		Fatigue is listed as an Aging Mechanism. "Existing AMP" entry states that components have been designed for fatigue for a 40 year design life in accordance with ASME III. This not strictly true; it has been assumed that the components in question will experience certain operational stresses during the plant lifetime, for example, less than or equal to 500 thermal cycles caused by plant startups and shutdowns. This number of transients will have originally been estimated with some consideration of a 40 year plant life, but there is no explicit correlation to a time period in this evaluation. Also, simply identifying Fatigue as a TLAA does not identify an AMP per se. More information should be provided, for example, relevant to Fatigue Monitoring Programs that calculate Cumulative Usage Factors based on partial cycles. In addition, if high-cycle Fatigue caused by flow-induced vibrations is the Aging Mechanism of concern, a discussion of TLAAs is not applicable and a program such as ISI or a plant-specific program may have to be credited.
IV.B3-8	Reactor Vessel Internals (PWR) – (Combustion Engineering)	B3.3.1 & B3.3.2	IV B3-16 & 17		Neutron irradiation embrittlement is a misnomer and a non-issue for most internals. Embrittlement does not occur in austenitic stainless steel after 80 years. There is some ductility loss, but the material retains considerable resistance to crack initiation and growth. This Aging Mechanism should not be considered. At most it should be required that applicants participate in industry programs to address the significance of this Aging Mechanism, and develop an aging management program if future experience warrants.

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IV

GALL SECTION	TITLE	ITEM NO	PAGE	REVIEWER	COMMENT
IV.B3-9	Reactor Vessel Internals (PWR) (Combustion Engineering)	B3.3.1 & B3.3.2	IV.B3-16 & 17		Neutron Irradiation Embrittlement is listed as an Aging Mechanism. (This mechanism was recommended to be removed from the GALL Report; see separate comments on the Aging Mechanism elsewhere. Comments to follow assume that the Aging Mechanism is retained in Report despite earlier comment.) Evaluation and Technical Basis entry, Item (4), states that "VT-3 may not be adequate to detect tight cracks". No information is provided as to what is considered adequate. The entry on IASCC suggests an enhanced VT-1 or, alternatively, a component-specific loading evaluation. (This program combination was accepted in NUREG-1705.) The inconclusive nature of this entry does not provide the prospective applicant with enough useful information. An entry along the lines of the entry for IASCC would be more helpful for the prospective applicant.

GALL SECTION	TITLE	ITEM NO	PAGE	REVIEWER	COMMENT
IV.B3-11	Reactor Vessel Internals (PWR) – (Combustion Engineering)	B3.5.2 and B3.5.5	IV B3-22 & 23		Stress Relaxation is listed as an Aging Mechanism for the FAP and CSC Bolts. Because the fuel alignment pins are pre-shrunk to fit into the core support plate, some industry documentation considers that stress relaxation could affect this component's function. Unlike a bolt or other thread fastener, the preload stresses on an insert pin are compressive and thus would not result in plastic deformation of the pin. Further, if any plastic deformation of the pin did occur, subsequent stress levels on the pin would decrease substantially, resulting in insufficient stress for further relaxation. Also, stress relaxation is unlikely because temperatures are too low for thermal relaxation and fluence too low for irradiation-induced relaxation. The pins only need to provide lateral support to the bundle and could still perform this function with some loss of preload. Because this ARDM could not affect the component's intended function, stress relaxation is not plausible for the fuel alignment pins. This position was accepted in NUREG-1705.

B4 Reactor Vessel Internals (PWR) - Babcock & Wilcox

B4.1 Plenum Cover and Plenum Cylinder

B4.2 CRA Guide Tube Assemblies

B4.2.1 Tubes

B4.2.2 Spacer Casting

B4.2.3 Spider Casting

B4.2.4 Bolts

B4.3 Upper Grid Assembly

B4.3.1 Upper Grid Rib Section

B4.3.2 Upper Grid Assembly Bolts

B4.3.3 Plenum Rib Pads

B4.4 Core Support Assembly

B4.4.1 Core Support Shield

B4.4.2 Core Support Shield Flange

B4.4.3 Core Support Shield to Core Barrel Bolts

B4.5 Vent Valve Assembly

B4.6 Core Barrel Assembly

B4.6.1 Core Barrel

B4.6.2 Core Barrel Bolts

B4.6.3 Core Barrel to Thermal Shield Bolts

B4.7 Lower Grid Assembly (LGA)

B4.7.4 Support Columns

B4.7.5 Cylinder Guide Blocks

B4.7.6 LGA Bolts

B4.7.7 Core Barrel Bolts

B4.7.8 Flow Distributor Bolts

B4.7.9 Thermal Shield Bolts

B4.8 Flow Distributor

B4.9 Thermal Shield

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

System, Structures, and Components

The system, structures, and components included in this table comprise the Babcock & Wilcox pressurized water reactor (PWR) reactor vessel internals and consist of the Plenum Cover and Plenum Cylinder, Control Rod Assemblies (CRA) Guide Tube Assemblies, Upper Grid Assembly, Core Support Assembly, Vent Valve Assembly, Core Barrel Assembly, and Lower Internal Assembly. All structures and components in the reactor vessel are classified as Group A or B Quality Standards.

System Interfaces

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

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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B4.1 IV B4-17	Plenum Cover and Plenum Cylinder	Plenum Cover and Plenum Cylinder	Stainless Steel (SS)	hemically reated orated ater up to 40°C	Crack Initiation & Growth	Stress Corrosion Cracking (SCC) IV B4-1	ASME Section XI, 1989 Edition. EPRI TR-105714. *Operating Experience* NRC IN 84-18. NRC IN 98-11. IV B4-18
B4.1	Plenum Cover and Plenum Cylinder	Plenum Cover and Plenum Cylinder	SS	hemically reated orated ater up to	Crack Initiation and Growth	Irradiation Assisted Stress Corrosion	ASME Section XI, 1989 Edition. EPRI TR-105714.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

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

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

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B4.1	Plenum Cover	Plenum Cover	SS	hemically	Changes in	Void	EPRI TR-107521.

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) greatly reduces susceptibility to IASCC. However, introduction of oxygen can occur during shutdown and potential exists for the formation of more aggressive chemistry conditions by radiolysis in creviced regions or in low-flow stagnant regions. Also for sufficiently high fluence levels, IASCC can occur even in low oxygen environments. The AMP must rely upon ISI in accordance with ASME Section XI to detect possible degradation. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of SCC on the intended function by detection and sizing of cracks by inservice inspection (ISI). Table IWB-2500, category B-N-3 specifies visual VT-3 examination of all accessible surfaces of reactor internals. ***(4) Detection of Aging Effects:*** Degradation due to IASCC can not occur without crack initiation and growth. VT-3 may not be adequate to detect tight cracks. The inspection technique, including the reliability in detecting the features of interest (crack appearance and size) in assuring the integrity of the component, should be specified. For example, enhancement of the visual VT-1 examination to achieve a 1/2-mil (0.0005 in.) resolution, with the conditions (lighting and surface cleanness) for the ISI bounded by those used to demonstrate the resolution of the inspection technique. Creviced regions are difficult to inspect visually and supplementary UT or other nondestructive examinations may be needed. As an alternate to enhanced inspection, perform a component-specific evaluation including a mechanical loading assessment to determine the maximum tensile loading on the component during ASME Code Level A, B, C, and D conditions. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with IWB-3520. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWB-3140. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Although the only cracking presently directly attributable to IASCC in PWRs occurred in SS fuel cladding used in early reactors, susceptibility to this degradation will increase as plant operating time and hence accumulated fluence levels increase. The role of IASCC in the more recent cracking that has occurred in SS baffle/former bolts in U.S. plants as well as foreign plants (IN 98-11) has not yet been universally agreed upon.	

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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B4.1	Plenum Cover and Plenum Cylinder	Plenum Cover and Plenum Cylinder	SS	hemically reated orated ater up to 40°C	Cumulative fatigue damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B4.2.1 thru B4.2.4	CRA Guide Tube Assemblies	Tubes, Spacer Casting, Spider Casting, Bolts	SS, Cast Austenitic Stainless Steel (CASS), Microbraze, Ni alloy	hemically reated orated ater up to 40°C	Crack initiation & growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B4.2.1 thru B4.2.4	CRA Guide Tube Assemblies	Tubes, Spacer Casting, Spider Casting, Bolts	SS, CASS, Microbraze, Ni alloy	hemically reated orated ater up to 40°C	Crack initiation & growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B4.2.1 thru B4.2.4	CRA Guide Tube Assemblies	Tubes, Spacer Casting, Spider Casting, Bolts	SS, CASS, Microbraze, Ni alloy	hemically reated orated ater up to 40°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.
B4.2.1 thru B4.2.3	CRA Guide Tube Assemblies	Tubes, Spacer Casting, Spider Casting	CASS	hemically reated orated ater up to 40°C	~~Loss of~~ Fracture Toughness REDUCTION	Thermal Aging and Neutron Irradiation Embrittle-ment	EPRI TR-106092. ASME Section XI, 1989 Edition.

IVB4-3
IVB4-6
IVB4-4
IVB4-2
IVB4-4

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA IV B4.2
Same as for the effect of Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.	*Same as for the effect of Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.*	Yes Element 4 should be further evaluated
Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.	*Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.*	Yes Element 4 should be further evaluated IV B4.2
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.	Plant specific aging management program is to be evaluated. The applicant must provide the basis for concluding that void swelling is not an issue for the component or must provide an AMP.	Yes TLAA

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

Item	Structure and Component	Region of Interest	Material	Environment	Aging Effect	Aging Mechanism	References
Determination of the susceptibility of CASS components to thermal aging embrittlement based on casting method, Mo content, and percent ferrite. For "potentially susceptible" components, based on the neutron fluence of the component, implement either a supplemental examination of the affected components as part of the applicant's 10-year inservice inspection (ISI) program during the license renewal term or a component-specific evaluation to determine the susceptibility to loss of fracture toughness.			***(1) Scope of Program:*** The program includes determination of the susceptibility of CASS components to thermal aging based on casting method, Mo content, and percent ferrite, and for "potentially susceptible" components, to account for the synergistic loss of fracture toughness due to neutron embrittlement and thermal aging embrittlement, implement either a supplemental examination of the affected components as part of a 10-year inservice inspection (ISI) program during the license renewal term or a component-specific evaluation to determine the susceptibility to loss of fracture toughness. ***(2) Preventive Actions:*** The program provides no guidance on methods to mitigate thermal aging or neutron embrittlement. ***(3) Parameters Monitored/ Inspected:*** The program specifics depend on the neutron fluence and ferrite content of the component. The criteria in EPRI TR-106092, with some modifications approved by the NRC staff, can be used to determine whether CASS components are potentially susceptibility to thermal aging embrittlement based on the casting method, Mo content and ferrite content. For low-Mo content (0.5 wt.% max.) steels, only static-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement, all centrifugal-cast and static-cast steels with ≤20% ferrite are not susceptible. For high-Mo content (2.0 to 3.0 wt.%) steels, static-cast steels with >14% ferrite and centrifugal-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement, static-cast steels with ≤14% ferrite and centrifugal-cast steels with ≤20% ferrite are not susceptible. Ferrite content will be calculated by the Hull's equivalent factors or a method producing an equivalent level of accuracy (±6% deviation between measured and calculated values). ***(4) Detection of Aging Effects:*** For all CASS components that have a neutron						No

IV B4-4

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

Existing Aging Management Program (AMP)			Evaluation and Technical Basis					Further Evaluation
B4.2.1	CRA Guide Tube	Tubes	SS, CASS,	hemically reated	Attrition	Wear		ASME Section XI, 1989 Edition.

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
			(continued from previous page) fluence of greater than 10^{17} n/cm^2 (E>1 MeV), implement a 10-year ISI program during the renewal period that includes supplemental inspection that covers portions of the susceptible components determined to be limiting from the standpoint of thermal aging susceptibility (Ferrite and Mo contents, casting process, and operating temperature), neutron fluence, and cracking susceptibility (applied stress, operating temperature, and environmental conditions). The inspection technique, including the reliability in detecting the features of interest (crack appearance and size) in assuring the integrity of the component, should be specified. For example, enhancement of the visual VT-1 examination to achieve a 1/2-mil (0.0005 in.) resolution, with the conditions (lighting and surface cleanness) for the ISI bounded by those used to demonstrate the resolution of the inspection technique. perform a component-specific , perform a component-specific evaluation including a mechanical loading assessment to determine the maximum tensile loading on the component under ASME Code Level A, B, C, and D conditions. If the loading is compressive or low enough (<5 ksi) to preclude fracture, then supplemental inspection of the component is not required. Failure to meet this criteria requires continued use of supplemental inspection program. For all CASS components that have a neutron fluence of less than 10^{17} n/cm^2 (E>1 MeV), implement the supplement inspection program if they are "potentially susceptible" to thermal aging; the existing ASME Section XI inspection requirements are considered adequate if the components are "not susceptible" to thermal aging. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 should provide timely detection of cracks. ***(6) Acceptance Criteria:*** Flaws detected in CASS components are evaluated in accordance with the applicable procedures of IWB-3600. ***(7) Corrective Actions:*** Repair is in conformance with IWA-4000 and IWB-4000, and replacement according to IWA-7000 and IWB-7000. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** The AMP based on susceptibility determination, neutron fluence level, and supplemental inspection is effective in managing the effects of synergistic loss of fracture toughness due to neutron and thermal aging embrittlement on the intended function of CASS components.				
Visual inspection (VT-3) is performed			***(1) Scope of Program:*** The program includes inservice				No

Existing Aging Management Program (AMP)			Evaluation and Technical Basis				Further Evaluation
B4.2.4	CRA Guide Tube Assemblies	Bolts	SS, Ni Alloy	hemically reated orated ater up to 40°C	Loss of Preload	Stress Relaxation	ASME Section XI, 1989 Edition.

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
			(continued from previous page) condition of components. ***(4) Detection of Aging Effects:*** Inspection can reveal indications of degradation due to wear such as the verification of clearances, settings, physical displacements, loose of missing parts, debris, corrosion, wear, erosion, or the loss of integrity at bolted or welded connections. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 is adequate for timely detection of wear. ***(6) Acceptance Criteria:*** Acceptance standard for the inspection is IWB-3520. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWB-3140. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** To date the AMP has been effective in managing the effects of wear on reactor internals.				
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.			***(1) Scope of Program:*** The program includes inservice inspection (ISI) to monitor the condition of the components that depend on preload, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** No practical preventative actions are possible. ***(3) Parameters Monitored/Inspected:*** The AMP utilizes inservice inspection (ISI) to monitor the effects of stress relaxation on the intended function of the component by detection and sizing of cracks that could be formed by excessive vibration etc. if the preload is lost. Table IWB-2500, category B-N-3 specifies visual VT-3 examination of all accessible surfaces of reactor internals. Because VT-3 inspection can only detect degradation that occurs after the loss of preload, it may be adequate if there is sufficient redundancy that loss of some bolting between inspections is acceptable. In some cases additional inspection may be required. ***(4) Detection of Aging Effects:*** As part of the AMP it may be possible to identify acceptable levels of preload and demonstrate under the fluence of interest whether less of acceptable preload is likely. VT-3 may not be adequate to detect tight cracks. Creviced regions are difficult to inspect visually. Supplementary inspections by techniques such as ultrasonic testing (UT) or other nondestructive methods may be needed to detect cracking in inaccessible regions.***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with IWB-3520. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWB-3140. ***(8 & 9) Confirmation***				Yes Elements 3 and 4 should be further evaluated

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

Existing Aging Management Program (AMP)			Evaluation and Technical Basis				Further Evaluation
B4.2.1 thru B4.2.4	CRA Guide Tube Assemblies	Tubes, Spacer Casting, Spider Casting, Bolts	SS, CASS, Nicrobraze, Ni alloy	hemically reated orated ater up to 40°C	Cumulative fatigue damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B4.3.1 thru B4.3.3	Upper Grid Assembly	Upper Grid Rib Section, Upper Grid Assembly Bolts, Fuel Guide Pads	SS, Ni alloy	hemically reated orated ater up to 40°C	Crack initiation & growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B4.3.1 thru B4.3.3	Upper Grid Assembly	Upper Grid Rib Section, Upper Grid Assembly Bolts, Fuel Guide Pads	SS, Ni alloy	hemically reated orated ater up to 40°C	Crack initiation & growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B4.3.1 thru B4.3.3	Upper Grid Assembly	Upper Grid Rib Section, Upper Grid Assembly Bolts, Fuel Guide Pads	SS, Ni alloy	hemically reated orated ater up to 40°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.
B4.3.1 thru B4.3.3	Upper Grid Assembly	Upper Grid Rib Section, Upper Grid Assembly Bolts, Fuel Guide Pads	SS, Ni alloy	hemically reated orated ater up to 40°C	Cumulative fatigue damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.

IV B4-7

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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B4.3.1 thru B4.3.3	Upper Grid Assembly	Upper Grid Rib Section, Upper Grid Assembly Bolts, Fuel Guide Pads	SS, Ni alloy	hemically reated orated ater up to 40°C	Loss of Fracture Toughness	Neutron Irradiation Embrittle-ment	ASME Section XI, 1989 Edition.

IV B4 - 7

Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Same as for the effect of Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.	*Same as for the effect of Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.*	Yes Element 4 should be further evaluated
Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.	*Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.*	Yes Element 4 should be further evaluated
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the	Plant specific aging management program is to be evaluated. The applicant must provide the basis for concluding that void swelling is not an issue for the component or must provide an AMP.	Yes TLAA

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	***(1) Scope of Program:*** The program includes inservice inspection (ISI) to detect cracking and/or failure, and repair and/or replacement as needed to maintain the capability to perform the intended function. ***(2) Preventive Actions:*** No practical preventative actions are possible. Stainless steels are susceptible to embrittlement under neutron irradiation. Fracture toughness will depend strongly on the fluence on a particular component. Components can be screened out if the maximum tensile loading on the component under ASME Code Level A, B, C, and D conditions is sufficiently low. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of neutron irradiation embrittlement on the intended function of the component by detection and sizing of cracks by inservice inspection (ISI). Table IWB-2500, category B-N-3 specifies visual VT-3 examination of all accessible surfaces of reactor internals. ***(4) Detection of Aging Effects:*** Loss of fracture toughness is of consequence only if cracks exist. Cracking is expected to initiate at the surface and should be detectable by ISI except for some crevice regions. VT-3 may not be adequate to detect tight cracks. Also, creviced regions are difficult to inspect visually and supplementary UT or other nondestructive examinations may be needed. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 is adequate for timely detection of cracks. ***(6) Acceptance Criteria:*** Any degradation is evaluated in accordance with IWB-3520.	Yes Elements 3 and 4 should be further evaluated

B4.3.3	Upper Grid Assembly	Plenum Rib Pads	SS	hemically reated orated ater up to 40°C	Attrition	Wear	ASME Section XI, 1989 Edition.
B4.3.2	Upper Grid Assembly	Upper Grid Assembly Bolts	SS, Ni alloy	hemically reated orated ater up to 40°C	Loss of Preload	Stress Relaxation	ASME Section XI, 1989 Edition.
B4.4.1 B4.4.2	Core Support Assembly	Core Support Shield,	SS	hemically reated	Crack initiation &	SCC	ASME Section XI, 1989 Edition.

IVB4-8

IVB4-9

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

Item	Structure and Component	Region of Interest	Material	Environment	Aging Effect	Aging Mechanism	References
B4.4.1 B4.4.2	Core Support Assembly	Core Support Shield, Core Support Shield Flange	SS	hemically reated orated ater up to 40°C	Crack initiation & growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B4.4.1 B4.4.2	Core Support Assembly	Core Support Shield, Core Support Shield Flange	SS	hemically reated orated ater up to 40°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.
B4.4.1 thru B4.4.3	Core Support Assembly	Core Support Shield, Core Support Shield Flange, Core Support Shield to Core Barrel Bolts	SS	hemically reated orated ater up to 40°C	Loss of Fracture Toughness	Neutron Irradiation Embrittlement	ASME Section XI, 1989 Edition.
B4.4.1, B4.4.2	Core Support Assembly	Core Support Shield, Core Support Shield Flange	SS	hemically reated orated ater up to 40°C	Cumulative fatigue damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B4.4.2	Core Support Assembly	Core Support Shield Flange	SS	hemically reated orated ater up to 40°C	Attrition	Wear	ASME Section XI, 1989 Edition.

WB4-9

	(continued from previous page) ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWB-3140. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** No instances of internals degradation recorded have been definitely attributed to irradiation embrittlement.	
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*Same as for the effect of Wear on Item B4.2.1 CRA guide tubes.*	No
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*Same as for the effect of Stress Relaxation on Item B4.2.2 CRA guide tubes bolts.*	Yes Elements 3 and 4 should be

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.	*Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.*	Yes Element 4 should be further evaluated
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.	Plant specific aging management program is to be evaluated. The applicant must provide the basis for concluding that void swelling is not an issue for the component or must provide an AMP.	Yes TLAA
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*Same as for the effect of Neutron Irradiation Embrittlement on Items B4.2.1-B4.2.3 upper grid assembly rib section, bolts, and plenum rib pads.*	Yes Elements 3 and 4 should be further evaluated
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*Same as the effect of Wear on Item B4.2.1, CRA guide tubes*	No

IVB4-10

B4.5	Vent Valve Assembly	Vent Valve Assembly	Ni alloy, SS, CASS, Stellite, Martensitic SS	hemically reated orated ater up to 40°C	Crack initiation & growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B4.5	Vent Valve Assembly	Vent Valve Assembly	Ni alloy, SS, CASS, Stellite, Martensitic SS	hemically reated orated ater up to 40°C	Crack initiation & growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B4.5	Vent Valve Assembly	Vent Valve Assembly	Ni alloy, SS, CASS, Stellite, Martensitic SS	hemically reated orated ater up to 40°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.
B4.5	Vent Valve Assembly	Vent Valve Assembly	Ni alloy, SS,	hemically reated	Crack initiation &	Neutron Irradiation	ASME Section XI, 1989 Edition.

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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B4.5	Vent Valve Assembly	Vent Valve Assembly	CASS	hemically reated orated ater up to 40°C	Loss of Fracture Toughness	Thermal Aging and Neutron Irradiation Embrittle-ment	EPRI TR-106092. ASME Section XI, 1989 Edition.
B4.6.1 thru B4.6.3	Core Barrel Assembly	Core Barrel (CB) (Baffle & Former Plates), CB Bolts, CB to Thermal Shield Bolts	SS, Ni alloy (bolts)	hemically reated orated ater up to 40°C	Crack initiation & growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B4.6.1 thru B4.6.3	Core Barrel Assembly	Core Barrel (CB) (Baffle & Former Plates), CB Bolts, CB to Thermal Shield Bolts	SS, Ni alloy (bolts)	hemically reated orated ater up to 40°C	Crack initiation & growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B4.6.1 thru B4.6.3	Core Barrel Assembly	Core Barrel (CB) (Baffle & Former Plates), CB Bolts, CB to Thermal Shield Bolts	SS, Ni alloy (bolts)	hemically reated orated ater up to 40°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.
B4.6.1 thru B4.6.3	Core Barrel Assembly	Core Barrel (CB) (Baffle & Former Plates), CB Bolts, CB to Thermal Shield Bolts	SS, Ni alloy (bolts)	hemically reated orated ater up to 40°C	Loss of Fracture Toughness	Neutron Irradiation Embrittle-ment	ASME Section XI, 1989 Edition.

IV B4-11

IV B4-11

Same as for the effect of Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.	*Same as for the effect of Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.*	Yes Element 4 should be further evaluated
Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.	*Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.*	Yes Element 4 should be further evaluated
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the	Plant specific aging management program is to be evaluated. The applicant must provide the basis for concluding that void swelling is not an issue for the component or must provide an AMP.	Yes TLAA

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Same as for the effect of Thermal Aging and Neutron Embrittlement on Items B4.2.1- B1.2.3 CRA guide tubes, spacer casting, and spider casting.	*Same as for the effect of Thermal Aging and Neutron Embrittlement on Items B4.2.1- B1.2.3 CRA guide tubes, spacer casting, and spider casting.*	No
Same as for the effect of Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.	*Same as for the effect of Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.*	Yes Element 4 should be further evaluated
Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.	*Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.*	Yes Element 4 should be further evaluated
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.	Plant specific aging management program is to be evaluated. The applicant must provide the basis for concluding that void swelling is not an issue for the component or must provide an AMP.	Yes TLAA
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*Same as for the effect of Neutron Irradiation Embrittlement on Items B4.2.1-B4.2.3 upper grid assembly rib section, bolts, and plenum rib pads.*	Yes Elements 3 and 4 should be further evaluated

B4.6.1 thru B4.6.3	Core Barrel Assembly	Core Barrel (CB) (Baffle & Former Plates), CB Bolts, CB to Thermal Shield Bolts	SS, Ni alloy (bolts)	hemically reated orated ater up to 40°C	Cumulative fatigue damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B4.6.2, B4.6.3	Core Barrel Assembly	CB Bolts, CB to Thermal Shield Bolts	SS, Ni alloy	hemically reated orated ater up to 40°C	Loss of Preload	Stress Relaxation	ASME Section XI, 1989 Edition.

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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B4.7.1 thru B4.7.9	Lower Grid Assembly (LGA)	Upper Grid, Fuel Guide Pads, Lower Grid, Support Columns, Cylinder Guide Blocks, LGA Bolts, Core Barrel Bolts, Flow Distributor Bolts, Thermal Shield Bolts	SS , Ni Alloy (bolts)	hemically reated orated ater up to 40°C	Crack initiation & growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B4.7.1 thru B4.7.9	Lower Grid Assembly (LGA)	Upper Grid, Fuel Guide Pads, Lower Grid, Support Columns, Cylinder Guide Blocks, LGA Bolts, Core Barrel Bolts, Flow Distributor Bolts, Thermal Shield Bolts	SS , Ni Alloy (bolts)	hemically reated orated ater up to 40°C	Crack initiation & growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B4.7.1 thru B4.7.9	Lower Grid Assembly (LGA)	Upper Grid, Fuel Guide Pads, Lower Grid, Support Columns, Cylinder Guide Blocks, LGA Bolts, Core Barrel Bolts, Flow Distributor Bolts, Thermal Shield Bolts	SS , Ni Alloy (bolts)	hemically reated orated ater up to 40°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.

IV B4-12

Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Same as for the effect of Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.	*Same as for the effect of Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.*	Yes Element 4 should be further evaluated
Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.	*Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.*	Yes Element 4 should be further evaluated
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.	Plant specific aging management program is to be evaluated. The applicant must provide the basis for concluding that void swelling is not an issue for the component or must provide an AMP.	Yes TLAA

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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B4.7.1 thru B4.7.9	Lower Grid Assembly (LGA)	Upper Grid, Fuel Guide Pads, Lower Grid, Support Columns, Cylinder Guide Blocks, LGA Bolts, [redacted] Bolts, Thermal Shield Bolts	SS, Ni Alloy (bolts)	hemically reated orated ater up to 40°C	Loss of Fracture Toughness	Neutron Irradiation Embrittle-ment	ASME Section XI, 1989 Edition.
B4.7.6 thru B4.7.9	Lower Grid Assembly (LGA)	LGA Bolts, [redacted] Bolts, Thermal Shield Bolts	SS, Ni Alloy	hemically reated orated ater up to 40°C	Loss of Preload	Stress Relaxation	ASME Section XI, 1989 Edition.
B4.7.2, B4.7.5	Lower Grid Assembly (LGA)	Fuel Guide Pads, Cylinder Guide Blocks	SS	hemically reated orated ater up to 40°C	Attrition	Wear	ASME Section XI, 1989 Edition.
B4.7.1 thru B4.7.9	Lower Grid Assembly (LGA)	Upper Grid, Fuel Guide Pads, Lower Grid, Support Columns, Cylinder Guide Blocks, LGA Bolts, [redacted] Thermal Shield Bolts	SS, Ni Alloy (bolts)	hemically reated orated ater up to 40°C	Cumulative fatigue damage	Fatigue	Original Licensing Criteria or ASME Section III, 1989 Edition. GSI-190.
B4.8	Flow Distributor	-	SS	hemically reated orated ater up to 40°C	Crack initiation & growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B4.8	Flow Distributor	-	SS	hemically reated orated ater up to 40°C	Crack initiation & growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.

IV B4-12

IV B4-13

IV B4-14

IV B4-14

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*Same as for the effect of Neutron Irradiation Embrittlement on Items B4.2.1-B4.2.3 upper grid assembly rib section, bolts, and plenum rib pads.*	Yes Elements 3 and 4 should be further evaluated
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*Same as for the effect of Stress Relaxation on Item B4.2.2 CRA guide tubes bolts.*	Yes Elements 3 and 4 should be further evaluated
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*Same as the effect of Wear on Item B4.2.1, CRA guide tubes*	No
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of the original licensing criteria or ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG.	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed. IVB4-14	Yes TLAA
Same as for the effect of Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.	*Same as for the effect of Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.*	Yes Element 4 should be further evaluated
Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.	*Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.*	Yes Element 4 should be further evaluated

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
B4.8	Flow Distributor	-	SS	hemically reated orated ater up to 40°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.
B4.8	Flow Distributor		SS	hemically reated orated ater up to 40°C	Loss of Fracture Toughness	Neutron Irradiation Embrittle-ment	ASME Section XI, 1989 Edition.
B4.9	Thermal Shield	-	SS	hemically reated orated ater up to 40°C	Crack initiation & growth	SCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B4.9	Thermal Shield	-	SS	hemically reated orated ater up to 40°C	Crack initiation & growth	IASCC	ASME Section XI, 1989 Edition. EPRI TR-105714.
B4.9	Thermal Shield	-	SS	hemically reated orated ater up to 40°C	Changes in Dimensions	Void Swelling	EPRI TR-107521.
B4.9	Thermal Shield	-	SS	hemically reated orated ater up to 40°C	Loss of Fracture Toughness	Neutron Irradiation Embrittle-ment	ASME Section XI, 1989 Edition.

