NUCLEAR ENERGY INSTITUTE

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Program Director, License Renewal and Environmental Impacts
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, DC 0555-0001

PROJECT NUMBER: 690

Dear Dr. Kuo:

We are providing with this letter comments on the NRC's January 31, 2005, update material for the Generic Aging Lessons Learned (GALL) Report and Standard Review Plan (SRP). We appreciate the opportunity to review this material, and wish to compliment the NRC on the extensive improvements to these documents. We believe that when the industry comments are addressed, licensees submitting future applications will be able to achieve an improved alignment with the GALL Report, thus improving LRA preparation and review efficiencies.

We are providing a summary of our most significant comments in this letter, and are attaching more detailed comments in the numerous attachments to the letter. The significant comments in the body of this letter address four general areas, and should have a high priority for resolution:

- Electrical components (GALL)
- Mechanical components (GALL)
- Civil/Structural components (GALL)
- SRP

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Electrical Components (GALL)

Principal comments related to electrical components are as follows.

1. ISG-17 and XI.E4, Aging Management Program for Bus Ducts

   The draft of NUREG-1801, Rev. 1 (GALL update) proposes a new aging management program (AMP), XI.E4, for "bus ducts." The proper electrical and industry standard (IEEE Std. 27-1974/ANSI C37.20-1969 and Supplements, ANSI/IEEE C37.100-1981) designation for this equipment is "Metal-Enclosed Bus" (MEB). The service life of a properly designed, installed, and maintained MEB can be unlimited. The industry presents an AMP for MEB (non-segregated phase, segregated phase, and isolated phase) along with supporting technical basis in the attached specific comments. The technical basis is actual plant operating experience, vendor design information, and industry standards. This AMP is an alternative to the new program (Section XI.E4) presented in the GALL Update.
   Since a MEB AMP was not required for the majority of nuclear plants with renewed licenses, provisions should exist in GALL for a licensee to show that their MEB materials and environment do not produce aging effects requiring management and an MEB AMP is therefore not required. Examples are provided in the alternate program. Likewise based on previously approved staff positions, provisions to not require an AMP should also apply to GALL Section XI.E5, "Aging Management Program for Fuse Holders."

   If an MEB AMP is needed, the proposed recommendation to check for proper torque (even on a sample basis) is contrary to vendor recommendations and good bolting practices [*Electrical Connectors Application Guidelines*, EPRI Technical Report 1003471, December 2002]. This recommendation should be deleted from the GALL. Supporting criteria are provided in the program presented by the industry. The alternative methods for checking connections proposed as recommendations by the GALL update, either resistance measurement across the connection or thermography of the connection, have practical considerations that limit or prevent their use. They should not be a recommendation of the GALL AMP.

2. <u>XI.E6, *Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements*</u>

The draft of NUREG-1801, Rev. 1 (GALL update) proposes a new AMP for electrical cable connections not subject to 10 CFR 50.49 environmental qualification requirements. This program would include "connections used to connect cables to other cables or electrical devices."

This program has not been proposed previously, was not identified in a formal or draft Interim Staff Guideline (ISG), and was not required during any prior licensee renewal application review. As described in the attached specific comments, there is no operating experience (OE) indicating a need for this program. It should be eliminated since the subcomponents identified for this program are either not subjected to adverse environments or stressors that could produce aging effects requiring management or are adequately managed by other AMPs.

Normal design and construction practice in the nuclear industry is that electrical power and instrumentation cables are one continuous run from the supply to the load. As an example, a power cable to a motor would have a connection in the switchgear and a connection in the motor connection box. Both of these terminations are part of an active component, and therefore not subject to aging management. Connections that may occur outside active components are splices (typically occurring at containment penetrations). Splices are designed for the environmental conditions found in the field and are addressed by XI.E1, *Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements.*

The metallic portions of electrical power and instrumentation connections are not exposed outside of active equipment and are not accessible while energized. End-to-end cable connections, connections to pigtails, or bolted connections at penetrations are taped or sleeved within heat-shrink splice materials already included in aging management program XI.E1. Systems that are in scope for license renewal where there may be connectors in the field are limited to Nuclear Instrumentation and Radiation Monitoring and these are included in XI.E2, *Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits*. Most of these connectors are covered by splice material, protected by a terminal box, or installed in a mild environment where stressors are not present.

Mechanical Components (GALL)

Principal comments related to mechanical components are the following.

1. ASME Section XI

   A footnote was added to several AMP descriptions related to ASME Section XI. The Bases Document states that the footnote was to permit reference to the future versions, which is a worthwhile objective since applicants must take exception to the programs when a specific version of the code is referenced. The footnote acknowledges that the ASME code required under 10 CFR 50.55a changes periodically; however, it does not clearly state the applicant can credit the code version applicable in their current and future 10 year ISI plans. As the footnote is currently written, applicants will still have to take exceptions to the programs. The footnote should be revised to indicate that the process to update 10 CFR 50.55a considers attributes of the ASME code relevant to the management of aging effects. This will allow ISI AMPs based on versions of the code endorsed in the future by 10 CFR 50.55a to be accepted as effective aging management programs.

2. EPRI Water Chemistry Guidelines

   EPRI water chemistry guidelines change with experience, and plant chemistry programs generally adopt the new guidelines as they evolve. Since later editions than those listed in GALL will clearly be used, applicants do not want to commit to a specific edition. We recognize that the NRC does not review EPRI water chemistry guidelines except in conjunction with the review of a license renewal application or other licensing issue. The GALL water chemistry program description already permits use of "later revisions or updates of these reports as approved by the staff." The changes proposed to AMP XI.M2, *Water Chemistry*, permit the applicant to credit later versions of the guidelines when reviewed and approved for implementation by the staff in applicant Safety Evaluation Reports. This will reduce the number of exceptions taken for this program.

3. <u>Bolting and Loss of Preload</u>

   Proper joint preparation and make-up in accordance with industry standards is expected to preclude loss of preload in closure bolting applications where stress relaxation due to metallic creep is not a concern. Factors other than high-temperature stress relaxation that could contribute to a loss of preload in closure bolting applications, such as vibration, should not result in loosening of a properly assembled bolted joint. The loosening of closure bolting due to operating conditions such as significant vibration is a design-driven or maintenance-driven occurrence caused by inadequate joint design or improper fastener installation rather than an aging effect.

4. <u>Aging Management Programs Replaced by Commitments to Submit Information in the Future</u>

   Two aging management programs (AMPs) described in GALL Chapter XI were deleted and text was added stating that licensees must commit to submit information in the future (after the renewed license is issued) rather than reference these two AMPs. The two AMPs were XI.M11, *Nickel-Alloy Nozzles and Penetration,* and XI.M16, *PWR Vessel Internals,* which have both been programs reviewed and approved by the NRC in previous license renewal applications. It appears inappropriate to delete two staff-approved programs from GALL, since the stated purpose of GALL is to document previously acceptable programs to reduce the burden on NRC review resources. We suggest the XI.M11 and XI.M16 AMPs be restored to GALL since they represent previously approved programs. We also suggest that the text regarding submitting future information be deleted since it is inconsistent with the majority of previously approved license renewal applications.

5. <u>One-Time Inspections</u>

   The industry is concerned that the revisions to the GALL description of GALL aging management program XI.M32 impose expectations that are unnecessary and overly prescriptive. Inspection techniques should be appropriate for the component, material, environment, aging effect combination, whereas the proposed revision to XI.M32 imposes ASME code inspection requirements on non-code components. Experience from license renewal applications indicates that the guidance in XI.M32 in the 2001 version of GALL was adequate to establish effective one-time inspections.

6. Proposed Aging Management Programs

Two new aging management program descriptions have been developed for inclusion in GALL Chapter XI. The External Surfaces Monitoring Program is proposed for visual monitoring of system external surfaces. The Flux Thimble Tube Inspection Program is proposed to monitor thinning of the flux thimble tube walls. These programs replace the plant-specific programs listed in numerous lines of the GALL tables.

7. Comments Affecting Multiple Line Items

A number of comments affect multiple lines in the GALL mechanical systems tables. These issues include:

a. New Line Items – Proposed new line items are listed at the end of each system table.

b. Stainless Steel in Treated Borated Water – GALL does not currently address the aging effect of loss of material for stainless steel in treated borated water. New line items have been proposed.

c. Water Chemistry Reference – The reference to the specific EPRI document should not be included in the Aging Management Program column. These details are addressed in the AMP write-ups.

d. Integration of CASS with Stainless Steel – To simplify GALL, CASS should be treated as a subset of stainless steel, with the exception of when CASS is subject to thermal embrittlement.

e. External Environments – Changes are proposed to move most external surface and external bolting lines to the external tables.

f. Heat Exchanger Components Description – The designation of the tube side or shell side of a heat exchanger unnecessarily limits the applicability of the GALL line item. Changes to eliminate the designators have been proposed.

Civil/Structural Comments (GALL)

Principal comments related to civil/structural elements and components are as follows.

1. Areas Of GALL Requiring Correction

   Two significant examples where corrections are needed are:

   a. The aging management program throughout Chapter III should be "Structures Monitoring Program" instead of "ASME Section XI, Subsection IWL," since this chapter of the GALL deals with structures other than containment and those structures are monitored by the "Structures Monitoring Program". ASME Section XI, Subsection IWL applies to concrete containments.

   b. Chapter III Group 6 (Water Control Structures) does not differentiate between accessible and inaccessible areas for concrete structures as do other groups in Chapter III. Also, statements in the AMP column indicate that if specific conditions are met, aging management of concrete is not required. This is inconsistent with the NRC position in ISG-3.

   Other corrections are as noted in the marked-up version provided.

2. Consolidation of GALL Sections IIIA and IIIB

   This comment is intended to eliminate duplication and provide for a more efficient review while maintaining the necessary detail of the GALL report. The proposed consolidation has been shown in the marked-up version provided as an attachment.

Standard Review Plan

Revised language is provided for SRP Section 4.4.1 to provide better characterization of the licensing bases of plants with respect to mechanical equipment qualification programs and how those programs would be treated in LR. Most plants do not have such programs. The revised language addresses the treatment of those programs in license renewal, but does not suggest that all plants have or should have such programs.

Changes to the GALL report as a result of the industry comments on that report should be reflected in changes to the SRP, as appropriate. No additional significant high-level changes to the SRP are proposed, but detailed comments are provided in an attachment.

We look forward to meeting with you on April 20 and 21 to discuss these comments. If you have any questions, please contact me (202-739-8080; am@nei.org) or Fred Emerson (202-739-8086; fae@nei.org).

Sincerely,

Alexander Marion

Alexander Marion

Attachments

c: Mr. Ken Chang, NRR
Mr. Jerry Dozier, NRR
NRC Document Control Desk

Table of Contents

Metal-Enclosed Bus

1. Purpose

To present an aging management program (AMP) for metal-enclosed bus[1] (non-segregated phase, segregated phase, and isolated phase) along with the supporting technical basis. The technical basis is actual plant operating experience, vendor design information, and industry standards. This AMP is an alternative to the program (Section XI.E4) proposed in the draft of NUREG-1801, Rev. 1. The recommended program uses industry standard terminology as defined within this document.

2. Overview

The service life of a properly designed and installed metal-enclosed bus can be unlimited if proper periodic inspections and maintenance are performed[2]. Industry operating experience (OE) provides insights into proper design, installation, operation, and preventive maintenance, as well as the degradation associated with the lack thereof. For example, industry experience indicates loosening of properly designed and installed bus bar bolted connections is not an industry problem for bus that is not overloaded[3]. No aging effects associated with bus bar sleeving/insulation have affected the intended function of metal-enclosed bus[4]. The association of credible aging mechanisms or stressors with specific aging effects will be addressed based on industry operating experience.

Proper design, installation, and operation of metal-enclosed bus should preclude aging effects that would require management for an unlimited service life. Periodic inspections and maintenance are performed to verify adequate design, installation, and operation of metal-enclosed bus. If periodic inspection and maintenance are not performed at the proper frequency, then potential aging effects may require management for license renewal. The timeline for bus failure when an aging effect is present involves many years of slow decline of bus condition that may be followed by rapid degradation within a few months.

Metal-enclosed bus is widely used in the power industry as well as other industrial installations. The design of metal-enclosed bus for nuclear power plants is not unique, but based on many prior years of non-nuclear operating experience. Numerous national standards apply to metal-enclosed bus that incorporate lessons learned from operating experience to ensure proper design, installation, maintenance, and operation of metal-enclosed bus.

3. Comparison of Industry position to NUREG-1801 (Rev. 1)

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[1] The term "bus duct" is misleading, since it can be interpreted to have different meanings. The generic term that will be used through this document is "metal-enclosed bus". [Ref. 3.5.2and 3.5.3]
[2] Based on information contained in EPRI TR-112784, *Isolated Phase Bus Maintenance Guide*, Final Report, May 1999, [Ref. 3.5.1]. Section 9, Visual Inspections, draws this conclusion.
[3], IN 2000-14 and IN 89-64 support this position. This position was accepted by the staff in NUREG-1796.
[4] This can be supported with information obtained from IEEE Std. 27, EPRI TR-112784, and NUREG-1796.

3.1. Industry Position on Metal-Enclosed Bus Program

Proper design, installation, and operation of metal-enclosed bus will ensure a long failure-free life. Routine inspections and maintenance should be performed to ensure that the design, installation and operation are in accordance with manufacturer specifications. If this has not been effective based on plant OE, then an AMP that addresses the identified aging concern may be required for license renewal. If an AMP is needed for license renewal, the use of a general visual inspection of the metal-enclose bus will be adequate to identify aging effects. Based on industry guidance, the entire bus, inside and out, should be visually inspected at least once every ten years. Accepted inspections and general maintenance requirements should be incorporated into an aging management program to maintain the intended functions of metal-enclosed bus. Plant-specific operating experience or specific vendor requirements could warrant a higher level of action on a plant-specific basis. Visual inspection is appropriate in the determination of bus conditions, including joint and insulator integrity. A 10 CFR 50, Appendix B, corrective action program is sufficient to resolve anomalies found during visual inspections. Testing, such as insulation resistance checks, thermography, continuity, over-potential, or partial discharge, can supplement visual inspection or can assist with anomaly resolution.

Since the purpose of NUREG-1801 is to provide aging management programs that have been previously reviewed and approved, a review of past precedents from license renewal SERs was performed. The majority of previous SERs show that an AMP for metal-enclosed bus is not always needed, and when an AMP is required the attributes are based on plant-specific conditions. Details obtained from the SER review are provided in Table 3.1-1 located in Attachment 1.

One of the issues related to the industry position on metal-enclosed bus is the identification of materials, credible aging mechanisms or stressors, and credible aging effects.

This section provides the technical basis for the aging management program (AMP) described in Section 3.2. The program description section of the aging management program is divided into four paragraphs that provide component descriptions (terminology), industry operating experience, conductor aging effects, and AMP purpose.

3.2. Industry Evaluation of Aging Effects

The License Renewal Electrical Handbook (LREH) [Ref. 5.4] provides information for metal-enclosed bus (i.e. phase bus) aging effects, and this document provides additional information, which will be added to the next revision of the LREH. All of the approved license renewal SERs, which are listed in Attachment 1, were reviewed for information associated with metal-enclosed bus[5].

3.3. Metal-Enclosed Bus Inspection Program

Program Description

Metal-enclosed bus is an assembly of rigid conductors (bus) with associated connections, joints, and insulating supports (bus support) within a grounded metal enclosure. Bus is a conductor, or group of conductors, that serve as a common electrical connection for two or more circuits (i.e. switchgear, transformers, generators, etc.). A bus support is an insulating support for a bus. In general, there are three basis types of metal-enclosed bus construction: nonsegregated phase, segregated phase, and isolated phase. Nonsegregated phase bus is one in which all phase conductors are in a common metal enclosure without barriers between phases. Segregated phase bus is one in which all phase conductors are in a common metal enclosure, but are segregated by metal barriers between phases. Isolated phase bus is one in which each phase conductor is enclosed by an individual metal housing separated from adjacent conductor housings by an air space. Metal-enclosed bus is used in power systems to connect various elements in electric power circuits such as switchgear, large transformers, the main generator, and diesel generators.

Industry operating experience indicates that most failures of metal-enclosed bus have been caused by external events or items impacting the metal-enclosed bus, maintenance activities, inadequate preventative maintenance (PM) and inspections, exceeding operating limits, and inadequate design margin. As indicated by operating experience, most failures are not the result of aging effects. Stressors such as thermal (ohmic) heating and moisture and debris internal to the bus housing should be considered in design and operation, but they should not cause aging effects that require management. The applicable aging effects are reduced insulation resistance and increased connection resistance.

Metal-enclosed bus exposed to excessive ohmic heating during operation may experience loosening of bolted connections resulting from the repeated cycling of connected loads. This phenomenon is a design issue, not an aging effect. A properly designed bus should not be heavily loaded. In addition, bus loads are not typically cycled, since this would create large voltage transients for the plant. The Sandia report SAND96-0344 [Ref. 5.9] does not address metal-enclosed bus, so this reference is inappropriate and misleading. The effects associated with bolted

[5] The following SERs had metal-enclosed bus that was subject to aging management review: NUREG-1723, NUREG-1769, NUREG-1779, NUREG-1782, NUREG-1772, NUREG-1785, NUREG-1786, and NUREG-1796.

connections for cable are not necessarily applicable to bolted bus connections. A reference for EPRI TR-104213 [Ref. 5.10] would provide appropriate guidance. NRC Information Notice 2000-14 [Ref. 5.8] and the associated LER were inconclusive regarding the cause of the event, but the most probable contributing cause was inadequate design margin. A single load cycle from an event is not sufficient to cause loosening of bolted connections.

The purpose of the aging management program is to manage the effects of aging on metal-enclosed bus if the plant-specific aging management review identifies aging effects requiring management. In this aging management program, a visual inspection of the interior portions of the metal-enclosed bus will be performed to identify aging degradation of insulating and conductive components. This inspection will also verify the absence of water or debris. The external portions of metal-enclosed bus and structural supports will be inspected in accordance with a plant-specific structures monitoring program. Based on industry recommendations in EPRI TR-104213, bolted connections will not and should not be checked for proper torque unless guidance is provided by the bus manufacturer, since overtorquing can cause loose connections resulting in increased connection resistance.

Evaluation

1. Scope of Program

   NUREG-1801 (Rev. 1), Scope

   "This program applies to bus ducts within the scope of license renewal."

   Industry Recommended Program Scope

   This program applies to metal-enclosed bus subject to aging management review that has aging effects requiring management.

   Industry Rationale

   Defined terminology is important in technical documents to ensure the exact meaning is understood by all involved. The term "bus duct" is not defined in the License Renewal Electrical Handbook (LREH) [Ref. 5.4], and its use is limited to a few occurrences in the document. The term "bus duct" is used in Table 5-1 and Chapter 7 as an example for the phase bus commodity group. Phase bus is used in the LREH to capture the examples for bus listed in App. B, Item 77 of NEI-95-10, cables and connections, bus, electrical portions of electrical and I&C penetration assemblies, which are isolated-phase bus, nonsegregated-phase bus, segregated-phase bus, and switchyard bus. The term "bus duct" has been used in certain plant-specific aging management programs. One plant-specific program used the term "bus duct" to mean only the bus bar enclosure, and another plant-specific program used the term "bus duct" to mean the bus bar. "Bus duct" while used at specific plants, is not the appropriate term for generic use because of the various meanings. The LREH is being revised to eliminate possible confusion with this terminology. IEEE-C2-1997, "National Electric Safety

Code" [Ref. 5.5], defines a duct as a single enclosed raceway for conductors or cable. IEEE-C2-1997, Section 181 refers to metal-enclosed bus and isolated-phase bus as well as the term busway, and Section 322 discusses ducts for underground conductors, but the term "bus duct" is not defined. The IEEE Standard for Switchgear Assemblies Including Metal-Enclosed Bus (IEEE Std. 27-1974) [Ref. 5.2] does not use the term "bus duct". This standard defines "metal-enclosed bus" as an assembly of rigid conductors with associated connections, joints, and insulating supports within a grounded metal enclosure. In general, three basic types of construction are used: nonsegregated phase, segregated phase, or isolated phase. The definition is also in accordance with IEEE Standard Definitions for Power Switchgear (IEEE C37-100-1981) [Ref. 5.3].

2. Preventive Actions

NUREG-1801 (Rev. 1), Preventive Actions

"This is an inspection program and no actions are taken as part of this program to prevent or mitigate aging degradation."

Industry Recommended Preventive Actions

This is an inspection program only, which requires no actions to prevent or mitigate aging effects.

Industry Rationale

Inspection and cleaning of dirt, dust, or other debris will prevent and mitigate aging degradation. If these preventive actions are being performed as routine maintenance separate from the AMP, then the AMP may not be needed. Preventive actions are not needed, if the inspection AMP is used.

3. Parameters Monitored/Inspected

NUREG-1801 (Rev. 1), Parameters Monitored/Inspected

"A sample of accessible bolted connections will be checked for proper torque or connection resistance using a low range ohmmeter. This program will also provide for the inspection of the internal portion of the bus ducts for cracks, corrosion, foreign debris, excessive dust buildup, and evidence of water intrusion. The bus insulating system will be inspected for signs of embrittlement, cracking, melting, swelling, or discoloration, which may indicate overheating or aging degradation. The (internal) bus supports will be inspected for structural integrity and signs of cracks."

Industry Recommended Parameters Monitored/Inspected

Parameters visually inspected are the bus insulating system and conductor condition, which includes bolted connections. Specifically, inspections should monitor for discoloration, cracking, surface contamination, foreign material, moisture, or corrosion.

Industry Rationale

NUREG1800, Appendix A, Section A.1.2.3, "Aging Management Program Elements", contains guidance for the ten program elements for AMPs. Section A.1.2.3.3, "Parameters Monitored or Inspected", states that, "The parameters monitored or inspected should be linked to the degradation of the particular structure and component intended function(s)." For a condition-monitoring program, the parameter monitored or inspected should detect the presence and extent of aging effects.

A properly designed bolted joint with proper torque should not become loose if operated within the limits of the design. Section 8.2 of TR-104213 discusses inspection of electrical bolted joints. Section 8.2 states:

> "Inspect bolted joints for evidence of overheating, signs of burning or discoloration, and indications of loose bolts. The bolts should not be retorqued unless the joint requires service or the bolts are clearly loose. Verifying the torque is not recommended. The torque required to turn the fastener in the tightening direction (restart torque) is not a good indicator of the preload once the fastener is in service. Due to relaxation of the parts of the joint, the final loads are likely to be lower than the installed loads. However, this load reduction has little effect on electrical conductivity or joint performance."

Proper design of bus conductor bolted connections is vital to long life. One of the most common mistakes is to over-torque bolts on aluminum connections, which causes rapid cold flow (creep) from under the bolts. The bolts seem to be loose and are tightened again, repeating the cycle. Manufacturer recommendations for joint assembly and bolt torque values must be followed unless problems develop, which will require a redesign of the bolted joint. This is further supported by NP-5067, TR-112784, and EPRI 1003471 [Ref. 5.11, 5.1, and 5.15].

As stated in EPRI TR-104213 and NP-5067, bus connections should not be torque checked, since over-torqued connections will create additional problems. The bolts should not be retorqued unless the joint requires service or the bolts are clearly loose based on visual inspection. The absence of discoloration, cracking, or surface contamination of the bolted joint or insulation covering the joint provides positive indication that the bolted connections are not loose. Industry experience indicates loosening of properly designed and installed bus bar bolted connections is not a problem for bus that is not overloaded. NUREG-1796 (page 3-481) [Ref. 5.12] accepts this position, and summarizes, "...there is no plant or industry operating experience that shows that there is a credible aging mechanism pertaining to the bus bar bolts; therefore, no aging management other than visual inspection is required."

4. Detection of Aging Effects

NUREG-1801 (Rev. 1), Detection of Aging Effects

"A sample of accessible bolted connections will be checked for proper torque or connection resistance. Internal surfaces of the bus ducts will be visually inspected for aging degradation of insulating material and for foreign debris and excessive dust buildup, and evidence of water intrusion. This program will be completed before the end of the initial 40-year license term and every 10 years thereafter. This is an adequate period to preclude failures of the bus ducts since experience has shown that bus duct aging degradation is a slow process. A 10-year inspection frequency will provide two data points during a 20-year period, which can be used to characterize the degradation rate."

Industry Recommended Detection of Aging Effects

A sample of accessible internal surfaces of the metal-enclosed bus will be visually inspected for aging degradation that can cause reduced insulation resistance or increased connection resistance. This program will be completed before the end of the initial 40-year license term and every 10 years thereafter. This is an adequate period to preclude failures of the bus since experience has shown that bus aging degradation is a slow process.

Industry Rationale

NUREG1800, Appendix A, Section A.1.2.3, "Aging Management Program Elements", contains guidance for the ten program elements for AMPs. Section A.1.2.3.4, "Detection of Aging Effects", states that, "Detection of aging effects should occur before there is a loss of the structure and component intended function(s). The parameters to be monitored or inspected should be appropriate to ensure that the structures and component function (s) will be adequately maintained for license renewal under all CLB design conditions. This includes aspects such as method or technique (e.g., visual, volumetric, surface inspection), frequency, sample size, data collection and timing of new/one-time inspections to ensure timely detection of aging effects. Provide information that links the parameter to be monitored or inspected to the aging effects being managed." This program element describes "when," "where," and "how" program data are collected (i.e., all aspects of activities to collect data as part of the program).

The method or technique and frequency may be linked to plant-specific or industry-wide operating experience. Provide justification, including codes and standards referenced, that the technique and frequency are adequate to detect the aging effects before a loss of system or component (SC) intended function. A program based solely on detecting SC failures is not considered an effective aging management program.

When sampling is used to inspect a group of SCs, provide the basis for the inspection population and sample size. The inspection population should be based on such aspects of the SCs as a similarity of materials of construction, fabrication, procurement, design, installation, operating environment, or aging effects. The sample size should be based on such aspects of the SCs as the specific aging effect, location, existing technical information, system and structure design, materials of construction, service environment, or previous failure history. The samples should be biased toward locations most susceptible to the specific aging effect of concern in the period of extended operation. Provisions should also be included on expanding the sample size when degradation is detected in the initial sample.

For metal-enclosed bus, the stressors and mechanisms discussed produce only two aging effects: reduced insulation resistance or increased connection resistance. For example, cracking or chipping of insulation or insulators, or accumulation of moisture or debris (surface contamination) of insulators can possibly reduce the BIL rating for bus, which is reduced insulation resistance (IR). Also, loosening of bolted connections, corrosion, or oxidation can increase the resistance of an electrical connection, which is increased connection resistance. Either aging effect can produce increased heating or arcing, which allows detection of the aging effect through visual inspections. As previously discussed torque checks should not be performed.

5. Monitoring and Trending

NUREG-1801 (Rev. 1), Monitoring and Trending

"Trending actions are not included as part of this program because the ability to trend inspection results is limited. Although not a part of the program, trending would provide additional information on the rate of degradation."

Industry Recommended Monitoring and Trending

Trending actions are not included as part of this program because the ability to trend inspection results is limited. Although not a part of the program, trending would provide additional information on the rate of degradation.

Industry Rationale

No changes recommended.

6. Acceptance Criteria

NUREG-1801 (Rev. 1), Acceptance Criteria

"Bolted connections need to meet minimum torque specifications or low resistance appropriate for the application. Bus ducts are to be free from unacceptable, visual indications of surface anomalies, which suggest that conductor insulation degradation exists. Additional acceptance criteria include no unacceptable indication of corrosion, cracks, foreign debris, excessive dust buildup or evidence of water intrusion. An unacceptable indication is defined as a noted condition or situation that, if left unmanaged, could lead to a loss of intended function."

Industry Recommended Acceptance Criteria

The accessible metal-enclosed bus is to be free from unacceptable visual indications. Unacceptable visual indications are insulation material surface anomalies, which suggest bolted connection or insulation degradation; excessive foreign material (e.g., dust, debris, etc.), surface contamination, or moisture, which could create tracking paths reducing IR. An unacceptable indication is defined as a noted condition or situation that, if left unmanaged, could lead to a loss of intended function.

Industry Rationale

Torque checks and measurement of connection resistance is not recommended. See previous discussion under Parameters Monitored/ Inspected. The acceptance criteria were modified to reflect the change in parameters monitored.

7. Corrective Actions

NUREG-1801 (Rev. 1), Corrective Actions

"Pursuant to 10 CFR Part 50, Appendix B, further investigation and evaluation are performed when an acceptance criterion is not met. Corrective actions may include but are not limited to increased inspection frequency, replacement, or rework of the affected bus duct insulation components. If an unacceptable condition or situation is identified, a determination is made as to whether the same condition or situation is applicable to other accessible or inaccessible bus ducts. As discussed in the appendix to this report, the staff finds the requirement of 10 CFR Part 50, Appendix B, acceptable to address corrective actions."

Industry Recommended Corrective Actions

Pursuant to 10 CFR Part 50, Appendix B, further investigation and evaluation are performed when an acceptance criterion is not met. Corrective actions may include but are not limited to increased inspection frequency, replacement, or rework of the affected metal-enclosed bus components. If an unacceptable condition or situation is identified, a determination is made as to whether the same condition or situation is applicable to other accessible or inaccessible metal-enclosed bus. As discussed in the appendix to this report, the staff finds the requirement of 10 CFR Part 50, Appendix B, acceptable to address corrective actions.

Industry Rationale

No changes recommended.

8. Confirmation Process

NUREG-1801 (Rev. 1), Confirmation Process

"As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process."

Industry Recommended Confirmation Process

As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process.

Industry Rationale

No changes recommended.

9. Administrative Controls

NUREG-1801 (Rev. 1), Administrative Controls

"As discussed in the appendix to this report, the staff finds the requirement of 10 CFR Part 50, Appendix B, acceptable to address administrative controls."

Industry Recommended Administrative Controls

As discussed in the appendix to this report, the staff finds the requirement of 10 CFR Part 50, Appendix B, acceptable to address administrative controls.

Industry Rationale

No changes recommended.

10. Operating Experience

NUREG-1801 (Rev. 1), Operating Experience

"Industry experience includes failures of bus ducts caused by cracked insulation and moisture or debris buildup internal to the bus duct. Experience has also shown that bus work in the bus ducts exposed to appreciable ohmic heating during operation may experience loosening of bolted connections related to repeated cycling of connected loads."

Industry Recommended Operating Experience

Adequate visual inspections and appropriate corrective actions (PMs) have been effective at preventing bus failures with the exception of failures directly attributable to inadequate design. Metal-enclose bus failures documented in IN 2000-14, IN 98-36, and IN 89-64 [Ref. 5.8, 5.7, and 5.6] were caused by inadequate design or maintenance activities, rather than resulting from aging effects.

Industry experience indicates that loosening of properly designed and installed bus bar bolted connections is not an industry problem for bus that is not overloaded. This is consistent with NUREG-1796 (page 3-481), which states, "...there is no plant or industry operating experience that shows that there is a credible aging mechanism pertaining to the bus bar bolts; therefore, no aging management other than visual inspection is required."

Industry Rationale

The NRC has issued 3 information notices (INs) related to metal-enclosed bus, which are referenced in the draft revision to NUREG-1801. [Ref. 5.8, 5.7, and 5.6]

The most recent NRC communication associated with metal-enclosed bus is IN 2000-14, which does not completely capture the information provided by PG&E in LER 2000-004-01 from Diablo Canyon, Unit 1. The LER states that the cause of the electrical fault could not be conclusively determined due to the absence of physical evidence (since the bus connection that failed was vaporized or melted), but is believed to be associated with long-term degradation, and/or inadequate preventative maintenance (PM) exacerbated by a marginal design. Corrective actions included a new PM program and upgrades to the 4kV and 12kV nonsegregated buses on both units. IN 2000-14 addresses a fault-based event, not aging. Multiple events, most of which were design related complicated one bus failure, which eventually lead to multiple failures. A combination of poor design factors was part of the root cause: mixture of aluminum and copper bus bars, poor silver-plating on the aluminum bar, corrosion induced on the aluminum bar due to the PVC boot material, and undersized splice plates of wrong material not centered on the bus bar, reducing contact area. The splice plates were undersized aluminum rather than larger copper plates, used by the vendor during tests to determine design temperature rise to meet IEEE 37.20-1969 of 65°C. The bus was routinely loaded to 2100 amps and sometimes to 2250 amps, its rating limit. This caused the bus to exceed design conditions for some time. Torque relaxation likely occurred due to the overheating and bus bar expansion and contraction. A 1995 explosion of an auxiliary transformer that physically displaced the bus could have also contributed to the low torque values. Inability to isolate the bus caused a small event to propagate to multiple buses and major damage.