IV B4-14

IV B4-16

IV B4-15

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.	Plant specific aging management program is to be evaluated. The applicant must provide the basis for concluding that void swelling is not an issue for the component or must provide an AMP.	Yes TLAA
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*Same as for the effect of Neutron Irradiation Embrittlement on Items B4.2.1-B4.2.3 upper grid assembly rib section, bolts, and plenum rib pads.*	Yes Elements 3 and 4 should be further evaluated
Same as for the effect of Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.	*Same as for the effect of Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.*	Yes Element 4 should be further evaluated
Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.	*Same as for the effect of Irradiation Assisted Stress Corrosion Cracking on Item B4.1, plenum cover and plenum cylinder.*	Yes Element 4 should be further evaluated
Plant specific aging management program. Based on EPRI TR-107521, estimates of void swelling vary from 14% for baffle-former assemblies over a 40-y plant life to less than 3% for the most highly irradiated sections of the internals at 60 y.	Plant specific aging management program is to be evaluated. The applicant must provide the basis for concluding that void swelling is not an issue for the component or must provide an AMP.	Yes TLAA
Visual inspection (VT-3) is performed according to Category B-N-3 of Subsection IWB, ASME Section XI.	*Same as for the effect of Neutron Irradiation Embrittlement on Items B4.2.1-B4.2.3 upper grid assembly rib section, bolts, and plenum rib pads.*	Yes Elements 3 and 4 should be further evaluated

NOW 14

GALL SECTION	TITLE	ITEM NO	PAGE	REVIEWER	COMMENT
IV	Reactor Vessel (PWR)	All			General Comment Suggest that aging effects that apply to all RV items (e g., loss of external material by boric acid wastage and cracking by cumulative fatigue damage be discussed at the very beginning and not repeated thereafter). At present, CUF is listed for selected items and not for all RV items.
Reactor Vessel Internals—Pressurized Water Reactor Babcock & Wilcox Designed					
IV B4-1	Reactor Vessel Internals (PWR) Babcock and Wilcox	B.4.1	IV B4-4		Aging Mechanism—IASCC is not a credible damage mechanism for the plenum cylinder and plenum cover owing to low fluence levels AMP- In accordance with the SER for BAW-2248A, the NRC has accepted a two-pronged approach to manage cracking by SCC and/or IASCC. The approach includes inspection and/or to perform tests and analysis of irradiated material. The inspection approach would be a supplemental examination of the items believed to be the limiting components for cracking, considering both the susceptibility of the items to the aging mechanism as well as the material properties (fracture toughness) and the operating stresses. The test and analysis approach allows for the license renewal applicant to evaluate the results of the Materials Reliability Program (MRP), or any other industry research activity, to determine the scope of inspection during the period of extended operation. The approach described above should be part of the AMP in the GALL report suggest under the topic entitled ***"Detection of Aging Effects."***

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IV

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV B4-2	Reactor Vessel Internals (PWR) Babcock and Wilcox	B4.1	IV B4-7		Void swelling is not a TLAA as it does not meet the definition in 10 CFR 54.3. That is, B&W has not performed any calculations or analyses that address void swelling as part of the CLB. This issue must resolved by an aging management program through 10 CFR 54.21 (a)(3). Suggest deleting reference to TLAA.
IV B4-3	Reactor Vessel Internals (PWR) – Babcock and Wilcox	B4.1	IV B4-8		The B&W-designed reactor vessel internals were designed and constructed prior to the development of ASME Code requirements for core support structures. As such, the only specific fatigue analyses performed in the original design were those that addressed high cycle fatigue. The high-cycle fatigue time limited aging analysis was evaluated in BAW-2248A, Section 4.5.1. Reactor vessel internals items that were explicitly evaluated for low-cycle and high-cycle fatigue are limited to high strength replacement bolting (i.e., selected thermal shield and core barrel bolts). The NRC did not require that the B&WOG address GSI-190 in BAW-2248A. It is not clear why the staff has referenced GSI-190 for internals items in the GALL report. GSI-190 is closed and all references to it should be deleted from GALL. Suggest that all entries that list "Cumulative fatigue damage" be deleted and a single row be inserted at the beginning of the table be added to address fatigue. Would recommend that the fatigue entry in the GALL report reflect the treatment of fatigue in the NRC's SER of BAW-2248A, Section 3.4.1.
IV B4-4	Reactor Vessel Internals (PWR) – Babcock and Wilcox	B.4.2.1 through B4.2.4	IV B4-8		Region of Interest – Spider castings are not part of the CRA guide tube assemblies. The spider castings, which are fabricated from CASS, are part of the lower internals assembly. In addition, tubes should be replaced with "guide tubes" and "guide sectors."

GALL SECTION	TITLE	ITEM NO	PAGE	REVIEWER	COMMENT
					Aging Effect—Reduction of fracture toughness by thermal aging and neutron embrittlement is applicable to the spacer castings only. Remaining CRGT items are fabricated from wrought SS and are not susceptible to thermal embrittlement. Evaluation and Technical Basis—Please see RV item A2.2.1 for a discussion of a correction to the ferrite screening criteria for CASS items. Further Evaluation—SCC, IASCC, and void swelling entries do not meet the definition of TLAA. They may in the future but do not now. In addition, it is not clear what is meant by "Element 4 should be further evaluated" for SCC and IASCC entries.
IV B4-5	Reactor Vessel Internals (PWR) Babcock and Wilcox	B.4.2.1	IV B4-10		Region of Interest – Tubes should be replaced with guide tubes and guide sectors Aging Effect – Attrition is not applicable to the guide tubes and guide sectors as discussed in Section 3.3 of BAW-2248A AMP – The guide tubes and C-sectors are not easily examined by VT-3 (see BAW-2248, Figure 2-6, page 2-33). Therefore, B-N-3 should not be credited for aging management.
IV B4-6	Reactor Vessel Internals (PWR) – Babcock and Wilcox	B4.2.1 thru B4.2.4	IV B4-14		Aging Effect– Suggest that fatigue be addressed at the beginning of the table.
IV B4-7	Reactor Vessel Internals (PWR) – Babcock and Wilcox	B4.3.1 thru B4.3.3	IV B4-15		Further Evaluation—SCC, IASCC, void swelling, and reduction of fracture toughness are not TLAA in accordance with the CLB.
IV B4-8	Reactor Vessel Internals (PWR) – Babcock and Wilcox	B4.3.3	IV B4-16		Region of Interest – Change "plenum rib pads" to "fuel assembly support pads." No such thing as "plenum rib pads" for the B&W design.

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IV

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV B4-9	Reactor Vessel Internals (PWR) – Babcock and Wilcox	B4.4.1, B4.4.2, B4.4.3, and B4.5	IV B4-16 thru IV B4-18		Structure and Component—The core support assembly includes the following sub-assemblies: (1) core support shield assembly, (2) core barrel assembly, and (3) lower internals assembly. Therefore, suggest that items B4.4.1 and B4.4.2, and B4.5 be identified as the "Core Support Shield Assembly" and not "Core Support Assembly."
IV B4-10	Reactor Vessel Internals (PWR) – Babcock and Wilcox	B4.5	IV B4-18		Material—CASS--add precipitation-hardening stainless steel Aging Effect—Loss of Fracture Toughness Evaluation and Technical Basis—Please see RV item A2.2.1 for a discussion of a correction to the ferrite screening criteria for CASS items.
IV B4-11	Reactor Vessel Internals (PWR) Babcock and Wilcox	B4.6.1 thru B4.6.3	IV B4-18		Region of Interest—suggest CB Bolts be defined. That is, are these the high strength bolts or do they include the baffle-to-baffle and baffle-to-former bolts or all of the above. Baffle bolts are not listed as a separate item. Material—bolts are fabricated from stainless steel and Ni alloy (selected B&W plants use NiCr Alloy X-750 bolts) Further Evaluation—SCC, IASCC, IE, and void swelling are not TLAA.
IV B4-12	Reactor Vessel Internals (PWR) – Babcock and Wilcox	B4.7.1 thru B4.7.9	IV B4-20		Structure and Component—The lower internals assembly consists of three sub-assemblies: lower grid assembly, flow distributor assembly, and incore guide tube assembly. Items B4.7.1 thru B4.7.9 discuss the LGA. Item B.4.8 captures the flow distributor assembly. The incore guide tube assembly is not discussed at all. Region of Interest—The terminology used to identify the items is not consistent with the B&W design. Suggest the following—lower

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
					grid rib section, LG flow distributor plate, LG shell forging, fuel assembly support pads, guide blocks, shock pads, support post pipes, LGA bolts, LIA-to-thermal shield bolts
IV B4-13	Reactor Vessel Internals (PWR) Babcock and Wilcox	B4.7.6 thru B4.7.9	IV B4-22		Region of Interest— Entry should be LGA Bolts and LIA-to-thermal shield bolts. Delete core barrel bolts and flow distributor bolts
IV B4-14	Reactor Vessel Internals (PWR) – Babcock and Wilcox	B.4.8	IV B4-22		Structure and Component—Change Flow Distributor to Flow Distributor Assembly Region of Interest—flow distributor head, FD flange, FD bolts, incore guide support plate Further Evaluation—Mechanisms assessments are not TLAA
IV B4-15	Reactor Vessel Internals (PWR) – Babcock and Wilcox	NA	IV B4-24		Structure and Component—Have omitted the Incore Guide Tube Assembly Regions of Interest—Incore Guide Tubes, Gussets, Guide Tube Nuts, and Spiders Material—All are SS; spiders are CASS
IV B4-16	Reactor Vessel Internals (PWR) – Babcock and Wilcox	B4.9	IV B4-24		The thermal shield should be deleted her and added as a region of interest under the "Core Barrel Assembly" Items B4.6.1 thru B4.6.3
IV B4-17	Reactor Vessel Internals (PWR) – Babcock and Wilcox		IV B4-4, 8, 14, 16, 18, 20, and 22.		Replace "thru" with "through" or use dashes, (e.g., A1.1.1-A1.1.3).

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IV

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV B4-18	Reactor Vessel Internals (PWR) - Babcock and Wilcox	B4.1, B4.2.1 – B4.2.4, B4.3.1 – B4.3.3, B4.4.1 – B4.4.2, B4.5, B4.6.1 – B4.6.3, B4.7.1 – B4.7.9, B4.8, and B4.9.	IV B4-4, 5, 8, 14, 16, 18, 20, 22, and 24.		Replace "EPRI TR-105714" with " the latest version of EPRI TR-105714" or "EPRI PWR Primary Chemistry Guidelines, Revision 4, March 1999, EPRI TR-105714.

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

C2.1 Reactor Coolant System Piping & Fittings

- C2.1.1 Cold–Leg
- C2.1.2 Hot–Leg
- C2.1.3 Surge Line
- C2.1.4 Spray Line
- C2.1.5 Small-Bore RCS Piping

C2.2 Connected Systems Piping & Fittings

- C2.2.1 Residual Heat Removal (RHR) or Low–Pressure Injection System (Decay Heat Removal (DHR)/ Shutdown System)
- C2.2.2 Core Flood System (CFS)
- C2.2.3 Chemical and Volume Control System or High-Pressure Injection System (Makeup & Letdown Functions)
- C2.2.4 Sampling System
- C2.2.5 Drains and Instrument Lines
- C2.2.6 Nozzles and Safe Ends
- C2.2.7 Small-Bore Piping in Connected Systems

C2.3 Reactor Coolant Pump

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

C2.4 Safety & Relief Valves

C2.5 Pressurizer

- C2.5.1 Shell/Heads
- C2.5.2 Spray Line Nozzle
- C2.5.3 Surge Line Nozzle
- C2.5.4 Spray Head
- C2.5.5 Thermal Sleeves
- C2.5.6 Instrument Nozzle
- C2.5.7 Safe Ends
- C2.5.8 Manway and Flanges
- C2.5.9 Manway and Flange Bolting
- C2.5.10 Heater Sheaths and Sleeves
- C2.5.11 Support Keys, Skirt, & Shear Lugs
- C2.5.12 Integral Support

C2.6 Pressurizer Relief Tank

- C2.6.1 Tank Shell and Heads
- C2.6.2 Flanges and Nozzles

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

System, Structures, and Components

IVC2-2

The system, structures, and components included in this table comprise the pressurized water reactor (PWR) primary coolant pressure boundary and consist of the reactor coolant system, pressurizer, pressurizer relief tank, and portions of other connected systems extending up to the first isolation valve outside of containment or to the first anchor point. The connected systems include residual heat removal (RHR) or low–pressure injection system, core flood spray (CFS) or safety injection tank (SIT), chemical and volume control system (CVCS) or high–pressure injection system, and sampling system. With the exception of pressurizer relief tank, which is classified as Group B Quality Standards, all systems, structures, and components in the reactor coolant system are classified as Group A Quality Standards. The recirculating pump seal water heat exchanger is discussed in Table V D1 "Emergency Core Cooling System." The aging management program for the variety of lines that penetrate the containment and associated isolation valves is reviewed in Table V C "Containment Isolation Components."

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

System Interfaces

IVC2-3

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

IVC2-4

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
C2.1.1, C2.1.2	Reactor Coolant System (RCS) Piping & Fittings	Cold-Leg, Hot-Leg	Stainless Steel (SS), Cast Austenitic SS (CASS), Carbon Steel (CS) with SS Cladding	hemically reated orated ater up to 40°C	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. ANSI B31.1. GSI-190.
C2.1.3, C2.1.4	Reactor Coolant System (RCS) Piping & Fittings	Surge Line, Spray Line	Surge Line: SS, CASS; Spray Line: SS	hemically reated orated ater up to 40°C	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. ANSI B31.1. GSI-190.
C2.1.1 thru C2.1.4	Reactor Coolant System (RCS) Piping & Fittings	Cold-Leg, Hot-Leg, Surge Line, Spray Line	SS, CS with SS Cladding	hemically reated orated ater up to 40°C	Crack Initiation and Growth	Stress Corrosion Cracking (SCC), Inter-granular Stress Corrosion Cracking (IGSCC)	ASME Section XI, 1989 Edition. Reg. Guide 1.43. Reg. Guide 1.44. Reg. Guide 1.45. EPRI TR-105714. Plant Technical Specifications. *Operating Experience* NRC IN 80-38. NRC IN 84-18. NRC IN 84-89. NRC IN 91-05. NRC IN 94-63. NRC IN 97-19.

IVC2-5

IVC2-7 WHOLE SECTION FOR SS

IVC2-6 WHOLE SECTION

IVC2-1

IVC2-9 & IVC2-11 & IVC2-12

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or ANSI B31.1, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or ANSI B31.1, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

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

Item	Structure and Component	Region of Interest	Material	Environment	Aging Effect	Aging Mechanism	References
Guidelines of Regulatory Guide (RG) 1.44 to avoid sensitization of stainless steels (SSs) and guidelines of RG 1.43 to prevent underclad cracking and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a, Subsection IWB, Table IWB 2500-1, examination category B-J for pressure retaining welds in piping and testing category B-P for system leakage. Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 to minimize the potential of crack initiation and growth.			***(1) Scope of Program:*** The program includes preventive measures to mitigate SCC of SS components and cladding, and inservice inspection (ISI) to monitor the effects of SCC on the intended function of piping and fittings of reactor coolant system and connected lines. ***(2) Preventive Actions:*** Selection of material in compliance with the guidelines of Regulatory Guide (RG) 1.44 prevent or mitigate SCC, and RG 1.43 provides assurance that production cladding complies with ASME Section III and XI requirements to prevent underclad cracking. Also, control of halogens, sulfates, and oxygen in the primary water to less than 0.05, 0.05, and 0.005 ppm, respectively, during operation, and monitor and control of water chemistry during shut down, mitigate the potential of SCC. However, inadvertent introduction of contaminants into the coolant system can occur, e.g., introduced through the free surface of spent fuel pool which can be a natural collector of airborne contaminants or the introduction of oxygen during cool down. Also, cladding regions with low delta ferrite may be sensitized during post-weld heat treatment and thus become susceptible to SCC. The AMP must therefore rely upon ISI in accordance with ASME Section XI to detect possible degradation. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of IGSCC on the intended function of piping and fittings by detection and sizing of cracks by ISI. Inspection requirements of Table IWB 2500-1, examination category B-J specifies for circumferential and longitudinal welds in each pipe or branch run NPS 4 or larger, volumetric and surface examination of ID region extending 1/4 in. on either side of the weld and 1/3 wall thickness deep, and surface examination of OD surface extending 1/2 in. on either side. Surface examination is conducted for circumferential and longitudinal welds in each pipe or branch run less than NPS 4. For socket welds, surface examination is specified of OD surface extending 1 in. on the buttered side and 1/2 in. on the other. Requirements for training and qualification of personnel and performance demonstration for procedures and equipment are in conformance with Appendices VII and VIII of ASME Section XI, or any other formal program approved by the NRC. Inspection requirements of testing category B-P					No

NC2-B

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
C2. REACTOR COOLANT SYSTEM AND CONNECTED LINES (Pressurized Water Reactor)

Existing Aging Management Program (AMP)			Evaluation and Technical Basis					Further Evaluation
C2.1.1, C2.1.2	Reactor Coolant System (RCS) Piping & Fittings	Cold-Leg, Hot-Leg (External Surfaces)	CS with SS Cladding	ir, eaking hemically reated orated ater	Loss of Material	Corrosion/ Boric Acid Wastage of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 Edition. Plant Technical Specifications.	

BALANCE OF SECTION

IVC2-10

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
			(continued from previous page) conducted according to IWA–5000 specify visual VT-2 (IWA-5240) examination of all pressure retaining components extending to and including the second closed valve at the boundary extremity, during system leakage test (IWB-5221) and system hydrostatic test (IWB-5222). Leakage detection is in conformance with Position C of Regulatory Guide 1.45. ***(4) Detection of Aging Effects:*** Degradation of the piping due to SCC can not occur without crack initiation and growth; extent and schedule of inspection assure detection of cracks before the loss of intended function of piping and fittings of reactor coolant system and connected lines. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 should provides timely detection of cracks. System leakage test is conducted prior to plant startup following each refueling outage, and hydrostatic test at or near the end of each inspection interval. ***(6) Acceptance Criteria:*** Any IGSCC degradation is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3514. Planar and liner flaws are sized according to IWA-3300 and IWA-3400. ***(7) Corrective Actions:*** Repair is in conformance with IWA-4000 and IWB-4000, replacement according to IWA-7000 and IWB-7000, and reexamination in accordance with IWA-2200. Continued operation without repair requires that crack growth calculation be performed according to the guidance of ASME Section XI or other approved procedures. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Although the primary pressure boundary piping of PWRs have generally not been found to be affected by SCC because of low dissolved oxygen levels and control of primary water chemistry, potential of SCC exists due to inadvertent introduction of contaminants into the primary coolant system from (a) unacceptable levels of contaminants in the boric acid, and (b) free surface of the spent fuel pool (IN 84–18). Also, SCC could occur at creviced or cold worked locations (IN 84-89). SCC has been observed in safety injection lines (IN 97-19 and 84-18), charging pump casing cladding (INs 80-38 and 94-63), and instrument nozzles in safety injection tanks (IN 91-05).		
Implementation of NRC Generic Letter 88-05, and based on plant technical specifications, inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a),			***(1) Scope of Program:*** The staff guidance of Generic Letter (GL) 88-05 provides assurances that a program has been implemented consisting of systematic measures to ensure that the effects of corrosion caused by leaking coolant containing boric acid does not lead to degradation				No

BALANCE OF SECTION

IVC2-10

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

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

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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
			(continued from previous page) (b) examinations requirements and procedures for locating small leaks, and (c) engineering evaluations and corrective actions. ***(2) Preventive Actions:*** Minimizing reactor coolant leakage by frequent monitoring of the locations where potential leakage could occur and timely repair, prevent or mitigate boric acid corrosion. Modifications in the design or operating procedures to reduce the probability of leaks at locations where they may cause corrosion damage and use of suitable corrosion resistant materials or the application of protective coatings. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of boric acid corrosion on the intended function of the component by detection of coolant leakage by implementing the program delineated in GL 88-05 including guidelines for locating small leaks, conducting examinations, and performing engineering evaluations. Inspection requirements of ASME Section XI specify visual VT-2 (IWA-5240) examination during system leakage test (IWB-5221) and system hydrostatic test (IWB-5222) of all pressure retaining Class 1 components, according to Table IWB 2500-1 category B-P and pressure retaining Class 2 components according to Table IWC 2500-1 category C-H. ***(4) Detection of Aging Effects:*** Degradation of the component due to boric acid wastage can not occur without leakage of coolant containing boric acid; extent and schedule of the program delineated in GL 88-05 will assure detection of leakage before the loss of intended function of the component. ***(5) Monitoring and Trending:*** The program delineated in GL 88-05 provides for timely detection of leakage. ***(6) Acceptance Criteria:*** Any leakage is evaluated in accordance with the program proposed by GL 88-05. ***(7) Corrective Actions:*** The leakage source and areas of general corrosion are located and corrective actions are in conformance with the program proposed by GL 88-05, and the requirements of ASME Section XI, IWB-3142 and IWA-5250. Components with local areas of corrosion that reduces the wall thickness by more than 10% require analytical evaluation to demonstrate acceptability. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Boric acid wastage observed in nuclear power plants may be classified into two types: (a) corrosion that increases the rate of leakage, e.g., corrosion of closure bolting or fasteners, and (b) corrosion that occurs some distance from the source of leakage. Some recent incidents of boric acid wastage (IN					

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
C2. REACTOR COOLANT SYSTEM AND CONNECTED LINES (Pressurized Water Reactor)

Existing Aging Management Program (AMP)			Evaluation and Technical Basis				Further Evaluation
C2.1.1 thru C2.1.3	Reactor Coolant System (RCS) Piping & Fittings	Cold-Leg, Hot-Leg, Surge Line	CASS	hemically reated orated ater up to 40°C	Crack Initiation and Growth	SCC, IGSCC	EPRI TR-105714. NUREG-0313, Rev. 2.
C2.1.1 thru C2.1.3	Reactor Coolant System (RCS) Piping & Fittings	Cold-Leg, Hot-Leg, Surge Line	CASS	hemically reated orated ater up to 40°C	~~Loss of~~ Fracture Toughness	Thermal Aging Embrittle-ment	EPRI TR-106092. ASME Section XI, 1989 Edition.

REDUCED

IVC2-13

IVC2-15

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
Monitoring and control of primary water chemistry in accordance with the EPRI guidelines in TR-105714 minimize the potential of SCC, and material selection according to the NUREG-0313, Rev. 2 guidelines of ≤0.035% C and ≥7.5% ferrite has reduced susceptibility to SCC. For CASS components that do not meet either one of the above guidelines, define plant-specific aging management program.				Plant-specific aging management program is to be evaluated for CASS components that do not meet either the water chemistry guidelines of EPRI TR-105714, or the C and ferrite content criteria of NUREG-0313, Rev. 2. The program will include (a) adequate inspection methods to ensure detection of cracks and (b) flaw evaluation methodology for CASS components that are susceptible to thermal aging embrittlement.			Yes, no AMP
Determination of the susceptibility of CASS piping to thermal aging embrittlement based on casting method, Mo content, and percent ferrite. For "potentially susceptible" piping, aging management is accomplished either through enhanced volumetric examination or plant/component-specific flaw tolerance evaluation. Additional inspection or evaluations to demonstrate that the material has adequate fracture toughness are not required for "not susceptible" piping. Because of low applied stresses for pump casings and valve bodies, screening for susceptibility to thermal aging. Also, the existing ASME Section XI inspection requirements, including the alternative requirements of ASME Code Case N-481 for pump casings, are considered adequate for all pump casings and valve bodies.				***(1) Scope of Program:*** The program includes determination of the susceptibility of CASS components to thermal aging based on casting method, Mo content, and percent ferrite, and for potentially susceptible components aging management is accomplished either through volumetric examination or plant/component-specific flaw tolerance evaluation. ***(2) Preventive Actions:*** The program provides no guidance on methods to mitigate thermal aging. ***(3) Parameters Monitored/ Inspected:*** Based on the criteria in EPRI TR-106092, with some modifications, the susceptibility to thermal aging embrittlement of CASS piping is determined in terms of casting method, Mo content, and ferrite content. For low-Mo content (0.5 wt.% max.) steels, only static-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement, all centrifugal-cast and static-cast steels with ≤20% ferrite are not susceptible. For high-Mo content (2.0 to 3.0 wt.%) steels, static-cast steels with >14% ferrite and centrifugal-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement, static-cast steels with ≤14% ferrite and centrifugal-cast steels with ≤20% ferrite are not susceptible. Ferrite content will be calculated by the Hull's equivalent factors or a method producing an equivalent level of accuracy (±6% deviation between measured and calculated values.) For pump casings and valve bodies, screening for susceptibility to thermal aging is not required. ***(4) Detection of Aging Effects:*** For "not susceptible" piping, no additional inspection or evaluations to demonstrate that the material has adequate fracture toughness are required. For "potentially susceptible" piping, because the base metal does not receive periodic inspection per ASME Section XI, volumetric examination should be performed on the base metal, with the scope of the inspection covering the portions determined to be limiting from the standpoint of applied stress, operating time, and environmental considerations. Alternatively, a plant/component- specific flaw tolerance evaluation, using specific geometry and stress information, can be used to demonstrate that the thermally-embrittled			No

Existing Aging Management Program (AMP)				Evaluation and Technical Basis			Further Evaluation
C2.1.5	Reactor Coolant System (RCS) Piping & Fittings	Small-Bore RCS Piping	SS, CS	hemically reated orated ater up to 40°C	Crack Initiation and Growth	Unantic pated Thermal and Mechani-cal Loading	ASME Section XI, 1989 Edition. EPRI TR-105714. Plant Technical Specifications. NRC IN 97-46.

IVC2-16

IVC2-17

IVC2-18

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
			(continued from previous page) requirements, including the alternative requirements of ASME Code Case N-481 for pump casings, are considered adequate. For valve bodies less than NPS 4, the adequacy of inservice inspection according to ASME Section XI has been demonstrated by a NRC performed bounding fracture analysis. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 should provide timely detection of cracks. ***(6) Acceptance Criteria:*** Flaws detected in CASS components are evaluated in accordance with the applicable procedures of IWB-3500. If aging management is accomplished through plant/component-specific flaw tolerance evaluation, e.g., for potentially susceptible piping, flaw evaluation for piping with <25% ferrite is performed according to the principles associated with IWB-3640 procedures for submerged arc welds (SAW), disregarding the Code restriction of 20% ferrite in IWB-3641(b)(1). Flaw evaluation for piping with >25% ferrite is performed on a case-by-case basis using fracture toughness data provided by the applicant. ***(7) Corrective Actions:*** Repair is in conformance with IWA-4000 and IWB-4000, and replacement according to IWA-7000 and IWB-7000. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** The AMP based on susceptibility determination and examination requirements or flaw tolerance evaluations of potentially susceptible components is effective in managing the effects of thermal aging on the intended function of CASS components.		
Inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination category B-J for pressure retaining welds in piping and testing category B-P for system leakage, and primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 to minimize the potential of crack initiation and growth.			***(1) Scope of Program:*** The program includes preventive measures to inhibit cracking and inservice inspection (ISI) to monitor the effects of cracking on the intended function of small-bore piping of reactor coolant system and connected lines. ***(2) Preventive Actions:*** Control of halogens, sulfates, and oxygen in the primary water to less than 0.05, 0.05, and 0.005 ppm, respectively, during operation, and monitor and control of water chemistry during shut down, mitigate potential of cracking. However, inadvertent introduction of contaminants into the coolant system can occur, and impurities can buildup in stagnant regions and crevices. The AMP must therefore rely upon ISI in accordance with ASME Section XI to detect possible degradation. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of cracking on the intended function of piping and fittings by detection cracks and leakage by ISI. Inspection requirements of Table IWB 2500-1, examination category			Yes Elements 3 and 4 should be further evaluated

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
C2. REACTOR COOLANT SYSTEM AND CONNECTED LINES (Pressurized Water Reactor)

Existing Aging Management Program (AMP)			Evaluation and Technical Basis				Further Evaluation
C2.2.1 thru C2.2.3	Connected Systems Piping & Fittings	Residual Heat Removal (RHR), Core Flood System (CFS), Chemical & Volume Control System(CVCS)	SS	hemically reated orated ater up to 40°C	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. ANSI B31.1. GSI-190.
C2.2.4, C2.2.5	Connected Systems Piping & Fittings	Sampling, Drains & In-strumentation	CS, SS	hemically reated orated ater up to 40°C	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. ANSI B31.1. GSI-190.

IVC2-19

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
			(continued from previous page) does not require volumetric examination of pipes less than NPS 4. A plant-specific destructive examination or a nondestructive examination (NDE) that permits inspection of the inside surfaces of the piping should be conducted to ensure that cracking has not occurred and the component intended function will be maintained during the extended period. ***(4) Detection of Aging Effects:*** Degradation of the piping due to cracking would result in leakage of coolant. A one-time inspection of a sample of locations most susceptible to cracking should be conducted to verify that service-induced weld cracking is not occurring in the small-bore piping less than NPS 4, including pipe, fittings, and branch connections. Actual inspection locations should be based on risk-informed approaches and physical accessibility, exposure levels, and NDE examinations techniques, and locations identified in NRC Information Notice (IN) 97-46. ***(5) Monitoring and Trending:*** System leakage test is conducted prior to plant startup following each refueling outage, and hydrostatic test at or near the end of each inspection interval. The results of one-time inspection will be used to dictate the frequency of future inspections. ***(6) Acceptance Criteria:*** Any relevant conditions that may be detected during the leakage tests are evaluated in accordance with IWC-3516. ***(7) Corrective Actions:*** Repair is in conformance with IWA-4000 and IWB-4000, replacement according to IWA-7000 and IWB-7000. If destructive examination is employed, repair and replacement are in accordance with ASME Section XI rules. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Cracking has occurred in unisolable section of combined makeup and high-pressure injection lines.				
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or ANSI B31.1, or other evaluations based on cumulative usage factor (CUF).			Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.				Yes TLAA
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in			Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.				Yes TLAA

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
C2. REACTOR COOLANT SYSTEM AND CONNECTED LINES (Pressurized Water Reactor)

Existing Aging Management Program (AMP)			Evaluation and Technical Basis				Further Evaluation
C2.2.6	Connected Systems Piping & Fittings	Nozzles & Safe Ends	SS, CASS	hemically reated orated ater up to 40°C	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. ANSI B31.1. GSI-190.
C2.2.4, C2.2.5	Connected Systems Piping & Fittings	Sampling, Drains & In-strumentation (External Surfaces)	CS	ir, eaking hemically reated orated ater	Loss of Material	Corrosion/ Boric Acid Wastage of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 Edition. Plant Technical Specifications.
C2.2.6	Connected Systems Piping & Fittings	Nozzles & Safe Ends	CASS	hemically reated orated ater up to 40°C	~~Loss of~~ Fracture Toughness REDUCED	Thermal Aging Embrittle-ment	EPRI TR-106092. ASME Section XI, 1989 Edition.
C2.2.1 thru C2.2.6	Connected Systems Piping & Fittings	RHR, CFS, CVCS, Sampling, Drains & In-strumentation, Nozzles & Safe Ends	SS	hemically reated orated ater up to 40°C	Crack Initiation and Growth	SCC	*Same as for the effect of Stress Corrosion Cracking on Items C2.1.1 thru C2.1.4 cold-leg, hot-leg, surge line, and spray line piping and fittings constructed of SS*
C2.2.6	Connected Systems Piping & Fittings	Nozzles & Safe Ends	CASS	hemically reated orated ater up to 40°C	Crack Initiation and Growth	SCC	EPRI TR-105714. NUREG-0313, Rev. 2.
C2.2.7	Connected Systems Piping & Fittings	Small-Bore Piping in Connected Systems	SS, CS	hemically reated orated ater up to 40°C	Crack Initiation and Growth	SCC, IGSCC	ASME Section XI, 1989 Edition. EPRI TR-105714. Plant Technical Specifications. NRC IN 97-46.
C2.3.1, C2.3.2	Reactor Coolant Pump	Bowl/Casings, Cover	Bowl: CASS SA351 CF-8 or CF-8M; Cover: SS	hemically reated orated ater up to 40°C	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. GSI-190.