Industry operating experience discussed in NRC IN 98-36 provides examples of bus failures as a result of maintenance activities near exposed conductors, and inadequate inspection and maintenance of the bus metal enclosure. The first example in not aging related, and the second example would be covered by a site-specific structural monitoring program. IN 98-36 was excluded since it addresses event-driven faults caused by impact of an external foreign object (roofing materials), direct water leakage or moisture intrusion, inadequate design or assembly, or mis-operation, including an event at PBNP where a bus duct heater breaker was open for an extended period of time (exceeding one year). These are all events due to causes other than aging. Therefore, no aging effects were presented in this IN.

Other industry operating experience is discussed in NRC IN 89-64, which is based on LERs 88-010, Palo Verde Unit 1; 88-001and 87-009, Kewaunee; 87-001-01, Millstone Unit 1; 83-067, Sequoyah Unit 1; and 89-008, Browns Ferry Unit 2. This industry OE identifies degradation of Noryl insulation plus the presence of contaminates (moisture or debris). In one instance, the insulation degradation was attributed to the type of joint compound used in the manufacturing process. IEEE Standard 27 does not require bus to be insulated, since bus supports (insulators) can provide greater limits (temperature and insulation level) than bus bar insulation. Typically only bus constructed of rectangular bar is sleeved. IEEE C2-1997 and IEEE Std. 27-1974 do not rely on bus bar insulation/sleeving to obtain the required basic impulse insulation level (BIL) rating. The designed air gap and the insulating supports (insulators) provide the required BIL rating for metal-enclosed bus. Bus bar sleeving or insulation provides additional insulation for phase-to-phase shorts from large debris, which should not occur in metal-enclosed bus. In many instances the application of sleeving, insulating boots, or tape is used to prevent oxidation of bolted connections. The failures noted in IN 89-64 resulted from either an accumulation of water or debris, inadequate design, manufacturing defect, or an environment that caused deterioration of insulation. This will have to be addressed in each plant-specific review, since the construction, materials, and bus bar environments at each plant could be different. Each plant must conclude that no aging effects require management for the metal-enclosed bus installed in their plant, or they will need to have an aging management program.

Unless a large object is inserted inside the metal-enclosure of a bus, a phase-to-phase tracking path requires a tracking path from one phase across the bus support (insulator) to the grounded enclosure, and then requires another tracking across a different phase bus support (insulator) to another phase. Cracking of the bus bar sleeving/insulation should not be affected by the accumulation of moisture or debris.

References

IEEE Std. P1205-2000, IEEE Guide for Assessing, Monitoring and Mitigating Aging Effects on Class 1E Equipment Used in Nuclear Power Generating Stations.

~~SAND 96-0344, Aging Management Guideline for Commercial Nuclear Power Plants – Electrical Cable and Terminations, prepared by Sandia National Laboratories for the U.S. Department of Energy, September 1996.~~

~~EPRI TR-109619, Guideline for the Management of Adverse Localized Equipment Environments, Electric Power Research Institute, Palo Alto, CA, June 1999.~~

Information Notice 89-64, "Electrical Bus Bar Failures."

Information Notice 98-36, "Inadequate or Poorly Controlled, Non-Safety-Related Maintenance Activities Unnecessary Challenged Safety Systems."

Information Notice 2000-14, "Non-Vital Bus Fault Leads to Fire and Loss of Offsite Power."

<u>Industry Rationale</u>

Add the references provided with this document, and remove the reference lined through above.

SAND96-0344 [Ref. <u>5.9</u>] was referenced for loosening of bolted connections for metal-enclosed bus in the draft of NUREG-1801. SAND96-0344 analyzes potential aging mechanisms and their effects on low- and medium-voltage electrical cable and terminations, and provides guidelines for managing significant degradation mechanisms. This document does not address metal-enclosed bus, so this reference is inappropriate and misleading. The effects associated with bolted connections for cable and cable connections are not necessarily applicable to bolted bus bar connections. Better references for bus bar bolted connections are EPRI TR-104213 and EPRI TR-112784 [Ref. 5.10 and 5.1]. Sections 6.12 and 7.2 of TR-104213 discuss electrical connections for copper and aluminum bus. Another reference, which provides similar information, is EPRI NP-5067 [Ref. 5.11]. EPRI 1003471 [Ref. 5.15] documents connection failures from 1975 to 1997 based on data from NPRDS, EPIX, LERs, Ins, and I&E notices, which supports this conclusion. TR-104213 states for bus bar that "loosening of bolted connections because of the repeated cycling of connected loads" is really a design and installation issue.

4. Definitions

Consistent with ANSI, IEEE, and industry standards.

barrier – A partition for the insulation or isolation of electric circuits or electric arcs.

basic impulse insulation level (BIL) – A reference impulse insulation strength expressed in terms of the crest value of the withstand voltage for a standard full voltage wave.

bus - A conductor, or group of conductors, that serve as a common connection for two or more circuits.

bus support - An insulating support for a bus. It includes one or more insulator units with fittings for fastening to the mounting structure and for receiving the bus.

conducting mechanical joint – The juncture of two or more conducting surfaces held together by mechanical means.

continuous-current tests – Tests made at rated current, until temperature rise ceases, to determine that the device or equipment can carry its rated continuous current without exceeding its allowable temperature rise.

duct - A single enclosed raceway for conductors or cable.

enclosure – A surrounding case or housing used to protect the contained equipment and to prevent personnel from accidentally contacting live parts.

fault – See short circuit.

isolated phase bus – One in which each phase conductor is enclosed by an individual metal housing separated from adjacent conductor housings by an air space.

live parts – Those parts which are designed to operate at voltage different from that of the earth.

metal-enclosed bus - An assembly of rigid conductors with associated connections, joints, and insulating supports within a grounded metal enclosure. NOTE: In general, three basic types of construction are used: nonsegregated phase, segregated phase, or isolated phase.

nonsegregated phase bus – One in which all phase conductors are in a common metal enclosure without barriers between phases.

segregated phase bus – One in which all phase conductors are in a common metal enclosure, but are segregated by metal barriers between the phases.

short circuit – An abnormal connection (including an arc) of relatively low impedance, whether made accidentally or intentionally, between two points of different potential.

silver surfaced or equivalent – The term indicates metallic materials having satisfactory long-term performance and which operate within the temperature rise limits established for silver-surfaced electrical contact parts and conducting mechanical joints.

withstand voltage – The specified voltage that, under specified conditions, can be applied to insulation without causing flashover or puncture.

5. References

5.1. EPRI TR-112784, Isolated Phase Bus Maintenance Guide, Final Report, May 1999

5.2. IEEE Std. 27-1974, *IEEE Standard for Switchgear Assemblies Including Metal-Enclosed Bus*, Approved February 28, 1974, [ANSI C37.20-1969, 20a-1970, 20b-1972, 20c-1974]

5.3. IEEE C37.100-1981, *IEEE Standard Definitions for Power Switchgear*, Approved August 29, 1980

5.4. EPRI 1003057, *License Renewal Electrical Handbook*, Final Report, December 2001

5.5. IEEE-C2-1997, *National Electric Safety Code*, Approved 6 June 1996, 1997 Edition.

5.6. NRC Information Notice 89-64, *Electrical Bus Bar Failures.*

5.7. NRC Information Notice 98-36, *Inadequate or Poorly Controlled, Non-Safety-Related Maintenance Activities Unnecessary Challenged Safety Systems*

5.8. NRC Information Notice 2000-14, *Non-Vital Bus Fault Leads to Fire and Loss of Offsite Power*

5.9. SAND96-0344, *Aging Management Guideline for Commercial Nuclear Power Plants – Electrical Cable and Terminations*, Printed September 1996.

5.10. EPRI TR-104213, *Bolted Joint Maintenance & Applications Guide*, Final Report, December 1995.

5.11. EPRI NP5067, Vol. 2, *Good Bolting Practices*, December 1990.

5.12. NUREG-1796, *Safety Evaluation Report Related to the License Renewal of the Dresden Nuclear Power Station, Units 2 and 3 and Quad Cities Nuclear Power Station, Units 1 and 2*, Date Published: October 2004.

5.13. NUREG-1800, *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants*, July 2001.

5.14. EPRI 1003056, *Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools*, Revision 3, Final Report, November 2001.

5.15. EPRI 1003471, *Electrical Connector Application Guidelines*, Final Report December 2002.

5.16. EPRI 1003056, *Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools*, Revision 3, November 2001.

5.17. Federal Register Document 04-28067, Vol. 69, No. 246, December 23, 2004, *Proposed Interim Staff Guidance (ISG)-17on; Periodic Inspection of Bus Ducts for License Renewal Solicitation of Public Comment.*

ATTACHMENT 1

Table 3.1-1								
Plant	**SER**	**Bus Subject to AMR**	**Routine Maintenance Precludes Aging Effects**	**Aging Effects Managed**	**Component Managed**	**AMP**	**AMP Type**	**Parameters Monitored**
Calvert Cliffs	NUREG-1705	No	N/A	N/A	N/A	N/A	N/A	N/A
Oconee	NUREG-1723	Yes	Yes	None	None	No	N/A	N/A
ANO, Unit 1	NUREG-1743	No	N/A	N/A	N/A	N/A	N/A	N/A
E. I. Hatch	NUREG-1803	Yes	N/A	None	None	No	N/A	N/A
Turkey Point	NUREG-1759	No	N/A	N/A	N/A	N/A	N/A	N/A
North Anna	NUREG-1766	Yes	N/A	None	None	No	N/A	N/A
Surry	NUREG-1766	Yes	N/A	None	None	No	N/A	N/A
Peach Bottom	NUREG-1769	Yes	Yes	None	None	No	N/A	N/A

ATTACHMENT 1

Table 3.1-1								
Plant	**SER**	**Bus Subject to AMR**	**Routine Maintenance Precludes Aging Effects**	**Aging Effects Managed**	**Component Managed**	**AMP**	**AMP Type**	**Parameters Monitored**
St. Lucie	NUREG-1779	Yes	Yes	Reduced IR, including moisture and dust/debris; and loss of preload	Bus Bar Insulation, and bolted connections	Yes	10 year – visual and bolt torque	Perform periodic visual internal inspections of a representative sample on the nonsegregated phase buses. These inspections will include a visual inspection of the bus bar insulation for age-related defects (e.g., discoloration, cracking) and an inspection of the interior of the bus ducts for moisture or dust/debris. These inspections will include verification of a representative sample of the bus bar bolting torque values.
Ft. Calhoun	NUREG-1782	Yes	Yes	Loss of preload and Reduced IR	Bus bar (SBO restoration path only)	Yes	Periodic Surveillance and PM Program	The program's activities check bus connectors for loss of torque and degradation of insulation wrap.
McGuire	NUREG-1772	Yes	Yes	None	None	No	N/A	N/A
Catawba	NUREG-1772	Yes	Yes	None	None	No	N/A	N/A

ATTACHMENT 1

Table 3.1-1								
Plant	**SER**	**Bus Subject to AMR**	**Routine Maintenance Precludes Aging Effects**	**Aging Effects Managed**	**Component Managed**	**AMP**	**AMP Type**	**Parameters Monitored**
H. B. Robinson	NUREG-1785	Yes	No	Oxidation, loss of torque, corrosion	Bus bar bolted connections, bus duct, bus bar, bus supports	Yes	10 year - Visual inspection and torque check	A sample of accessible bolted connections will be checked for proper torque. Visual inspection of bus ducts, bus bar, and internal bus supports.
R. E. Ginna	NUREG-1786	Yes	Yes	Reduced IR, electrical failure, moisture intrusion	Electrical phase bus components	Yes	One time inspection and Periodic Surveillance and PM Program	The staff determined that the one-time inspection of the phase bus was not sufficient, so the other program was added. A visual inspection programs is performed under the PM program.
V. C. Summer	NUREG-1787	No	N/A	N/A	N/A	N/A	N/A	N/A

ATTACHMENT 1

Table 3.1-1								
Plant	**SER**	**Bus Subject to AMR**	**Routine Maintenance Precludes Aging Effects**	**Aging Effects Managed**	**Component Managed**	**AMP**	**AMP Type**	**Parameters Monitored**
Dresden	NUREG-1796	Yes	Yes for bolted connections, No for insulation resistance.	embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced insulation resistance (IR); electrical failure as the aging effects	insulators in the bus ducts	Yes	10 year visual	Accessible normally energized non-segregated bus duct internal components are visually inspected for insulator and bus bar insulation material surface anomalies, such as embrittlement, discoloration, cracking, chipping, or surface contamination. Internal components such as insulation material, bus duct support pieces, gaskets, insulating boots, taped connections, and bus bar sleeves are inspected. The visual inspections also check for evidence of water and dirt accumulation and presence of foreign material.
Quad Cities	NUREG-1796	Yes	Yes for bolted connections, No for insulation resistance.	Same as Dresden	insulators in the bus ducts	Yes	10 year visual	Same as Dresden

ATTACHMENT 1

Chapter II CONTAINMENT STRUCTURES

General Comments

1. Composite Component Descriptions

The combination of some lines to produce generic lines resulted in structure/component descriptions that included all the components previously (GALL 2001) listed in the individual lines. These comprehensive lists include components that do not apply to all system/structure tables. For example, in table II.B2.2 for Mark II BWR containments, line C-19 addresses steel elements including "Drywell; torus; drywell head; embedded shell and sand pocket regions; drywell support skirt; torus ring girder; downcomers; ECCS suction header." Some of these components (e.g., torus and torus ring girder) are not applicable to Mark II BWR containments and may lead to confusion when comparing plant AMR results to this line of GALL.

The compound names will create confusion because specific component types are listed in structures that don't contain those components. The descriptions in the rolled up lines should be split to their original configuration so that the list of structural component types matches the structure.

2. Mechanism/Program Mismatch

The aging mechanism in 'Aging Effect/Mechanism' column does not match with description in 'AMP' column for some lines. For example, item II.A2-7 calls out 'corrosion of embedded steel' in Mechanism but describes 'aggressive chemical attack' in AMP. Corrections have been proposed.

II.A1 Concrete Containments (Reinforced and Prestressed)

II CONTAINMENT STRUCTURES
A1 Concrete Containments (Reinforced and Prestressed)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
II.A1-3 (C-02)	II.A1.1-b	Concrete Dome; wall; basemat; ring girder; buttresses	Concrete	Water – flowing	Increase in porosity, permeability/ leaching of calcium hydroxide	Chapter XI.S2, "ASME Section XI, Subsection IWL" Accessible areas: Inspections performed in accordance with IWL will indicate the presence of increase in porosity, and permeability due to leaching of calcium hydroxide. Inaccessible Areas: ~~,~~An aging management program is not necessary, even if reinforced concrete is exposed to flowing water, if there is documented evidence that confirms the in-place concrete was constructed in accordance with the recommendations in ACI 201.2R-77.	No, if concrete was constructed as stated for inaccessible areas	Corrected typo Eliminated "period"(.) This comment also applies to II.A2-3 (C-02) and II.B2.2-2, (C-02), II.B3.1-2 (C-02), II.B3.2-3 (C-02).
II.A1-4 (C-08)	II.A1.1-h	Concrete Dome; wall; ~~concrete fill-in annulus,~~ basemat; ring girder; buttresses	Concrete	Air – indoor uncontrolled or air - outdoor	Reduction of strength and modulus/ elevated temperature (>150°F general; >200°F local)	Plant-specific aging management program The implementation of 10 CFR 50.55a and IWL would not be able to identify the reduction of strength and modulus due to elevated temperature. Thus, for any portions of concrete containment that exceed specified temperature limits, further evaluations are warranted.	Yes, if applicable	NRC Basis Document does not define this component. Industry does not recognize this term.

II CONTAINMENT STRUCTURES
A1 Concrete Containments (Reinforced and Prestressed)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						Subsection CC-3400 of ASME Section III, Division 2, specifies the concrete temperature limits for normal operation or any other long-term period. The temperatures shall not exceed 150°F except for local areas, such as around penetrations, which are not allowed to exceed 200°F. If significant equipment loads are supported by concrete at temperatures exceeding 150°F, an evaluation of the ability to withstand the postulated design loads is to be made. Higher temperatures than given above may be allowed in the concrete if tests and/or calculations are provided to evaluate the reduction in strength and this reduction is applied to the design allowables.		

II CONTAINMENT STRUCTURES
A1 Concrete Containments (Reinforced and Prestressed)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
II.A1-6 (C-04)	II.A1.1-d	Concrete: Dome; wall; basemat; ring girders; buttresses	Concrete	Any	~~Expansion and c~~ Cracking ***due to expansion/*** reaction with aggregates	Chapter XI.S2, "ASME Section XI, Subsection IWL." Accessible Areas: Inspections performed in accordance with IWL will indicate the presence of cracking due to reaction with aggregates. Inaccessible Areas: ***As described in NUREG-1557, investigations, tests, and petrographic examinations of aggregates performed in accordance with ASTM C295-54 or ASTM C227-50 can demonstrate that those aggregates do not react within reinforced concrete. For potentially reactive aggregates, aggregate-reinforced concrete reaction is not significant if the concrete was constructed in accordance with ACI 201.2R-77. Therefore, if these conditions are satisfied, aging management is not necessary.*** ~~Evaluation is needed if testing and petrographic examinations of aggregates performed in accordance with or *similar to* ASTM C295-54, ASTM C227-50, or ACI 201.2R-77 (NUREG-1557) demonstrate that the aggregates are reactive.~~	No, if stated conditions are satisfied for inaccessible areas	Consistent with previous GALL version.

II CONTAINMENT STRUCTURES
A1 Concrete Containments (Reinforced and Prestressed)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
II.A1-7 (C-05)	II.A1.1-e	Concrete: Dome; wall; basemat; ring girders; buttresses; ~~reinforcing steel~~	Concrete; ~~steel~~	Air – indoor uncontrolled or air – outdoor	Cracking, loss of bond, and loss of material (spalling, scaling)/ corrosion of embedded steel	Chapter XI.S2, "ASME Section XI, Subsection IWL." Accessible Areas: Inspections performed in accordance with IWL will indicate the presence of ~~increase in porosity and permeability,~~ ***cracking, loss of bond, and*** loss of material (spalling, scaling) due to ~~aggressive chemical attack (Partha Ghosal)~~ ***corrosion of embedded steel.*** Inaccessible Areas: Examination of representative samples of below-grade concrete, when excavated for any reason, is to be performed, if the below-grade environment is aggressive (pH < 5.5, chlorides > 500ppm, or sulfates > 1,500 ppm). Periodic monitoring of below-grade water chemistry (including consideration of potential seasonal variations) is an acceptable approach to demonstrate that the below-grade environment is aggressive or non-aggressive.	Yes, if environment is aggressive	The aging effect is for concrete – not steel. Incorrect aging mechanism listed under the AMP for the aging mechanism specified in the previous column. This comment also applies to II.A2-7, II.B2.2-6, II.B3.1-6 and II.B3.2-7

II CONTAINMENT STRUCTURES
A1 Concrete Containments (Reinforced and Prestressed)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
II.A1-11 (C-09)	II.A1.2-a	Steel elements: Liner; liner anchors; integral attachments	Steel	Air – indoor uncontrolled ~~or air – outdoor~~	Loss of material/ general~~, pitting, and crevice~~ corrosion	Chapter XI.S1, "ASME Section XI, Subsection IWE" For inaccessible areas (embedded containment steel shell or liner), loss of material due to corrosion is not significant if the following conditions are satisfied: *1.* Concrete meeting the requirements of ACI 318 or 349 and the guidance of 201.2R was used for the containment concrete in contact with the embedded containment shell or liner. *2.* The concrete is monitored to ensure that it is free of penetrating cracks that provide a path for water seepage to the surface of the containment shell or liner. *3.* The moisture barrier, at the junction where the shell or liner becomes embedded, is subject to aging management activities in accordance with IWE requirements. *4.* Borated water spills and water ponding on the containment concrete floor are not common and when detected are cleaned up in a timely manner. If any of the above conditions cannot be satisfied, then a plant-specific aging management program for	Yes, if corrosion is significant for inaccessible areas	Liners are only exposed to an air – indoor uncontrolled environment. Loss of material due to pitting and crevice corrosion is not applicable for air – indoor uncontrolled environment. Air – indoor uncontrolled environment is defined as a normally dry environment and crevice and pitting occur in a wetted environment per GALL Chapter IX. Itemized as given originally. Reads better this way.

II CONTAINMENT STRUCTURES
A1 Concrete Containments (Reinforced and Prestressed)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						corrosion is necessary. Chapter XI.S4, "10 CFR Part 50, Appendix J" and If a coatings program is credited for managing loss of material due to corrosion during the current licensing term (e.g., relief request from IWE), then it is to be continued during the period of extended operation. See Chapter XI.S8, "Protective Coating Monitoring and Maintenance Program."	No No	

II.A2 Steel Containments

II CONTAINMENT STRUCTURES A2 Steel Containments								
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
II.A2-1 (C-03)	II.A2.2-c	Concrete Dome; wall; basemat; ~~ring girder; buttresses~~	Concrete	Aggressive environment	Increase in porosity and permeability, cracking, loss of material (spalling, scaling)/ aggressive chemical attack	Chapter XI.S2, "ASME Section XI, Subsection IWL." Accessible Areas: Inspections performed in accordance with IWL will indicate the presence of increase in porosity and permeability, cracking, or loss of material (spalling, scaling) due to aggressive chemical attack. Inaccessible Areas: Examination of representative samples of below-grade concrete, when excavated for any reason, is to be performed, if the below-grade environment is aggressive (pH < 5.5, chlorides > 500ppm, or sulfates > 1,500 ppm). Periodic monitoring of below-grade water chemistry (including consideration of potential seasonal variations) is an acceptable approach to demonstrate that the below-grade environment is aggressive or non-aggressive.	Yes, if environment is aggressive	This item deals with free-standing steel containment PWRs. Ring girder and buttresses are not appropriate for this type of containment design. This is apparent since C-10 and C-11 for prestressing system do not apply to Section II.A2. This comment also applies to II.A2-2, II.A2-3, II.A2-4, II.A2-6, II.A2-7.

II CONTAINMENT STRUCTURES
A2 Steel Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
II.A2-2 (C-01)	II.A2.2-a	Concrete Dome; wall; basemat; ~~ring girder; buttresses~~	Concrete	Air – outdoor	Loss of material (spalling, scaling) and cracking/ freeze-thaw	Chapter XI.S2, "ASME Section XI, Subsection IWL" Accessible areas: Inspections performed in accordance with IWL will indicate the presence of loss of material (spalling, scaling) and cracking due to freeze-thaw. Inaccessible Areas: Evaluation is needed for plants that are located in moderate to severe weathering conditions (weathering index >100 day-inch/yr) (NUREG-1557). Documented evidence confirms that where the existing concrete had air content of 3% to 6%, subsequent inspection did not exhibit degradation related to freeze-thaw. Such inspections should be considered a part of the evaluation. The weathering index for the continental US is shown in ASTM C33-90, Fig. 1.	No, if stated conditions are satisfied for inaccessible areas	See comment in II.A2-1
II.A2-3 (C-02)	II.A2.2-b	Concrete Dome; wall; basemat~~; ring girder; buttresses~~	Concrete	Water – flowing	Increase in porosity, permeability/ leaching of calcium hydroxide	Chapter XI.S2, "ASME Section XI, Subsection IWL" Accessible areas: Inspections performed in accordance with IWL will indicate the presence of increase in porosity, and permeability due to leaching of calcium hydroxide. Inaccessible Areas: . An aging management program is not necessary, even if reinforced concrete is exposed to flowing water,	No, if concrete was constructed as stated for inaccessible areas	See comment in II.A2-1

II CONTAINMENT STRUCTURES
A2 Steel Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						if there is documented evidence that confirms the in-place concrete was constructed in accordance with the recommendations in ACI 201.2R-77.		
II.A2-4 (C-08)	II.A2.2-h	Concrete Dome; wall; ~~concrete fill-in annulus;~~ basemat; ~~ring girder; Buttresses~~	Concrete	Air – indoor uncontrolled or air – outdoor	Reduction of strength and modulus/ elevated temperature (>150°F general; >200°F local)	Plant-specific aging management program The implementation of 10 CFR 50.55a and IWL would not be able to identify the reduction of strength and modulus due to elevated temperature. Thus, for any portions of concrete containment that exceed specified temperature limits, further evaluations are warranted. Subsection CC-3400 of ASME Section III, Division 2, specifies the concrete temperature limits for normal operation or any other long-term period. The temperatures shall not exceed 150°F except for local areas, such as around penetrations, which are not allowed to exceed 200°F. If significant equipment loads are supported by concrete at temperatures exceeding 150°F, an evaluation of the ability to withstand the postulated design loads is to be made. Higher temperatures than given above may be allowed in the concrete if tests and/or calculations are provided to evaluate the reduction in strength and this reduction is applied to the design allowables.	Yes, if applicable	NRC Basis Document does not define concrete fill-in annulus. Industry does not recognize this term. This item deals with free-standing steel containment PWRs. Ring girder and buttresses are not appropriate for this type of containment design. This is apparent since C-10 and C-11 for prestressing system do not apply to Section II.A2.

II CONTAINMENT STRUCTURES
A2 Steel Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
II.A2-6 (C-04)	II.A2.2-d	Concrete: Dome; wall; basemat; ring girders; buttresses	Concrete	Any	Expansion and cracking/ reaction with aggregates	Chapter XI.S2, "ASME Section XI, Subsection IWL." Accessible Areas: Inspections performed in accordance with IWL will indicate the presence of cracking due to reaction with aggregates. Inaccessible Areas: ***As described in NUREG-1557, investigations, tests, and petrographic examinations of aggregates performed in accordance with ASTM C295-54 or ASTM C227-50 can demonstrate that those aggregates do not react within reinforced concrete. For potentially reactive aggregates, aggregate-reinforced concrete reaction is not significant if the concrete was constructed in accordance with ACI 201.2R-77. Therefore, if these conditions are satisfied, aging management is not necessary.*** ~~Evaluation is needed if testing and petrographic examinations of aggregates performed in accordance with ASTM C295-54, ASTM C227-50, or ACI 201.2R-77 (NUREG-1557) demonstrate that the aggregates are reactive.~~	No, if stated conditions are satisfied for inaccessible areas	To be consistent with previous GALL version.

II CONTAINMENT STRUCTURES
A2 Steel Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
II.A2-7 (C-05)	II.A2.2-e	Concrete: Dome; wall; basemat; ~~ring girders;~~ ~~buttresses~~; ~~reinforcing steel~~	Concrete; ~~steel~~	Air – indoor uncontrolled or air – outdoor	Cracking, loss of bond, and loss of material (spalling, scaling)/ corrosion of embedded steel	Chapter XI.S2, "ASME Section XI, Subsection IWL." Accessible Areas: Inspections performed in accordance with IWL will indicate the presence of ~~increase in~~ ***cracking, loss of bond, and*** ~~porosity and permeability, cracking, or~~ loss of material (spalling, scaling) due to ***corrosion of embedded steel***~~aggressive chemical attack~~. Inaccessible Areas: Examination of representative samples of below-grade concrete, when excavated for any reason, is to be performed, if the below-grade environment is aggressive (pH < 5.5, chlorides > 500ppm, or sulfates > 1,500 ppm). Periodic monitoring of below-grade water chemistry (including consideration of potential seasonal variations) is an acceptable approach to demonstrate that the below-grade environment is aggressive or non-aggressive.	Yes, if environment is aggressive	This item deals with free-standing steel containment PWRs. Ring girder and buttresses are not appropriate for this type of containment design. This is apparent since C-10 and C-11 for prestressing system do not apply to Section II.A2. The aging effect is for concrete – not steel. Incorrect aging mechanism listed under the AMP for the aging mechanism specified in the previous column.

II CONTAINMENT STRUCTURES
A2 Steel Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
II.A2-9 (C-09)	II.A2.1-a	Steel elements: Liner; liner anchors; integral attachments	Steel	Air – indoor uncontrolled ~~or air – outdoor~~	Loss of material/ general~~, pitting, and crevice~~ corrosion	Chapter XI.S1, "ASME Section XI, Subsection IWE" For inaccessible areas (embedded containment steel shell or liner), loss of material due to corrosion is not significant if the following conditions are satisfied: 1. Concrete meeting the requirements of ACI 318 or 349 and the guidance of 201.2R was used for the containment concrete in contact with the embedded containment shell or liner. 2. The concrete is monitored to ensure that it is free of penetrating cracks that provide a path for water seepage to the surface of the containment shell or liner. 3. The moisture barrier, at the junction where the shell or liner becomes embedded, is subject to aging management activities in accordance with IWE requirements. 4. Borated water spills and water ponding on the containment concrete floor are not common and when detected are cleaned up in a timely manner.	Yes, if corrosion is significant for inaccessible areas	Liners are only exposed to an air – indoor uncontrolled environment. Loss of material due to pitting and crevice corrosion is not applicable for air – indoor uncontrolled environment. Air – indoor uncontrolled environment is defined as a normally dry environment and crevice and pitting occur in a wetted environment per GALL Chapter IX. Itemized as given originally. Reads better this way.

II CONTAINMENT STRUCTURES
A2 Steel Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						If any of the above conditions cannot be satisfied, then a plant-specific aging management program for corrosion is necessary. Chapter XI.S4, "10 CFR Part 50, Appendix J" and If a coatings program is credited for managing loss of material due to corrosion during the current licensing term (e.g., relief request from IWE), then it is to be continued during the period of extended operation. See Chapter XI.S8, "Protective Coating Monitoring and Maintenance Program."	No No	

II.A3 Common Components

II CONTAINMENT STRUCTURES A3 Common Components								
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
II.A3-2 (C-15)	II.A3.1-d	Penetration sleeves; penetration bellows	Stainless steel; dissimilar metal welds	Air – indoor uncontrolled ***or air outdoor***	Cracking/ stress corrosion cracking	Chapter XI.S1, "ASME Section XI, Subsection IWE" and Chapter XI.S4, "10 CFR Part 50, Appendix J" Evaluation of 10 CFR 50.55a/IWE is augmented as follows: (4) Detection of Aging Effects: Transgranular Stress corrosion cracking (TGSCC) is a concern for dissimilar metal welds. In the case of bellows assemblies, SCC may cause aging effects particularly if the material is not shielded from a corrosive environment. Subsection IWE covers inspection of these items under examination categories E-B, E-F, and E-P (10 CFR Part 50, Appendix J pressure tests). 10 CFR 50.55a identifies examination categories E-B and E-F as optional during the current term of operation. For the extended period of operation, Examination Categories E-B & E-F, and additional appropriate examinations to detect SCC in bellows assemblies and dissimilar metal welds are warranted to address this issue. (10) Operating Experience:	Yes, detection of aging effects is to be evaluated	This item (component) is same as II.A3-1 which lists "Air – indoor uncontrolled or air outdoor" environment. This comment also applies to II.B4-2.