IVC2-19

IVC2-20

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
C2. REACTOR COOLANT SYSTEM AND CONNECTED LINES (Pressurized Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or ANSI B31.1, or other evaluations based on cumulative usage factor (CUF).			Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.				Yes TLAA
Same as for the effect of corrosion/boric acid wastage of external surfaces of Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.			*Same as for the effect of corrosion/boric acid wastage of external surfaces of Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.*				No
Same as for the effect of Thermal Aging Embrittlement on Items C2.1.1 thru C2.1.3 Cold-Leg, Hot-Leg, and Surge Line Piping and Fittings.			*Same as for the effect of Thermal Aging Embrittlement on Items C2.1.1 thru C2.1.3 Cold-Leg, Hot-Leg, and Surge Line Piping and Fittings.*				No
Same as for the effect of Stress Corrosion Cracking on Items C2.1.1 thru C2.1.4 cold-leg, hot-leg, surge line, and spray line piping and fittings constructed of SS.			*Same as for the effect of Stress Corrosion Cracking on Items C2.1.1 thru C2.1.4 cold-leg, hot-leg, surge line, and spray line piping and fittings constructed of SS.*				No
Monitoring and control of primary water chemistry in accordance with the EPRI guidelines in TR-105714 minimize the potential of SCC, and material selection according to the NUREG-0313, Rev. 2 guidelines of ≤0.035% C and ≥7.5% ferrite has reduced susceptibility to SCC. For CASS components that do not meet either one of the above guidelines, define plant-specific aging management program.			Plant-specific aging management program is to be evaluated for CASS components that do not meet either the water chemistry guidelines of EPRI TR-105714, or the C and ferrite content criteria of NUREG-0313, Rev. 2. The program will include (a) adequate inspection methods to ensure detection of cracks and (b) flaw evaluation methodology for CASS components that are susceptible to thermal aging embrittlement.				Yes, no AMP
Same as for the effect of Stress Corrosion Cracking on Item C2.1.7 small-bore RCS piping. IVC2-21			*Same as for the effect of Stress Corrosion Cracking on Item C2.1.7 small-bore RCS piping.*				Yes, Elements 3 and 4 should be further evaluated
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or other evaluations based on cumulative usage factor (CUF).			Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.				Yes TLAA

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

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

Existing Aging Management Program (AMP)			Evaluation and Technical Basis					Further Evaluation
C2.3.1	Reactor Coolant Pump	Bowl/Casings	CASS SA351 CF-8 or CF-8M	hemically reated orated ater up to 40°C	Crack Initiation and Growth	SCC	EPRI TR-105714. NUREG-0313, Rev. 2. ASME Section XI, 1989 Edition. *Operating Experience* NRC IN 84-18.	

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
Monitoring and control of primary water chemistry in accordance with the EPRI guidelines in TR-105714 minimize the potential of SCC, and material selection according to the NUREG-0313, Rev. 2 guidelines of ≤0.035% C and ≥7.5% ferrite have reduced susceptibility to SCC. For CASS components that do not meet either one of the above guidelines, inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination categories B-L-1 for pump casing welds and B-L-2 for pump casings, and testing category B-P for system leakage.							
			(1) Scope of Program: Control of primary water chemistry in accordance with EPRI TR-105714 guidelines and material selection according to the guidance in NUREG-0313, Rev. 2 mitigate the potential of SCC in CASS components. For components that do not meet either one of the above guidelines, inservice inspection (ISI) to monitor the effects of SCC on the intended function of the pump. ***(2) Preventive Actions:*** Control of primary water chemistry and selection of appropriate materials prevents or mitigates SCC. However, inadvertent introduction of contaminants into the coolant system can occur through the free surface of spent fuel pool or the introduction of oxygen during cool down. The AMP must therefore rely upon ISI in accordance with ASME Section XI to detect possible degradation. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of SCC on the intended function of the pump by detection and sizing of cracks by ISI. Inspection requirements of Table IWB 2500-1, examination category B-L-1 specify volumetric examination of all welds extending 1/2 in. on either side of the weld and through wall thickness, and category B-L-2 specifies visual VT-3 examination of internal surfaces of the pump. Requirements for training and qualification of personnel and performance demonstration for procedures and equipment are in conformance with Appendices VII and VIII of ASME Section XI, or any other formal program approved by the NRC. Inspection requirements of category B-P specify visual VT-2 (IWA-5240) examination of all pressure retaining boundary of the pump during system leakage test (IWB-5221) and system hydrostatic test (IWB-5222). ***(4) Detection of Aging Effects:*** Degradation of the pump due to SCC can not occur without crack initiation and growth; extent and schedule of inspection assures detection of cracks before the loss of the intended function of the component. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 should provide timely detection of cracks. All welds are inspected each inspection period from at least one pump in each group performing similar functions in the system. Visual examination is required only when the pump is disassembled for maintenance, repair, or volumetric examination, but at least once during the period. System leakage test is conducted prior to plant startup following each refueling outage, and hydrostatic test at or near the end of each inspection interval. ***(6) Acceptance Criteria:*** Any SCC degradation is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400; IWB-3518 for volumetric examination of welds and 3519 for visual examination of pump internal surfaces. Planar and liner flaws are sized according to IWA-3300 and IWA-3400. Supplementary					No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
C2. REACTOR COOLANT SYSTEM AND CONNECTED LINES (Pressurized Water Reactor)

Existing Aging Management Program (AMP)			Evaluation and Technical Basis				Further Evaluation
C2.3.1	Reactor Coolant Pump	Bowl/Casings	CASS SA351 CF-8 or CF-8M	hemically reated orated ater up to 40°C	~~Loss of~~ Fracture Toughness REDUCED	Thermal Aging Embrittle-ment	EPRI TR-106092. ASME Section XI, 1989 Edition.
C2.3.3	Reactor Coolant Pump	Closure Bolting	High Strength Low-Alloy Steel (HSLAS) SA540 GrB23, SA193 GrB7	ir, Leaking hemically reated orated ater nd/or team up to 40°C	Cumulative Fatigue Damage INC2-13	Fatigue	ASME Section III, 1989 Edition. GSI-190.
C2.3.3	Reactor Coolant Pump	Closure Bolting	HSLAS SA540 GrB23, SA193 GrB7	ir, Leaking hemically reated orated ater or team up to 40°C	Crack Initiation and Growth	SCC INC2-22	NUREG-1339 EPRI NP-5769 NRC GL 91-17 ASME Section XI, 1989 Edition. *Operating Experience* IEB 82-02

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
			(continued from previous page) ***Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Although the primary pressure boundary piping of PWRs have generally not been found to be affected by SCC because of low dissolved oxygen levels, potential of SCC exists from inadvertent introduction of contaminants into the primary coolant system (IN 84–18).				
Same as for the effect of Thermal Aging Embrittlement on Items C2.1.1 thru C2.1.3 Cold-Leg, Hot-Leg, and Surge Line Piping and Fittings.			*Same as for the effect of Thermal Aging Embrittlement on Items C2.1.1 thru C2.1.3 Cold-Leg, Hot-Leg, and Surge Line Piping and Fittings.*				No
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or other evaluations based on cumulative usage factor (CUF).			Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.				Yes TLAA
Recommendations for a comprehensive bolting integrity program delineated in NUREG–1339 on resolution of Generic Safety Issue 29 and implemented through NRC Generic Letter 91-17; additional details on bolting integrity outlined in EPRI NP-5769; and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination categories B-G–1 or B-G-2 for pressure retaining bolting, and testing category B-P for system leakage.			***(1) Scope of Program:*** The staff guidance of NRC Generic Letter (GL) 91-17 provides assurance that plant specific comprehensive bolting integrity programs have been implemented to ensure bolting reliability. The NRC staff recommendations and guidelines for a comprehensive bolting integrity program are delineated in NUREG-1339, and the industry's technical basis for the program is outlined in EPRI NP-5769. ***(2) Preventive Actions:*** Selection of bolting material and the use of lubricants and sealants in accordance with guidelines of EPRI NP-5769 and additional requirements of NUREG 1339, prevent or mitigate degradation and failure of all safety-related closure bolting. ***(3) Parameter Monitored/ Inspected:*** The AMP monitors the effects of aging degradation on the intended function of closure bolting by detection of coolant leakage, and by detection and sizing of cracks by inservice inspection (ISI). Inspection requirements of ASME Section XI, Table IWB 2500-1, examination category B-G-1 for bolting greater than 2 in. in diameter specify volumetric examination of studs and bolts from top of the nut to bottom of the flange hole, and visual VT-1 examination of surfaces of nuts, washers, bushings, and flanges including an 1 in. annular surface of flange surrounding each stud. Examination category B-G-2 for bolting 2 in. or smaller specifies only visual VT-1 examination of surfaces of bolts, studs, and nuts.				No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

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

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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
			(continued from previous page) connection is disassembled, or when bolting is removed. Requirements for training and qualification of personnel and performance demonstration for procedures and equipment are in conformance with Appendices VII and VIII of ASME Section XI, and additional requirements of EPRI NP-5769. Inspection requirements of ASME Section XI testing category B-P specify visual VT-2 (IWA-5240) examination of all pressure retaining components extending to and including the second closed valve at the boundary extremity, during system leakage test (IWB-5221) and system hydrostatic test (IWB-5222). ***(4) Detection of Aging Effects:*** Degradation of the closure bolting due to crack initiation, loss of prestress, or attrition of the closure bolting would result in leakage. The extent and schedule of inspection assure detection of aging degradation before the loss of intended function of closure bolting. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI is effective and adequate for timely detection of cracks and leakage. Inspection schedule in accordance with IWB-2400 requires inspection of all bolts, studs, nuts, bushing, and flange surfaces every 10 y. System leakage test is conducted prior to plant startup following each refueling outage, and hydrostatic test at or near the end of each inspection interval. ***(6) Acceptance Criteria:*** Any cracks in closure bolting are evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3515 and 3517. Any relevant conditions that may be detected during the leakage and hydrostatic tests are compared with the acceptance standards of IWB-3522. ***(7) Corrective Actions:*** Repair and replacement is according to IWB-4000 and IWB-7000, respectively, and recommendations and guidelines of EPRI NP-5769. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Significant number of incidents have been reported on bolts and nuts that have failed or become degraded because of boric acid wastage or SCC. Examples of affected fasteners include (a) steam generator and pressurizer manway closures, (b) valve bonnets and pump flange connections on lines 6 in. or greater, and (c) control rod drive and pressurizer heater connections. The bolting integrity programs developed and implemented in accordance with commitments made in response to NRC communications on bolting events have provided effective means of ensuring bolting reliability.						

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
C2. REACTOR COOLANT SYSTEM AND CONNECTED LINES (Pressurized Water Reactor)

Existing Aging Management Program (AMP)			Evaluation and Technical Basis					Further Evaluation
C2.3.3	Reactor Coolant Pump	Closure Bolting	High Strength Low-Alloy Steel (HSLAS) SA540 GrB23, SA193 GrB7	ir, Leaking hemically reated orated ater or team up to 40°C	Loss of Material	Corrosion/ Boric Acid Wastage of External Surfaces		NUREG-1339. EPRI NP-5769. NRC GL 91-17. NRC GL 88-05. ASME Section XI, 1989 Edition.
C2.3.3	Reactor Coolant Pump	Closure Bolting	HSLAS SA540 GrB23, SA193 GrB7	ir, Leaking hemically reated orated ater nd/or team up to 40°C	Loss of Prestress	Stress Relaxation	IVC2-24	NUREG-1339. EPRI NP-5769. NRC GL 91-17. ASME Section XI, 1989 Edition.
C2.3.3	Reactor Coolant Pump	Closure Bolting	HSLAS SA540 GrB23, SA193 GrB7	ir, Leaking hemically reated orated ater nd/or team up to 40°C	Attrition	Wear	IVC2-25	NUREG-1339. EPRI NP-5769. NRC GL 91-17. ASME Section XI, 1989 Edition.
C2.4.1, C2.4.2	Valves (Check, Control, Hand, Motor-Operated, and Relief Valves)	Bodies, Bonnet	CASS SA351 CF-8M, SA182 F316, SA582 Type 416	hemically reated orated ater up to 40°C	Cumulative Fatigue Damage	Fatigue	IVC2-26 IVC2-27	ASME Section III, 1989 Edition. GSI-190.
C2.4.1	Valves (Check, Control, Hand, Motor-Operated, and Relief Valves)	Bodies	CASS SA351 CF-8M	hemically reated orated ater up to 40°C	Crack Initiation and Growth	SCC		EPRI TR-105714. NUREG-0313, Rev. 2. ASME Section XI, 1989 Edition. *Operating Experience* NRC IN 84-18.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
C2. REACTOR COOLANT SYSTEM AND CONNECTED LINES (Pressurized Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
Recommendations for a comprehensive bolting integrity program delineated in NUREG–1339 on resolution of Generic Safety Issue 29 and implemented through NRC Generic Letter 91-17; additional details on bolting integrity outlined in EPRI NP-5769; implementation of NRC Generic Letter 88-05; and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination categories B-G–1 or B-G-2 for pressure retaining bolting, and testing category B-P for system leakage.			*Combination of (a) the effect of stress corrosion cracking on Item C2.3.3 closure bolting for reactor coolant pump, and (b) the effect of corrosion/boric acid wastage on external surfaces of Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.*				No
Same as for the effect of Stress Corrosion Cracking on Item C2.3.3 closure bolting for reactor coolant pump			*Same as for the effect of Stress Corrosion Cracking on Item C2.3.3 closure bolting for reactor coolant pump*				No
Same as for the effect of Stress Corrosion Cracking on Item C2.3.3 closure bolting for reactor coolant pump			*Same as for the effect of Stress Corrosion Cracking on Item C2.3.3 closure bolting for reactor coolant pump*				No
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or other evaluations based on cumulative usage factor (CUF).			Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.				Yes TLAA
Monitoring and control of primary water chemistry in accordance with the EPRI guidelines in TR-105714 minimize the potential of SCC, and material selection according to the NUREG-0313, Rev. 2 guidelines of ≤0.035% C and ≥7.5% ferrite has reduced susceptibility to SCC. For CASS components that do not meet either one of the above guidelines, inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination			***(1) Scope of Program:*** Control of primary water chemistry in accordance with EPRI TR-105714 guidelines and material selection according to the guidance in NUREG-0313, Rev. 2 mitigate the potential of SCC in CASS components. For components that do not meet either one of the above guidelines, inservice inspection (ISI) should be performed to monitor the effects of SCC on the intended function of the pump. ***(2) Preventive Actions:*** Control of primary water chemistry and selection of appropriate materials prevents or mitigates SCC. However, inadvertent introduction of contaminants into the coolant system can occur through the free surface of spent fuel pool or the introduction of oxygen during cool down. The AMP must therefore rely upon ISI in				No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

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

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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
			(continued from previous page) examination extending 1/2 in. on either side of the weld and through wall thickness, and for welds less than NPS 4, surface examination of OD surface extending 1/2 in. on either side of the weld. Category B-M-2 specifies visual VT-3 examination of internal surfaces of the valve. Requirements for training and qualification of personnel and performance demonstration for procedures and equipment are in conformance with Appendices VII and VIII of ASME Section XI, or any other formal program approved by the NRC. Inspection requirements of testing category B-P specify visual VT-2 (IWA-5240) examination of all pressure retaining components, during system leakage test (IWB-5221) and system hydrostatic test (IWB-5222). ***(4) Detection of Aging Effects:*** Degradation of the valve bodies due to SCC can not occur without crack initiation and growth; extent and schedule of inspection assure detection of cracks before the loss of intended function of the valves. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2400 should provide timely detection of cracks. All welds are inspected each inspection period from at least one valve in each group with similar design and performing similar functions in the system. Visual examination is required only when the valve is disassembled for maintenance, repair, or volumetric examination, but at least once during the period. System leakage test is conducted prior to plant startup following each refueling outage, and hydrostatic test at or near the end of each inspection interval. ***(6) Acceptance Criteria:*** Any SCC degradation is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400; IWB-3518 for volumetric and surface examination of welds and 3519 for visual examination of valve internal surfaces. Planar and liner flaws are sized according to IWA-3300 and IWA-3400. Supplementary surface examination may be performed on interior and/or exterior surfaces when flaws are detected in volumetric examination. ***(7) Corrective Actions:*** Repair is in conformance with IWA-4000 and IWB-4000, replacement according to IWA-7000 and IWB-7000, and reexamination in accordance with requirements of IWA-2200. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Although the primary pressure boundary piping of PWRs have generally not been found to be affected by SCC because of low dissolved oxygen levels, potential of SCC exists from inadvertent introduction of contaminants into the primary				

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
C2. REACTOR COOLANT SYSTEM AND CONNECTED LINES (Pressurized Water Reactor)

Existing Aging Management Program (AMP)			Evaluation and Technical Basis					Further Evaluation
C2.4.3	Valves	Closure Bolting	HSLAS	ir, Leaking hemically reated orated ater or team up to 40°C	Cumulative Fatigue Damage	Fatigue		ASME Section III, 1989 Edition. GSI-190.
C2.4.3	Valves	Closure Bolting	HSLAS	ir, Leaking hemically reated orated ater or team up to 40°C	Crack Initiation and Growth	SCC	IVC2-22	NUREG-1339. EPRI NP-5769. NRC GL 91-17. ASME Section XI, 1989 Edition.
C2.4.3	Valves	Closure Bolting	HSLAS	ir, Leaking hemically reated orated ater or team up to 40°C	Loss of Material	Corrosion/ Boric Acid Wastage of External Surfaces	IVC2-23	NUREG-1339. EPRI NP-5769. NRC GL 91-17. NRC GL 88-05. ASME Section XI, 1989 Edition.
C2.4.3	Valves	Closure Bolting	HSLAS	ir, Leaking hemically reated orated ater or team	Loss of Prestress	Stress Relaxation		NUREG-1339. EPRI NP-5769. NRC GL 91-17. ASME Section XI, 1989 Edition.
C2.4.3	Valves	Closure Bolting	HSLAS	ir, Leaking hemically reated orated ater or team	Attrition	Wear		NUREG-1339. EPRI NP-5769. NRC GL 91-17. ASME Section XI, 1989 Edition.
C2.5.1	Pressurizer	Shell/Heads	Low-Alloy Steel (LAS) with SS or Alloy 600 Cladding	hemically reated orated ater or aturated team 90-343°C	Cumulative Fatigue Damage	Fatigue		ASME Section III, 1989 Edition. GSI-190.
C2.5.1	Pressurizer	Shell/Heads (Outer surfaces)	LAS	ir, Leaking hemically reated orated	Loss of Material	Corrosion/ Boric Acid Wastage of External		NRC GL 88-05. ASME Section XI, 1989 Edition. Plant Technical

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or other evaluations based on cumulative usage factor (CUF).			Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.				Yes TLAA
Same as for the effect of stress corrosion cracking on Item C2.3.3 closure bolting for reactor coolant pump.			*Same as for the effect of stress corrosion cracking on Item C2.3.3 closure bolting for reactor coolant pump.*				No
Recommendations for a comprehensive bolting integrity program delineated in NUREG–1339 on resolution of Generic Safety Issue 29 and implemented through NRC Generic Letter 91-17; additional details on bolting integrity outlined in EPRI NP-5769; implementation of NRC Generic Letter 88-05; and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination categories B-G–1 or B-G-2 for pressure retaining bolting, and testing category B-P for system leakage.			*Combination of (a) the effect of stress corrosion cracking on Item C2.3.3 closure bolting for reactor coolant pump, and (b) the effect of corrosion/boric acid wastage on external surfaces of Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.*				No
Same as for the effect of stress corrosion cracking on Item C2.3.3 closure bolting for reactor coolant pump.			*Same as for the effect of stress corrosion cracking on Item C2.3.3 closure bolting for reactor coolant pump.*				No
Same as for the effect of stress corrosion cracking on Item C2.3.3 closure bolting for reactor coolant pump.			*Same as for the effect of stress corrosion cracking on Item C2.3.3 closure bolting for reactor coolant pump.*				No
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or other evaluations based on cumulative usage factor (CUF).			Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.				Yes TLAA
Same as for the effect of corrosion/boric acid wastage of external surfaces of Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.			*Same as for the effect of corrosion/boric acid wastage of external surfaces of Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.*				No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

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

Existing Aging Management Program (AMP)			Evaluation and Technical Basis				Further Evaluation
C2.5.1	Pressurizer	Shell/Heads	LAS with Types 308, 308L or 309 SS or Alloy 600 Cladding	hemically reated orated ater or aturated team 90-343°C	Crack Initiation and Growth	SCC/ IGSCC	ASME Section XI, 1989 Edition. Reg. Guide 1.43. Reg. Guide 1.44. Reg. Guide 1.45. EPRI TR-105714. *Operating Experience* NRC IN 84-18. NRC IN 84-89.

NC2-28

NC2-30

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
Guidelines of Regulatory Guide (RG) 1.44 to avoid sensitization of stainless steels (SSs) and requirements of RG 1.43 to prevent underclad cracking and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination category B-P for the pressurizer pressure boundary, category B-B for pressure-retaining welds, category B-D for full-penetration welds of nozzles, category B-E for pressure retaining partial retaining welds, and category B-F for pressure retaining dissimilar metal welds in nozzles. Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 to minimize the potential of crack initiation and growth.							
			(1) Scope of Program: The program includes preventive measures to mitigate SCC and inservice inspection (ISI) to monitor the effects of SCC on the intended function of SS components and cladding. Cracks in the pressurizer cladding could propagate from cyclic loading into the ferrite base metal and weld metal. However, because the weld metal between the surge nozzle and the vessel lower head is subjected to the maximum stress cycles and the area is periodically inspected as part of the ISI program, the existing AMP is adequate for managing the effect of pressurizer clad cracking. ***(2) Preventive Actions:*** Selection of material in compliance with the requirements of Regulatory Guide (RG) 1.44 prevent or mitigate SCC, and RG 1.43 provides assurance that production cladding complies with ASME Section III and XI requirements to prevent underclad cracking. Also, control of halogens, sulfates, and oxygen in the primary water to less than 0.05, 0.05, and 0.005 ppm, respectively, during operation, and monitor and control of water chemistry during shut down, mitigate potential of SCC. However, inadvertent introduction of contaminants into the coolant system can occur through the free surface of spent fuel pool or the introduction of oxygen during cool down. Also, cladding regions with low delta ferrite may be sensitized during post-weld heat treatment and thus become susceptible to SCC. The AMP must therefore rely upon ISI in accordance with ASME Section XI to detect possible degradation. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of SCC/IGSCC on the intended function of pressurizer components by detection and sizing of cracks by ISI. Inspection requirements of Table IWB 2500-1, specify the extent and schedule for inspection of the components making up the pressurizer pressure boundary. Requirements for training and qualification of personnel and performance demonstration for procedures and equipment are in conformance with Appendices VII and VIII of ASME Section XI, or any other formal program approved by the NRC. Inspection requirements of testing category B-P conducted according to IWA–5000 specify visual VT-2 (IWA-5240) examination of all pressure retaining components extending to and including the second closed valve at the boundary extremity, during system leakage test (IWB-5221) and system hydrostatic test (IWB-5222). Leakage detection is in conformance with Position C of Regulatory Guide 1.45. ***(4) Detection of Aging Effects:*** Degradation of the pressurizer pressure boundary due to SCC can not occur without crack initiation and growth; extent and schedule of inspection assure detection of cracks before the loss of intended function of the pressurizer components. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWB-2500 should provide timely						No

Existing Aging Management Program (AMP)			Evaluation and Technical Basis				Further Evaluation
C2.5.2	Pressurizer	Spray Line Nozzle	CS or LAS with SS Cladding	hemically reated orated ater or aturated team 90-343°C	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. GSI-190.
C2.5.3	Pressurizer	Surge Line Nozzle	CS or LAS with SS Cladding, CASS	hemically reated orated ater up to 40°C	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. GSI-190.
C2.5.4	Pressurizer	Spray Head	Alloy 600, SS, CASS	hemically reated orated ater or aturated team 90-343°C	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. GSI-190.
C2.5.5 thru C2.5.7	Pressurizer	Thermal Sleeves, Instru-mentation Penetrations, Safe Ends	Thermal Sleeves: Alloy 600; Penetration : A 600, SS; Safe Ends: SS	hemically reated orated ater or aturated team 90-343°C	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. GSI-190.
C2.5.2, C2.5.3, C2.5.6	Pressurizer	Spray Line Nozzle, Surge Line Nozzle,	CS or LAS with SS Cladding; or	hemically reated orated	Crack Initiation and Growth	SCC	*Same as for the effect of SCC on C2.5.1 pressurizer*

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
			(continued from previous page) flaws are sized according to IWA-3300 and IWA-3400. ***(7) Corrective Actions:*** Repair is in conformance with IWA-4000 and IWB-4000, replacement according to IWA-7000 and IWB-7000, and reexamination in accordance with requirements of IWA-2200. Continued operation without repair requires that crack growth calculation be performed according to the guidance of ASME Section XI or other approved procedures. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Although the pressurizer primary pressure boundary has generally not been found to be affected by SCC because of low dissolved oxygen levels potential of SCC exists from inadvertent introduction of contaminants into the primary coolant system (IN 84–18). Also, SCC could occur at creviced or cold worked locations (IN 84-89).				
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or other evaluations based on cumulative usage factor (CUF).			Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.				Yes TLAA
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or other evaluations based on cumulative usage factor (CUF).			Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.				Yes TLAA
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or other evaluations based on cumulative usage factor (CUF).			Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.				Yes TLAA
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or other evaluations based on cumulative usage factor (CUF).			Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.				Yes. TLAA
Same as for the effect of SCC on C2.5.1 pressurizer shell/heads.			*Same as for the effect of SCC on C2.5.1 pressurizer shell/heads.*				No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
C2. REACTOR COOLANT SYSTEM AND CONNECTED LINES (Pressurized Water Reactor)

Existing Aging Management Program (AMP)			Evaluation and Technical Basis				Further Evaluation
C2.5.7	Pressurizer	Safe Ends	SS	hemically reated orated ater or aturated team 90-343°C	Crack Initiation and Growth	SCC	*Same as for the effect of Stress Corrosion Cracking on Items C2.1.1 thru C2.1.4 cold-leg, hot-leg, surge line, and spray line piping and fittings constructed of SS.*
C2.5.3	Pressurizer	Surge Line Nozzle	CASS	hemically reated orated ater up to 40°C	Crack Initiation and Growth	SCC	EPRI TR-105714. NUREG-0313, Rev. 2.
C2.5.4	Pressurizer	Spray Head	Alloy 600, SS, CASS	hemically reated orated ater or aturated team 90-343°C	Crack Initiation and Growth	SCC	Plant Technical Specifications
C2.5.6	Pressurizer	Instru-mentation Penetrations	Alloy 600	hemically reated orated ater or aturated team 90-343°C	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. Reg. Guide 1.45. NRC GL 97-01. EPRI TR-105714. *Operating Experience* NRC IN 90-10. NRC IN 96-11.

IVC2-31

IVC2-32

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
Same as for the effect of Stress Corrosion Cracking on Items C2.1.1 thru C2.1.4 cold-leg, hot-leg, surge line, and spray line piping and fittings constructed of SS.			*Same as for the effect of Stress Corrosion Cracking on Items C2.1.1 thru C2.1.4 cold-leg, hot-leg, surge line, and spray line piping and fittings constructed of SS.*				No
Monitoring and control of primary water chemistry in accordance with the EPRI guidelines in TR-105714 minimize the potential of SCC, and material selection according to the NUREG-0313, Rev. 2 guidelines of ≤0.035% C and ≥7.5% ferrite has reduced susceptibility to SCC. For CASS components that do not meet either one of the above guidelines, define plant-specific aging management program.			Plant-specific aging management program is to be evaluated for CASS components that do not meet either the water chemistry guidelines of EPRI TR-105714, or the C and ferrite content criteria of NUREG-0313, Rev. 2. The program will include (a) adequate inspection methods to ensure detection of cracks and (b) flaw evaluation methodology for CASS components that are susceptible to thermal aging embrittlement.				Yes, no AMP
Plant specific aging management program.			Plant specific aging management program is to be evaluated.				Yes, No AMP
Inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination categories B-E for pressure retaining partial penetration welds and testing category B-P for system leakage. Primary water chemistry is monitored and maintained in accordance with EPRI guidelines in TR-105714 to minimize the potential of crack initiation and growth.			***(1) Scope of Program:*** The program includes preventive measures to mitigate stress corrosion cracking (SCC) and inservice inspection (ISI) to detect cracks or leakage. However, the program is inadequate to manage the effects of SCC on the intended function of Ni-alloy components. The applicant should review previous evaluations performed in response to Generic Letter (GL) 97-01, as updated with the most current industry susceptibility model and inspection results. ***(2) Preventive Actions:*** Control of halogens, sulfates, and oxygen in the primary water to less than 0.05, 0.05, and 0.005 ppm, respectively, during operation, and monitor and control of water chemistry during shut down, mitigate potential of SCC. However, based on the experiences described in NRC Information Notices (IN) 90-10 and 96-11, and Generic Letter (GL) 97-01, the susceptibility of Ni-alloys and welds to primary water SCC has not been addressed adequately, particularly when demineralizer resins contaminate the reactor coolant system. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of SCC on the intended function of instrument nozzles and penetrations by performance monitoring and detection of cracks by inservice inspection (ISI). Testing category B-P specify visual VT-2 (IWA-5240) examination of all				Yes, Elements 2 to 6 should be further evaluated

Existing Aging Management Program (AMP)			Evaluation and Technical Basis				Further Evaluation
C2.5.3, C2.5.4	Pressurizer	Surge Line Nozzle, Spray head	CASS	hemically reated orated	Loss of Fracture Toughness	Thermal Aging Embrittle-	EPRI TR-106092. ASME Section XI, 1989 Edition.