II CONTAINMENT STRUCTURES
A3 Common Components

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						IN 92-20 describes an instance of containment bellows cracking, resulting in loss of leak tightness.		
II.A3-3 (C-14)	II.A3.1-c	Penetration sleeves; *penetration* bellows	Steel; Stainless steel; Dissimilar metal welds	Air – indoor uncontrolled ***or air outdoor***	*Cracking/ cyclic* loading (CLB fatigue analysis does not exist)	Chapter XI.S1, "ASME Section XI, Subsection IWE " and Chapter XI.S4, "10 CFR Part 50, Appendix J" Evaluation of 10 CFR 50.55a/IWE is augmented as follows: (4) Detection of Aging Effects: VT-3 visual inspection may not detect fine cracks.	*Yes, detection of aging effects is to be evaluated*	This item *(component)* is same as II.A3-1 which lists "*Air – indoor* uncontrolled or air outdoor" environment. This comment also applies to II.B4-3.
II.A3-4 (C-13)	II.A3.1-b	Penetration sleeves; *penetration* bellows	Steel; Stainless steel; Dissimilar metal welds	Air – indoor uncontrolled ***or air outdoor***	Cumulative fatigue damage/ fatigue (Only if CLB fatigue analysis exists)	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.6, "Containment Liner Plate and Penetration Fatigue Analysis" for *acceptable methods for meeting the requirements of* 10 CFR 54.21(c)(1).	Yes, TLAA	This item (component) is same as II.A3-1 which lists "Air – indoor uncontrolled or air outdoor" environment. This comment also applies to II.B4-4.

II.B1 Mark I Containments
II.B1.1 Steel Containments

II B1.1	CONTAINMENT STRUCTURES Steel Containments							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
II.B1.1-1 (C-23)	II.B1.1.1-e	Steel elements: Drywell head; downcomers	Steel; ~~Graphite plate~~	Air – indoor uncontrolled	Fretting or lockup/ mechanical wear	Chapter XI.S1, "ASME Section XI, Subsection IWE"	No	Industry experience and EPRI Civil Tools indicates this aging effect is not applicable to graphite plate (lubrite). This is consistent with past approved applications. See NUREG-1759, Turkey Point SER, NUREG-1769, Peach Bottom SER, NUREG-1785, H.B. Robinson SER NUREG-1766, North Anna/Surry SER This comment applies to IIB2.1-3 (C-23) & II.B2.2-10 (C-23)
II.B1.1-2 (C-19)	II.B1.1.1-a	Steel elements: Drywell; torus; drywell head; embedded shell and sand pocket regions; drywell support	Steel	Air – indoor uncontrolled ***or treated water (as applicable)***	Loss of material/ general, pitting, and crevice corrosion	Chapter XI.S1, "ASME Section XI, Subsection IWE" For inaccessible areas (embedded containment steel shell or liner), loss of material due to corrosion is not significant if the following conditions are satisfied: Concrete meeting the specifications of ACI 318 or 349 and the guidance of 201.2R was used for the containment concrete in contact with the embedded containment shell or liner.	Yes, if corrosion is significant for inaccessible areas	Add "or treated water" to environment column to account for the wetted portion of the torus and other components inside the torus.

II CONTAINMENT STRUCTURES
B1.1 Steel Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
		skirt; torus ring girder; downcomers; ECCS suction header NOTE: Inspection of containment supports is addressed by ASME Section XI, Subsection IWF (see III.B1.3)				The concrete is monitored to ensure that it is free of penetrating cracks that provide a path for water seepage to the surface of the containment shell or liner. The moisture barrier, at the junction where the shell or liner becomes embedded, is subject to aging management activities in accordance with IWE requirements. Borated water spills and water ponding on the containment concrete floor are not common and when detected are cleaned up in a timely manner. If any of the above conditions cannot be satisfied, then a plant-specific aging management program for corrosion is necessary. Chapter XI.S4, "10 CFR Part 50, Appendix J" and If a coatings program is credited for managing loss of material due to corrosion during the current licensing term (e.g., relief request from IWE), then it is to be continued during the period of extended operation. See Chapter XI.S8, "Protective Coating Monitoring and Maintenance Program."	 No No	

II.B2 Mark II Containments
II.B2.1 Steel Containments

II	CONTAINMENT STRUCTURES
B2.1	Steel Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
II.B2.1-1 (C-14)	II.B2.1.1-b	~~Penetration sleeves;~~ ~~penetration bellows~~ ***Suppression pool shell, unbraced downcomers***	Steel; Stainless steel; Dissimilar metal welds	Air – indoor uncontrolled	Cracking/ cyclic loading (CLB fatigue analysis does not exist)	Chapter XI.S1, "ASME Section XI, Subsection IWE " and Chapter XI.S4, "10 CFR Part 50, Appendix J" Evaluation of 10 CFR 50.55a/IWE is augmented as follows: (4) Detection of Aging Effects: VT-3 visual inspection may not detect fine cracks.	Yes, detection of aging effects is to be evaluated	This item as it is presented is a duplicate of II.B4-3. Therefore, change component to match original GALL..
II.B2.1-2 (C-13)	II.B2.1.1-c	~~Penetration sleeves;~~ ~~penetration bellows~~ ***Suppression pool shell, unbraced downcomers***	Steel; Stainless steel; Dissimilar metal welds	Air – indoor uncontrolled	Cumulative fatigue damage/ fatigue (Only if CLB fatigue analysis exists)	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.6, "Containment Liner Plate and Penetration Fatigue Analysis" for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA	This item as it is presented is a duplicate of II.B4-4. Therefore, change component to match original GALL..
II.B2.1-3 (C-23)	II.B2.1.1-d	Steel elements: Drywell head; downcomers	Steel; ~~Graphite plate~~	Air – indoor uncontrolled	Fretting or lockup/ mechanical wear	Chapter XI.S1, "ASME Section XI, Subsection IWE"	No	Industry experience and EPRI Civil Tools indicates this aging effect is not applicable to graphite plate (lubrite). This is consistent with past approved applications. See NUREG-1759, Turkey Point SER, NUREG-1769, Peach Bottom SER, NUREG-1785, H.B.

II CONTAINMENT STRUCTURES
B2.1 Steel Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
								Robinson SER NUREG-1766, North Anna/Surry SER
II.B2.1-4 (C-19)	II.B2.1.1-a	Steel elements: ~~Drywell; torus; drywell head; embedded shell and sand pocket regions; drywell support skirt; torus ring girder; downcomers; ECCS suction header~~ ***Drywell; suppression chamber; drywell head; embedded shell and sand pocket regions; support skirt; downcomer pipes; region shielded by diaphragm floor*** NOTE: Inspection of	Steel	Air – indoor uncontrolled ***or treated water***	Loss of material/ general, pitting, and crevice corrosion	Chapter XI.S1, "ASME Section XI, Subsection IWE" For inaccessible areas (embedded containment steel shell or liner), loss of material due to corrosion is not significant if the following conditions are satisfied: Concrete meeting the specifications of ACI 318 or 349 and the guidance of 201.2R was used for the containment concrete in contact with the embedded containment shell or liner. The concrete is monitored to ensure that it is free of penetrating cracks that provide a path for water seepage to the surface of the containment shell or liner. The moisture barrier, at the junction where the shell or liner becomes embedded, is subject to aging management activities in accordance with IWE requirements. Borated water spills and water ponding on the containment concrete floor are not common and when detected are cleaned up in a timely manner. If any of the above conditions cannot be satisfied, then a plant-specific aging management program for corrosion is necessary.	Yes, if corrosion is significant for inaccessible areas	GALL II.B2 is for a Mark II BWR which *does not* have a torus or torus ring girder. Change component to match original GALL. Add "or treated water" to environment column to account for the wetted portion of the suppression chamber. .

II CONTAINMENT STRUCTURES
B2.1 Steel Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
		containment supports is addressed by ASME Section XI, Subsection IWF (see III.B1.3)				Chapter XI.S4, "10 CFR Part 50, Appendix J" and If a coatings program is credited for managing loss of material due to corrosion during the current licensing term (e.g., relief request from IWE), then it is to be continued during the period of extended operation. See Chapter XI.S8, "Protective Coating Monitoring and Maintenance Program."	No No	

II.B2 Mark II Containments

II.B2.2 Concrete Containments

II *B2.2*	CONTAINMENT STRUCTURES ***Concrete Containments***~~B2.1 Steel Containments~~							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
II.B2.2-1 (C-03)	II.B2.2.1-b	Concrete ~~Dome; wall; basemat; ring girder; buttresses~~ ***Containment, wall, basemat***	Concrete	Aggressive environment	Increase in porosity and permeability, cracking, loss of material (spalling, scaling)/ aggressive chemical attack	Chapter XI.S2, "ASME Section XI, Subsection IWL". Accessible Areas: Inspections performed in accordance with IWL will indicate the presence of increase in porosity and permeability, cracking, or loss of material (spalling, scaling) due to aggressive chemical attack. Inaccessible Areas: Examination of representative samples of below-grade concrete, when excavated for any reason, is to be performed, if the below-grade environment is aggressive (pH < 5.5, chlorides > 500ppm, or sulfates > 1,500 ppm). Periodic monitoring of below-grade water chemistry (including consideration of potential seasonal variations) is an acceptable approach to demonstrate that the below-grade environment is aggressive or non-aggressive.	Yes, if environment is aggressive	GALL II.B2.2 is for a Mark II BWR concrete containment, the component listed is for a PWR concrete containment. Change component to match original GALL. This comment also applies to II.B2.2-2, II.B2.2-3, II.B2.2-5, II.B2.2-6
II.B2.2-2 (C-02)	II.B2.2.1-a	Concrete ~~Dome; wall; basemat; ring~~	Concrete	Water – flowing	Increase in porosity, permeability/ leaching of calcium	Chapter XI.S2, "ASME Section XI, Subsection IWL" Accessible areas: Inspections performed in accordance	No, if concrete was constructed as stated for inaccessible areas	GALL II.B2.2 is for a Mark II BWR concrete containment, the component listed is for a PWR concrete

II CONTAINMENT STRUCTURES *B2.2 Concrete Containments*~~B2.1 Steel Containments~~								
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
		~~girder; buttresses~~ ***Containment, wall, basemat***			hydroxide	with IWL will indicate the presence of increase in porosity, and permeability due to leaching of calcium hydroxide. Inaccessible Areas: . An aging management program is not necessary, even if reinforced concrete is exposed to flowing water, if there is documented evidence that confirms the in-place concrete was constructed in accordance with the recommendations in ACI 201.2R-77.		containment. Change component to match original GALL.
II.B2.2-3 (C-08)	II.B2.2.1-g	Concrete ~~Dome; wall; concrete fill-in annulus, basemat; ring girder; buttresses~~ ***Containment, wall, basemat***	Concrete	Air – indoor uncontrolled or air - outdoor	Reduction of strength and modulus/ elevated temperature (>150°F general; >200°F local)	Plant-specific aging management program The implementation of 10 CFR 50.55a and IWL would not be able to identify the reduction of strength and modulus due to elevated temperature. Thus, for any portions of concrete containment that exceed specified temperature limits, further evaluations are warranted. Subsection CC-3400 of ASME Section III, Division 2, specifies the concrete temperature limits for normal operation or any other long-term period. The temperatures shall not exceed 150°F except for local areas, such as around penetrations, which are not allowed to exceed 200°F. If significant equipment loads are supported by concrete at temperatures exceeding 150°F, an evaluation of the ability to withstand the postulated design loads is to be made.	Yes, if applicable	GALL II.B2.2 is for a Mark II BWR concrete containment, the component listed is for a PWR concrete containment. Change component to match original GALL.

II CONTAINMENT STRUCTURES
B2.2 Concrete Containments~~B2.1 Steel Containments~~

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						Higher temperatures than given above may be allowed in the concrete if tests and/or calculations are provided to evaluate the reduction in strength and this reduction is applied to the design allowables.		
II.B2.2-5 (C-04)	II.B2.2.1-c	Concrete: ~~Dome; wall; basemat; ring girders; buttresses~~ ***Containment, wall, basemat***	Concrete	Any	~~Expansion and c~~ Cracking ***due to expansion/*** reaction with aggregates	Chapter XI.S2, "ASME Section XI, Subsection IWL." Accessible Areas: Inspections performed in accordance with IWL will indicate the presence of cracking due to reaction with aggregates. Inaccessible Areas: ***As described in NUREG-1557, investigations, tests, and petrographic examinations of aggregates performed in accordance with ASTM C295-54 or ASTM C227-50 can demonstrate that those aggregates do not react within reinforced concrete. For potentially reactive aggregates, aggregate-reinforced concrete reaction is not significant if the concrete was constructed in accordance with ACI 201.2R-77. Therefore, if these conditions are satisfied, aging management is not necessary.*** ~~Evaluation is needed if testing and petrographic examinations of~~	No, if stated conditions are satisfied for inaccessible areas	GALL II.B2.2 is for a Mark II BWR concrete containment, the component listed is for a PWR concrete containment. Change component to match original GALL. AMP should be consistent with previous GALL version.

II CONTAINMENT STRUCTURES
B2.2 Concrete Containments~~B2.1 Steel Containments~~

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						~~aggregates performed in accordance with or *similar to* ASTM C295-54, ASTM C227-50, or ACI 201.2R-77 (NUREG-1557) demonstrate that the aggregates are reactive.~~		
II.B2.2-6	II.B2.2.1-d	Concrete: ~~Dome; wall; basemat; ring girders; buttresses; reinforcing steel~~ ***Containment, wall, basemat***	Concrete; ~~steel~~	Air – indoor uncontrolled or air – outdoor	Cracking, loss of bond, and loss of material (spalling, scaling)/ corrosion of embedded steel	Chapter XI.S2, "ASME Section XI, Subsection IWL". Accessible Areas: Inspections performed in accordance with IWL will indicate the presence of ~~increase in~~ ***cracking, loss of bond, and*** ~~porosity and permeability, cracking, or~~ loss of material (spalling, scaling) due to ***corrosion of embedded steel***~~aggressive chemical attack~~ Inaccessible Areas: Examination of representative samples of below-grade concrete, when excavated for any reason, is to be performed, if the below-grade environment is aggressive (pH < 5.5, chlorides > 500ppm, or sulfates > 1,500 ppm). Periodic monitoring of below-grade water chemistry (including consideration of potential seasonal variations) is an acceptable approach to demonstrate that the below-grade environment is aggressive or non-aggressive.	Yes, if environment is aggressive	GALL II.B2.2 is for a Mark II BWR concrete containment, the component listed is for a PWR concrete containment. Change component to match original GALL Incorrect aging mechanism listed under the AMP for the aging mechanism specified in the previous column.
II.B2.2-10 (C-23)	II.B2.2.2-e	Steel elements: Drywell head;	Steel; ~~Graphite plate~~	Air – indoor uncontrolled	Fretting or lockup/ mechanical	Chapter XI.S1, "ASME Section XI, Subsection IWE"	No	Industry experience and EPRI Civil Tools indicates this aging

II CONTAINMENT STRUCTURES
B2.2 Concrete Containments~~B2.1 Steel Containments~~

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
		downcomers			wear			effect is not applicable to graphite plate (lubrite). This is consistent with past approved applications. See NUREG-1759, Turkey Point SER, NUREG-1769, Peach Bottom SER, NUREG-1785, H.B. Robinson SER NUREG-1766, North Anna/Surry SER
II.B2.2-11 (C-19)	II.B2.2.2-a	Steel elements: ~~Drywell; torus; drywell head; embedded shell and sand pocket regions; drywell support skirt; torus ring girder; downcomers; ECCS suction header~~ ***Drywell; suppression chamber; drywell head; embedded shell and sand pocket***	Steel	Air – indoor uncontrolled	Loss of material/ general, pitting, and crevice corrosion	Chapter XI.S1, "ASME Section XI, Subsection IWE" For inaccessible areas (embedded containment steel shell or liner), loss of material due to corrosion is not significant if the following conditions are satisfied: Concrete meeting the specifications of ACI 318 or 349 and the guidance of 201.2R was used for the containment concrete in contact with the embedded containment shell or liner. The concrete is monitored to ensure that it is free of penetrating cracks that provide a path for water seepage to the surface of the containment shell or liner. The moisture barrier, at the junction where the shell or liner becomes embedded, is subject to aging management activities in accordance with IWE requirements. Borated water spills and water	Yes, if corrosion is significant for inaccessible areas	GALL II.B2 is for a Mark II BWR which *does not* have a torus or torus ring girder. Change component to match original GALL. Add "or treated water" to environment column to account for the wetted portion of the suppression chamber.