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
			(continued from previous page) penetration welds during the hydrostatic test. However, the applicant should perform a susceptibility study of all Ni-alloy components to identify the most susceptible locations and to determine whether an augmented inspection program, including a combination of surface and volumetric examination, is necessary. ***(4) Detection of Aging Effects:*** Aging effects degradation of the vessel penetrations can not occur without crack initiation. Based on GL 97-01, the applicant should review the scope and schedule of inspection, including leakage detection system, to assure detection of cracks before the loss of intended function of the penetrations. ***(5) Monitoring and Trending:*** System leakage test is conducted prior to plant startup following each refueling outage, and hydrostatic test at or near the end of each inspection interval. Applicant should either provide the technical basis that justifies the adequacy of the program or develop an integrated long-term program which includes periodic inspection of the most susceptible locations to detect the occurrence of SCC. The frequency of subsequent inspections should be based on the finding of the initial inspections and crack growth rate models for Ni alloys. ***(6) Acceptance Criteria:*** Any SCC degradation is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and 3522 for visual examination. Acceptance of the component for continued service is based on the results of visual examination, supplemental volumetric and surface examination (IWB-3200), corrective measure or repair, analytical evaluation , or replacement (IWB-3142). However, the applicant should provide information on crack initiation and crack growth models and the data used to validate these models to verify adequacy of the inspection program and acceptance criteria (GL 97-01). ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWA-4000 and IWB-4000, and reexamination in accordance with requirements of IWA-2200. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** SCC of Alloy 600 and austenitic SSs has occurred in domestic and foreign PWRs (IN 90-10). Very little attention has been given to the inspection for SCC in Alloy 600 applications other than that associated with steam generator tubes.		
Same as for the effect of Thermal Aging Embrittlement on Items C2.1.1 thru C2.1.3 Cold-Leg, Hot-Leg, and Surge			*Same as for the effect of Thermal Aging Embrittlement on Items C2.1.1 thru C2.1.3 Cold-Leg, Hot-Leg, and Surge Line Piping and Fittings.*				No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
C2. REACTOR COOLANT SYSTEM AND CONNECTED LINES (Pressurized Water Reactor)

Existing Aging Management Program (AMP)			Evaluation and Technical Basis				Further Evaluation
C2.5.4	Pressurizer	Spray head	Alloy 600, SS, CASS	hemically reated orated ater or aturated team 90-343°C	Wall Thinning	Erosion	Plant Technical Specifications.
C2.5.8	Pressurizer	Manway and Flange	LAS with Types 308, 308L, or 30 SS; or Alloy 600 Cladding	hemically reated orated ater or aturated team 90-343°C	Crack Initiation and Growth	SCC	*Same as for the effect of SCC on Item C2.5.1 pressurizer shell/heads*
C2.5.9	Pressurizer	Manway and Flange Bolting	HSLAS	ir, Leaking hemically reated orated ater or team up to 40°C	Crack Initiation and Growth	SCC	NUREG-1339. EPRI NP-5769. NRC GL 91-17. ASME Section XI, 1989 Edition.
C2.5.9	Pressurizer	Manway and Flange Bolting	HSLAS	ir, Leaking hemically reated orated ater or team up to 40°C	Loss of Material	Corrosion / Boric Acid Wastage of External Surfaces	NUREG-1339. EPRI NP-5769. NRC GL 91-17. NRC GL 88-05. ASME Section XI, 1989 Edition.
C2.5.9	Pressurizer	Manway and Flange Bolting	HSLAS	ir, Leaking hemically reated orated ater or team up to 40°C	Loss of Prestress	Stress Relaxation	NUREG-1339. EPRI NP-5769. NRC GL 91-17. ASME Section XI, 1989 Edition.
C2. 5.10	Pressurizer	Heater Sheaths and Sleeves	Alloy 600 or Austenitic SS	hemically reated orated ater up to 40°C	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. GSI-190.
C2. 5.10	Pressurizer	Heater Sheaths and	Austenitic SS	hemically reated	Crack Initiation	SCC	*Same as for the effect of SCC on*

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
C2. REACTOR COOLANT SYSTEM AND CONNECTED LINES (Pressurized Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
Plant-specific aging management program.			Plant-specific aging management program is to be evaluated.				Yes No AMP
Same as for the effect of SCC on Item C2.5.1 pressurizer shell/heads.			*Same as for the effect of SCC on Item C2.5.1 pressurizer shell/heads.*				No
Same as for the effect of stress corrosion cracking on Item C2.3.3 closure bolting for reactor coolant pump.			*Same as for the effect of stress corrosion cracking on Item C2.3.3 closure bolting for reactor coolant pump.*				No
Recommendations for a comprehensive bolting integrity program delineated in NUREG–1339 on resolution of Generic Safety Issue 29 and implemented through NRC Generic Letter 91-17; additional details on bolting integrity outlined in EPRI NP-5769; implementation of NRC Generic Letter 88-05; and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination categories B-G–1 or B-G-2 for pressure retaining bolting, and testing category B-P for system leakage.			*Combination of (a) the effect of stress corrosion cracking on Item C2.3.3 closure bolting for reactor coolant pump, and (b) the effect of corrosion/boric acid wastage on external surfaces of Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.*				No
Same as for the effect of Stress Corrosion Cracking on Item C2.3.3 closure bolting for reactor coolant pump.			*Same as for the effect of Stress Corrosion Cracking on Item C2.3.3 closure bolting for reactor coolant pump.*				No
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or other evaluations based on cumulative usage factor (CUF).			Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.				Yes, TLAA
Same as for the effect of SCC on Item C2.5.1 pressurizer shell/heads.			*Same as for the effect of SCC on Item C2.5.1 pressurizer shell/heads.*				No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
C2. REACTOR COOLANT SYSTEM AND CONNECTED LINES (Pressurized Water Reactor)

Existing Aging Management Program (AMP)			Evaluation and Technical Basis				Further Evaluation
C2. 5.10	Pressurizer	Heater Sheaths and Sleeves	Alloy 600	hemically reated orated ater up to 40°C	Crack Initiation and Growth	SCC	*Same as for the effect of SCC on Item C2.5.6 pressurizer instrumentation penetrations constructed of Alloy 600*
C2. 5.10	Pressurizer	Heater Sheaths and Sleeves	Alloy 600 or Austenitic SS	hemically reated orated ater up to 40°C	Attrition	Wear	lant Technical Specifications.
C2. 5.11	Pressurizer	Support Keys, Skirts, and Shear Lugs	CS, LAS	ir, with etal emperature up to 40°C	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition.
C2. 5.12	Pressurizer	Integral Support	CS	ir, eaking hemically reated orated ater	Loss of Material	Corrosion/ Boric Acid Wastage of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 Edition. NRC IN 86-108 S 3.

INC2-34

INC2-35

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
C2. REACTOR COOLANT SYSTEM AND CONNECTED LINES (Pressurized Water Reactor)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
Same as for the effect of SCC on Item C2.5.6 pressurizer instrumentation penetrations constructed of Alloy 600.			*Same as for the effect of SCC on Item C2.5.6 pressurizer instrumentation penetrations constructed of Alloy 600.*				Yes, Elements 2 to 6 should be further evaluated
Plant-specific aging management program.			Plant-specific aging management program is to be evaluated.				Yes, no AMP
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or other evaluations based on cumulative usage factor (CUF).			Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal.				Yes, TLAA
Implementation of NRC Generic Letter 88-05, and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination category B-K-1 for integral attachments for piping, pumps, and valves.			***(1) Scope of Program:*** The staff guidance of Generic Letter (GL) 88-05 provides assurance that a program has been implemented consisting of systematic measures to ensure that the effects of corrosion caused by leaking coolant containing boric acid does not lead to degradation and provides assurance that the reactor coolant pressure boundary will have a extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture. The program includes (a) determination of principal location, (b) examinations requirements and procedures for locating small leaks, and (c) engineering evaluations and corrective actions. ***(2) Preventive Actions:*** Minimizing reactor coolant leakage by frequent monitoring of the locations where potential leakage could occur and repairing the leaky components as soon as possible, prevent or mitigate boric acid corrosion. ***(3) Parameters Monitored/Inspected:*** The AMP monitors the effects of boric acid corrosion on the intended function of the component by detection of flaws and coolant leakage by inservice inspection (ISI). Inspection requirements of ASME Section XI, Table IWB 2500-1, examination category B-K-1 specify volumetric or surface examination extending 1/2 in. on either side of the weld. Inspection includes the welded attachments of piping required to be examined by examination category B-J and the welded attachments to associated pumps and valves integral to the piping. ***(4) Detection of Aging Effects:*** Degradation of the component due to boric acid wastage can not occur without leakage of coolant containing boric acid; extent and schedule of inspection assure detection of flaws and leakage before the loss of intended function of the component. ***(5) Monitoring and Trending:***				No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
C2. REACTOR COOLANT SYSTEM AND CONNECTED LINES (Pressurized Water Reactor)

Existing Aging Management Program (AMP)			Evaluation and Technical Basis				Further Evaluation
C2. 5.12	Pressurizer	Integral Support	CS, SS	ir	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition.
C2.6.1, C2.6.2	Pressurizer Relief Tank	Tank Shell and Heads, Flanges and Nozzles	CS with Type 304 SS Cladding	hemically reated orated ater at 4°C	Cumulative Fatigue Damage	Fatigue	Original Licensing Basis or ASME Section III, 1989 Edition.
C2.6.1	Pressurizer Relief Tank	Tank Shells and Heads (External Surfaces)	CS	4°C, air, hemically reated orated ater	Loss of Material	Corrosion/ Boric Acid Wastage of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 Edition. NRC IN 86-108 S 3. Plant Technical Specifications.
C2.6.1, C2.6.2	Pressurizer Relief Tank	Tank Shell and Heads, Flanges and Nozzles	CS with Type 304 SS Cladding	hemically reated orated ater at 4°C	Crack Initiation and Growth	SCC	ASME Section XI, 1989 Edition. Regulatory Guide 1.43. Regulatory Guide 1.44. Regulatory Guide 1.45.

NC2-36

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
			(continued from previous page) and replacement are in conformance with IWA-4000 and IWB-4000. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Boric acid wastage observed in nuclear power plants may be classified into two distinct types: (a) corrosion that increases the rate of leakage, e.g., corrosion of closure bolting or fasteners in reactor coolant pressure boundary, and (b) corrosion that occurs some distance from the source of leakage. Some recent incidents of boric acid wastage (IN 86-108 S3) indicate that, although implementation of GL 88-05 ensures timely detection of leakage, there may still be a lack of awareness of the conditions that can lead to boric acid wastage.				
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NG, or other evaluations based on cumulative usage factor (CUF).			Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal.				Yes, TLAA
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NC, or other evaluations based on cumulative usage factor (CUF).			Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal.				Yes, TLAA
Same as for the effect of corrosion/boric acid wastage of external surfaces of Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.			*Same as for the effect of corrosion/boric acid wastage of external surfaces of Items C2.1.1 and C2.1.2 cold-leg and hot-leg piping and fittings.*				No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Guidelines of Regulatory Guide (RG) 1.44 to avoid sensitization of stainless steels (SSs) and guidelines of RG 1.43 to prevent underclad cracking and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWC, Table IWC 2500-1, examination category C-A for pressure-retaining welds, category C-B for pressure-retaining nozzle welds, and category C-H for all pressure-retaining components.	**(1) Scope of Program:** The program includes preventive measures to mitigate SCC of SS components and cladding, and inservice inspection (ISI) to monitor the effects of SCC on the intended function of clad components. **(2) Preventive Actions:** Selection of material in compliance with the guidelines of Regulatory Guide (RG) 1.44 prevent or mitigate SCC, and RG 1.43 provides assurance that production cladding complies with ASME Section III and XI requirements to prevent underclad cracking. Also, control of halogens, sulfates, and oxygen in the primary water to less than 0.05, 0.05, and 0.005 ppm, respectively, during operation, and monitor and control of water chemistry during shut down, mitigate potential of SCC. However, inadvertent introduction of contaminants into the coolant can occur, and cladding regions with low delta ferrite may be sensitized during post-weld heat treatment and thus become susceptible to SCC. Nonetheless, oxygen-induced SCC is not anticipated for this component because of the low operating	No

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

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

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

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

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IV

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV.C2-1	Reactor Coolant System and Connected Lines (PWR)	All			General Comment --Suggest that aging effects that apply to all RCS piping items (e.g., loss of external material by boric acid wastage of ferritic items and cumulative fatigue damage be discussed at the very beginning and not repeated thereafter). At present, CUF is listed for selected items and not for all RCS piping items.
IV.C2-2	Reactor Coolant System and Connected Lines (PWR)	C2	3		The Systems, Structures and Components section states this section includes the connecting systems up to the first isolation valve outside containment. This is not consistent with the section VII.E1 CVCS that includes the Regeneration Heat Exchanger that is located inside Containment. This also contradicts section V.D1 ECCS (PWR) that includes the Core Flood System that is contained in Containment. The scope should be consistent with the ASME code break from class one to class two.
IV.C2-3	Reactor Coolant System and Connected Lines (PWR))	C2	3		The System Interfaces write-up is not consistent with the Systems, Structures and Components section. CVCS section VII.E1 is also a system interface.
IV.C2-4	Reactor Coolant System and Connected Lines (PWR)	C2	4		Chapter IV should reference the new EPRI PWR Primary Water Chemistry Guidelines Rev. 4, March 1999, EPRI TR-105714. This reference is missing completely at the end of the chapter.

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV.C2-5	Reactor Coolant System and Connected Lines (PWR)	C2	Various		Fatigue is listed as an Aging Mechanism. The Existing AMP entry states that components have been designed for fatigue for a 40 year design life in accordance with ASME III. This not strictly true; it has been assumed that the components in question will experience certain operational stresses during the plant lifetime, for example, less than or equal to 500 thermal cycles caused by plant startups and shutdowns. This number of transients will have originally been estimated with some consideration of a 40 year plant life, but there is no explicit correlation to a time period in this evaluation. Also, not all plants are licensed to ASME III. This statement should be removed. Also, simply identifying Fatigue as a TLAA does not identify an AMP per se. More information should be provided, for example, relevant to Fatigue Monitoring Programs that calculate Cumulative Usage Factors based on partial cycles.
IV.C2-6	Reactor Coolant System and Connected Lines (PWR)	C2.1.1 through C2.1.4, C2.2.7	4		The use of combined terms such as "SCC, IGSCC" for the aging mechanism is confusing. If the authors trying to indicate that the general mechanism is SCC, but the more specific mechanism is IGSCC as opposed to TGSCC, IDSCC, IASCC, PWSCC, ODSCC, they should just use the specific applicable SCC term. The aging mechanism should either refer to only the generic term "SCC" or the specific applicable phenomena such as IGSCC, TGSCC or IDSCC. Sometimes a very highly specific SCC term such as "ODSCC" is used acceptably. Section B of Chapter IV simplifies aging mechanisms by using only single terms like SCC or IASCC, while Section D1 again sometimes refers to a combined term of "SCC, IGSCC, PWSCC." In this case, PWSCC would probably be sufficient. Also, IGSCC is often used with low alloy steels (LAS) that typically crack by TGSCC.

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV.C2-7	Reactor Coolant System and Connected Lines (PWR)	C2.1.1 through C2.1.4, C2.2.1 through C2.2.6, C2.2.7, C2.5.1, C2.5.2, C2.5.3, C2.5.6, C2.5.7, C2.5.8, C2.5.10, C2.6.1, C2.6.2	4, 16, 30, 32, 34, 38, 42		SCC or SCC/IGSCC (non-CASS) is listed as an Aging Mechanism. SCC is not a concern for SS components in treated borated water where chemistry controls maintain halides < 150 ppb or sulfates < 100 ppb (BAW-2270). Chemistry controls in accordance with industry guidelines assure this requirement is met. Therefore, for SCC, chemistry programs in accordance with industry guidelines alone should be credited. A similar position was accepted in NUREG-1705. The References, Existing AMP, Evaluation and Technical Basis, and Further Evaluation entries should be rewritten to correspond to the provided example.
IV.C2-8	Reactor Coolant System and Connected Lines (PWR)	C2	4		SCC/IGSCC (non-CASS) is listed as an Aging Mechanism. (See separate comments on Programs for this Aging Mechanism relating to the entire section elsewhere.) Reg. Guide 1.43 and 1.44 Guidelines are not an AMP, they are a design feature and as such should not be listed or should be listed in the Material entry. Also, specifically in Item IV.C2.5.1 for the Pressurizer shell and heads, the "Evaluation and Technical Basis" entry, Item (1), discusses cracks in the clad propagating from cyclic loading. This sounds more like Fatigue than SCC. Either the Aging Mechanism entry for this Item needs to also include Fatigue, another entry needs to incorporate this information, or if this information does not actually describe a legitimate Aging Mechanism, it should be removed.

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IV

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV C2-9	Reactor Coolant System and Connected Lines (PWR)	C2	4		SCC/IGSCC (non-CASS) is listed as an Aging Mechanism. (See separate comments on Programs for this Aging Mechanism relating to the entire section elsewhere. Comment to follow assumes that the previous comment regarding crediting only Chemistry Program is not accepted.) If the staff maintains that cracking of ferritic material is attributed to SCC or IGSCC of the stainless steel cladding then it should be made clear in the table. If this is the case, all cladding and all underlying ferritic steel (welds and base metal) could be susceptible to cracking and ASME Section XI, Examination Category B-J, focuses only on the welded joints. The GALL report should specify under what conditions Examination Category B-J as modified by Code Case N-560 to incorporate risk-informed inspections (see the ANO-1 LRA) would be acceptable for license renewal. Two risk-informed methods may be used to select piping welds for inspection: EPRI and ASME. Both methods require that the revised set of piping inspections not increase the Core Damage Frequency relative to the 89 Edition of Section XI, B-J. See the ANO-1 LRA for a discussion of the applicability of risk-informed inspection for license renewal. In addition, aging management of cracking in dissimilar metal welds that connect vessel nozzles to safe ends is through ASME Section XI, Examination Category B-F. Examination Category B-J covers dissimilar metal welds that connect branch connections to connected piping.

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IV

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV.C2-10	Reactor Coolant System and Connected Lines (PWR)	C2.1.1, C2.1.2, C2.2.4, C2.2.5, C2.3.3, C2.4.3, C2.5.1, C2.5.9, C2.6.1	6, 16, 24, 28, 38, 40, 42		Corrosion/Boric Acid Wastage is listed as an Aging Mechanism. The Existing AMP entry lists both a Boric Acid Corrosion Inspection Program (meeting the requirements of GL 88-05) and ISI. The Boric Acid Corrosion Inspection Program itself was deemed adequate for aging management in NUREG-1705. Specifically, in Items IV.C2.3.3, C2.4.3, C2.5.9, Corrosion/Boric Acid Wastage, (Closure Bolting), the "Existing AMP" entry lists numerous programs in combination for this Aging Mechanism: NUREG-1339, GSI 29, GL 91-17, EPRI NP-5769, GL 88-05, and ISI. This seems to be unnecessarily excessive considering the above information concerning NUREG-1705, especially since these are not all actually "programs". The References, Existing AMP, Evaluation and Technical Basis, and Further Evaluation entries should be rewritten to correspond to the provided example.
IV.C2-11	Reactor Coolant System and Connected Lines (PWR)	C2	10		SCC/IGSCC (CASS) is listed as an Aging Mechanism. (See separate comments on Programs for this Aging Mechanism relating to the entire section elsewhere.) NUREG-0313 Guidelines are not an AMP, they are a design feature and as such should not be listed or should be included in the Material entry. Also, most plants were constructed prior to the issuance of the guidelines, so inclusion of this NUREG in the Report should be qualified.

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IV

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV.C2-12	Reactor Coolant System and Connected Lines (PWR)	C2.1.1 through C2.1.3, C2.2.6, 2.3.1, C2.4.1, C2.5.3	10, 16, 18, 24, 34		SCC/IGSCC (CASS) is listed as an Aging Mechanism. This entry suggests that some PWRs will not meet the EPRI water chemistry guidelines and NUREG-0313, rev. 2 on carbon and ferrite content. This appears to be a concern about low temperature sensitization of CASS with C > 0.035 %, ferrite < 7.5 % (not likely), and intergranular attack from an aggressive environment. A technical basis for this assumption should be provided or these entries should be removed. Chemistry programs in accordance with industry guidelines alone should be credited. A similar position was accepted in NUREG-1705. The References, Existing AMP, Evaluation and Technical Basis, and Further Evaluation entries should be rewritten to correspond to the provided example.
IV.C2-13	Reactor Coolant System and Connected Lines (PWR)	C2.1.1 thru C2.1.32, C2.3.1, C2.4.1	10, 20, 26		Change "Loss of" Fracture Toughness to "Reduced" Fracture Toughness
IV.C2-14	Reactor Coolant System and Connected Lines (PWR)	C2.1.1 thru C2.1.3	10		Thermal Embrittlement (CASS) is listed as an Aging Mechanism. Sentence beginning "Because of low applied stresses" under "Existing AMP" is not completed.

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV.C2-15	Reactor Coolant System and Connected Lines (PWR)	C2.1.1 thru C2.1.3	10		The ferrite screening criteria is not consistent with the NRC-specified screening criteria contained in the Oconee license renewal application. Reduction of fracture toughness by thermal embrittlement is an applicable aging effect for CASS items under the following circumstances: **High-Mo Content** Statically Cast and ferrite > 10% Centrifugally Cast and Ferrite > 20% **Low-Mo Content** Statically Cast and Ferrite > 20% Low Moly Centrifugally Cast Not Susceptible to Thermal Embrittlement • Flaws in CASS with ferrite levels less than 25% and no niobium may be evaluated using ASME Section XI, IWB-3640 procedures • Flaws in CASS with ferrite levels exceeding 25% must be evaluated using IWB-3640 procedures. If such flaws occur, fracture toughness data will be presented on a case-by-case basis.
IV.C2-16	Reactor Coolant System and Connected Lines (PWR)	C2.1.5	12		Small Bore RCS Piping has CS (carbon steel) listed as a material. This is not a valid material for this application. Also, exactly what is meant to constitute smallbore piping is not defined. In accordance with BAW-2243A, small bore piping includes pipe, fittings, and branch connections--NPS less than 4 inches.

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IV

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV.C2-17	Reactor Coolant System and Connected Lines (PWR)	C2.1.5	12		Reactor Coolant System Small-Bore Piping lists "Crack Initiation and Growth" as an Aging Effect that results from the Aging Mechanism "Unanticipated Thermal and Mechanical Loading". It is not clear what this Aging Mechanism refers to. It seems similar to Fatigue. It seems that this "Aging Mechanism" does not actually result from aging but involves discrete events that exploit design or maintenance flaws. If this entry actually describes an Aging Mechanism, and Cracking is the Aging Effect, it appears that it would be appropriate to describe the Aging Mechanism as either Fatigue or SCC, and if this is not possible, then perhaps it is not really an Aging Mechanism.
IV.C2-18	Reactor Coolant System and Connected Lines (PWR)	C2.1.5	12		(See previous comment regarding Aging Effects/ Mechanisms for this entry.) The technical issue of degradation and failure of small-bore Class 1 piping has been subsumed under the aging mechanism of unanticipated thermal and mechanical loading, with crack initiation and growth as the effects. Existing programs are found to be adequate with the exception of Elements 3 (Parameters Monitored/Inspected) and 4 (Detection of Aging Effects). The NRC staff are asking the industry to destructively or volumetrically examine, on a one-time basis, a sample of locations for piping NPS 4 and smaller. The NRC staff should acknowledge the fact that the industry Materials Reliability Project is addressing the issue generically, and that individual applicants will be able to take advantage of the findings for that generic program in the next few years. The safety significance of small-bore piping cracking and leakage, considering ASME Code Examination Category B-P visual examinations and plant leakage detection capability, does not warrant the imposition of this augmented inspection requirement.
IV.C2-19	Reactor Coolant System and Connected Lines (PWR)	C2.2.4, C2.2.5	14		Sampling, Drains & Instrumentation has CS (carbon steel) listed as a material. This is not a valid material for this application.

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV.C2-20	Reactor Coolant System and Connected Lines (PWR)	C2.2.7	16		Small Bore Piping in Connected Systems has CS (carbon steel) listed as a material. This is not a valid material for this application. Also, exactly what is meant to constitute smallbore piping is not defined. In accordance with BAW-2243A, small bore piping includes pipe, fittings, and branch connections--NPS less than 4 inches.
IV.C2-21	Reactor Coolant System and Connected Lines (PWR)	C2.2.7	16		SCC/IGSCC is listed as an Aging Mechanism for Connected Systems Small-Bore Piping. The Existing AMP entry refers to a previous entry on SCC/IGSCC for RCS Small-Bore Piping (using an incorrect section number C2.1.7) which does not exist. Does this actually refer to the IV.C2.1.5 entry on Unanticipated Thermal and Mechanical Loading?
IV.C2-22	Reactor Coolant System and Connected Lines (PWR)	C2.3.3, C2.4.3, C2.5.9	20		Items list SCC as an Aging Mechanism for RCP Bolting, Valve Bolting, and Manway Flange Bolting. In the "Evaluation and Technical Basis", Item 10, it is indicated that Boric Acid Wastage is a parallel concern for such bolting. Since it is likely that the contaminants and the aqueous environment required for SCC would be provided in this instance by borated water leakage, and since boric acid corrosion would propagate at a much faster rate than SCC, and since the application of these components does not result in high stresses, and since programs to monitor boric acid corrosion typically inspect potentially degraded areas each refueling interval, it would be more appropriate to characterize the Aging Mechanism of concern in this case as only Corrosion/Boric Acid Wastage, and delete this entry in favor of the following entry. This position was accepted in NUREG-1705.

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IV

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV.C2-23	Reactor Coolant System and Connected Lines (PWR)	C2	20		SCC is listed as an Aging Mechanism for Closure Bolting. (See Previous comment on this Aging Mechanism for Closure Bolting. Comment to follow assumes that the Aging Mechanism is retained in Report despite earlier comment.) In the "Evaluation and Technical Basis", Item (4), mention is made of loss of prestress and attrition, neither of which effect should be of concern for this entry. Remove this information from the entry.
IV.C2-24	Reactor Coolant System and Connected Lines (PWR)	C2.3.3, C2.4.3, C2.5.9	24		Items list Stress Relaxation as an Aging Mechanism for RCP Closure Bolting, RCS Valve Bolting, Pressurizer Manway Bolting, and SG Manway and Handhold Bolting. The operating temperature of the RCS is well below the threshold for creep, the precursor for Stress Relaxation, to occur. Additionally, the neutron field around all components other than the vessel is less than that required to reduce the temperature threshold. This Aging Mechanism should not be considered. This position was accepted in NUREG-1705.
IV.C2-25	Reactor Coolant System and Connected Lines (PWR)	C2	24		Stress Relaxation is listed as an Aging Mechanism for Closure Bolting. (This mechanism was considered Not Plausible for CCNPP; see separate comments on the Aging Mechanism elsewhere. Comments to follow assume that the Aging Mechanism is retained in Report despite earlier comment.) This entry refers back to a previous entry for the RCP Closure Bolting regarding SCC. These Aging Mechanisms are very different. Why are the program attributes for these different mechanisms combined into one entry? It will be difficult for the prospective applicant to segregate which information is presented for which Aging Mechanism.
IV.C2-26	Reactor Coolant System and Connected Lines (PWR)	C2.3.3, C2.4.3	24	J C LaBorde	Wear does not appear to be a valid Aging Mechanism for Closure Bolting. Normal maintenance practices should not result in aging of these components due to Wear. These entries should be removed.

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IV

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV.C2-27	Reactor Coolant System and Connected Lines (PWR)	C2	24		Wear is listed as an Aging Mechanism for Closure Bolting. (See separate comments on the Aging Mechanism elsewhere. Comments to follow assume that the Aging Mechanism is retained in Report despite earlier comment.) This entry refers back to a previous entry for the RCP Closure Bolting regarding SCC. These Aging Mechanisms are very different. Why are the program attributes for these different mechanisms combined into one entry? It will be difficult for the prospective applicant to segregate which information is presented for which Aging Mechanism.
IV.C2-28	Reactor Coolant System and Connected Lines (PWR)	C.2.5.1	28		The Pressurizer shell and heads are fabricated from stainless steel clad carbon steel for B&W operating plants. Carbon steel may be added to the Material entry with no revisions necessary to the other information
IV.C2-29	Reactor Coolant System and Connected Lines (PWR)	C.2.5.1	30		SCC/IGSCC (non-CASS) is listed as an Aging Mechanism. (See separate comments on Programs for this Aging Mechanism relating to the entire section elsewhere. This comment applies only if substitution of separate material for Existing AMP and Evaluation and Technical Basis information is not performed per prior comment.) Examination Categories B-B and B-P are the only categories that apply to the shell/heads. Other exam Categories apply to other items.

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV.C2-30	Reactor Coolant System and Connected Lines (PWR)	C2	30		SCC is listed as an Aging Mechanism for Alloy 600 components. The discussion in the "Evaluation and Technical Basis" entry provides an excellent summary of the arguments put forward by industry on SCC and IGSCC of the stainless steel clad pressurizer shell and head. However, it seems the "Evaluation and Technical Basis" entry is telling the prospective applicant that they need to establish an "Alloy 600 Program", but this is not clearly stated. Various elements of an adequate "Alloy 600 Program" are scattered throughout the "Evaluation and Technical Basis" entry. It would be more helpful if these elements of an adequate Alloy 600 Program were segregated from the other information. Also, wouldn't the evaluations conducted for GL 97-01 constitute part of the "Existing AMP", much the same as for GL 88-05 regarding Boric Acid Corrosion? The determination that ASME Code Section XI inservice inspections are adequate to manage these mechanisms or effects provide the basis for closure of a major technical issue.
IV.C2-31	Reactor Coolant System and Connected Lines (PWR)	C2.5.4	34		The Spray Head has been determined to be not within the scope of license renewal for CCNPP. This conclusion likely applies to some other plants as well. It may be appropriate to remove this entry.
IV.C2-32	Reactor Coolant System and Connected Lines (PWR)	C2.5.6	35		SCC is listed as an Aging Mechanism for the Alloy 600 Pressurizer Instrument Penetrations. There is no technical basis for augmentation of Elements 2 through 6 in the column on "Evaluation and Technical Basis." The ASME Code Section XI inservice inspection program, in particular Examination Category B-P, together with adherence to the EPRI water chemistry guidelines, should be adequate to manage the effects of PWSCC for Ni-Fe-Cr alloy penetrations in the pressurizer. The column on "Further Evaluation" should be changed to read "No" and the related information should be deleted from the "Evaluation and Technical Basis" entry

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV.C2-33	Reactor Coolant System and Connected Lines (PWR)	C2.5.4	38		(See previous comment on Scoping. Comments to follow assume that the component is retained in Report despite earlier comment) Erosion does not appear to be a valid Aging Mechanism for Pzr spray heads. These materials are resistant and the normal flow rates are not sufficient to be concerned with erosion.
IV.C2-34	Reactor Coolant System and Connected Lines (PWR)	C2.5.10	40		Item lists wear as an Aging Mechanism for the Pressurizer Heater Sheaths and Sleeves. Wear should not occur for these components because there is no relative motion between them or any other parts of the Pressurizer. This Aging Mechanism should not be considered. This position was accepted in NUREG-1705.
IV.C2-35	Reactor Coolant System and Connected Lines (PWR)	C2.5.10	40		Wear is listed as an Aging Mechanism for the Pressurizer Heater Sheath and Sleeve. (This mechanism was considered Not Plausible for CCNPP; see separate comments on the Aging Mechanism elsewhere. Comments to follow assume that the Aging Mechanism is retained in Report despite earlier comment.) Since no program elements are provided to demonstrate how to manage Wear of Pressurizer Heater Sheaths and Sleeves, it is not clear how wear may be occurring. Since the prospective applicant may not desire to credit any programs for Wear of Pressurizer Heater Sheaths and Sleeves, based on their plant-specific operating experience and other aging evaluations, this entry is of no use.
IV.C2-36	Reactor Coolant System and Connected Lines (PWR)	C2.6	42		The Pressurizer Relief Tank does not meet the criteria for inclusion into the scope of License Renewal. Although the tank may be Safety Related it is not required to remain functional during and following a design basis event. Remove this item from the Report.

D1 Steam Generator (Recirculating)

D1.1 Pressure Boundary and Structural

D1.1.1 Top Head

D1.1.2 Steam Nozzle and Safe End

D1.1.3 Upper and Lower Shell

D1.1.4 Transition Cone

D1.1.5 Feedwater Nozzle and Safe End

D1.1.6 Feedwater Impingement Plate and Support

D1.1.7 Secondary Manways and Bolting

D1.1.8 Secondary Handholes and Bolting

D1.1.9 Instrument Nozzles

D1.1.10 Primary Manways and Bolting

D1.2 Tube Bundle

D1.2.1 Tubes

D1.2.2 Tube Support Plates

D1.2.3 Tube Support Lattice Bars (Combustion Engineering)

D1.2.4 Tube Plugs

D1.2.5 Tube Repair Sleeves

D1.3 Upper Assembly and Separators

D1.3.1 Feed Water Inlet Ring

D1.4 Piping and Fittings

D1.5 Safety and Relief Valves

D1.5.1 Body

D1. Steam Generator (Recirculating)

IV D1-2

System, Structures, and Components

The system, structures, and components included in this table consist of the recirculating-type steam generators, as found in Westinghouse and Combustion Engineering pressurized water reactors (PWRs), including all internal components, external primary and secondary water/steam nozzles and safe ends, and portions of main steam, feedwater, and auxiliary feedwater systems extending from the steam generator up to the first isolation valve outside of containment or to the first anchor point. Based on US Nuclear Regulatory Commission Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," the primary water side (tube side) of the steam generator is classified as Group A Quality Standards and secondary water side, including portions of main steam, feedwater, and auxiliary feedwater systems extending up to the first isolation valve outside of containment, is classified as Group B Quality Standard. The aging management program for the lines that penetrate the containment, and associated isolation valves, is reviewed in Table V C.

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

System Interfaces

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

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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.1.1, D1.1.2	Pressure Boundary and Structural	Top Head, Steam Nozzle & Safe End	Low-alloy Steel (LAS)	p to 300°C team	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. GSI-190.
D1.1.2	Pressure Boundary and Structural	Steam Nozzle & Safe End	LAS	p to 300°C team	Wall Thinning	Erosion/ Corrosion (E/C)	NUREG-1344. EPRI NSAC-202L-R2. NRC IN 93-21. EPRI TR-102134. *Operating Experience* NRC BL 87-01. NRC GL 89-08. NRC IN 89-53. NRC IN 91-18. NRC IN 91-18 S1. NRC IN 91-28. NRC IN 92-35. NRC IN 95-11. NRC IN 97-84.

IVD1-2

IVD1-1

IVD1-5

IVD1-4

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

D1 STEAM GENERATOR (Recirculating)

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a) or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed. IVD1-6	Yes TLAA
Program delineated in NUREG-1344 for single phase lines and implemented through NRC Generic Letter 89-08; CHECWORKS Code; EPRI guidelines of NSAC-202L-R2 for effective erosion/corrosion program; and water chemistry program based on EPRI guidelines for secondary water chemistry (EPRI TR-102134).	***(1) Scope of Program:*** The AMP includes NUMARC program delineated in Appendix A of NUREG-1344 and implemented through NRC Generic Letter (GL) 89-08; CHECWORKS computer Code; and EPRI guidelines of NSAC-202L-R2. The program includes the following recommendations: (a) conduct appropriate analysis and limited baseline inspection, (b) determine the extent of thinning and repair/replace components, and (c) perform follow-up inspections to confirm or quantify and take longer term corrective actions. Technical aspects of the CHECWORKS Code, including the parameters and inputs, are acceptable. However, the EPRI guidance document NSAC-202L-R2 (April 1999) is too general to ensure applicant's flow-accelerated corrosion program will be effective in managing aging in safety-related systems. ***(2) Preventive Actions:*** The rate of E/C is affected by piping material, geometry and hydrodynamic conditions, and operating conditions such as temperature, pH, and dissolved oxygen content. Mitigation is by selecting material considered resistant to E/C, adjusting water chemistry and operating conditions, and improving hydrodynamic conditions through design modifications. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of E/C on the intended function of piping by measuring wall thickness by nondestructive examination and performing analytical evaluations. The inspection program delineated in NUREG-1344 requires ultrasonic or radiographic testing of 10 most susceptible locations and 5 additional locations based on unique operating conditions or special considerations. For each location outside the acceptance guidelines, the inspection sample is expanded based on engineering judgment. Analytical models such as those incorporated into the CHECWORKS code are used to predict E/C in piping systems based on specific plant data including material and hydrodynamic and operating conditions. The inspection data is used to calibrate and benchmark the models and code. ***(4) Detection of Aging Effects:*** Aging degradation of piping and fittings occurs by wall thinning; extent and schedule of inspection assure detection of wall thinning before the loss of intended function of the piping. ***(5) Monitoring and Trending:*** Inspection schedule of NUREG-1344 and EPRI guidelines should provide for timely detection of leakage. Inspections and analytical evaluations are performed during plant outage.	Yes, Element 1 should be further evaluated

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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.1.3 thru D1.1.6	Pressure Boundary and Structural	Upper & Lower Shell, Transition Cone, FW Nozzle & Safe End, FW Impingement Plate Support	Carbon stee (CS), LAS	p to 300°C econdary-ide Water hemistry t 5.3-7.2 Pa	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. GSI-190. N01-7
D1.1.3, D1.1.4	Pressure Boundary and Structural	Upper & Lower Shell, Transition Cone	CS, LAS	p to 300°C econdary-ide Water hemistry t 5.3-7.2 Pa	Loss of Material	Crevice and Pitting Corrosion	ASME Section XI, 1989 Edition. EPRI TR-102134. *Operating Experience* NRC IN 82-37.

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

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

D1. STEAM GENERATOR (Recirculating)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
The program includes preventive measures to mitigate crevice corrosion by recommendations on secondary water chemistry of EPRI TR-102134 and, based on plant specifications, inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a) Table IWC 2500-1, examination category C-A for pressure retaining welds in pressure vessels, and examination category C-H for pressure retaining Class 2 components.			***(1) Scope of Program:*** The program relies on preventive measures to mitigate crevice or pitting corrosion and inservice inspection (ISI) to monitor the effects of corrosion on the intended function of the steam generator shell. ***(2) Preventive Actions:*** Stringent control of secondary water chemistry in accordance with the guidance of EPRI TR-102134, frequent monitoring, and timely corrective action when specified impurity levels are exceeded, prevent or mitigate corrosion. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of corrosion by detection and sizing of flaws in pressure retaining welds and by detection of coolant leakage by inservice inspection (ISI). Inspection requirements of Table IWC 2500-1, examination					Yes Element 4 should be further evaluated

IV01-9

IV01-9

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

Existing Aging Management Program (AMP)			Evaluation and Technical Basis					Further Evaluation
D1.1.5	Upper Assembly and Separators	FW Nozzle & Safe End	LAS	p to 225°C econdary-ide Water hemistry	Wall Thinning	Erosion/ Corrosion (E/C)	*Same as the effect of Erosion/ Corrosion on Item D1.1.2 Steam Nozzle and Safe End*	
D1.1.6	Upper Assembly and Separators	Feedwater Impingement Plate Support	CS	p to 300°C econdary-ide Water hemistry	Loss of section thickness	Erosion	Plant Technical Specifications	

IV01-8

IV01-10

(continued from previous page)
category C-A specify volumetric examination, extending 1/2 in. each side, of all circumferential and tubesheet-to-shell welds, and visual VT-2 (IWA-5240) examination during system leakage and hydrostatic tests of all pressure retaining Class 2 components. Requirements for training and qualification of personnel and performance demonstration for procedures and equipment is in conformance with Appendix VII and VIII of ASME Section XI, or any other formal program approved by the NRC. ***(4) Detection of Aging Effects:*** The extent and schedule of the inspections prescribed by the program are designed to ensure that flaws cannot attain a depth sufficient to threaten the integrity of the welds. However, based on NRC Information notice 90-04 where general corrosion pitting of the shell exists, the program requirements may not be sufficient to differentiate isolated cracks from inherent geometric conditions, and additional inspection procedures may be required. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI should provide for timely detection of leakage. System leakage test is conducted prior to plant startup following each refueling outage, and hydrostatic test at or near the end of each inspection interval. ***(6) Acceptance Criteria:*** Any defect detected is compared with the requirements of Section XI, IWC 3510. Any relevant conditions that may be detected during the leakage and hydrostatic tests are evaluated in accordance with IWC-3410 and IWC-3516. ***(7) Corrective Actions:*** Welds containing flaws that exceed the maximum permissible size must be repaired. Repair and replacement are in conformance with IWA-4000. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls, in conjunction with NRC oversight, are implemented in accordance with the requirements of Appendix B of 10 CFR 50. These requirements, in conjunction with

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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
Same as the effect of Erosion/Corrosion on Item D1.1.2 Steam Nozzle and Safe End			*Same as the effect of Erosion/Corrosion on Item D1.1.2 Steam Nozzle and Safe End*				No
Plant-specific aging management program			Plant-specific aging management program will be evaluated				Yes no AMP

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

Existing Aging Management Program (AMP)			Evaluation and Technical Basis					Further Evaluation
D1.1.7, D1.1.8	Pressure Boundary and Structural	Secondary Manway & Bolting, Secondary Handhold & Bolting	CS, LAS	p to 300°C econdary-ide Water hemistry t 5.3-7.2 Pa	Attrition	Wear		NUREG-1339. EPRI NP-5769. NRC GL 91-17. ASME Section XI, 1989 Edition. *Operating Experience* IEB 82-02.

IV01-12

IV01-13

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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
Recommendations for a comprehensive bolting integrity program delineated in NUREG–1339 on resolution of Generic Safety Issue 29 and implemented through NRC Generic Letter 91-17; additional details on bolting integrity outlined in EPRI NP-5769; and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWC 2500-1, examination categories C-D for pressure retaining bolting in Class 2 components, and testing category C-H for system leakage.			***(1) Scope of Program:*** The staff guidance of NRC Generic Letter (GL) 91-17 provide assurance that plant specific comprehensive bolting integrity programs have been implemented to ensure bolting reliability. The NRC staff recommendations and guidelines for a comprehensive bolting integrity program are delineated in NUREG-1339, and the industry's technical basis for the program is outlined in EPRI NP-5769. ***(2) Preventive Actions:*** Selection of bolting material and the use of lubricants and sealants in accordance with guidelines of EPRI NP-5769 and additional requirements of NUREG 1339, prevent or mitigate degradation and failure of all safety-related closure bolting. ***(3) Parameter Monitored/ Inspected:*** The AMP monitors the effects of aging degradation on the intended function of closure bolting by detection of coolant leakage, and by detection and sizing of cracks by inservice inspection (ISI). Inspection requirements of ASME Section XI, Table IWC 2500-1, examination category C-D for pressure retaining bolting greater than 2 in. in diameter specify volumetric examination of the entire length of bolts. However, because most failures have occurred in fasteners 2 in. or smaller, based on IE Bulletin 82-02, enhanced inspection and improved techniques are recommended. These examinations may be conducted on one component among a group of components with similar design and performing similar functions in the system. Requirements for training and qualification of personnel and performance demonstration for procedures and equipment are in conformance with Appendices VII and VIII of ASME Section XI, and additional requirements of EPRI NP-5769. Requirements of Table IWC 2500-1 category C-H specify visual VT-2 (IWA-5240) examination during system leakage test and hydrostatic test of all pressure retaining Class 2 components. ***(4) Detection of Aging Effects:*** Degradation of the closure bolting can not occur without crack initiation. Also, loss of prestress or attrition of the closure bolting would result in leakage. The extent and schedule of inspection assure detection of aging degradation before the loss of intended function of closure bolting. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI is effective and adequate for timely detection of cracks and leakage. System leakage test is conducted prior to plant startup following each refueling outage, and hydrostatic test is conducted at or near the end of each inspection interval. ***(6) Acceptance Criteria:*** Any cracks in closure bolting are evaluated in accordance with IWC-3100 by comparing ISI results with the acceptance standards of IWC-3400 and IWC-3513 and 3517. Any relevant conditions that may be detected during the leakage and hydrostatic tests are compared with the acceptance standards of IWC-					No

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

Existing Aging Management Program (AMP)			Evaluation and Technical Basis					Further Evaluation
D1.1.7, D1.1.8	Pressure Boundary and Structural	Primary Manways Bolting, Secondary Manways Handholes Bolting	LAS	ir, with etal emperature up to 40°C	Loss of Preload	Stress Relaxation		NUREG-1339. EPRI NP-5769. NRC GL 91-17. ASME Section XI, 1989 Edition.

IVD1-14

IVD1-15

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
			(continued from previous page) administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Significant number of incidents have been reported on bolts and nuts that have failed or become degraded because of SCC. Examples of affected fasteners include (a) steam generator and pressurizer manway closures, (b) valve bonnets and pump flange connections on lines 6 in. or greater, and (c) control rod drive and pressurizer heater connections. The bolting integrity programs developed and implemented in accordance with commitments made in response to NRC communications on bolting events have provided effective means of ensuring bolting reliability.				
Recommendations for a comprehensive bolting integrity program delineated in NUREG–1339 on resolution of Generic Safety Issue 29 and implemented through NRC Generic Letter 91-17; additional details on bolting integrity outlined in EPRI NP-5769. Inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, testing categories B-G-1 and B-G-2 for pressure-retaining bolting and B-P for system leakage (Class 1 components); Subsection IWC, Table IWC 2500-1, testing category C-D for all bolting >2 inches in diameter (Class 2 components).			***(1) Scope of Program:*** The staff guidance of NRC Generic Letter (GL) 91-17 provides assurances that plant specific comprehensive bolting integrity programs have been implemented to ensure bolting reliability. The NRC staff recommendations and guidelines for a comprehensive bolting integrity program are delineated in NUREG-1339, and the industry's technical basis for the program is outlined in EPRI NP-5769. ***(2) Preventive Actions:*** Selection of bolting material and the use of lubricants and sealants in accordance with guidelines of EPRI NP-5769 and additional requirements of NUREG 1339, prevent or mitigate degradation and failure of all safety-related closure bolting. ***(3) Parameter Monitored/ Inspected:*** The AMP monitors the effects of aging degradation on the intended function of closure bolting by detection of coolant leakage. Inspection requirements of ASME Section XI testing category B-P specifies visual VT-2 (IWA-5240) examination of Class 1 pressure retaining components during system leakage and hydrostatic testing. Inspection requirements of ASME Section XI testing categories C-D and C-H specify volumetric inspection of bolting >2 inches in diameter and visual VT-2 (IWA-5240) examination of all pressure retaining components (Class 2 components). ***(4) Detection of Aging Effects:*** Degradation of the closure bolting due to loss of prestress or attrition would result in leakage. The extent and schedule of inspection assure detection of aging degradation before the loss of intended function of closure bolting. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI are effective and adequate for timely detection of cracks and leakage. System leakage test is typically conducted at 40-month intervals. ***(6) Acceptance Criteria:*** Inspection results are evaluated in accordance with IWB-3100 (Class 1 components) and IWC-3100 (Class 2 components). Any relevant conditions that may be detected during the leakage and hydrostatic tests are compared with the				No

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

Existing Aging Management Program (AMP)			Evaluation and Technical Basis				Further Evaluation
D1.1.9	Pressure Boundary and Structural	Instrument Nozzles	Alloy 600	hemically reated orated ater at emperature up to 40°C and 5.5 MPa	Crack Initiation an Growth	SCC, IGSCC, Primary Water Stress Corrosion Cracking (PWSCC)	ASME Section XI, 1989 Edition. Reg. Guide 1.45. NRC GL 97-01, EPRI TR-105714, Plant Technical Specifications. *Operating Experience* NRC IN 90-10. NRC IN 96-11.

NDI-17

NDI-16

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
			(continued from previous page) approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Significant number of incidents have been reported on bolts and nuts that have failed or become degraded. Examples of affected fasteners include (a) steam generator and pressurizer manway closures, (b) valve bonnets and pump flange connections on lines 6 in. or greater, and (c) control rod drive and pressurizer heater connections. The bolting integrity programs developed and implemented in accordance with commitments made in response to NRC communications on bolting events have provided effective means of ensuring bolting reliability.				
Operational limits on leakage from Plant Technical Specifications. Inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination category B-E for pressure-retaining partial penetration welds and testing category B-P for system leakage. PWR primary water chemistry guidelines from EPRI TR-105714.			***(1) Scope of Program:*** The program is focused on managing the effects of stress corrosion cracking (SCC) on the intended function of vessel penetrations. ***(2) Preventive Actions:*** Control of halogens, sulfates, and oxygen in the primary water to less than 0.05, 0.05, and 0.005 ppm, respectively, during operation, and monitor and control of water chemistry during shut down, mitigate potential of SCC, based upon guidelines given in EPRI TR-105714. However, based on the experiences described in NRC Information Notices (IN) 90-10 and 96-11, and Generic Letter (GL) 97-01, the susceptibility of Ni-alloys and welds to primary water SCC has not been addressed adequately, particularly when demineralizer resins contaminate the reactor coolant system. As described in IN 96-11 and implemented through GL 97-01, applicant should review the primary coolant system water chemistry to identify any primary system resin bed intrusions that exceed the PWR primary water chemistry guidelines. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of IGSCC on the intended function of instrument nozzles by performance monitoring and detection of cracks by inservice inspection (ISI). Testing category B-P specifies visual VT-2 (IWA-5240) examination of all pressure retaining components extending to and including the second closed valve at the boundary extremity, during system leakage test (IWB-5221) and system hydrostatic test (IWB-5230). Leakage detection is in conformance with Position C of Regulatory Guide 1.45. However, in compliance with GL 97-01, applicant should determine whether an augmented inspection program is necessary. ***(4) Detection of Aging Effects:*** Aging effects degradation of the vessel penetrations can not occur without crack initiation. In accordance with GL 97-01, applicant should review whether the scope and schedule of inspection, including				Yes Elements 2 thru 6 should be further evaluated

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

Existing Aging Management Program (AMP)			Evaluation and Technical Basis				Further Evaluation
D1. 1.10	Pressure Boundary and Structural	Primary Manway & Bolting	CS, LAS	hemically reated orated ater at emperature up to 40°C and 5.5 MPa	Attrition	Wear	ASME Section XI, 1989 Edition. Reg. Guide 1.45.

IVD1-12

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
			(continued from previous page) program or develop an integrated long-term program which includes periodic inspection and performance monitoring. ***(6) Acceptance Criteria:*** Any IGSCC degradation is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and 3522 for visual examination. Acceptance of the component for continued service is based on the results of visual examination, supplemental volumetric and surface examination (IWB-3200), corrective measure or repair, analytical evaluation, or replacement (IWB-3132). However, in accordance with GL 97-01, the applicant should provide information on crack initiation and crack growth models and the data used to validate these models to verify adequacy of the inspection program and acceptance criteria. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWA-4000 and IWB-4000, and reexamination in accordance with requirements of IWA-2200. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** SCC of Alloy 600 has occurred in domestic and foreign PWRs (IN 90-10). Very little attention has been given to the inspection for SCC in Alloy 600 applications other than that associated with steam generator tubes.		
Inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, testing category B-P for pressure retaining components. Supplemental inspection requirements may be contained in plant-specific Technical Specifications.			***(1) Scope of Program:*** The program is focused on managing the effects of wear due to deterioration of the sealing surfaces of manes and handholds, and closure bolting. ***(2) Preventive Actions:*** Verification of the sealing function by frequent monitoring and timely corrective action prevent or mitigate wear damage of manway and handhold. ***(3) Parameter Monitored/ Inspected:*** The AMP monitors the effects of attrition on the sealing function of manway and handhold by detection of leakage or attrition by inservice inspection (ISI). Inspection requirements of ASME Section XI, Table IWB 2500-1, testing category B-P specify visual VT-2 (IWA-5240) examination of all Class 1 pressure retaining components during system leakage and hydrostatic tests. Leakage detection is in conformance with Position C of Regulatory Guide 1.45. ***(4) Detection of Aging Effects:*** Aging degradation of manway and handhold can not occur without attrition and leakage; extent and schedule of inspection assure detection of leakage or attrition before the loss of the sealing function of manway and handhold. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI should provide for timely		No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

D1. STEAM GENERATOR (Recirculating)

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

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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1. 1.10	Pressure Boundary and Structural	Primary Manway & Bolting	CS, LAS	ir, eaking hemically reated orated ater nd/or team at emperature up to 40°C	Loss of Material	Corrosion/ Boric Acid Wastage of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 Edition.

	(continued from previous page) needed to meet the requirements of IWB-3142 and IWA-5250. The leakage source and areas of wear are located. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval	

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

D1. STEAM GENERATOR (Recirculating)

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Implementation of NRC Generic Letter 88-05, and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWC, Table IWB 2500-1, testing category B-P. IV01-18	***(1) Scope of Program:*** The staff guidance of Generic Letter (GL) 88-05 provides assurances that a program has been implemented consisting of systematic measures to ensure that the effects of corrosion caused by leaking coolant containing boric acid does not lead to degradation of the assurance that the reactor coolant pressure boundary will have a extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture. The program includes (a) determination of principal location, (b) examinations requirements and procedures for locating small leaks, and (c) engineering evaluations and corrective actions. ***(2) Preventive Actions:*** Minimizing reactor coolant leakage by frequent monitoring of the locations where potential leakage could occur and repairing the leaky components as soon as possible, prevent or mitigate boric acid corrosion. ***(3) Parameters Monitored/Inspected:*** The AMP detects coolant leakage by inservice inspection (ISI). Inspection requirements of ASME Section XI, Table IWB 2500-1, testing category B-P specifies visual inspection of all Class 1 pressure-retaining components during system leakage and hydrostatic tests. ***(4) Detection of Aging Effects:*** Aging degradation of the component can not occur without leakage of coolant containing boric acid; extent and schedule of inspection assure detection of leakage before the loss of intended function of the component. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI should provide for timely detection of leakage. System leakage test is conducted at ≈40-month intervals. ***(6) Acceptance Criteria:*** Any relevant conditions that may be detected during the leakage and hydrostatic tests are evaluated in accordance with IWB-3100 and acceptance standards of NRC Generic Letter 88-05. ***(7) Corrective Actions:*** Prior to service, corrective measures are needed to meet the requirements of NRC Generic Letter 88-05. The leakage source and areas of general corrosion are located. Components with local areas of corrosion that reduces the wall thickness by more than 10% require analytical evaluation to demonstrate acceptability. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:***	No

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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1. 1.10	Pressure Boundary and Structural	Primary Manway Bolting	LAS	ir, eaking hemically reated orated ater nd/or team at emperature up to 40°C	Crack Initiation and Growth	Stress Corrosion Cracking (SCC) IV01-19	NUREG-1339. EPRI NP-5769. NRC GL 91-17. IEB 82-02. ASME Section XI, 1989 Edition.

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

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

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

IND1-21

IND1-22

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) accordance with IWB-2400 requires inspection of all bolts, studs, nuts, bushing, and flange surfaces every 10 y. System leakage test is conducted at each refueling outage. ***(6) Acceptance Criteria:*** Any cracks in closure bolting are evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3515 and IWB-3517. Any relevant conditions that may be detected are compared with the acceptance standards of IWB-3522. ***(7) Corrective Actions:*** Repair and replacement is in conformance with IWA-4000 and recommendations and guidelines of EPRI NP-5769. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Significant number of incidents have been reported on bolts and nuts that have failed or become degraded because of boric acid wastage or SCC. Examples of affected fasteners include (a) steam generator and pressurizer manway closures, (b) valve bonnets and pump flange connections on lines 6 in. or greater, and (c) control rod drive and pressurizer heater connections. The present AMP has provided effective means of ensuring bolting reliability.	
Inservice inspection in conformance with Plant Technical Specifications and NRC Regulatory Guide (RG) 1.83 ("Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes." Plugging or repair of defective tubes is governed by NRC Regulatory Guide 1.121 ("Bases for Plugging Degraded PWR Steam Generator Tubes"), but defects at the tube sheet may be shown to satisfy the 1.121 requirements using the P*, F*, or L* criterion. Alternative criteria applicable to cracking at the tube support plates in Westinghouse-designed steam generators under certain circumstances are provided in NRC Generic Letter (GL) 95-05 ("Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking"). Specific guidance on the inservice inspection of tubes is also provided in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5" and NEI 97-06.	***(1) Scope of Program:*** The program deals with the effects of SCC of the steam generator tubes on the integrity of the pressure boundary. Inservice inspection in accordance with Plant Technical Specifications and RG 1.83 deal with the detection of flaws, and RG 1.121 and NRC GL 95-05 describe the basis for determining when flawed tubes must be plugged or repaired. ***(2) Preventive Actions:*** Primary water chemistry guidelines given in EPRI TR-105714 provide guidance on the prevention of PWSCC. ***(3) Parameters Monitored/Inspected:*** The inspection activity in the program monitors flaw size and depth, or, alternatively, remaining sound wall thickness. ***(4) Detection of Aging Effects:*** The extent and schedule of the inspections prescribed by the program are designed to ensure that flaws cannot attain a depth sufficient to threaten the integrity of the tubes. Problems with tube inspection (IN 97-88), e.g., failures to detect some flaws, uncertainties in flaw sizing, inaccuracies in flaw locations, and inability to detect some cracks at locations with dents are addressed by current inspection guidelines given in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5" and NEI 97-06. ***(5) Monitoring and Trending:*** Required inspection intervals (typically ≈18 months or each refueling or maintenance/repair outage) provide for timely detection	No, provided Plant Technical Specifications conform to EPRI inspection guidelines and NEI 97-06 and are adhered to.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

D1. STEAM GENERATOR (Recirculating)

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) ***Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls, in conjunction with NRC oversight, are implemented in accordance with the requirements of Appendix B of 10 CFR 50. Recent experience indicates the importance of performing a complete inspection using the best available techniques and equipment for the reliable detection of tube degradation and to ensure that new forms of degradation are detected. ***(10) Operating Experience:*** The present AMP has been effective in ensuring timely detection and correction of PWSCC degradation in steam generator tubes.	
Inservice inspection in conformance with Plant Technical Specifications and NRC Regulatory Guide 1.83 ("Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes". Plugging or repair of defective tubes is governed by NRC Regulatory Guide 1.121 ("Bases for Plugging Degraded PWR Steam Generator Tubes"), but defects at the tube sheet may be shown to satisfy the 1.121 requirements using the P*, F*, or L* criterion. Alternative criteria applicable to cracking at the tube support plates in Westinghouse-designed steam generators under certain circumstances are provided in NRC Generic Letter 95-05 ("Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking"). Recommendations on secondary water chemistry given in EPRI TR-102134 ("PWR Secondary Water Chemistry Guidelines—Revision 3"). Specific guidance on the inservice inspection of tubes is also provided in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5" and NEI 97-06. Supplemental inspection requirements and plugging criteria may be contained in plant-specific Technical Specifications.	***(1) Scope of Program:*** The program deals with the effects of SCC of the steam generator tubes on the integrity of the pressure boundary. Plant Technical Specifications and Reg. Guide 1.83 deal with the detection of flaws, and Reg. Guide 1.121 and GL 95-05 describe the basis for determining when flawed tubes must be plugged or repaired. ***(2) Preventive Actions:*** Secondary water chemistry guidelines given in EPRI TR-102134 provide guidance on the prevention of ODSCC. ***(3) Parameters Monitored/Inspected:*** The inspection activity in the program monitors flaw size and depth, or, alternatively, remaining sound wall thickness. ***(4) Detection of Aging Effects:*** The extent and schedule of the inspections prescribed by the program are designed to ensure that flaws cannot attain a depth sufficient to threaten the integrity of the tubes. Problems with tube inspection, e.g., failures to detect some flaws, uncertainties in flaw sizing, and inaccuracies in flaw locations, are addressed in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5" and NEI 97-06. ***(5) Monitoring and Trending:*** Required inspection intervals (typically ≈18 months or each refueling or maintenance/repair outage) provide for timely detection of SCC. ***(6) Acceptance Criteria:*** Any cracking detected is compared with the requirements of the Plant Technical Specifications, Reg. Guide 1.121, and GL 95-05. ***(7) Corrective Actions:*** Tubes containing flaws that exceed the maximum permissible size must be plugged or repaired. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls, in conjunction with NRC oversight, are implemented in accordance with the requirements of Appendix B of 10 CFR 50. Recent experience indicates the importance of performing a complete inspection using the best available techniques and equipment for the reliable detection of	No, provided Plant Technical Specifications conform to EPRI inspection guidelines and NEI 97-06 and are adhered to.