II CONTAINMENT STRUCTURES

B2.2 Concrete Containments~~B2.1 Steel Containments~~

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
		regions; support skirt; downcomer pipes; region shielded by diaphragm floor NOTE: Inspection of containment supports is addressed by ASME Section XI, Subsection IWF (see III.B1.3)				ponding on the containment concrete floor are not common and when detected are cleaned up in a timely manner. If any of the above conditions cannot be satisfied, then a plant-specific aging management program for corrosion is necessary. Chapter XI.S4, "10 CFR Part 50, Appendix J" and If a coatings program is credited for managing loss of material due to corrosion during the current licensing term (e.g., relief request from IWE), then it is to be continued during the period of extended operation. See Chapter XI.S8, "Protective Coating Monitoring and Maintenance Program."	No No	
II.B2.2-12 (C-20)	II.B2.2-c	Steel elements: ~~Torus; vent line; vent header; vent line bellows; downcomers~~ ***Vent header, downcomers***	Stainless steel; steel	Air – indoor uncontrolled	Cracking/ cyclic loading (CLB fatigue analysis does not exist)	Chapter XI.S1, "ASME Section XI, Subsection IWE " and Chapter XI.S4, "10 CFR Part 50, Appendix J" Evaluation of 10 CFR 50.55a/IWE is augmented as follows: (4) Detection of Aging Effects: VT-3 visual inspection may not detect fine cracks.	Yes, detection of aging effects is to be evaluated	GALL II.B2.2 is for a Mark II BWR concrete containment, the component listed is for a BWR Mark I. Change component to match original GALL..
II.B2.2-13 (C-21)	II.B2.2-d	Steel elements: ~~Torus; vent line; vent header; vent~~	Stainless steel; steel	Air – indoor uncontrolled	Cumulative fatigue damage/ fatigue (Only if CLB	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.6, "Containment Liner Plate and Penetration Fatigue Analysis" for	Yes, TLAA	GALL II.B2.2 is for a Mark II BWR concrete containment, the component listed is for a BWR Mark I. Change component to

II CONTAINMENT STRUCTURES
B2.2 Concrete Containments~~B2.1 Steel Containments~~

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
		~~line bellows; downcomers~~ ***Vent header, downcomers***			fatigue analysis exists)	acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).		match original GALL..
II.B2.2-14 (C-22)	II.B2.2.2-b	Steel elements: ~~Vent line bellows~~ ***Suppression chamber liner (interior surface)***	***Steel or*** Stainless steel	Air – indoor uncontrolled ***or treated water***	***Loss of material/ general, pitting, and crevice corrosion*** ~~Cracking/ stress corrosion cracking~~	Chapter XI.S1, "ASME Section XI, Subsection IWE " and Chapter XI.S4, "10 CFR Part 50, Appendix J" ~~Evaluation of 10 CFR 50.55a/IWE is augmented as follows:~~ ~~(4) Detection of Aging Effects: Stress corrosion cracking (SCC) is a concern for dissimilar metal welds. In the case of bellows assemblies, SCC may cause aging effects particularly if the material is not shielded from a corrosive environment. Subsection IWE covers inspection of these items under examination categories E-B, E-F, and E-P (10 CFR Part 50, Appendix J pressure tests). 10 CFR 50.55a identifies examination categories E-B and E-F as optional during the current term of operation. For the extended period of operation, Examination Categories E-B and E-F, and additional appropriate examinations to detect SCC in bellows assemblies and dissimilar metal welds are warranted to address this issue.~~ ~~(10) Operating Experience: IN 92-20~~	Yes, detection of aging effects is to be evaluated	GALL II.B2.2 is for a Mark II BWR that does not have vent line bellows. Change component to match original GALL. Add steel to account for suppression chamber with carbon steel liner. Add treated water to account for the wetted portion of the suppression pool. Add loss of material aging effect. Delete cracking due to SCC because suppression chamber temperature is <140F. Delete text on SCC in the AMP column that dealt with bellows because Mark II BWR does not have this component.

II CONTAINMENT STRUCTURES

B2.2 Concrete Containments~~B2.1 Steel Containments~~

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						~~describes an instance of containment bellows cracking, resulting in loss of leak tightness.~~		

II.B3 Mark III Containments
II.B3.1 Steel Containments

II B3.1	CONTAINMENT STRUCTURES Steel Containments							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
II.B3.1-5 (C-04)	II.B3.1.2-c	Concrete: Dome; wall; basemat; ring girders; buttresses	Concrete	Any	~~Expansion and c~~ Cracking ***due to expansion/*** reaction with aggregates	Chapter XI.S2, "ASME Section XI, Subsection IWL." Accessible Areas: Inspections performed in accordance with IWL will indicate the presence of cracking due to reaction with aggregates. Inaccessible Areas: ***As described in NUREG-1557, investigations, tests, and petrographic examinations of aggregates performed in accordance with ASTM C295-54 or ASTM C227-50 can demonstrate that those aggregates do not react within reinforced concrete. For potentially reactive aggregates, aggregate-reinforced concrete reaction is not significant if the concrete was constructed in accordance with ACI 201.2R-77. Therefore, if these conditions are satisfied, aging management is not necessary.*** ~~Evaluation is needed if testing and petrographic examinations of aggregates performed in accordance with or *similar to* ASTM C295-54, ASTM C227-50, or ACI 201.2R-77 (NUREG-1557) demonstrate that the aggregates are reactive.~~	No, if stated conditions are satisfied for inaccessible areas	Consistent with previous GALL version.
II.B3.1-6	II.B3.1.2-d	Concrete:	Concrete; ~~steel~~	Air – indoor uncontrolled	Cracking, loss of bond, and loss	Chapter XI.S2, "ASME Section XI, Subsection IWL."	Yes, if environment is	GALL II.B3.1 is for a Mark III BWR steel

II CONTAINMENT STRUCTURES B3.1 Steel Containments								
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
(C-05)		~~Dome; wall; basemat; ring girders; buttresses; reinforcing steel~~ ***Containment, wall, basemat***		or air – outdoor	of material (spalling, scaling)/ corrosion of embedded steel	Accessible Areas: Inspections performed in accordance with IWL will indicate the presence of ~~increase in porosity and permeability,~~ ***cracking, loss of bond, and*** loss of material (spalling, scaling) due to ~~aggressive chemical attack~~ ***corrosion of embedded steel.*** Inaccessible Areas: Examination of representative samples of below-grade concrete, when excavated for any reason, is to be performed, if the below-grade environment is aggressive (pH < 5.5, chlorides > 500ppm, or sulfates > 1,500 ppm). Periodic monitoring of below-grade water chemistry (including consideration of potential seasonal variations) is an acceptable approach to demonstrate that the below-grade environment is aggressive or non-aggressive.	aggressive	containment, the component listed is for a PWR concrete containment. Change component to match original GALL The *aging effect* is for concrete – not steel. Incorrect aging mechanism listed under the AMP for the aging mechanism specified in the previous column.
II.B3.1-8 (C-19)	II.B3.1.1-a	Steel elements: ~~Drywell; torus; drywell head; embedded shell and sand pocket regions; drywell support skirt; torus ring~~	Steel	Air – indoor uncontrolled ***or treated water***	Loss of material/ general, pitting, and crevice corrosion	Chapter XI.S1, "ASME Section XI, Subsection IWE" For inaccessible areas (embedded containment steel shell or liner), loss of material due to corrosion is not significant if the following conditions are satisfied: Concrete meeting the specifications of ACI 318 or 349 and the guidance of 201.2R was used for the containment	Yes, if corrosion is significant for inaccessible areas	GALL II.B3.1 is for a Mark III BWR concrete containment, the component listed is for a BWR Mark I. Change component to match original GALL. Add "or treated water" to environment column to account for the wetted portion of the suppression chamber.

II CONTAINMENT STRUCTURES
B3.1 Steel Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
		~~girder; downcomers; ECCS suction header~~ ***Containment shell; suppression chamber shell; basemat liner; liner anchors*** NOTE: Inspection of containment supports is addressed by ASME Section XI, Subsection IWF (see III.B1.3)				concrete in contact with the embedded containment shell or liner. The concrete is monitored to ensure that it is free of penetrating cracks that provide a path for water seepage to the surface of the containment shell or liner. The moisture barrier, at the junction where the shell or liner becomes embedded, is subject to aging management activities in accordance with IWE requirements. Borated water spills and water ponding on the containment concrete floor are not common and when detected are cleaned up in a timely manner. If any of the above conditions cannot be satisfied, then a plant-specific aging management program for corrosion is necessary. Chapter XI.S4, "10 CFR Part 50, Appendix J" and If a coatings program is credited for managing loss of material due to corrosion during the current licensing term (e.g., relief request from IWE), then it is to be continued during the period of extended operation. See Chapter XI.S8, "Protective Coating Monitoring and Maintenance Program."	No No	

II.B3 Mark III Containments
II.B3.2 Concrete Containments

II B3.2	CONTAINMENT STRUCTURES Concrete Containments							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
II.B3.2-6 (C-04)	II.B3.2.1-d	Concrete: Dome; wall; basemat; ring girders; buttresses	Concrete	Any	~~Expansion and c~~ Cracking ***due to expansion/*** reaction with aggregates	Chapter XI.S2, "ASME Section XI, Subsection IWL." Accessible Areas: Inspections performed in accordance with IWL will indicate the presence of cracking due to reaction with aggregates. Inaccessible Areas: ***As described in NUREG-1557, investigations, tests, and petrographic examinations of aggregates performed in accordance with ASTM C295-54 or ASTM C227-50 can demonstrate that those aggregates do not react within reinforced concrete. For potentially reactive aggregates, aggregate-reinforced concrete reaction is not significant if the concrete was constructed in accordance with ACI 201.2R-77. Therefore, if these conditions are satisfied, aging management is not necessary.*** ~~Evaluation is needed if testing and petrographic examinations of aggregates performed in accordance with or *similar to* ASTM C295-54, ASTM C227-50, or ACI 201.2R-77~~	No, if stated conditions are satisfied for inaccessible areas	Consistent with previous GALL version.

II CONTAINMENT STRUCTURES
B3.2 Concrete Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						~~(NUREG-1557) demonstrate that the aggregates are reactive.~~		
II.B3.2-7 (C-05)	II.B3.2.1-e	Concrete: ~~Dome; wall; basemat; ring girders; buttresses; reinforcing steel~~ ***Containment, wall, basemat***	Concrete; ~~steel~~	Air – indoor uncontrolled or air – outdoor	Cracking, loss of bond, and loss of material (spalling, scaling)/ corrosion of embedded steel	Chapter XI.S2, "ASME Section XI, Subsection IWL." Accessible Areas: Inspections performed in accordance with IWL will indicate the presence of ~~increase in porosity and permeability,~~ ***cracking, loss of bond, and*** loss of material (spalling, scaling) due to ~~aggressive chemical attack~~ ***corrosion of embedded steel.*** Inaccessible Areas: Examination of representative samples of below-grade concrete, when excavated for any reason, is to be performed, if the below-grade environment is aggressive (pH < 5.5, chlorides > 500ppm, or sulfates > 1,500 ppm). Periodic monitoring of below-grade water chemistry (including consideration of potential seasonal variations) is an acceptable approach to demonstrate that the below-grade environment is aggressive or non-aggressive.	Yes, if environment is aggressive	GALL II.B3.2 is for a Mark III BWR concrete containment, the component listed is for a PWR concrete containment. Change component to match original GALL The aging effect is for concrete – not steel. Incorrect aging mechanism listed under the AMP for the aging mechanism specified in the previous column.

II.B4 Common Components

II B4	CONTAINMENT STRUCTURES Common Components							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
II.B4-2 (C-15)	II.B4.1-d	Penetration sleeves; penetration bellows	Stainless steel; dissimilar metal welds	Air – indoor uncontrolled ***or air outdoor***	Cracking/ stress corrosion cracking	Chapter XI.S1, "ASME Section XI, Subsection IWE" and Chapter XI.S4, "10 CFR Part 50, Appendix J" Evaluation of 10 CFR 50.55a/IWE is augmented as follows: (4) Detection of Aging Effects: Transgranular Stress corrosion cracking (TGSCC) is a concern for dissimilar metal welds. In the case of bellows assemblies, SCC may cause aging effects particularly if the material is not shielded from a corrosive environment. Subsection IWE covers inspection of these items under examination categories E-B, E-F, and E-P (10 CFR Part 50, Appendix J pressure tests). 10 CFR 50.55a identifies examination categories E-B and E-F as optional during the current term of operation. For the extended period of operation, Examination Categories E-B & E-F, and additional appropriate examinations to detect SCC in bellows assemblies and dissimilar metal welds are warranted to address this issue. (10) Operating Experience:	Yes, detection of aging effects is to be evaluated	This item (component) is same as II.A3-1 which lists "Air – indoor uncontrolled or air outdoor" environment.

II CONTAINMENT STRUCTURES
B4 Common Components

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						IN 92-20 describes an instance of containment bellows cracking, resulting in loss of leak tightness.		
II.B4-3 (C-14)	II.B4.1-c	Penetration sleeves; penetration bellows	Steel; Stainless steel; Dissimilar metal welds	Air – indoor uncontrolled ***or air outdoor***	Cracking/ cyclic loading (CLB fatigue analysis does not exist)	Chapter XI.S1, "ASME Section XI, Subsection IWE " and Chapter XI.S4, "10 CFR Part 50, Appendix J" Evaluation of 10 CFR 50.55a/IWE is augmented as follows: (4) Detection of Aging Effects: VT-3 visual inspection may not detect fine cracks.	Yes, detection of aging effects is to be evaluated	This item (component) is same as II.A3-1 which lists "Air – indoor uncontrolled or air outdoor" environment.
II.B4-4 (C-13)	II.B4.1-b	Penetration sleeves; penetration bellows	Steel; Stainless steel; Dissimilar metal welds	Air – indoor uncontrolled ***or air outdoor***	Cumulative fatigue damage/ fatigue (Only if CLB fatigue analysis exists)	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.6, "Containment Liner Plate and Penetration Fatigue Analysis" for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA	This item (component) is same as II.A3-1 which lists "Air – indoor uncontrolled or air outdoor" environment.

CHAPTER III STRUCTURES AND COMPONENT SUPPORTS

General Comments

1. Consolidation of Chapter III Tables

The tables of Chapter III for the various structures and component supports are highly repetitive. As shown in the following table, many of the generic GALL lines (listed by their alpha-numeric designator in the left column) are the same for each structure table (listed by section identifier across the top).

GALL SECTION III A (CLASS I AND CLASS II STRUCTURES) COMMON ITEM MATRIX									
Section	A1	A2	A3	A4	A5	A6	A7	A8	A9
	BWR Rx, PWR Shld	BWR Rx w/stl Struc	Aux, DG, etc	Cont Int Struc	FS Facility	WC Struc	Conc Tank	Steel Tank	Vent Stack
T-01	X (A1.1-a)	X (A2.1-a)	X (A3.1-a)		X (A5.1-a)		X (A7.1-a)	X (A8.1-a)	X (A9.1-a)
T-02	X (A1.1-b)	X (A2.1-b)	X (A3.1-b)		X (A5.1-b)		X (A7.1-b)	X (A8.1-b)	X (A9.1-b)
T-03	X (A1.1-c)	X (A2.1-c)	X (A3.1-c)	X (A4.1-b)	X (A5.1-c)		X (A7.1-c)	X (A8.1-c)	X (A9.1-c)
T-04	X (A1.1-d)	X (A2.1-d)	X (A3.1-d)	X (A4.1-d)	X (A5.1-d)		X (A7.1-d)		X (A9.1-d)
T-05	X (A1.1-e)	X (A2.1-e)	X (A3.1-e)		X (A5.1-e)		X (A7.1-e)	X (A8.1-d)	X (A9.1-e)
T-06	X (A1.1-f)	X (A2.1-f)	X (A3.1-f)	X (A4.1-a)	X (A5.1-f)		X (A7.1-f)		X (A9.1-f)
T-07	X (A1.1-g)	X (A2.1-g)	X (A3.1-g)		X (A5.1-g)		X (A7.1-g)	X (A8.1-e)	X (A9.1-g)
T-08	X (A1.1-h)	X (A2.1-h)	X (A3.1-h)		X (A5.1-h)	X (A6.1-f)	X (A7.1-h)	X (A8.1-f)	X (A9.1-h)
T-09	X (A1.1-i)	X (A2.1-i)	X (A3.1-i)		X (A5.1-i)	X (A6.1-g)	X (A7.1-i)	X (A8.1-g)	X (A9.1-i)
T-10	X (A1.1-j)	X (A2.1-j)	X (A3.1-j)	X (A4.1-c)	X (A5.1-j)				
T-11	X (A1.2-a)	X (A2.2-a)	X (A3.2-a)	X (A4.2-a)	X (A5.2-a)		X (A7.2-a)	X (A8.2-a)	
T-12	X (A1.3-a)	X (A2.3-a)	X (A3.3-a)		X (A5.3-a)	X (A6.2-a)			
T-13				X (A4.2-b)					
T-14					X (A5.2-b)				
T-15						X (A6.1-a)			
T-16						X (A6.1-b)			
T-17						X (A6.1-c)			
T-18						X (A6.1-d)			
T-19						X (A6.1-e)			
T-20						X (A6.1-h)			
T-21						X (A6.2-a)			
T-22						X (A6.4-a)			
T-23							X (A7.2-b)	X (A8.2-b)	

Similarly, the following table demonstrates that many of the generic GALL lines are the same for each component support table.