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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.2.1	Tube Bundle	Steam Generator Tubes	Alloy 600	p to 300°C econdary-ide Water hemistry t 5.3-7.2 Pa	Crack Initiation and Growth	Intergranu-lar Attack (IGA)	Plant Technical Specifications. Reg. Guide 1.83. Reg. Guide 1.121. NRC GL 95-05, EPRI document "PWR Steam Generator Examination Guidelines: Rev. 5", NEI 97-06
D1.2.1	Tube Bundle	Steam Generator Tubes	Alloy 600	D hemically reated orated ater up to 40°C; D up to 00°C econdary-	Cumulative Fatigue Damage	Fatigue	Plant Technical Specifications. GSI-190.

IVD1-23

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Inservice inspection in conformance with Plant Technical Specifications and NRC Regulatory Guide 1.83 ("Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes". Plugging or repair of defective tubes is governed by NRC Regulatory Guide 1.121 ("Bases for Plugging Degraded PWR Steam Generator Tubes"), but defects at the tube sheet may be shown to satisfy the 1.121 requirements using the P*, F*, or L* criterion. Alternative criteria applicable to cracking at the tube support plates in Westinghouse-designed steam generators under certain circumstances are provided in NRC Generic Letter 95-05 ("Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking"). Recommendations on secondary water chemistry given in EPRI TR-102134 ("PWR Secondary Water Chemistry Guidelines—Revision 3"). Specific guidance on the inservice inspection of tubes is also provided in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5" and NEI 97-06. Supplemental inspection requirements and plugging criteria may be contained in plant-specific Technical Specifications.	***(1) Scope of Program:*** The program deals with the effects of IGA of the steam generator tubes on the integrity of the pressure boundary. Plant Technical Specifications and Reg. Guide 1.83 deal with the detection of flaws, and Reg. Guide 1.121 and GL 95-05 describe the basis for determining when flawed tubes must be plugged or repaired. ***(2) Preventive Actions:*** Secondary water chemistry guidelines given in EPRI TR-102134 provide guidance on the prevention of IGA. ***(3) Parameters Monitored/Inspected:*** The inspection activity in the program monitors the extent of degradation or, alternatively, remaining sound wall thickness. ***(4) Detection of Aging Effects:*** The extent and schedule of the inspections prescribed by the program are designed to ensure that flaws cannot attain a depth sufficient to threaten the integrity of the tubes. Problems with tube inspection, e.g., failures to detect and properly characterize IGA, are addressed in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5" and NEI 97-06. ***(5) Monitoring and Trending:*** Required inspection intervals (typically ≈18 months or each refueling or maintenance/repair outage) provide for timely detection of IGA. ***(6) Acceptance Criteria:*** Any degradation detected is compared with the requirements of the Plant Technical Specifications, Reg. Guide 1.121, and GL 95-05. ***(7) Corrective Actions:*** Tubes containing flaws that exceed the maximum permissible size must be plugged or repaired. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls, in conjunction with NRC oversight, are implemented in accordance with the requirements of Appendix B of 10 CFR 50. Recent experience indicates the importance of performing a complete inspection using the best available techniques and equipment for the reliable detection of tube degradation and to ensure that new forms of degradation are detected. ***(10) Operating Experience:*** The present AMP has been effective in ensuring timely detection and correction of IGA degradation in steam generator tubes.	No, provided Plant Technical Specifications conform to EPRI inspection guidelines and NEI 97-06 and are adhered to.
Components have been designed or evaluated for fatigue for a 40 y design life, according to Plant Technical Specifications or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

D1. STEAM GENERATOR (Recirculating)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.2.1	Tube Bundle	Steam Generator Tubes	Alloy 600	p to 300°C econdary- ide Water hemistry t 5.3-7.2 Pa	Loss of section thickness	Fretting and wear	Plant Technical Specifications. Reg. Guide 1.83. Reg. Guide 1.121. EPRI document "PWR Steam Generator Examination Guidelines: Rev. 5", NEI 97-06.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

D1. STEAM GENERATOR (Recirculating)

Existing Aging Management Program (AMP)			Evaluation and Technical Basis				Further Evaluation
D1.2.1	Tube Bundle	Steam Generator Tubes	Alloy 600	p to 300°C econdary- ide Water hemistry t 5.3-7.2 Pa	Loss of material	General and Pitting corrosion (OD) IVD1-24	lant Technical pecifications. Reg. Guide 1.83. Reg. Guide 1.121. EPRI document "PWR Steam Generator Examination Guidelines: Rev. 5", NEI 97-06.

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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
Inservice inspection in conformance with Plant Technical Specifications and NRC Regulatory Guide 1.83 ("Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes". Plugging or repair of defective tubes is governed by NRC Regulatory Guide 1.121 ("Bases for Plugging Degraded PWR Steam Generator Tubes"). Specific guidance on the inservice inspection of tubes is also provided in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5." Specific guidance on the inservice inspection of tubes is also provided in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5" and NEI 97-06. Supplemental inspection requirements and plugging criteria may be contained in plant-specific Technical Specifications.							***(1) Scope of Program:*** The program deals with the effects of the fretting and wear of the steam generator tubes on the integrity of the pressure boundary. Plant Technical Specifications and Reg. Guide 1.83 deal with the detection of flaws, and Reg. Guide 1.121 describes the basis for deciding when flawed tubes must be plugged or repaired. ***(2) Preventive Actions:*** The program provides no guidance or recommendations on measures to prevent fretting and wear. ***(3) Parameters Monitored/Inspected:*** The inspection activity in the program monitors flaw size and depth, or, alternatively, remaining sound wall thickness. ***(4) Detection of Aging Effects:*** The extent and schedule of the inspections prescribed by the program are designed to ensure that flaws cannot attain a depth sufficient to threaten the integrity of the tubes. Guidance on tube inspection is given in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5" and NEI 97-06. ***(5) Monitoring and Trending:*** Required inspection intervals (typically ≈18 months or each refueling or maintenance/repair outage) should provide for timely detection of fretting and wear. ***(6) Acceptance Criteria:*** Any cracks detected are compared with the requirements of the Plant Technical Specifications and Reg. Guide 1.121. ***(7) Corrective Actions:*** Tubes containing flaws that exceed the maximum permissible size must be plugged or repaired. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls, in conjunction with NRC oversight, are implemented in accordance with the requirements of Appendix B of 10 CFR 50. This should continue to ensure the timely detection and correction of fretting and wear but cannot be depended upon to prevent further degradation. ***(10) Operating Experience:*** To date, the present program has been effective in ensuring the timely detection and correction of fretting and wear degradation in steam generator tubes.	No, provided Plant Technical Specifi-cations conform to EPRI inspection guidelines and NEI 97-06 and are adhered to.
Inservice inspection in conformance with Plant Technical Specifications and NRC Regulatory Guide 1.83 ("Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes". Plugging or repair of defective tubes is governed by NRC Regulatory Guide 1.121 ("Bases for Plugging Degraded PWR Steam Generator Tubes"). Recommendations on secondary water chemistry given in EPRI TR-102134 ("PWR Secondary Water Chemistry Guidelines—Revision 3"). Specific guidance on the inservice inspection of tubes is also provided in the EPRI document "PWR Steam							***(1) Scope of Program:*** The program deals with the effects of pitting of the steam generator tubes on the integrity of the pressure boundary. Plant Technical Specifications and Reg. Guide 1.83 deal with the detection of flaws, and Reg. Guide 1.121 describes the basis for deciding when flawed tubes must be plugged or repaired. ***(2) Preventive Actions:*** Guidance provided in EPRI TR-102134 can significantly reduce secondary-side pitting. ***(3) Parameters Monitored/Inspected:*** The inspection activity in the program monitors flaw size and depth, or, alternatively, remaining sound wall thickness. ***(4) Detection of Aging Effects:*** The extent and schedule of the inspections prescribed by the program are designed to ensure that flaws cannot attain a depth sufficient to threaten the integrity of the tubes. Specific guidance on	No, provided Plant Technical Specifi-cations conform to EPRI inspection guidelines and NEI 97-06 and are adhered to.

EVANS, Bob

From: epatel@peco-energy.com
Sent: Wednesday, May 31, 2000 4:23 PM
To: rdyle@msn.com; rce@nei.org; fpolaski@peco-energy.com; djw@nei.org; rldyle@southernco.com; mgsemmle@duke-energy.com
Cc: epatel@peco-energy.com
Subject: Conference call with NEI

I have, as discussed set up the conference call between NEI, Fred and myself, Robin Dyle and Mike Semmler for Friday, June 2, 2000 at 2:00pm EDT.

Please call the following number to set up the call: 1-888-271-0949.
The participant code is : 974927.

The purpose is to discuss the upcoming meeting with NRC on June 6, 2000 to address GALL Section IVA1, B1 and C1 industry comments.

Erach Patel
Senior Engineer
License Renewal Project
PECO Energy Company
610-640-6417/ Fax 610-640-6006
epatel@peco-energy.com

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

Existing Aging Management Program (AMP)			Evaluation and Technical Basis				Further Evaluation
D1.2.1	Tube Bundle	Steam Generator Tubes	Alloy 600	p to 300°F econdary- de Water hemistry t 5.3-7.2 Pa	Deforma-tion (denting)	Corrosion of tube support plate at intersec-tions with tubes	Plant Technical Specifications. Reg. Guide 1.83. Reg. Guide 1.121. NRC GL 95-05.

IVD1-21

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
			(continued from previous page) outage) should provide for timely detection of virtually all forms of pitting. ***(6) Acceptance Criteria:*** Any cracks detected are compared with the requirements of the Plant Technical Specifications and Reg. Guide 1.121. ***(7) Corrective Actions:*** Tubes containing flaws that exceed the maximum permissible size must be plugged or repaired (e.g., by sleeving). ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls, in conjunction with NRC oversight, are implemented in accordance with the requirements of Appendix B of 10 CFR 50. This should continue to ensure the timely detection and correction of virtually all forms of pitting but cannot be depended upon to prevent further degradation. ***(10) Operating Experience:*** To date, the present program has been effective in ensuring the timely detection and correction of pitting in steam generator tubes.	
		Inservice inspection in conformance with Plant Technical Specifications and NRC Regulatory Guide 1.83 ("Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes". Plugging or repair of defective tubes is governed by NRC Regulatory Guide 1.121 ("Bases for Plugging Degraded PWR Steam Generator Tubes"). Recommendations on secondary water chemistry given in EPRI TR-102134 ("PWR Secondary Water Chemistry Guidelines—Revision 3"). Specific guidance on the inservice inspection of tubes is also provided in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5" and NEI 97-06. Supplemental inspection requirements and plugging criteria may be contained in plant-specific Technical Specifications.	***(1) Scope of Program:*** The program deals with the effects of denting of the steam generator tubes on the integrity of the pressure boundary. Plant Technical Specifications and Reg. Guide 1.83 deal with the detection of flaws, and Reg. Guide 1.121 describes the basis for deciding when degraded tubes must be plugged or repaired. ***(2) Preventive Actions:*** Guidance provided in EPRI TR-102134 can significantly reduce denting. ***(3) Parameters Monitored/Inspected:*** The inspection activity in the program monitors extent of degradation. ***(4) Detection of Aging Effects:*** The extent and schedule of the inspections prescribed by the program are designed to ensure that degradation cannot proceed to an extent sufficient to threaten the integrity of the tubes. ***(5) Monitoring and Trending:*** Required inspection intervals (typically ≈18 months or each refueling or maintenance/repair outage) should provide for timely detection of denting. ***(6) Acceptance Criteria:*** Any denting detected is compared with the requirements of the Plant Technical Specifications. ***(7) Corrective Actions:*** Tubes containing denting that exceed the maximum permissible size must be plugged or repaired. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls, in conjunction with NRC oversight, are implemented in accordance with the requirements of Appendix B of 10 CFR 50. This should continue to ensure the timely detection and correction of virtually all forms of denting degradation but cannot be depended upon to prevent further degradation. ***(10) Operating Experience:*** The present AMP has been effective in ensuring timely detection and correction of	No

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

Existing Aging Management Program (AMP)			Evaluation and Technical Basis				Further Evaluation
D1.2.4	Tube Bundle	Tube Mechanical Plugs (Westinghouse)	Alloy 600, Alloy 690	hemically reated orated ater at emperature up to 40°C and 5.5 MPa	Crack Initiation and Growth	PWSCC	Plant Technical Specifications; NRC IN 89-33; NRC BL 89-01 & 89-01, S 1 and S 2; NRC IN 94-87; WCAP-12244; EPRI TR-102134
D1.2.4	Tube Bundle	Tube Mechanical Plugs (Babcock & Wilcox)	Alloy 600, Alloy 690	hemically reated orated ater at emperature up to 40°C and 5.5 MPa	Crack Initiation and Growth	PWSCC	Plant Technical Specifications; NRC Reg. Guide 1.83; NRC IN 89-65

IVD1-25

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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
	Inservice inspection in conformance with Plant Technical Specifications and NRC Regulatory Guide 1.83 ("Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes". Correlations for estimating plug life contained in Westinghouse reports WCAP-12244 and WCAP 12245. Specific guidance on the inservice inspection of plugs is provided in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5" and NEI 97-06. Supplemental inspection requirements may be contained in plant-specific Technical Specifications.				***(1) Scope of Program:*** The program deals with the periodic inspection of steam generator tube plugs. ***(2) Preventive Actions:*** Guidance on primary water chemistry provided in EPRI TR-102134 can significantly reduce PWSCC. The program also recommends that certain susceptible heats be avoided. ***(3) Parameters Monitored/ Inspected:*** The inspection activity in the program monitors flaw size and depth. ***(4) Detection of Aging Effects:*** The extent and schedule of the inspections prescribed by the program are designed to ensure that flaws cannot attain a depth sufficient to threaten the integrity of the tubes. Past problems with failure to detect flaws in plugs have led to leaking or failed plugs in several plants ((BL 89-01; 89-01, S. 1; 89-01, S. 2; IN 89-33, 94-87). However, improved detection procedures are provided in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5" and NEI 97-06. ***(5) Monitoring and Trending:*** Required inspection intervals (typically ≈18 months or each refueling or maintenance/repair outage) are intended to provide for timely detection of SCC. However, note item (4) immediately above ***(6) Acceptance Criteria:*** Any cracking detected requires replacement. Plug lives can be estimated using procedures in WCAP-12244 and 12245 and compared with the limits given in those reports. ***(7) Corrective Actions:*** Tube plugs containing flaws or having insufficient estimated lives must be replaced. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls, in conjunction with NRC oversight, are implemented in accordance with the requirements of Appendix B of 10 CFR 50. This should ensure the timely detection and correction of cracking. ***(10) Operating Experience:*** Problems appear to have been related to susceptible heats of material and improper heat treatment. However, any Alloy 600 mechanical plugs remaining in service must be considered susceptible (BL 89-01; 89-01, S 1; 89-01, S 2; IN 89-33, 94-87).	No, provided Plant Technical Specifi-cations conform to EPRI inspection guidelines and NEI 97-06 and are adhered to.
	Inservice inspection in conformance with Plant Technical Specifications and NRC Regulatory Guide 1.83 ("Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes". Specific guidance on the inservice inspection of plugs is provided in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5" and NEI 97-06. Supplemental inspection requirements may be contained in plant-specific				***(1) Scope of Program:*** The program deals with the periodic inspection of steam generator tube plugs. ***(2) Preventive Actions:*** Guidance on primary water chemistry provided in EPRI TR-102134 can significantly reduce PWSCC. The program also recommends that certain susceptible heats be avoided. ***(3) Parameters Monitored/ Inspected:*** The inspection activity in the program monitors flaw size and depth. ***(4) Detection of Aging Effects:*** The extent and schedule of the inspections prescribed by the program are designed to ensure that flaws cannot attain a depth sufficient to	No, provided Plant Technical Specifi-cations conform to EPRI inspection guidelines and NEI

Existing Aging Management Program (AMP)				Evaluation and Technical Basis			Further Evaluation
D1.2..5	Tube Bundle	Tubes in the Region of Repair Sleeves	Alloy 600	hemically reated orated ater at emperature up to 40°C and 5.5 MPa	Crack Initiation and Growth	PWSCC	Plant Technical Specifications; NRC Reg. Guide 1.83; NRC IN 94-05; EPRI document "PWR Steam Generator Examination Guidelines: Rev. 5"; NEI 97-06
D1.3.1	Upper Assembly and Separators	Feedwater Inlet Ring and Supports	CS	p to 300°C econdary-ide Water hemistry t 5.3-7.2 Pa	Loss of Material	Erosion/ corrosion	ASME Section XI, 1989 Edition.; NRC IN 91-19; NRC LER 50-362/90-05-01; Combustion Engineering Info-

IVD1-27

IVD1-26

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

	(continued from previous page) compared with the requirements of the Plant Technical Specifications. ***(7) Corrective Actions:*** Tube plugs containing flaws must be replaced. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls, in conjunction with NRC oversight, are implemented in accordance with the requirements of Appendix B of 10 CFR 50. This should continue to ensure the timely detection and correction of cracking. ***(10) Operating Experience:*** Problems appear to have been related to susceptible heats of material and improper heat treatment. However, any Alloy 600 mechanical plugs remaining in service must be considered susceptible.	
Inservice inspection in conformance with Plant Technical Specifications and NRC Regulatory Guide 1.83 ("Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes". Specific guidance on the inservice inspection of repair sleeves is provided in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5" and NEI 97-06. Supplemental inspection requirements may be contained in plant-specific Technical Specifications.	***(1) Scope of Program:*** The program deals with the periodic inspection of steam generator tube repair sleeves and the associated parent tubing material. ***(2) Preventive Actions:*** Guidance on primary water chemistry provided in EPRI TR-102134 can significantly reduce PWSCC. ***(3) Parameters Monitored/Inspected:*** The inspection activity in the program monitors flaw size and depth. ***(4) Detection of Aging Effects:*** The extent and schedule of the inspections prescribed by the program are designed to ensure that flaws cannot attain a depth sufficient to threaten the integrity of the tubes. ***(5) Monitoring and Trending:*** Required inspection intervals (typically ≈18 months or each refueling or maintenance/repair outage) provide for timely detection of SCC. ***(6) Acceptance Criteria:*** Any cracking detected is compared with the requirements of the Plant Technical Specifications. ***(7) Corrective Actions:*** Tubes containing flaws must be repaired or plugged. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls, in conjunction with NRC oversight, are implemented in accordance with the requirements of Appendix B of 10 CFR 50. This should continue to ensure the timely detection and correction of cracking. ***(10) Operating Experience:*** Problems appear to have been related to susceptible heats of material and improper heat treatment (NRC Information Notice 94-05).	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

D1. STEAM GENERATOR (Recirculating)

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Periodic inservice inspections in conformance with Plant Technical Specifications. Additional recommendations on inservice inspection are contained in Combustion Engineering Info-bulletin 90-04: ("Feedwater Distribution System Degradation"), and supplemental inspection requirements may be included in plant-specific Technical Specifications.	***(1) Scope of Program:*** The program deals with the periodic inspection of pressure retaining components and integral attachments (including feedwater distribution system components). ***(2) Preventive Actions:*** The program provides no guidance or recommendations on measures to prevent degradation. ***(3) Parameters Monitored/Inspected:*** The program monitors wall thinning and degradation of supports and attachments. ***(4) Detection of Aging Effects:*** The extent and schedule of the inspections prescribed by the program should ensure that thinning and degradation cannot proceed sufficiently to threaten the integrity of the welds. ***(5) Monitoring and Trending:*** Required inspection intervals should provide for timely detection of all	No

D1.4.1	Piping and Fittings	Main Steam	CS	p to 300°C team	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. ANSI B31.1, GSI-190.
D1.4.1	Piping and Fittings	Main Steam	CS	p to 300°C team	Wall Thinning	E/C	*Same as the effect of Erosion/Corrosion on Item D1.1.2 Steam Nozzle and Safe End*
D1.4.2	Piping and Fittings	Feedwater	CS	p to 225°C econdary-	Cumulative Fatigue	Fatigue	ASME Section III, 1989 Edition.

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

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D1.4.2	Piping and Fittings	Feedwater	CS	p to 225°C econdary-ide Water hemistry	Wall Thinning	E/C	*Same as the effect of Erosion/Corrosion on Item D1.1.2 Steam Nozzle and Safe End*
D1.4.3	Piping and Fittings	Auxiliary Feedwater	CS	90°C econdary-ide Water hemistry	Wall Thinning	E/C	*Same as the effect of Erosion/Corrosion on Item D1.1.2 Steam Nozzle and Safe End*

	(continued from previous page) forms of degradation. ***(6) Acceptance Criteria:*** Any defect detected is compared with the requirements of the Plant Technical Specifications and with the recommendations of Combustion Engineering Info-bulletin 90-04. These requirements should ensure structural integrity. ***(7) Corrective Actions:*** Excessively degraded components must be repaired or replaced ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls, in conjunction with NRC oversight, are implemented in accordance with the requirements of Appendix B of 10 CFR 50. These requirements should continue to ensure the timely detection and correction of all forms of degradation. ***(10) Operating Experience:*** This form of degradation has been detected only in certain Combustion Engineering System 80 steam generators, where it has been successfully dealt with (NRC Information Notice I91-19; NRC LER 50-362/90-05-01).	
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), ANSI B31.1, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Same as the effect of Erosion/Corrosion on Item D1.1.2 Steam Nozzle and Safe End	*Same as the effect of Erosion/Corrosion on Item D1.1.2 Steam Nozzle and Safe End*	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

D1. STEAM GENERATOR (Recirculating)

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a), Subsection NB, or ANSI B31.1, or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Same as the effect of Erosion/Corrosion on Item D1.1.2 Steam Nozzle and Safe End	*Same as the effect of Erosion/Corrosion on Item D1.1.2 Steam Nozzle and Safe End*	No
Same as the effect of Erosion/Corrosion on Item D1.1.2 Steam Nozzle and Safe End	*Same as the effect of Erosion/Corrosion on Item D1.1.2 Steam Nozzle and Safe End*	No

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

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

ND1-28

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
The program includes preventive measures to mitigate crevice corrosion by control of secondary water chemistry and, based on plant specifications, inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Table IWC 2500-1, examination category C-H for pressure retaining Class 2 components.	***(1) Scope of Program:*** The program relies on preventive measures to mitigate general, crevice, or pitting corrosion, and inservice inspection (ISI) to monitor the effects of corrosion on the intended function of the main steam system components. ***(2) Preventive Actions:*** Mitigation is by monitoring and control of secondary water chemistry and demineralized water chemistry to minimize concentration of corrosive impurities such as chlorides, sulfates, and oxygen. Chemical species and impurities are monitored either continuous, daily, weekly, or as required based on plant operating conditions. Corrective actions are taken when specified values are exceeded. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of corrosion by detection of coolant leakage by inservice inspection (ISI). Inspection requirements of ASME Section XI specify visual VT-2 (IWA-5240) examination during system leakage test and hydrostatic test of all pressure retaining Class 2 components up to the first normally closed valve, according to Table IWC 2500-1 category C-H. ***(4) Detection of Aging Effects:*** Degradation of the component due to corrosion can not occur without leakage of coolant; extent and schedule of inspection assure detection of leakage before the loss of intended function of the component. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI should provide for timely detection of leakage. System leakage test is typically conducted at 40 month intervals. ***(6) Acceptance Criteria:*** Any relevant conditions that may be detected during the leakage and hydrostatic tests are evaluated in accordance with IWC-3100 and acceptance standards of IWC-3400 and IWC-3516. ***(7) Corrective Actions:*** Prior to service, corrective measures are needed to meet the requirements of IWC-3132 and IWA-5250. Repair and replacement are in conformance with IWA-4000 and IWA-4000. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** No significant corrosion related problem has been reported for piping and fittings in secondary side systems of steam generators.	No
Same as the effect of Crevice and Pitting Corrosion on Item D1.4.1 Piping and Fittings in Main Steam System	*Same as the effect of Crevice and Pitting Corrosion on Item D1.4.1 Piping and Fittings in Main Steam System*	No
Same as the effect of Erosion/Corrosion on Item D1.1.2 Steam Nozzle and Safe End	*Same as the effect of Erosion/Corrosion on Item D1.1.2 Steam Nozzle and Safe End*	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

D1. STEAM GENERATOR (Recirculating)

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

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IV

GALL SECTION	TITLE	ITEM NO	PAGE	REVIEWER	COMMENT
IV.D1-1	Steam Generator (Recirculating)	Various			Erosion Corrosion is listed as an Aging Mechanism for various subcomponents throughout this section. Comments on the applicability of this Aging Mechanism to these subcomponents are included in the specific comments to follow. For any subcomponents that are retained with Erosion Corrosion listed as an Aging Mechanism, and that are managed by the Flow Accelerated Corrosion Program, the References, Existing AMP, Evaluation and Technical Basis, and Further Evaluation entries should be revised to correspond to the provided example.
IV.D1-2	Steam Generator (Recirculating)		4		It is not clear how the SG primary side is considered. There appears to be no entries for the lower head in this section or in section IV.C2. In any event, there is no consideration of stainless steel as a material. This omission needs to be clarified.
IV.D1-3	Steam Generator (Recirculating)	D1.1.2	4		Steam Nozzle & Safe End should not list Erosion/ Corrosion as an Aging Mechanism. Erosion/ Corrosion is not a valid Aging Mechanism for the dry steam coming out of the S/G.
IV.D1-4	Steam Generator (Recirculating)	D1.1.2, D1.1.3, D1.1.4	4		Chapter IV should reference the new EPRI PWR Primary Water Chemistry Guidelines, Rev. 4, November 1999, EPRI TR-102134. This reference should be updated at the end of the chapter.
IV.D1-5	Steam Generator (Recirculating)	D1.1.2	4		Erosion Corrosion is listed as an Aging Mechanism for the SG Outlet Nozzles. (See separate comments on the Aging Mechanism elsewhere. Comments to follow assume that the Aging Mechanism is retained in Report despite earlier comment.) The "Existing AMP" entry concentrates solely on the Erosion Corrosion Program for this Aging Mechanism. CCNPP credited the ISI program in this instance, and this position was accepted in the SER. The Report should consider this option also. For the actual Flow Assisted Corrosion Program details, the References, Existing AMP, Evaluation and Technical Basis, and Further Evaluation entries should be rewritten to correspond to the provided example

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IV

GALL SECTION	TITLE	ITEM NO	PAGE	REVIEWER	COMMENT
IV.D1-6	Steam Generator (Recirculating)		5		Fatigue is listed as an Aging Mechanism. "Existing AMP" entry states that components have been designed for fatigue for a 40 year design life in accordance with ASME III. This not strictly true; it has been assumed that the components in question will experience certain operational stresses during the plant lifetime, for example, less than or equal to 500 thermal cycles caused by plant startups and shutdowns. This number of transients will have originally been estimated with some consideration of a 40 year plant life, but there is no explicit correlation to a time period in this evaluation. Also, simply identifying Fatigue as a TLAA does not identify an AMP per se. More information should be provided, for example, relevant to Fatigue Monitoring Programs that calculate Cumulative Usage Factors based on partial cycles.
IV.D1-7	Steam Generator (Recirculating)	D1.1.3, D1.1.4, D1.4.1,	6, 38		CC/Pitting is listed as an Aging Mechanism for the listed sections. BAW-2270 indicates that crevice corrosion and pitting are only plausible for carbon steel in treated water if oxygen levels are > 100 ppb. Chemistry controls in accordance with industry guidelines, assure this requirement is met. Pitting is also identified as requiring stagnant or low flow conditions (flow velocity < 3 ft/sec), and halides or copper > 150 ppb or sulfates > 100 ppb. Therefore, for Crevice Corrosion and Pitting, the applicable chemistry controls in accordance with industry guidelines alone should be credited as the Existing AMP in the Report. A similar position was accepted in the Oconee SER. For the actual Chemistry Program details, the References, Existing AMP, Evaluation and Technical Basis, and Further Evaluation entries should be rewritten to correspond to the provided example.
IV.D1-8	Steam Generator (Recirculating)	D1.1.5, D1.1.6	8		Structure and Component section should be 'Pressure Boundary and Structure' not 'Upper Assembly and Separators.'
IV.D1-9	Steam Generator (Recirculating)	D1.1.5	8		Erosion / Corrosion is not a significant Aging Mechanism for the SG Feedwater nozzle due to material resistance and low relative flow. This position was accepted in NUREG-1705.

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IV

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV.D1-10	Steam Generator (Recirculating)	D1.1.5	8		Erosion Corrosion is listed as an Aging Mechanism for the Feedwater Nozzles. (This mechanism was considered Not Plausible for CCNPP; see separate comments on the Aging Mechanism elsewhere. Comments to follow assume that the Aging Mechanism is retained in Report despite earlier comment.) The "Existing AMP" entry concentrates solely on the Erosion Corrosion Program for this Aging Mechanism. A prospective applicant may wish to credit the ISI Program instead. The Report should consider this option also. For the actual Flow Assisted Corrosion Program or ISI Program details, the References, Existing AMP, Evaluation and Technical Basis, and Further Evaluation entries should be rewritten to correspond to the provided example.
IV.D1-11	Steam Generator (Recirculating)	D1.1.6	8		The Feedwater Impingement Plate and Supports perform no passive intended function. The only passive intended function for the Steam Generator is the pressure boundary function. These items should be removed from the Report.
IV.D1-12	Steam Generator (Recirculating)	D1.1.7, D1.1.8, D1.1.10	10		Items list Wear as an Aging Mechanism for SG Manways and Handhole and Bolting. Wear is not plausible for these items because they do not experience relative motion during normal operation. Wear that results from maintenance activities during outages should not be considered an aging effect for license renewal. Such damage caused by poor work practices should not be equated with aging. Therefore, Wear should be not listed for these items. This position was accepted in NUREG-1705 (see also letter BGE to NRC 2/4/99).