GALL SECTION III B (COMPONENT SUPPORTS) COMMON ITEM MATRIX							
Section	B1 (ASME PIPING & COMP)			B2	B3	B4	B5
	B1.1	B1.2	B1.3	CT, Cond,	Anchorage	DG, Mech Equip	Platforms,
	Class 1	Class 2 & 3	Class MC	HVAC	Racks, Cabinet	HVAC Equip	PWR, Masonry
T-24	X	X	X				
T-25	X	X		X	X	X	X
T-26	X	X	X				
T-27	X						
T-28	X	X	X				
T-29	X	X	X	X	X	X	X
T-30				X	X	X	X
T-31						X	
TP-1				X		X	
TP-2				X		X	
TP-3	X	X	X	X	X	X	X
TP-4	X	X	X	X	X	X	X
TP-5	X	X	X	X	X	X	X
TP-6				X		X	
TP-7							
TP-8	X	X	X	X	X	X	X

To simplify these GALL tables, the nine structures tables have been consolidated into three tables. Tables A1, A2, A3, A4, A5 and A9 comprise a new Group 1 Structures Table; Table A6 becomes a new Group 2 Structures Table; and Tables A7 and A8 are joined to become a new Group 3 Structures Table. Similarly, the three ASME piping supports tables B1.1, B1.2 and B1.3 are combined as a table for supports for ASME piping and components and Class MC (BWR Containment Supports); and the remaining four tables have been combined as a table for all other supports.

Because the consolidation changes more than just the tables themselves, the entire chapter is presented below showing how it should appear; not all deletions are marked. The bases for consolidation is not discussed for each line. Comments in addition to the consolidation maintain the convention of bold italics for additions and strikethrough for deletions and a basis for the change is provided.

2. New Line Items

Proposed new lines items are listed at the end of each table. The new lines are designated "New III.X(Y) where X is the table identifier and Y is a sequential number for new lines in that table.

3. **Inappropriate AMP**

   The wrong AMP is designated in the AMP column of some lines. For example Item III.A1-7 for Class I (other than Containment) concrete is managed by SMP not IWL. Corrections are suggested.

4. **AMP Modifications for Consistency**

   'Accessible Areas and Inaccessible Areas' were introduced for Water Control Structures concrete AMP to make it consistent with rest of the GALL. For example see Item III.A2-3.

5. **Alternate AMPs**

   Added 'and/or Structures Monitoring Program'. Section XI mentions SMP as an alternative program; we are adding it in GALL so that we can credit it as 'consistent with GALL'. For example see Item III.A2-3.

CHAPTER III

STRUCTURES AND COMPONENT SUPPORTS

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STRUCTURES AND COMPONENT SUPPORTS

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CLASS 1 *AND CLASS 2* STRUCTURES

A1. Group 1 Structures (BWR Reactor Bldg., PWR Shield Bldg., Control Rm./Bldg., ***Auxiliary Bldg., Diesel Generator Bldg., Radwaste Bldg., Turbine Bldg., Switchgear Rm., Yard Structures such as AFW Pumphouse, Utility/Piping Tunnels, Security/Lighting Poles, Manholes, Duct Banks; SBO Structures such as Transmission Towers, Startup Towers Circuit Breaker Foundation, Electrical Enclosure, Containment Internal Structures, Fuel Storage Facility, Refueling Canal, BWR Unit Vent Stack***)

A2. Group *2* Structures (Water-Control Structures)

A3. Group *3* Structures (Tanks ***(concrete and steel)*** and Missile Barriers)

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A1. GROUP 1 STRUCTURES (BWR REACTOR BLDG., PWR SHIELD BLDG., CONTROL RM./BLDG. *AUXILIARY BLDG., DIESEL GENERATOR BLDG., RADWASTE BLDG., TURBINE BLDG., SWITCHGEAR RM., YARD STRUCTURES SUCH AS AFW PUMPHOUSE, UTILITY/PIPING TUNNELS, SECURITY/LIGHTING POLES, MANHOLES, DUCT BANKS; SBO STRUCTURES SUCH AS TRANSMISSION TOWERS, STARTUP TOWERS CIRCUIT BREAKER FOUNDATION, ELECTRICAL ENCLOSURE, CONTAINMENT INTERNAL STRUCTURES, FUEL STORAGE FACILITY, REFUELING CANAL, BWR UNIT VENT STACK*)

Systems, Structures, and Components

In-scope Class 1 *and Class 2* structures are organized into *three* groups and are discussed separately under subheadings A1 through A3. This section addresses the elements of BWR reactor building, PWR shield building, and control room/building, *Auxiliary Bldg., Diesel Generator Bldg., Radwaste Bldg., Turbine Bldg., Switchgear Rm., Yard Structures such as AFW Pumphouse, Utility/Piping Tunnels, Security/Lighting Poles, Manholes, Duct Banks; SBO Structures such as Transmission Towers, Startup Towers Circuit Breaker Foundation, Electrical Enclosure, Containment Internal Structures, Fuel Storage Facility, Refueling Canal*, *BWR Unit Vent Stack.* For this group, the applicable structural elements are concrete, steel, and masonry walls. The aging management review is presented for each applicable combination of structural element and aging effect.

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System Interfaces

Physical interfaces exist with any system or component that either penetrates the structure wall or is supported by the structure wall, floor, and roof. The direct interface is through the system or component supports that are anchored to the structure. Structures also protect housed systems or components from internal and external design basis events. In the case of tanks, there is a functional interface with the associated system. Water-control structures are integral parts of the systems that provide plant cooling water and residual heat removal.

III STRUCTURES AND COMPONENT SUPPORTS
A1 Group 1 Structures ~~(BWR Reactor Bldg., PWR Shield Bldg., Control Rm./Bldg.)~~

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
III.A1-1 (T-10)	III.A1.1-j; ***III.A2.1-j;*** ***III.A3.1-j;*** ***III.A4.1-c;*** ***III.A5.1-j***	Concrete: All	Reinforced concrete	Air – indoor uncontrolled	Reduction of strength and modulus/ elevated temperature (>150°F general; >200°F local)	Plant-specific aging management program For any concrete elements that exceed specified temperature limits, further evaluations are warranted. Appendix A of ACI 349-85 specifies the concrete temperature limits for normal operation or any other long-term period. The temperatures shall not exceed 150°F except for local areas which are allowed to have increased temperatures not to exceed 200°F.	Yes, if applicable	
III.A1-2 (T-03)	III.A1.1-c; ***III.A2.1-c;*** ***III.A3.1-c;*** ***III.A4.1-b;*** ***III.A5.1-c;*** ***III.A9.1-c***	Concrete: All	Reinforced concrete	Any	Expansion and cracking/ reaction with aggregates	Chapter XI.S6, "Structures Monitoring Program" Accessible Areas: Inspections/evaluations performed in accordance with "Structures Monitoring Program" will indicate the presence of expansion and cracking due to reaction with aggregates. Inaccessible Areas: ***As described in NUREG-1557, investigations, tests, and petrographic examinations of aggregates performed in accordance with ASTM C295-54 or ASTM C227-50 can***	No, if within the scope of the applicant's structures monitoring program and stated conditions are satisfied for inaccessible areas	To make this item match the previous version of the GALL. NOTE: This comment also applies to sections III.A2-2, III.A3-2, III.A4-2, III.A5-2, III.A7-1, III.A8-1,& III.A9-1 if consolidation is not accepted.

III	STRUCTURES AND COMPONENT SUPPORTS							
A1	Group 1 Structures ~~(BWR Reactor Bldg., PWR Shield Bldg., Control Rm./Bldg.)~~							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
						demonstrate that those aggregates do not react within reinforced concrete. For potentially reactive aggregates, aggregate-reinforced concrete reaction is not significant if the concrete was constructed in accordance with ACI 201.2R-77. Therefore, if these conditions are satisfied, aging management is not necessary. ~~Evaluation is needed if testing and petrographic examinations of aggregates performed in accordance with ASTM C295-54, ASTM C227-50, or ACI 201.2R-77 (NUREG-1557) demonstrate that the aggregates are reactive.~~		

III STRUCTURES AND COMPONENT SUPPORTS
A1 Group 1 Structures ~~(BWR Reactor Bldg., PWR Shield Bldg., Control Rm./Bldg.)~~

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
III.A1-3 (T-08)	III.A1.1-h; ***III.A2.1-h;*** ***III.A3.1-h;*** ***III.A5.1-h;*** ***III.A9.1-h***	Concrete: All	Reinforced concrete	Soil	Cracks and distortion due to increased stress levels from settlement	Chapter XI.S6, "Structures Monitoring Program" The initial licensing basis for some plants included a program to monitor settlement. If no settlement was evident during the first decade or so, the NRC may have given the licensee approval to discontinue the program. However, if a de-watering system is relied upon for control of settlement, then the licensee is to ensure proper functioning of the de-watering system through the period of extended operation.	No, if within the scope of the applicant's structures monitoring program	
III.A1-4 (T-05)	III.A1.1-e; ***III.A2.1-e;*** ***III.A3.1-e;*** ***III.A5.1-e;*** ***III.A9.1-e***	Concrete: Below-grade exterior; foundation	Reinforced concrete	Aggressive environment	Cracking, loss of bond, and loss of material (spalling, scaling)/ corrosion of embedded steel	Chapter XI.S~~2~~6, "***Structures Monitoring Program (SMP)***~~ASME Section XI, Subsection IWL~~". Accessible Areas: Inspections performed in accordance with ***SMP***~~IWL~~ will indicate the presence of ~~increase in porosity and permeability,~~ cracking, ***loss of bond, and*** ~~or~~ loss of material (spalling, scaling) due to ***corrosion of embedded steel*** ~~aggressive chemical attack~~ Inaccessible Areas:	Yes, if environment is aggressive	Incorrect AMP specified. IWL is for inspection of the reinforced concrete Containment and is not used to inspect other structures. The SMP is utilized for the inspection of these other structures in the sections listed, which is consistent with previous versions of the GALL.

III STRUCTURES AND COMPONENT SUPPORTS
A1 Group 1 Structures ~~(BWR Reactor Bldg., PWR Shield Bldg., Control Rm./Bldg.)~~

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						Examination of representative samples of below-grade concrete, when excavated for any reason, is to be performed, if the below-grade environment is aggressive (pH < 5.5, chlorides > 500ppm, or sulfates > 1,500 ppm). Periodic monitoring of below-grade water chemistry (including consideration of potential seasonal variations) is an acceptable approach to demonstrate that the below-grade environment is aggressive or non-aggressive.		Incorrect aging mechanism listed under the AMP for the aging mechanism specified in the previous column. NOTE: This comment also applies to III.A2-4, III.A3-4, III.A5-4, III.A7-3, III.A8-3, and III.A9-3 if consolidation is not accepted.
III.A1-5 (T-07)	III.A1.1-g; ***III.A2.1-g; III.A3.1-g; III.A5.1-g; III.A9.1-g***	Concrete: Below-grade exterior; foundation	Reinforced concrete	Aggressive environment	Increase in porosity and permeability, cracking, loss of material (spalling, scaling)/ aggressive chemical attack	Inaccessible Areas: Examination of representative samples of below-grade concrete, when excavated for any reason, is to be performed, if the below-grade environment is aggressive (pH < 5.5, chlorides > 500ppm, or sulfates > 1,500 ppm). Periodic monitoring of below-grade water chemistry (including consideration of potential seasonal variations) is an acceptable approach to demonstrate that the below-grade environment is aggressive or non-aggressive.	Yes, if environment is aggressive	

III STRUCTURES AND COMPONENT SUPPORTS
A1 Group 1 Structures ~~(BWR Reactor Bldg., PWR Shield Bldg., Control Rm./Bldg.)~~

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
III.A1-6 (T-01)	III.A1.1-a; ***III.A2.1-a; III.A3.1-a; III.A5.1-a; III.A9.1-a***	Concrete: Exterior above and below grade; foundation	Reinforced concrete	Air – outdoor	Loss of material (spalling, scaling) and cracking/ freeze-thaw	Chapter XI.S6, "Structures Monitoring Program" Accessible Areas: Inspections performed in accordance with "Structures Monitoring Program" will indicate the presence of loss of material (spalling, scaling) and cracking due to freeze-thaw. Inaccessible Areas: Evaluation is needed for plants that are located in moderate to severe weathering conditions (weathering index > 100 day-inch/yr) (NUREG-1557). Documented evidence to confirm that existing concrete has air content of 3% to 6% and subsequent inspections did not exhibit degradation related to freeze-thaw, should be considered a part of the evaluation. The weathering index for the continental US is shown in ASTM C33-90, Fig.1.	No, if within the scope of the applicant's structures monitoring program and stated conditions are satisfied for inaccessible areas	

III STRUCTURES AND COMPONENT SUPPORTS
A1 Group 1 Structures ~~(BWR Reactor Bldg., PWR Shield Bldg., Control Rm./Bldg.)~~

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
III.A1-7 (T-02)	III.A1.1-b; ***III.A2.1-b; III.A3.1-b; III.A5.1-b; III.A9.1-b***	Concrete: Exterior above and below grade; foundation	Reinforced concrete	Water – flowing	Increase in porosity and permeability, loss of strength/ leaching of calcium hydroxide	Chapter XI.S~~2~~6, "***Structures Monitoring Program (SMP)***~~ASME Section XI, Subsection IWL~~". Accessible areas: Inspections performed in accordance with ~~IWL~~ ***SMP*** will indicate the presence of increase in porosity, and permeability due to leaching of calcium hydroxide. Inaccessible Areas: ~~-~~ An aging management program is not necessary, even if reinforced concrete is exposed to flowing water, if there is documented evidence that confirms the in-place concrete was constructed in accordance with the recommendations in ACI 201.2R-77.	No,if concrete was constructed as stated for inaccessible areas	Incorrect AMP specified. IWL is for inspection of the reinforced concrete Containment and is not used to inspect other structures. The SMP is utilized for the inspection of these other structures in the sections listed, which is consistent with previous versions of the GALL. NOTE: This comment also applies to III.A2-7, III.A3-7, III.A5-7, III.A7-6, III.A8-6, and III.A9-6 if consolidation is not accepted.