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IV

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV.D1-13	Steam Generator (Recirculating)	D1.1.7, D1.1.8	10		Wear is listed as an Aging Mechanism for the Secondary Manhole/Handhold & Bolting. (This mechanism was considered Not Plausible for CCNPP; see separate comments on the Aging Mechanism elsewhere. Comments to follow assume that the Aging Mechanism is retained in Report despite earlier comment.) The Aging Mechanism of concern for this entry is Wear but the "Evaluation and Technical Basis" information extensively discusses SCC (cracking) and Stress Relaxation. There appears to be little or no specific information in the "Evaluation and Technical Basis" entry on aging management for Wear. This entry does not appear to offer any useful information for the prospective applicant. Remove the information on cracking and Stress Relaxation and add information on Wear.
IV.D1-14	Steam Generator (Recirculating)	D1.7, D1.8	12		Items list Stress Relaxation as an Aging Mechanism for SG Manway and Handhold Bolting. The operating temperature of the RCS is well below the threshold for creep, the precursor for Stress Relaxation, to occur. Additionally, the neutron field around all components other than the vessel is less than that required to reduce the temperature threshold. This Aging Mechanism should not be considered. This position was accepted in NUREG-1705.
IV.D1-15	Steam Generator (Recirculating)		12		Stress Relaxation is listed as an Aging Mechanism for Bolting. (This mechanism was considered Not Plausible for CCNPP; see separate comments on the Aging Mechanism elsewhere. Comments to follow assume that the Aging Mechanism is retained in Report despite earlier comment.) The Aging Mechanism of concern for this entry is Stress Relaxation but the "Evaluation and Technical Basis" information also discusses Cracking which has nothing to do with Stress Relaxation. Information on cracking should be removed from the Report. Also, the "Evaluation and Technical Basis", Item (2), includes selection of bolting as part of the AMP. Material selection is not an AMP; it is a design feature and as such should not be listed or should be included in the Material entry.

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IV

GALL SECTION	TITLE	ITEM NO	PAGE	REVIEWER	COMMENT
IV D1-16	Steam Generator (Recirculating)	D1.1.9, D1.2.1	14		Chapter IV should reference the new EPRI PWR Primary Water Chemistry Guidelines Rev. 4 March 1999, EPRI TR-105714. This reference is missing completely at the end of the chapter.
IV.D1-17	Steam Generator (Recirculating)	D1.1.9	14		SCC/IGSCC/PWSCC is listed as an Aging Mechanism for Alloy 600 Instrument Nozzles. This entry seems to be analogous to Item IV.C2.5.6 for SCC of Pressurizer Instrument Nozzles. While some of the conclusions appear to be the same, the verbiage is completely different. For the sake of consistency, it would seem to benefit the prospective applicant if these entries were identical. This comment likely applies in numerous other cases throughout the Report.
IV.D1-18	Steam Generator (Recirculating)	D1.1.10	18		Corrosion/Boric Acid Wastage is listed as an Aging Mechanism. The "Existing AMP" entry lists both a Boric Acid Corrosion Inspection Program (meeting the requirements of GL 88-05) and ISI. The Boric Acid Corrosion Inspection Program itself was deemed adequate for aging management in NUREG-1705. The References, Existing AMP, Evaluation and Technical Basis, and Further Evaluation entries should be rewritten to correspond to the provided example.
IV.D1-19	Steam Generator (Recirculating)	D1.1.10	20		SCC is listed as an Aging Mechanism for the Primary Manway & Bolting. In the "Evaluation and Technical Basis", Item (4), mention is made of loss of prestress and attrition, neither of which effect should be of concern for this entry. This information should be removed from the Report. More importantly, in the "Evaluation and Technical Basis", Item 10, it is indicated that Boric Acid Wastage is a parallel concern for such bolting. Since it is likely that the contaminants and the aqueous environment required for SCC would be provided in this instance by borated water leakage, and since boric acid corrosion would propagate at a much faster rate than SCC, and since programs to monitor boric acid corrosion typically inspect potentially degraded areas each refueling interval, it would be more appropriate to characterize the Aging Mechanism of concern in this case as only Corrosion/Boric Acid Wastage, and delete this entry in favor of the previous entry.

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IV

GALL SECTION	TITLE	ITEM NO	PAGE	REVIEWER	COMMENT
IV.D1-20	Steam Generator (Recirculating)	D1.2.1	22 thru 30		For the Aging Mechanisms PWSCC, ODSCC, IGA, Fretting & Wear, General and Pitting Corrosion, and Denting, a single comprehensive SG inspection program to manage all of these effects is contained in NEI 97-06 (latest revision). This document includes all other references currently presented in the "Existing AMP" column, and also provides the details to be included under the "Technical Basis" column. Remove the other information and retain only NEI 97-06.
IV.D1-21	Steam Generator (Recirculating)	D1.2.1	22		For the PWSCC entry for SG Tubes and Sleeves, Environment is given as Secondary side water but should be Primary side.
IV.D1-22	Steam Generator (Recirculating)	D1.2.1	22 thru 30		For all Aging Mechanisms for the SG Tubes, no mention is made of the specific type of inspections performed. They are just referred to as "inspection activity" or "inspections". It seems the entries are referring to eddy current testing. If this is the case it should be acknowledged.
IV.D1-23	Steam Generator (Recirculating)	D1.2.1	26		Item lists Fatigue as an Aging Mechanism for SG Tubes. Fatigue of tubes is not plausible for CE steam generators. Tube failures due to high-cycle fatigue have not been observed in CE steam generators. Remove Fatigue as an Aging Mechanism for SG Tubes. This position was accepted in NUREG-1705.
IV.D1-24	Steam Generator (Recirculating)	D1.2.1	28		Item lists General Corrosion as an Aging Mechanism for Alloy 600 SG Tubes. General corrosion of Alloy 600 tubes should only be plausible for plants that use or have used phosphate steam generator chemistry control. General corrosion (or phosphate wastage) is not plausible if this form of chemistry control was never used. This distinction should be included in the Report
IV.D1-25	Steam Generator (Recirculating)	D1.2.4	32		Item lists PWSCC as an Aging Mechanism for Alloy 600 and 690 SG Tube Plugs. This mechanism is plausible for plugs fabricated from Alloy 600 but it is not plausible for plugs fabricated from alloy 690. Alloy 690 should not be listed for these entries. This position was accepted in NUREG-1705. (See also letter BGE to NRC 11/19/98.)

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IV

GALL SECTION	TITLE	ITEM NO	PAGE	REVIEWER	COMMENT
IV.D1-26	Steam Generator (Recirculating)	D1.3.1	34		The Feed Water Inlet Ring and Supports perform no passive intended function. The only passive intended function for the Steam Generator is the pressure boundary function. These items should be removed from the Report.
IV.D1-27	Steam Generator (Recirculating)	D1.3	34		(See previous comment regarding scoping of Feedwater Inlet Ring. Comments to follow assume that decision is made to consider internals and the component is retained in Report despite earlier comment.) The Upper Assembly and Separators entry for Erosion/ Corrosion should include the separator components themselves identified as components.
IV.D1-28	Steam Generator (Recirculating)	D1.5.1	38		Item lists Erosion Corrosion as an Aging Mechanism for Safety and Relief Valves. Erosion Corrosion is not plausible for relief and safety valves because they are not exposed to steam flow for significant periods. Minor seat leakage, possibly caused by steam cutting, should not be considered a pressure boundary failure. Therefore, Erosion Corrosion should not be listed for these valves. This position was accepted in NUREG-1705.

D2. Steam Generator (Once–Through)

D2.1 Pressure Boundary and Structural

- D2.1.1 Upper & Lower Heads
- D2.1.2 Tube Sheets
- D2.1.3 Primary Nozzles & Safe Ends
- D2.1.4 Shell
- D2.1.5 Feed Water and Auxiliary Feed Water Nozzles & Safe Ends
- D2.1.6 Steam Nozzles & Safe Ends
- D2.1.7 Instrument & Drain Nozzles
- D2.1.8 Primary Manways & Bolting
- D2.1.9 Secondary Manways Handholes & Bolting

D2.2 Tube Bundle

- D2.2.1 Tubes

D2.3 Piping and Fittings

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

D2.4 Safety and Relief Valves

- D1.5.1 Body

D2. Steam Generator (Once-Through)

IVD2-2

System, Structures, and Components

The system, structures, and components included in this table consist of the once-through type steam generators, as found in Babcock & Wilcox pressurized water reactors (PWRs), including all internal components, external primary and secondary water/steam nozzles and safe ends, and portions of main steam, feedwater, and auxiliary feedwater systems extending from the steam generator up to the first isolation valve outside of containment or to the first anchor point. Based on US Nuclear Regulatory Commission Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," the primary water side (tube side) of the steam generator is classified as Group A Quality Standards and secondary water side, including portions of main steam, feedwater, and auxiliary feedwater systems extending up to the first isolation valve outside of containment, is classified as Group B Quality Standard. The aging management program for the lines that penetrate the containment, and associated isolation valves, is reviewed in Table V C.

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

System Interfaces

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

IVD2-13 BALANCE OF SECTION

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D2 STEAM GENERATOR (Once-Through)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D2.1.1, D2.1.2	Pressure Boundary and Structural	Upper & Lower Heads, Tube Sheets	Low-Alloy Steel (LAS) with Stainless Steel (SS) Cladding	Up to 300°C Secondary -side Water Chemistry at 5.3-7.2 MPa	Crack Initiation and Growth	Stress Corrosion Cracking (SCC), Intergranu-lar Stress Corrosion Cracking (IGSCC)	ASME Section XI 1989 Edition. Reg. Guide 1.43, Reg. Guide 1.44, EPRI TR-102134

IVD2-4

IVD2-3

IVD2-1

IVD2-10 BALANCE OF SECTION

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
12. STEAM GENERATOR (Once-Through)

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Guidelines of Regulatory Guide (RG) 1.44 to avoid sensitization of stainless steels (SSs) and requirements of RG 1.43 to prevent underclad cracking; and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), subsection IWC, Table IWC 2500-1, examination category C-H for pressure retaining welds and category C-H for all pressure retaining components. Secondary water chemistry guidelines of EPRI TR-102134. Supplemental inspection requirements may be contained in Plant Technical Specifications.	***(1) Scope of Program:*** The program includes preventive measures to mitigate stress corrosion cracking (SCC) of SS cladding and inservice inspection (ISI) to monitor the effects of SCC of the cladding on the intended function of upper and lower heads and tubesheets. ***(2) Preventive Actions:*** Selection of material in compliance with the requirements of Regulatory Guide (RG) 1.44 prevent or mitigate IGSCC, and requirements of RG 1.43 prevent underclad cracking. In addition, secondary water chemistry guidelines given in EPRI TR-105714 provide guidance on the prevention of SCC. Also, control of halogens and oxygen in the primary water to less than 5 and 0.01 ppm, respectively, during operation, and monitor and control of water chemistry during shut down, mitigate the potential of SCC. However, inadvertent introduction of contaminants into the coolant system can occur, and cladding regions with low delta ferrite may be sensitized during post-weld heat treatment and thus become susceptible to SCC. The AMP must therefore rely upon ISI in accordance with ASME Section XI to detect possible degradation. ***(3) Parameters Monitored/ Inspected:*** The program monitors flaw size and depth, or, alternatively, remaining wall thickness in pressure retaining welds. Inspection requirements of Table IWC 2500-1, examinatio category C-H specifies visual VT-2 inspection of all pressure retaining components. Requirements for training and qualification of personnel and performance demonstration for procedures and equipment is in conformance with Appendices VII and VIII of ASME Section XI, or any other formal program approved by the NRC. ***(4) Detection of Aging Effects:*** The extent and schedule of the inspections prescribed by the program are designed to ensure that flaws cannot attain a depth sufficient to threaten the integrity of the welds. However, the extent and schedule of inspection does not assure detection of cracks because ASME Section XI inspection requires examination of only the welds and weld regions, the potential of cracking in cladding remote from welds is not addressed. ***(5) Monitoring and Trending:*** Required inspection intervals should provide for timely detection of virtually all forms of degradation. ***(6) Acceptance Criteria:*** Any defect detected is compared with the requirements of Section XI, IWC 3516. These requirements should ensure structural integrity. ***(7) Corrective Actions:*** Welds containing flaws that exceed the maximum permissible size must be repaired. Repair and replacement are in conformance with IWA-4000 and IWA-4000. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls, in conjunction with NRC over-sight, are implemented in accordance with the require-ments of Appendix B of 10	Yes Element 4 should be further evaluated

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D2. STEAM GENERATOR (Once-Through)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D2.1.1	Pressure Boundary and Structural	Upper & Lower Heads (External Surfaces)	LAS	Air, Leaking Chemically Treated Borated Water and/or Steam at temperatures up to 340°C	Loss of Material	Corrosion/ Boric Acid Wastage of External Surfaces	NRC GL 88-05. ASME Section XI, 1989 Edition.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D2. STEAM GENERATOR (Once-Through)

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Implementation of NRC Generic Letter 88-05, and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWC, Table IWC 2500-1, testing category C-H for all pressure retaining components.	***(1) Scope of Program:*** The staff guidance of Generic Letter (GL) 88-05 provides assurances that a program has been implemented consisting of systematic measures to ensure that the effects of corrosion caused by leaking coolant containing boric acid does not lead to degradation of the assurance that the reactor coolant pressure boundary will have a extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture. The program includes (a) determination of principal location, (b) examinations requirements and procedures for locating small leaks, and (c) engineering evaluations and corrective actions. ***(2) Preventive Actions:*** Minimizing reactor coolant leakage by frequent monitoring of the locations where potential leakage could occur and repairing the leaky components as soon as possible, prevent or mitigate boric acid corrosion. ***(3) Parameters Monitored/Inspected:*** The AMP detects coolant leakage by inservice inspection (ISI). Inspection requirements of ASME Section XI, Table IWC 2500-1, testing category C-H specify visual VT-2 (IWA-5240) examination of all pressure retaining components. ***(4) Detection of Aging Effects:*** Aging effects degradation of the component can not occur without leakage of coolant containing boric acid; extent and schedule of inspection assure detection of leakage before the loss of intended function of the component. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI should provide for timely detection of leakage. System leakage test is typically conducted at 40-month intervals. ***(6) Acceptance Criteria:*** Any relevant conditions that may be detected during the leakage and hydrostatic tests are evaluated in accordance with IWC-3100 and acceptance standards of NRC GL 88-05. ***(7) Corrective Actions:*** Prior to service, corrective measures are needed to meet the requirements of GL 88-05. The leakage source and areas of general corrosion are located. Components with local areas of corrosion that reduces the wall thickness by more than 10% require analytical evaluation to demonstrate acceptability. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Boric acid wastage in nuclear power plants may either be classified as (a) corrosion that increases the rate of leakage, e.g., corrosion of closure bolting, or (b) corrosion that occurs some distance from the source of leakage. Some recent incidents of boric acid wastage (IN 86-108 S3) indicate that, although implementation of GL 88-05 ensures	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D2. STEAM GENERATOR (Once-Through)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a) or other evaluations based on cumulative usage factor (CUF).			Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.				Yes TLAA

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D2. STEAM GENERATOR (Once-Through)

Existing Aging Management Program (AMP)			Evaluation and Technical Basis				Further Evaluation
D2.1.3	Pressure Boundary and Structural	Primary Nozzles and Safe Ends	Carbon Steel (CS) with SS cladding, Safe Ends: SS (NiCrFe buttering, and SS or NiCrFe weld)	Chemically Treated Borated Water at temperatures up to 340°C and 15.2 MPa	Crack Initiation and Growth	SCC, IGSCC	ASME Section XI, 1989 Edition. Reg. Guide 1.43. Reg. Guide 1.44. NRC IN 84-18.

IV02-12 BALANCE OF SECTION

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D2. STEAM GENERATOR (Once-Through)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
Guidelines of Regulatory Guide (RG) 1.44 to avoid sensitization and RG 1.43 to prevent underclad cracking. Primary water chemistry guidelines of EPRI TR-105714. Inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB 2500-1, examination category B-D for nozzle-to-vessel welds, category B-F for pressure-retaining dissimilar metal welds in vessel nozzles, and testing category B-P for system leakage			***(1) Scope of Program:*** The program includes preventive measures to mitigate stress corrosion cracking (SCC) of SS cladding and inservice inspection (ISI) to monitor the effects of SCC of the cladding on the intended function of reactor vessel nozzles. ***(2) Preventive Actions:*** Selection of material in compliance with the requirements of Regulatory Guide (RG) 1.44 prevent or mitigate IGSCC, and requirements of RG 1.43 prevent underclad cracking. In addition, primary water chemistry guidelines given in EPRI TR-102134 provide guidance on the prevention of SCC. Also, control of halogens and oxygen in the primary water to less than 5 and 0.01 ppm, respectively, during operation, and monitor and control of water chemistry during shut down, mitigate the potential of SCC. However, inadvertent introduction of contaminants into the coolant system can occur, and cladding regions with low delta ferrite may be sensitized during post-weld heat treatment and thus become susceptible to SCC. The AMP must therefore rely upon ISI in accordance with ASME Section XI to detect possible degradation. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of IGSCC on the intended function of reactor vessel nozzles by detection and sizing of cracks by ISI. Inspection requirements of Table IWC 2500-1, examination category C-A specifies volumetric inspection for all pressure retaining welds. Requirements for training and qualification of personnel and performance demonstration for procedures and equipment is in conformance with Appendices VII and VIII of ASME Section XI, or any other formal program approved by the NRC. ***(4) Detection of Aging Effects:*** Aging degradation of the reactor vessel nozzles can not occur without crack initiation. However, the extent and schedule of inspection does not assure detection of cracks because ASME Section XI inspection requires examination of only the welds and weld regions, the potential of cracking in cladding remote from welds is not addressed. Also, based on NRC Information Notices (INs) 90-10 and 90-30 applicant should review Ni-alloy applications in primary coolant and implement an augmented inspection program and evaluate choice of transducers for ultrasonic examination of dissimilar metal welds. ***(5) Monitoring and Trending:*** Inspection schedule in accordance with IWC-2400 should provide timely detection of cracks. System leakage test is typically conducted at 40-month intervals. ***(6) Acceptance Criteria:*** Any IGSCC degradation is evaluated in accordance with IWC-3100 by comparing ISI results with the acceptance standards of IWC-3410. Planar and liner flaws are sized according to IWA-3300 and IWA-3400. ***(7) Corrective Actions:*** Repair and replacement are in conformance with IWA-4000, and reexamination in accordance with requirements of IWA-					Yes Element 4 should be further evaluated

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D2. STEAM GENERATOR (Once-Through)

Existing Aging Management Program (AMP)			Evaluation and Technical Basis				Further Evaluation
D2.1.4, D2.1.5	Pressure Boundary and Structural	Shell Assembly, Feedwater (FW) and Auxiliary FW (AFW) Nozzles & Safe Ends	CS	Up to 300°C Secondary -side Water Chemistry at 5.3-7.2 MPa	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. GSI-190.

N025

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D2. STEAM GENERATOR (Once-Through)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D2.1.4	Pressure Boundary and Structural	Shell Assembly	CS	Up to 300°C Secondary -side Water Chemistry at 5.3-7.2 MPa	Loss of Material	Crevice and Pitting Corrosion	ASME Section XI, 1989 Edition. EPRI TR-102134. NRC IN 82-37. NRC IN 85-65. NRC IN 90-04.

	(continued from previous page) will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** The primary pressure boundary components of PWRs are generally not affected by SCC because of low dissolved oxygen levels and control of primary water chemistry. The potential of SCC exists from inadvertent introduction of contaminants into the primary coolant system; sources are (a) contaminants in boric acid and (b) spent fuel pool exposed to airborne contaminants (IN 84–18).	

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D2. STEAM GENERATOR (Once-Through)

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a) or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
The program includes preventive measures to mitigate crevice corrosion by recommendations on secondary water chemistry of EPRI TR-102134 and, based on Plant Technical Specifications, inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a) Table IWC 2500-1, examination category C-A for pressure retaining welds in pressure vessels, and examination category C-H for pressure retaining Class 2 components.	***(1) Scope of Program:*** The program relies on preventive measures to mitigate crevice or pitting corrosion and inservice inspection (ISI) to monitor the effects of corrosion on the intended function of the steam generator shell. ***(2) Preventive Actions:*** Stringent control of secondary water chemistry in accordance with the guidance of EPRI TR-102134, frequent monitoring, and timely corrective action when specified impurity levels are exceeded, prevent or mitigate corrosion. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of corrosion by detection and sizing of flaws in pressure retaining welds and by detection of coolant leakage by inservice inspection (ISI). Inspection requirements of Table IWC 2500-1, examination category C-A specify volumetric examination, extending 1/2 in. each side, of all circumferential and tubesheet-to-shell welds, and visual VT-2 (IWA-5240) examination during system leakage and hydrostatic tests of all pressure retaining Class 2 components. Requirements for training and qualification of personnel and performance demonstration for procedures and equipment is in conformance with Appendix VII and VIII of ASME Section XI, or any other formal program approved by the NRC. ***(4) Detection of Aging Effects:*** The extent and schedule of the inspections prescribed by the program are designed to ensure that flaws cannot attain a depth sufficient to threaten the integrity of the welds. However, based on NRC Information notice 90-04 where general corrosion pitting of the shell exists, the program requirements may not be sufficient to differentiate isolated cracks from inherent geometric conditions, and additional inspection procedures may be required. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI should provide for timely detection of leakage. System leakage test is typically conducted at 40-month intervals. ***(6) Acceptance Criteria:*** Any defect detected is compared with the requirements of Section XI, IWC 3510. Any relevant conditions that may be detected during the leakage and hydrostatic tests are evaluated in accordance with IWC-3400 and IWC-3516. ***(7) Corrective Actions:*** Welds containing flaws that exceed the maximum	Yes Element 4 should be further evaluated

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D2. STEAM GENERATOR (Once-Through)

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

IV D2-11
BALANCE
OF
SECTION

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) permissible size must be repaired. Repair and replacement are in conformance with IWA-4000. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls, in conjunction with NRC oversight, are implemented in accordance with the requirements of Appendix B of 10 CFR 50. These requirements, in conjunction with enhanced UT techniques, should continue to ensure the timely detection and correction of virtually all forms of this type of degradation but cannot be depended upon to prevent further degradation. ***(10) Operating Experience:*** This AMP has resulted in the timely detection and correction of corrosion in several steam generators in the U.S. (NRC Information Notices 82-37 & 85-65).	
Program delineated in NUREG-1344 for single phase lines and implemented through NRC Generic Letter 89-08; CHECWORKS Code; EPRI guidelines of NSAC-202L-R2 for effective erosion/corrosion program; and water chemistry program based on EPRI guidelines for secondary water chemistry (EPRI TR-102134).	***(1) Scope of Program:*** The AMP includes NUMARC program delineated in Appendix A of NUREG-1344 and implemented through NRC Generic Letter (GL) 89-08; CHECWORKS computer Code; and EPRI guidelines of NSAC-202L-R2. The program includes the following recommendations: (a) conduct appropriate analysis and limited baseline inspection, (b) determine the extent of thinning and repair/replace components, and (c) perform follow-up inspections to confirm or quantify and take longer term corrective actions. Technical aspects of the CHECWORKS Code, including the parameters and inputs, are acceptable. However, the EPRI guidance document NSAC-202L-R2 (April 1999) is too general to ensure applicant's flow-accelerated corrosion program will be effective in managing aging in safety-related systems. ***(2) Preventive Actions:*** The rate of E/C is affected by piping material, geometry and hydrodynamic conditions, and operating conditions such as temperature, pH, and dissolved oxygen content. Mitigation is by selecting material considered resistant to E/C, adjusting water chemistry and operating conditions, and improving hydrodynamic conditions through design modifications. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of E/C on the intended function of piping by measuring wall thickness by nondestructive examination and performing analytical evaluations. The inspection program delineated in NUREG-1344 requires ultrasonic or radiographic testing of 10 most susceptible locations and 5 additional locations based on unique operating conditions or special considerations. For each location outside the acceptance guidelines, the inspection sample is expanded based on engineering judgment. Analytical models such as those incorporated into the CHECWORKS code are used to predict E/C in piping systems based on specific plant data including material and hydrodynamic and operating conditions. The	Yes, Element 1 should be further evaluated

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D2. STEAM GENERATOR (Once-Through)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D2.1.6	Pressure Boundary and Structural	Steam Nozzles & Safe Ends	CS	Up to 300°C Steam	Cumulative Fatigue Damage	Fatigue	ASME Section III, 1989 Edition. GSI-190.
D2.1.6	Pressure Boundary and Structural	Steam Nozzles & Safe Ends	CS	Up to 300°C Steam	Wall Thinning	Erosion/ Corrosion (E/C)	*Same as the effect of Erosion/ Corrosion on Item*

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) guidelines should provide for timely detection of leakage. Inspections and analytical evaluations are performed during plant outage. If analysis shows unacceptable conditions, inspection of initial sample is performed within 6 months. ***(6) Acceptance Criteria:*** Inspection results are used to calculate number of refueling or operating cycles remaining before the component reaches Code minimum allowable wall thickness. If calculations indicate that an area will reach Code minimum (plus 10% margin), the component must be repaired or replaced. However, NRC staff has identified the problems in implementing E/C program that pertain to weakness or errors in (a) using predictive models, (b) calculating minimum wall thickness acceptance criteria, (c) analyzing the results of UT examinations, and (d) assessment of E/C program activities (NRC Information Notice IN 93-21). ***(7) Corrective Actions:*** Prior to service, repair or replace to meet the requirements of NUREG-1344. Follow-up inspections are performed to confirm or quantify thinning and take longer term corrective actions such as adjustment of chemistry and operating parameters, or selection of materials resistant to E/C. However, NRC staff has identified weakness or errors in (a) dispositioning components after reviewing the results of the inspection analysis, and (b) repairing or replacing components that failed to meet the acceptance criteria (IN 93-21). ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Wall-thinning problems in single-phase systems have occurred in feedwater and condensate systems (NRC Bulletin No. 87-01, INs 81-28, 92-35, 95-11), and in two-phase piping in extraction steam lines (INs 89-53, 97-84) and moisture separation reheater and feedwater heater drains (INs 89-53, 91-18, 93-21, 97-84). The AMP outlined in NUREG-1344 and EPRI report and implemented through GL 89-08 has provided effective means of ensuring the structural integrity of all high-energy carbon steel systems.	
Components have been designed or evaluated for fatigue for a 40 y design life, according to the requirements of ASME Section III (edition specified in 10 CFR 50.55a) or other evaluations based on cumulative usage factor (CUF).	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.	Yes TLAA
Same as the effect of Erosion/Corrosion on Item D2.1.5 FW and AFW Nozzles	*Same as the effect of Erosion/Corrosion on Item D2.1.5 FW and AFW Nozzles and Safe Ends*	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D2. STEAM GENERATOR (Once-Through)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
D2.1.7	Pressure Boundary and Structural	Instruments & Drain Nozzles 1V02-6	Alloy 600	Chemically Treated Borated Water at temperatures up to 340°C and 15.2 MPa	Crack Initiation and Growth	SCC, IGSCC, Primary Water Stress Corrosion Cracking (PWSCC)	ASME Section XI, 1989 Edition. Reg. Guide 1.45. NRC IN 90-10. NRC IN 96-11. NRC GL 97-01, Plant Technical Specifications

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D2. STEAM GENERATOR (Once-Through)

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
Operational limits from Plant Technical Specifications. Inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWB, Table IWB-2500-1, examination category B-E for pressure-retaining partial penetration welds and testing category B-P for system leakage. PWR primary water chemistry guidelines from EPRI TR-105714.	***(1) Scope of Program:*** The program is focused on managing the effects of stress corrosion cracking (SCC) on the intended function of vessel penetrations. ***(2) Preventive Actions:*** Control of halogens and oxygen in the primary water to less than 5 and 0.01 ppm, respectively, during operation, and monitor and control of water chemistry during shut down, mitigate potential of SCC, based upon guidelines given in EPRI TR-105714. However, based on the experiences described in NRC Information Notices (IN) 90-10 and 96-11, and Generic Letter (GL) 97-01, the susceptibility of Ni-alloys and welds to primary water SCC has not been addressed adequately, particularly when demineralizer resins contaminate the reactor coolant system. As described in IN 96-11 and implemented through GL 97-01, applicant should review the primary coolant system water chemistry to identify any primary system resin bed intrusions that exceed the PWR primary water chemistry guidelines. ***(3) Parameters Monitored/ Inspected:*** The AMP monitors the effects of IGSCC on the intended function of instrument nozzles by performance monitoring and detection of cracks by inservice inspection (ISI). Testing category B-P specifies visual VT-2 (IWA-5240) examination of all pressure retaining components extending to and including the second closed valve at the boundary extremity, during system leakage test (IWB-5221) and system hydrostatic test (IWB-5230). Leakage detection is in conformance with Position C of Regulatory Guide 1.45. However, in compliance with GL 97-01, applicant should determine whether an augmented inspection program is necessary. ***(4) Detection of Aging Effects:*** Aging effects degradation of the vessel penetrations can not occur without crack initiation. In accordance with GL 97-01, applicant should review whether the scope and schedule of inspection, including leakage detection system, assure detection of cracks before the loss of intended function of the penetrations. ***(5) Monitoring and Trending:*** System leakage test is conducted prior to plant startup following each refueling outage, and hydrostatic test at or near the end of each inspection interval. Applicant should either provide the technical basis that justifies the adequacy of the program or develop an integrated long-term program which includes periodic inspection and performance monitoring. ***(6) Acceptance Criteria:*** Any IGSCC degradation is evaluated in accordance with WB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and 3522 for visual examination. Acceptance of the component for continued service is based on the results of visual examination, supplemental volumetric and surface examination (IWB-3200), corrective measure or repair, analytical evaluation, or replacement (IWB-3132). However, in accordance with GL 97-01, the	Yes Elements 2 thru 6 should be further evaluated

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D2. STEAM GENERATOR (Once-Through)

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

Existing Aging Management Program (AMP)	Evaluation and Technical Basis	Further Evaluation
	(continued from previous page) reexamination in accordance with requirements of IWA-2200. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** SCC of Alloy 600 has occurred in domestic and foreign PWRs (IN 90-10). Very little attention has been given to the inspection for SCC in Alloy 600 applications other than that associated with steam generator tubes.	
Implementation of NRC Generic Letter 88-05, and inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWC, Table IWC 2500-1, testing category C-H for all pressure retaining components.	*Same as the effect of Corrosion/Boric Acid Wastage on Items D2.1.1 Upper and Lower Heads*	No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D2. STEAM GENERATOR (Once-Through)

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

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D2. STEAM GENERATOR (Once-Through)

Existing Aging Management Program (AMP)			Evaluation and Technical Basis					Further Evaluation
D2.1.9	Pressure Boundary and Structural	Secondary Manways and Handholes	CS	Air, Leaking Secondary -side Water and/or Steam at temperatures up to 300°C	Wall Thinning	Erosion/ Corrosion (E/C)	ASME Section XI, 1989 Edition. Reg. Guide 1.45	

	(continued from previous page) ***(7) Corrective Actions:*** Repair and replacement is in conformance with IWA-4000 and recommendations and guidelines of EPRI NP-5769. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** Significant number of incidents have been reported on bolts and nuts that have failed or become degraded. Examples of affected fasteners include (a) steam generator and pressurizer	

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D2. STEAM GENERATOR (Once-Through)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
Inservice inspection in conformance with ASME Section XI (edition specified in 10 CFR 50.55a), Subsection IWC, Table IWC 2500-1, testing category C-H for pressure retaining components.			***(1) Scope of Program:*** The program is focused on managing the effects of erosion damage to the sealing surfaces of manway and handhole covers due to localized steam leakage. ***(2) Preventive Actions:*** Verification of the sealing function by frequent monitoring and timely corrective action prevent or mitigate erosion damage. ***(3) Parameter Monitored/ Inspected:*** The AMP monitors the effects of erosion on the sealing function of manway and handhole covers by detection of leakage or erosion damage by inservice inspection (ISI). Inspection requirements of ASME Section XI, Table IWC 2500-1, testing category C-H specify visual VT-2 (IWA-5240) examination of all pressure retaining components. Leakage detection is in conformance with Position C of Regulatory Guide 1.45. ***(4) Detection of Aging Effects:*** Aging degradation of manway and handhole covers can not occur without erosion damage and steam leakage; extent and schedule of inspection assure detection of leakage or erosion damage before the loss of sealing function of the flange. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI should provide for timely detection of leakage. System leakage test is typically conducted at 40-month intervals. ***(6) Acceptance Criteria:*** Any relevant conditions that may be detected during the leakage and hydrostatic tests are evaluated in accordance with IWC-3100 and acceptance standards of IWC-3410 and IWC-3516. ***(7) Corrective Actions:*** Prior to service, corrective measures are needed to meet the requirements of IWC-3132. The leakage source and areas of erosion damage are located. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** The present AMP has been effective in managing the effects of erosion due to localized steam leakage on the sealing function of secondary manways and handholes.					No

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D2. STEAM GENERATOR (Once-Through)

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

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D2. STEAM GENERATOR (Once-Through)

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
Inservice inspection in conformance with Plant Technical Specifications and NRC Regulatory Guide (RG) 1.83 ("Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes." Plugging or repair of defective tubes is governed by NRC Regulatory Guide 1.121 ("Bases for Plugging Degraded PWR Steam Generator Tubes"). Specific guidance on the inservice inspection of tubes is also provided in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5" and NEI 97-06. Supplemental inspection requirements and plugging criteria may be contained in Plant Technical Specifications.			***(1) Scope of Program:*** The program deals with the effects of SCC of the steam generator tubes on the integrity of the pressure boundary. Inservice inspection in accordance with Plant Technical Specifications and RG 1.83 deal with the detection of flaws, and RG 1.121 and NRC GL 95-05 describe the basis for determining when flawed tubes must be plugged or repaired. ***(2) Preventive Actions:*** Primary water chemistry guidelines given in EPRI TR-105714 provide guidance on the prevention of PWSCC. ***(3) Parameters Monitored/Inspected:*** The inspection activity in the program monitors flaw size and depth, or, alternatively, remaining sound wall thickness. ***(4) Detection of Aging Effects:*** The extent and schedule of the inspections prescribed by the program are designed to ensure that flaws cannot attain a depth sufficient to threaten the integrity of the tubes. Problems with tube inspection (IN 97-88), e.g., failures to detect some flaws, uncertainties in flaw sizing, inaccuracies in flaw locations, and inability to detect some cracks at locations with dents are addressed by current inspection guidelines given in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5" and NEI 97-06. ***(5) Monitoring and Trending:*** Required inspection intervals (typically ≈18 months or each refueling or maintenance/repair outage) provide for timely detection of SCC. ***(6) Acceptance Criteria:*** Any cracking detected is compared with the requirements of the Plant Technical Specifications, RG 1.121, and GL 95-05. ***(7) Corrective Actions:*** Tubes containing flaws that exceed the maximum permissible size must be plugged or repaired. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls, in conjunction with NRC oversight, are implemented in accordance with the requirements of Appendix B of 10 CFR 50. Recent experience indicates the importance of performing a complete inspection using the best available techniques and equipment for the reliable detection of tube degradation and to ensure that new forms of degradation are detected. ***(10) Operating Experience:*** The present AMP has been effective in ensuring timely detection and correction of PWSCC degradation in steam generator tubes.				No, provided Plant Technical Specifi-cations conform to EPRI inspection guidelines and NEI 97-06 and are adhered to.
Inservice inspection in conformance with Plant Technical Specifications and NRC Regulatory Guide 1.83 ("Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes". Plugging or repair of defective tubes is governed by NRC Regulatory Guide 1.121 ("Bases for Plugging Degraded PWR Steam Generator Tubes"). Recommendations			***(1) Scope of Program:*** The program deals with the effects of SCC of the steam generator tubes on the integrity of the pressure boundary. Plant Technical Specifications and Reg. Guide 1.83 deal with the detection of flaws, and Reg. Guide 1.121 and GL 95-05 describe the basis for determining when flawed tubes must be plugged or repaired. ***(2) Preventive Actions:*** Secondary water chemistry guidelines given in EPRI TR-102134 provide guidance on the prevention of ODSCC. ***(3) Parameters***				No, provided Plant Technical Specifi-cations conform to EPRI inspection

Existing Aging Management Program (AMP)			Evaluation and Technical Basis				Further Evaluation
D2.2.1	Tube Bundle (Babcock & Wilcox)	Steam Generator Tubes	Alloy 600	Up to 300°C Secondary -side Water Chemistry at 5.3-7.2 MPa	Crack Initiation and Growth	Intergranu -lar Attack (IGA)	Plant Technical Specifications. Reg. Guide 1.83. Reg. Guide 1.121. NRC GL 95-05, EPRI document "PWR Steam Generator Examination Guidelines: Rev. 5", NEI 97-06

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
(continued from previous page) tubes is also provided in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5" and NEI 97-06. Supplemental inspection requirements and plugging criteria may be contained in plant-specific Technical Specifications.			*(continued from previous page)* cannot attain a depth sufficient to threaten the integrity of the tubes. Problems with tube inspection, e.g., failures to detect some flaws, uncertainties in flaw sizing, and inaccuracies in flaw locations, are addressed in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5" and NEI 97-06. ***(5) Monitoring and Trending:*** Required inspection intervals (typically ≈18 months or each refueling or maintenance/repair outage) provide for timely detection of SCC. ***(6) Acceptance Criteria:*** Any cracking detected is compared with the requirements of the Plant Technical Specifications, Reg. Guide 1.121, and GL 95-05. ***(7) Corrective Actions:*** Tubes containing flaws that exceed the maximum permissible size must be plugged or repaired. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls, in conjunction with NRC oversight, are implemented in accordance with the requirements of Appendix B of 10 CFR 50. Recent experience indicates the importance of performing a complete inspection using the best available techniques and equipment for the reliable detection of tube degradation and to ensure that new forms of degradation are detected. ***(10) Operating Experience:*** The present AMP has been effective in ensuring timely detection and correction of SCC degradation in steam generator tubes.		
Inservice inspection in conformance with Plant Technical Specifications and NRC Regulatory Guide 1.83 ("Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes". Plugging or repair of defective tubes is governed by NRC Regulatory Guide 1.121 ("Bases for Plugging Degraded PWR Steam Generator Tubes"). Recommendations on secondary water chemistry given in EPRI TR-102134 ("PWR Secondary Water Chemistry Guidelines—Revision 3"). Specific guidance on the inservice inspection of tubes is also provided in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5" and NEI 97-06. Supplemental inspection requirements and plugging criteria may be contained in plant-specific Technical Specifications.			***(1) Scope of Program:*** The program deals with the effects of IGA of the steam generator tubes on the integrity of the pressure boundary. Plant Technical Specifications and Reg. Guide 1.83 deal with the detection of flaws, and Reg. Guide 1.121 and GL 95-05 describe the basis for determining when flawed tubes must be plugged or repaired. ***(2) Preventive Actions:*** Secondary water chemistry guidelines given in EPRI TR-102134 provide guidance on the prevention of IGA. ***(3) Parameters Monitored/Inspected:*** The inspection activity in the program monitors the extent of degradation or, alternatively, remaining sound wall thickness. ***(4) Detection of Aging Effects:*** The extent and schedule of the inspections prescribed by the program are designed to ensure that flaws cannot attain a depth sufficient to threaten the integrity of the tubes. Problems with tube inspection, e.g., failures to detect and properly characterize IGA, are addressed in the EPRI document "PWR Steam Generator Examination Guidelines: Revision 5" and NEI 97-06. ***(5) Monitoring and Trending:*** Required inspection intervals (typically ≈18 months or each refueling or maintenance/repair outage) provide for timely detection of IGA. ***(6) Acceptance Criteria:*** Any degradation detected is compared with the requirements			No, provided Plant Technical Specifications conform to EPRI inspection guidelines and NEI 97-06 and are adhered to.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D2. STEAM GENERATOR (Once-Through)

Existing Aging Management Program (AMP)			Evaluation and Technical Basis				Further Evaluation
D2.2.1	Tube Bundle (Babcock & Wilcox)	Steam Generator Tubes	Alloy 600	Up to 300°C Secondary -side Water Chemistry at 5.3-7.2 MPa	Crack Initiation and Growth	Fatigue	Plant Technical Specifications, GSI-190
D2.3.1	Piping and Fittings	Main Steam	CS	Up to 300°C Steam	Cumulative Fatigue Damage	Fatigue	ANSI B31.1
D2.3.1	Piping and Fittings	Main Steam	CS	Up to 300°C Steam	Wall Thinning	E/C	*Same as the effect of Erosion/ Corrosion on Item D2.1.5 FW and AFW Nozzles and Safe Ends*
D2.3.2	Piping and Fittings	Feedwater	CS	Up to 225°C Secondary -side Water Chemistry	Cumulative Fatigue Damage	Fatigue	ANSI B31.1
D2.3.2	Piping and Fittings	Feedwater	CS	Up to 225°C Secondary -side Water Chemistry	Wall Thinning	E/C	*Same as the effect of Erosion/ Corrosion on Item D2.1.5 FW and AFW Nozzles and Safe Ends*
D2.3.3	Piping and Fittings	Auxiliary Feedwater	CS	<90°C Secondary -side Water Chemistry	Wall Thinning	E/C	*Same as the effect of Erosion/ Corrosion on Item D2.1.5 FW and AFW Nozzles and Safe Ends*
D2.3.1	Piping and Fittings	Main Steam	CS	Up to 300°C Steam	Loss of Material	Crevice and Pitting Corrosion	ASME Section XI, 1989 Edition; Plant Technical Specifications

WDZ-8

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
			(continued from previous page) accordance with the requirements of Appendix B of 10 CFR 50. Recent experience indicates the importance of performing a complete inspection using the best available techniques and equipment for the reliable detection of tube degradation and to ensure that new forms of degradation are detected. ***(10) Operating Experience:*** The present AMP has been effective in ensuring timely detection and correction of IGA degradation in steam generator tubes.				
Components have been designed or evaluated for fatigue for a 40 y design life, according to Plant Technical Specifications or other evaluations based on cumulative usage factor (CUF).			Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal, and Generic Safety Issue (GSI)-190 is to be addressed.				Yes TLAA
Components have been designed for fatigue for a 40 y design life according to the requirements of ANSI B31.1.			Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal; check Code limits for allowable cycles (less than 7000 cycles) of thermal stress range.				Yes TLAA
Same as the effect of Erosion/Corrosion on Item D2.1.5 FW and AFW Nozzles and Safe Ends			*Same as the effect of Erosion/Corrosion on Item D2.1.5 FW and AFW Nozzles and Safe Ends*				No
Components have been designed for fatigue for a 40 y design life according to the requirements of ANSI B31.1.			Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of license renewal; check Code limits for allowable cycles (less than 7000 cycles) of thermal stress range.				Yes TLAA
Same as the effect of Erosion/Corrosion on Item D2.1.5 FW and AFW Nozzles and Safe Ends			*Same as the effect of Erosion/Corrosion on Item D2.1.5 FW and AFW Nozzles and Safe Ends*				No
Same as the effect of Erosion/Corrosion on Item D2.1.5 FW and AFW Nozzles and Safe Ends			*Same as the effect of Erosion/Corrosion on Item D2.1.5 FW and AFW Nozzles and Safe Ends*				No
The program includes preventive measures to mitigate crevice corrosion by control of secondary water chemistry and, based on Plant Technical Specifications, inservice inspection in			***(1) Scope of Program:*** The program relies on preventive measures to mitigate general, crevice, or pitting corrosion, and inservice inspection (ISI) to monitor the effects of corrosion on the intended function of the auxiliary feedwater system components. ***(2) Preventive Actions:***				No

N02-8

Existing Aging Management Program (AMP)			Evaluation and Technical Basis				Further Evaluation
D2.3.2, D2.3.3	Piping and Fittings	Feedwater, Auxiliary Feedwater	CS	Up to 300°C Secondary -side Water Chemistry	Loss of Material	Crevice and Pitting Corrosion	ASME Section XI, 1989 Edition
D2.4.1	Safety and Relief Valves	Body	CS	Up to 300°C Steam or Secondary -side Water Chemistry	Wall Thinning	E/C	*Same as the effect of Erosion/ Corrosion on Item D2.1.5 FW and AFW Nozzles and Safe Ends*

Item	Structure and Component	Region of Interest	Material	Environ-ment	Aging Effect	Aging Mechanism	References
			(continued from previous page) The AMP monitors the effects of corrosion by detection of coolant leakage by inservice inspection (ISI). Inspection requirements of ASME Section XI specify visual VT-2 (IWA-5240) examination during system leakage test and hydrostatic test of all pressure retaining Class 2 components up to the first normally closed valve, according to Table IWC 2500-1 category C-H. ***(4) Detection of Aging Effects:*** Degradation of the component due to corrosion can not occur without leakage of coolant; extent and schedule of inspection assure detection of leakage before the loss of intended function of the component. ***(5) Monitoring and Trending:*** Inspection schedule of ASME Section XI should provide for timely detection of leakage. System leakage test is typically conducted at 40-month intervals. ***(6) Acceptance Criteria:*** Any relevant conditions that may be detected during the leakage and hydrostatic tests are evaluated in accordance with IWC-3100 and acceptance standards of IWC-3400 and IWC-3516. ***(7) Corrective Actions:*** Prior to service, corrective measures are needed to meet the requirements of IWC-3132 and IWA-5250. Repair and replacement are in conformance with IWA-4000. ***(8 & 9) Confirmation Process and Administrative Controls:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of license renewal. ***(10) Operating Experience:*** No significant corrosion related problem has been reported for piping and fittings in secondary side systems of steam generators.		
			Same as the effect of Crevice and Pitting Corrosion on Item D2.3.1 Piping and Fittings in Main Steam System				No
			Same as the effect of Erosion/Corrosion on Item D2.1.5 FW and AFW Nozzles and Safe Ends				No

References

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

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

ASME OM S/G, Part 2, *Requirements for Performance Testing of Nuclear Power Plant Closed Cooling Water Systems*, Standards and Guides for Operation and Maintenance of Nuclear Power Plants, The American Society of Mechanical Engineers, New York, NY.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, 1989 Edition, The ASME Boiler and Pressure Vessel Code, The American Society of Mechanical Engineers, New York, NY, July 1, 1989.

ASTM D95-83, *Standard Test Method for Water in Petroleum Products and Bituminous Materials by Distillation*, American Society for Testing and Materials, West Conshohocken, PA, 1983.

BWRVIP-03, *RPV Internals Examination Guidelines*, November 10, 1995, and as amended April 16, 1996 and March 30,1999.

BWRVIP-06, *Safety Assessment of BWR Reactor Internals*, October 5, 1995.

BWRVIP-07, *Guidelines for Reinspection of BWR Core Shrouds*, February 29, 1996.

BWRVIP-14, *Evaluation of Crack Growth in BWR Stainless Steel RPV Internals*, March 18, 1996, and as amended November 24, 1998.

BWRVIP-16, *Internal Core Spray Piping and Sparger Replacement Design Criteria*, March 18, 1997.

BWRVIP-18, *Core Spray Internals and Inspection and Flaw Evaluation Guidelines*, July 26, 1996, and as amended January 11, 1999.

BWRVIP-19, *Internal Core Spray Piping and Sparger Repair Design Criteria*, September 17, 1996.

BWRVIP-26, *Top Guide Inspection and Flaw Evaluation Guideline*, December 27, 1996, and as amended August 5, 1999.

BWRVIP-41, *BWR Jet Pump Assembly Inspection and FLaw Evaluation Guidelines*, October 15,1997.

BWRVIP-42, *BWR LPCI Coupling Injection and Flaw Evaluation Guidelines*, December 11, 1997.

BWRVIP-44, *Underwater Weld Repair of Nickel Alloy Reactor Vessel Internals*, October 27, 1997.

BWRVIP-45, *Weldability of Irradiated LWR Structural Components*, October 27, 1997.

BWRVIP-49, *Instrument Penetration Inspection and Flaw Evaluation Guidelines*, March 13, 1998.

BWRVIP-50, *Top Guide/Core Plate Repair Design Criteria*, May 14, 1998.

BWRVIP-51, *Jet Pump Repair Design Criteria*, May 14, 1998.

BWRVIP-57, *Instrument Penetration Repair Design Criteria*, December 16, 1998.

BWRVIP-59, *Evaluation of Crack Growth in BWR Nickel-Base Austenitic Alloys in RPV Internals*, December 23, 1998.

BWRVIP-60, *Evaluation of Crack Growth in BWR Low Alloy Steel RPV Internals*, March 30, 1999, and as amended July 30, 1999.

BWRVIP-62, *Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection*, December 31, 1998.

BWRVIP-63, *Shroud Vertical Weld Inspection and Evaluation Guidelines*, July 1, 1999.

EPRI NP-5769, *Degradation and Failure of Bolting in Nuclear Power Plants*, Volumes 1 and 2, Electric Power Research Institute, Palo Alto, CA, April 1988.

EPRI TR-102134, *PWR Secondary Water Chemistry Guidelines—Revision 3*, Electric Power Research Institute, Palo Alto, CA, May 1993.

EPRI TR-103515, *BWR Water Chemistry Guidelines-Revision 3, Normal and Hydrogen Water Chemistry*, Electric Power Research Institute, Palo Alto, CA, February 1994.

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

NRC Bulletin 87-01, *Thinning of Pipe Walls in Nuclear Power Plants*, July 9, 1987.

NRC Generic Letter 89-13, *Service Water System Problems Affecting Safety-Related Equipment*, July 18, 1989.

NRC Generic Letter 89-13, Supplement 1, *Service Water System Problems Affecting Safety-Related Equipment*, April 4, 1990.

NRC Generic Letter 90-06, *Resolution of Generic Issue 70, "Power-Operated Relief Valve and Block Valve Reliability ," and Generic Issue 94, "Additional Low-Temperature Overpressure Protection for Light Water Reactors," Pursuant to 10 CFR 50.54(f)*, June 25, 1990.

NRC Generic Letter 91-17, *Generic Safety Issue 79, "Bolting Degradation or Failure in Nuclear Power Plants"*, October 17, 1991.

NRC Generic Letter 96-05, *Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves*, September 18, 1996.

NRC Information Notice 81-21, *Potential Loss of Direct Access to Ultimate Heat Sink*, July 21, 1981.

NRC Information Notice 85-24, *Failures of Protective Coatings in Pipes and Heat Exchangers*, March 26, 1985.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IV

GALL SECTION	TITLE	ITEM NO	PAGE	REVIEWER	COMMENT
IV D2-1	Steam Generator (Once-Through)	All	D2-4	M. A. Rinckel	General Comment—Suggest that aging effects that apply to all OTSG items (e.g., loss of external material by boric acid wastage of ferritic items and cumulative fatigue damage be discussed at the very beginning and not repeated thereafter). At present, CUF is listed for selected items and not for all OTSG items.
IV D2-2	Steam Generator (Once-Through)	Section D2.	IV D2-3	M.A. Rinckel	The secondary water side, excluding portions of the main steam system, feedwater system, and auxiliary feedwater system, is classified as Class A Quality Standard. The main and auxiliary feedwater headers and riser piping were designed to ANSI B31.1.
IV D2-3	Steam Generator (Once-Through)		D2-4	M. A. Rinckel	It appears that the OTSG support skirt transition ring was omitted. Suggest that this item be added. It is fabricated from low alloy steel. Aging effects include cracking at weld and loss of material. AMP includes Exam. Category B-H and Boric Acid Wastage.
IV D2-4	Steam Generator (Once-Through)	D2.1.1. D2.1.2	IV D2-4	M. A. Rinckel	Material—the tube sheets are clad with Alloy 82/182. The heads are clad with stainless steel Environment—Chemically treated borated water up to 340 °C for upper and lower heads. Both chemically treated borated water up to 340 °C and secondary-side water chemistry for tubesheets. Aging Effect—Suggest cracking at welded joints Aging Mechanism—see discussion of items C2.1.1 thru C2.1.4 for RCS piping & fittings. If SCC and IGSCC apply to SS and Alloy 82/182 cladding with subsequent propagation to the base metal then say so. Aging Management Program—ASME Section XI, Subsection IWB, Examination Category B-B for primary pressure boundary welds.

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IV

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV D2-5	Steam Generator (Once-Through)	D2.1.3	IV D2-6	M. A. Rinckel	Aging Effect-Cumulative Fatigue Damage See the general comment regarding fatigue at the beginning of this section. A resolution path for GSI-190 for license renewal should be discussed in the GALL report. Either evaluation of NUREG-6260 items or alternate solution per closure of GSI-190.
IV D2-6	Steam Generator (Once-Through)	D2.1.7	IV D2-16	M. A. Rinckel	This entry is confusing since the primary pressure boundary has only one Alloy 600 drain nozzle and no instrument nozzles. The drain nozzle is fabricated from 1-inch Schedule 160 Alloy 600 pipe. The secondary pressure boundary contains vents, drains, and instrumentation connections fabricated from Alloy 600 and carbon steel. Suggest a separate entry for secondary nozzles as Examination Category B-E and B-P do not apply to these items.
IV D2-7	Steam Generator (Once-Through)	D2.1.8, D2.1.9	IV D2-18	M. A. Rinckel	Aging Management Program—AMP should list and describe Examination Category B-G-2 since the primary manway bolting is 2-inch diameter and less.
IV D2-8	Steam Generator (Once-Through)	D2.3.2	IV D2-26	M. A. Rinckel	Aging Effect—Cracking at Welded Joints Region of Interest—Aux. And main Feedwater Header and Piping AMP—ASME Section XI, examination Category C-F-2 should be credited for aging management of cracking in piping
IV D2-9	Steam Generator (Once-Through)		IV D2-18	B. Gordon	Replace "thru" with "through" or use dashes, (e.g., A1.1.1 – A1.1.3)
IV D2-10	Steam Generator (Once-Through)	D2.1.1 D2.1.2 D2.1.3 D2.1.7	IV D2-5, 8 - 11, & 16 – 19.	B. Gordon	Replace "SCC, IGSCC" or "Stress Corrosion Cracking (SCC), Intergranular Stress Corrosion Cracking (IGSCC)" with "SCC".

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IV

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV D2-11	Steam Generator (Once-Through)	D2.1.5 D2.1.6 D2.1.9 D2.3.1 D2.3.3 D2.4.1	IV D2-12, 13, 14, 15, 20, 21, 26, and 28.	B. Gordon	Replace "Erosion/Corrosion (E/C)" with "Flow-Accelerated Corrosion (FAC)".
IV D2-12	Steam Generator (Once-Through)	D2.1.3 D2.1.7 D2.2.1	IV D2-8, 9, 10, 11, 16, 17, 18, 19, and 22.	B. Gordon	Replace "EPRI TR-105714" with "the latest version of EPRI TR-105714" or "EPRI PWR Primary Water Chemistry Guidelines, Revision 4, March 1999, EPRI TR-105714".
IV D2-13	Steam Generator (Once-Through)	D2.1.1 D2.1.2 D2.1.4 D2.1.5 D2.2.1	IV D2-4, 5, 10, 11, 12, 13, 14, 15, 24, 25, 26, and 27	B. Gordon	Replace "EPRI TR-102134" with " the latest version of EPRI TR-102134" or "EPRI PWR Secondary Water Chemistry Guidelines, Revision 4, November 1999, EPRI TR-102134".

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV D2-1	Steam Generator (Once-Through)	All	D2-4		General Comment—Suggest that aging effects that apply to all OTSG items (e.g., loss of external material by boric acid wastage of ferritic items and cumulative fatigue damage be discussed at the very beginning and not repeated thereafter). At present, CUF is listed for selected items and not for all OTSG items.
IV D2-2	Steam Generator (Once-Through)	Section D2.	IV D2-3		The secondary water side, excluding portions of the main steam system, feedwater system, and auxiliary feedwater system, is classified as Class A Quality Standard. The main and auxiliary feedwater headers and riser piping were designed to ANSI B31.1.
IV D2-3	Steam Generator (Once-Through)		D2-4		It appears that the OTSG support skirt transition ring was omitted. Suggest that this item be added. It is fabricated from low alloy steel. Aging effects include cracking at weld and loss of material. AMP includes Exam. Category B-H and Boric Acid Wastage.
IV D2-4	Steam Generator (Once-Through)	D2.1.1. D2.1.2	IV D2-4		Material—the tube sheets are clad with Alloy 82/182. The heads are clad with stainless steel Environment—Chemically treated borated water up to 340 °C for upper and lower heads. Both chemically treated borated water up to 340 °C and secondary-side water chemistry for tubesheets. Aging Effect—Suggest cracking at welded joints Aging Mechanism—see discussion of items C2.1.1 thru C2.1.4 for RCS piping & fittings. If SCC and IGSCC apply to SS and Alloy 82/182 cladding with subsequent propagation to the base metal then say so. Aging Management Program—ASME Section XI, Subsection IWB, Examination Category B-B for primary pressure boundary welds.

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV D2-5	Steam Generator (Once-Through)	D2.1.3	IV D2-6		Aging Effect-Cumulative Fatigue Damage See the general comment regarding fatigue at the beginning of this section. A resolution path for GSI-190 for license renewal should be discussed in the GALL report. Either evaluation of NUREG-6260 items or alternate solution per closure of GSI-190.
IV D2-6	Steam Generator (Once-Through)	D2.1.7	IV D2-16		This entry is confusing since the primary pressure boundary has only one Alloy 600 drain nozzle and no instrument nozzles. The drain nozzle is fabricated from 1-inch Schedule 160 Alloy 600 pipe. The secondary pressure boundary contains vents, drains, and instrumentation connections fabricated from Alloy 600 and carbon steel. Suggest a separate entry for secondary nozzles as Examination Category B-E and B-P do not apply to these items.
IV D2-7	Steam Generator (Once-Through)	D2.1.8, D2.1.9	IV D2-18		Aging Management Program—AMP should list and describe Examination Category B-G-2 since the primary manway bolting is 2-inch diameter and less.
IV D2-8	Steam Generator (Once-Through)	D2.3.2	IV D2-26		Aging Effect—Cracking at Welded Joints Region of Interest—Aux. And main Feedwater Header and Piping AMP--ASME Section XI, examination Category C-F-2 should be credited for aging management of cracking in piping
IV D2-9	Steam Generator (Once-Through)		IV D2-18		Replace "thru" with "through" or use dashes, (e.g., A1.1.1 – A1.1.3)
IV D2-10	Steam Generator (Once-Through)	D2.1.1 D2.1.2 D2.1.3 D2.1.7	IV D2-5, 8 - 11, & 16 – 19.		Replace "SCC, IGSCC" or "Stress Corrosion Cracking (SCC), Intergranular Stress Corrosion Cracking (IGSCC)" with "SCC".

GALL REPORT – MECHANICAL DISCIPLINE COMMENTS
SECTION IV

GALL SECTION	TITLE	ITEM NO.	PAGE	REVIEWER	COMMENT
IV D2-11	Steam Generator (Once-Through)	D2.1.5 D2.1.6 D2.1.9 D2.3.1 D2.3.3 D2.4.1	IV D2-12, 13, 14, 15, 20, 21, 26, and 28.		Replace "Erosion/Corrosion (E/C)" with "Flow-Accelerated Corrosion (FAC)".
IV D2-12	Steam Generator (Once-Through)	D2.1.3 D2.1.7 D2.2.1	IV D2-8, 9, 10, 11, 16, 17, 18, 19, and 22.		Replace "EPRI TR-105714" with "the latest version of EPRI TR-105714" or "EPRI PWR Primary Water Chemistry Guidelines, Revision 4, March 1999, EPRI TR-105714".
IV D2-13	Steam Generator (Once-Through)	D2.1.1 D2.1.2 D2.1.4 D2.1.5 D2.2.1	IV D2-4, 5, 10, 11, 12, 13, 14, 15, 24, 25, 26, and 27		Replace "EPRI TR-102134" with " the latest version of EPRI TR-102134" or "EPRI PWR Secondary Water Chemistry Guidelines, Revision 4, November 1999, EPRI TR-102134".