III STRUCTURES AND COMPONENT SUPPORTS
A1 Group 1 Structures ~~(BWR Reactor Bldg., PWR Shield Bldg., Control Rm./Bldg.)~~

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
III.A1-8 (T-09)	III.A1.1-i; ***III.A2.1-i; III.A3.1-i; III.A5.1-i; III.A9.1-i***	Concrete: Foundation; subfoundation	Reinforced concrete; porous concrete	Water - flowing under foundation	Reduction in foundation strength, cracking, differential settlement/ erosion of porous concrete subfoundation	Chapter XI.S6, "Structures Monitoring Program" Erosion of cement from porous concrete subfoundations beneath containment basemats is described in IN 97-11. IN 98-26 proposes Maintenance Rule Structures Monitoring for managing this aging effect, if applicable. If a de-watering system is relied upon for control of erosion of cement from porous concrete subfoundations, then the licensee is to ensure proper functioning of the de-watering system through the period of extended operation.	No, if within the scope of the applicant's structures monitoring program	
III.A1-9 (T-06)	III.A1.1-f; ***III.A2.1-f; III.A3.1-f; III.A4.1-a; III.A5.1-f; III.A9.1-f***	Concrete: Interior and above-grade exterior	Reinforced concrete	Aggressive environment	Increase in porosity and permeability, cracking, loss of material (spalling, scaling)/ aggressive chemical attack	Chapter XI.S6, "Structures Monitoring Program" Accessible Areas: Inspections performed in accordance with "Structures Monitoring Program" will indicate the presence of increase in porosity and permeability, cracking, or loss of material (spalling, scaling) due to aggressive chemical attack.	No, if within the scope of the applicant's structures monitoring program	

III STRUCTURES AND COMPONENT SUPPORTS
A1 Group 1 Structures ~~(BWR Reactor Bldg., PWR Shield Bldg., Control Rm./Bldg.)~~

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
III.A1-10 (T-04)	III.A1.1-d; ***III.A2.1-d;*** ***III.A3.1-d;*** ***III.A4.1-d;*** ***III.A5.1-d;*** ***III.A9.1-d***	Concrete: Interior and above-grade exterior	Reinforced concrete	Air – indoor uncontrolled or air - outdoor	Cracking, loss of bond, and loss of material (spalling, scaling)/ corrosion of embedded steel	Chapter XI.S6, "Structures Monitoring Program" Accessible areas: Inspections performed in accordance with "Structures Monitoring Program" will indicate the presence of cracking, loss of bond, and loss of material (spalling, scaling) due to corrosion of embedded steel.	No, if within the scope of the applicant's structures monitoring program	
III.A1-11 (T-12)	III.A1.3-a; ***III.A2.3-a;*** ***III.A3.3-a;*** ***III.A5.3-a***	Masonry walls: All	Concrete block	Air – indoor uncontrolled or air - outdoor	Cracking due to restraint shrinkage, creep, and aggressive environment	Chapter XI.S5, "Masonry Wall Program" ***Note: May be part of XI.S6, "Structures Monitoring Program".***	No	Many plants use SMP for Masonry Walls.
III.A1-12 (T-11)	III.A1.2-a; ***III.A2.2-a;*** ***III.A3.2-a;*** ***III.A4.2-a;*** ***III.A5.2-a***	Steel components: All structural steel	Steel	Air – indoor uncontrolled or air - outdoor	Loss of material/ corrosion	Chapter XI.S6, "Structures Monitoring Program" If protective coatings are relied upon to manage the effects of aging, the structures monitoring program is to include provisions to address protective coating monitoring and maintenance.	No, if within the scope of the applicant's structures monitoring program	

III STRUCTURES AND COMPONENT SUPPORTS
A1 Group 1 Structures ~~(BWR Reactor Bldg., PWR Shield Bldg., Control Rm./Bldg.)~~

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
~~III.A4-6~~ ***III.A1-13*** (T-13)	III.A4.2-b	Steel components: Radial beam seats in BWR drywell; RPV support shoes for PWR with nozzle supports; Steam generator supports	Lubrite	Air – indoor uncontrolled	~~Lock-up/ wear~~ ***None***	~~Chapter XI.S3, "ASME Section IX, Subsection IWF" or Chapter XI.S6, "Structures Monitoring Program"~~	No, if within the scope of Section XI, IWF or structures monitoring program	Industry experience and EPRI Civil Tools indicates this aging effect is not applicable to graphite plate (lubrite). This is consistent with past approved applications. See NUREG-1759, Turkey Point SER, NUREG-1769, Peach Bottom SER, NUREG-1785, H.B. Robinson SER NUREG-1766, North Anna/Surry SER
~~III.A5-13~~ ***III.A1-14*** (T-14)	III.A5.2-b	Steel components: Fuel pool liner	Stainless steel	~~Water-Standing~~ ***Treated water or treated borated water***	Cracking/ stress corrosion cracking Loss of material/pitting and crevice corrosion	Chapter XI.M2, "Water Chemistry Program" and monitoring of the spent fuel pool water level	No	Treated water or Treated borated water environment is a better description of spent fuel pool environment.

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A2. GROUP *2* STRUCTURES (WATER-CONTROL STRUCTURES)

Systems, Structures, and Components

In-scope Class 1 ***and Class 2*** structures are organized into ***three*** groups and are discussed separately under subheadings A1 through A*3*. This section addresses the elements of water-control structures. For this group, the applicable structural elements are identified: concrete, steel, masonry walls, and earthen water-control structures. The aging management review is presented for each applicable combination of structural element and aging effect.

System Interfaces

Physical interfaces exist with any system or component that either penetrates the structure wall or is supported by the structure wall, floor, and roof. The direct interface is through the system or component supports that are anchored to the structure. Structures also protect housed systems and components from internal and external design basis events. In the case of tanks, there is a functional interface with the associated system. Water-control structures are integral parts of the systems that provide plant cooling water and residual heat removal.

III STRUCTURES AND COMPONENT SUPPORTS
A*2* Group *2* Structures (Water-Control Structures)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
~~III.A6-1~~ ***III.A2-1*** (T-19)	III.A6.1-e	Concrete: All	Reinforced concrete	Aggressive environment	Increase in porosity and permeability, cracking, loss of material (spalling, scaling)/ aggressive chemical attack	Chapter XI.S7, "Regulatory Guide 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants" or the FERC / US Army Corp of Engineers dam inspections and maintenance ***and/or Chapter XI.S6 "Structures Monitoring Program".*** ~~As described in NUREG-1557, aggressive chemical attack on interior and above-grade exterior reinforced concrete is not significant if the concrete is not exposed to an aggressive environment (pH <5.5), or to chloride or sulfate solutions beyond defined limits (>500 ppm chloride, or >1500 ppm sulfate). Therefore, if these conditions are satisfied, aging management is not necessary.~~ ***Accessible Areas: Inspections performed in accordance with "Regulatory Guide 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants" or the FERC / US Army Corp of Engineers dam inspections and maintenance or the Structures***	~~No~~ ***Yes, if environment is aggressive***	Change recognizes that SMP is also commonly used for aging management of concrete in water control structures. As previously written, text was not consistent with other sections of GALL because there was no discussion of accessible and inaccessible concrete. It is also not consistent with the NRC's requirement in ISG-3 and other sections of the GALL.

III STRUCTURES AND COMPONENT SUPPORTS
A2 Group 2 Structures (Water-Control Structures)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						Monitoring Program will indicate the presence of increase in porosity and permeability, cracking, or loss of material (spalling, scaling) due to aggressive chemical attack. *Inaccessible Areas: Examination of representative samples of below-grade concrete, when excavated for any reason, is to be performed, if the below-grade environment is aggressive (pH < 5.5, chlorides > 500ppm, or sulfates > 1,500 ppm).* *Periodic monitoring of below-grade water chemistry (including consideration of potential seasonal variations) is an acceptable approach to demonstrate that the below-grade environment is aggressive or non-aggressive.*		

III A2	STRUCTURES AND COMPONENT SUPPORTS Group 2 Structures (Water-Control Structures)							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
~~III.A6-2~~ ***III.A2-2*** (T-18)	III.A6.1-d	Concrete: All	Reinforced concrete	Air – indoor uncontrolled or air – outdoor	Cracking, loss of bond, and loss of material (spalling, scaling)/ corrosion of embedded steel	Chapter XI.S7, "Regulatory Guide 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants" or the FERC / US Army Corp of Engineers dam inspections and maintenance ***and/or Chapter XI.S6 "Structures Monitoring Program"*** ~~As described in NUREG-1557, corrosion of exterior above-grade and interior embedded steel is not significant if the steel is not exposed to an aggressive environment (concrete pH <11.5 or chlorides >500 ppm). If such steel is exposed to an aggressive environment, corrosion is not significant if the concrete in which the steel is embedded has a low water-to-cement ratio (0.35-0.45), adequate air entrainment (3-6%), low permeability, and is designed in accordance with ACI 318-63 or ACI 349-85. Therefore, if these conditions are satisfied, aging management is not necessary.~~ ***Accessible areas: Inspections performed in***	~~No~~ ***Yes, if environment is aggressive***	Change recognizes that SMP is also commonly used for aging management of concrete in water control structures. As previously written, text was not consistent with other sections of GALL because there was no discussion of accessible and inaccessible concrete. It is also not consistent with the NRC's requirement in ISG-3 and other sections of the GALL.

III STRUCTURES AND COMPONENT SUPPORTS
A2 Group 2 Structures (Water-Control Structures)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						accordance with "Regulatory Guide 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants" or the FERC / US Army Corp of Engineers dam inspections and maintenance *or the Structures Monitoring Program* ***will indicate the presence of cracking, loss of bond, and loss of material (spalling, scaling) due to corrosion of embedded steel.*** ***Inaccessible Areas: Examination of representative samples of below-grade concrete, when excavated for any reason, is to be performed, if the below-grade environment is aggressive (pH < 5.5, chlorides > 500ppm, or sulfates >*** *1,500 ppm).* ***Periodic monitoring of below-grade water chemistry (including consideration of potential seasonal variations) is an acceptable approach to demonstrate that the below-grade environment is aggressive or non-aggressive.***		

III	STRUCTURES AND COMPONENT SUPPORTS
A*2*	Group *2* Structures (Water-Control Structures)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
~~III.A6-3~~ ***III.A2-3*** (T-17)	III.A6.1-c	Concrete: All	Reinforced concrete	Any	Expansion and cracking/ reaction with aggregates	Chapter XI.S7, "Regulatory Guide 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants" or the FERC / US Army Corp of Engineers dam inspections and maintenance ***and/or Chapter XI.S6 "Structures Monitoring Program".*** ***Accessible Areas:*** ***Inspections/evaluations performed in accordance with "Regulatory Guide 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants" or the FERC / US Army Corp of Engineers dam inspections and maintenance or Structures Monitoring Program will indicate the presence of expansion and cracking due to reaction with aggregates.*** ***Inaccessible areas:*** As described in NUREG-1557, investigations, tests, and petrographic examinations of aggregates performed in	No, ***if within the scope of the applicant's , "Regulatory Guide 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants" or the FERC / US Army Corp of Engineers dam inspections and maintenance or Structures Monitoring Program and stated conditions are satisfied for inaccessible areas.***	Change recognizes that SMP is also commonly used for aging management of concrete in water control structures. As previously written, text was not consistent with other sections of GALL because there was no discussion of accessible and inaccessible concrete. It is also not consistent with the NRC's requirement in ISG-3 and other sections of the GALL. Clarification provided under further evaluation column.

III A2	STRUCTURES AND COMPONENT SUPPORTS Group 2 Structures (Water-Control Structures)							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
						accordance with ASTM C295-54 or ASTM C227-50 can demonstrate that those aggregates do not react within reinforced concrete. For potentially reactive aggregates, aggregate-reinforced concrete reaction is not significant if the concrete was constructed in accordance with ACI 201.2R-77. Therefore, if these conditions are satisfied, aging management is not necessary.		
~~III.A6-4~~ ***III.A2-4*** (T-08)	III.A6.1-f	Concrete: All	Reinforced concrete	Soil	Cracks and distortion due to increased stress levels from settlement	Chapter XI.S6, "Structures Monitoring Program" The initial licensing basis for some plants included a program to monitor settlement. If no settlement was evident during the first decade or so, the NRC may have given the licensee approval to discontinue the program. However, if a de-watering system is relied upon for control of settlement, then the licensee is to ensure proper functioning of the de-watering system through the period of extended operation.	No, if within the scope of the applicant's structures monitoring program	

III	STRUCTURES AND COMPONENT SUPPORTS
A2	Group 2 Structures (Water-Control Structures)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
~~III.A6-5~~ ***III.A2-5*** (T-15)	III.A6.1-a	Concrete: Exterior above and below grade; foundation; ~~interior slab~~	Reinforced concrete	Air – outdoor	Loss of material (spalling, scaling) and cracking/ freeze-thaw	Chapter XI.S7, "Regulatory Guide 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants" or the FERC / US Army Corp of Engineers dam inspections and maintenance ***and/or Chapter XI.S6 "Structures Monitoring Program".*** ~~As described in NUREG-1557, freeze-thaw does not cause loss of material from reinforced concrete in foundations, and in above- and below-grade exterior concrete, for plants located in a geographic region of negligible weathering conditions (weathering index <100 day-inch/yr). Loss of material from such concrete is not significant at plants located in areas in which weathering conditions are severe (weathering index >500 day-inch/yr) or moderate (100-500 day-inch/yr), provided that the concrete mix design meets the air content (entrained air 3-6%) and water-to-cement ratio (0.35-0.45) specified in ACI 318-~~	No, ***if within the scope of the applicant's "Regulatory Guide 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants," the FERC / US Army Corp of Engineers dam inspections and maintenance or the Structures Monitoring Program and stated conditions are satisfied for inaccessible areas***	Interior slabs are not exposed to air – outdoor. Change recognizes that SMP is also commonly used for aging management of concrete in water control structures. As previously written, text was not consistent with other sections of GALL because there was no discussion of accessible and inaccessible concrete. It is also not consistent with the NRC's requirement in ISG-3 and other sections of the GALL.

III STRUCTURES AND COMPONENT SUPPORTS
A2 Group 2 Structures (Water-Control Structures)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						~~63 or ACI 349-85. Therefore, if these conditions are satisfied, aging management is not necessary.~~ ~~The weathering index is defined in ASTM C33-90, Table 3, Footnote E. Fig. 1 of ASTM C33-90 illustrates the various weathering index regions throughout the U.S.~~ ***Accessible Areas: Inspections performed in accordance with "Regulatory Guide 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants," the "FERC / US Army Corp of Engineers dam inspections and maintenance" or the Structures Monitoring Program will indicate the presence of loss of material (spalling, scaling) and cracking due to freeze-thaw.*** ***Inaccessible Areas: Evaluation is needed for plants that are located in moderate to severe weathering conditions***		

III STRUCTURES AND COMPONENT SUPPORTS
A2 Group 2 Structures (Water-Control Structures)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						(weathering index > 100 day-inch/yr) (NUREG-1557). Documented evidence to confirm that existing concrete has air content of 3% to 6%, and subsequent inspections did not exhibit degradation related to freeze-thaw, should be considered a part of the evaluation. The weathering index for the continental US is shown in ASTM C33-90, Fig.1.		

III	STRUCTURES AND COMPONENT SUPPORTS							
A2	Group *2* Structures (Water-Control Structures)							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
~~III.A6-6~~ ***III.A2-6*** (T-16)	III.A6.1-b	Concrete: Exterior above and below grade; foundation; interior slab	Reinforced concrete	Water – flowing	Increase in porosity and permeability, loss of strength/ leaching of calcium hydroxide	Chapter XI.S7, "Regulatory Guide 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants" or the FERC / US Army Corp of Engineers dam inspections and maintenance ***and/or Chapter XI.S6 "Structures Monitoring Program".*** ~~As described in NUREG-1557, leaching of calcium hydroxide from reinforced concrete becomes significant only if the concrete is exposed to flowing water. Even if reinforced concrete is exposed to flowing water, such leaching is not significant if the concrete is constructed to ensure that it is dense, well-cured, has low permeability, and that cracking is well controlled. Cracking is controlled through proper arrangement and distribution of reinforcing bars. All of the above characteristics are assured if the concrete was constructed with the guidance of ACI 201.2R-77. Therefore, if these conditions are satisfied, aging management is~~	No, ***if concrete was constructed as stated for inaccessible areas***	Change recognizes that SMP is also commonly used for aging management of concrete in water control structures. As previously written, text was not consistent with other sections of GALL because there was no discussion of accessible and inaccessible concrete. It is also not consistent with the NRC's requirement in ISG-3 and other sections of the GALL. Clarification provided under further evaluation column.

III STRUCTURES AND COMPONENT SUPPORTS
A2 Group 2 Structures (Water-Control Structures)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						~~not necessary.~~ ***Accessible Areas:*** ***Inspections performed in accordance with "Regulatory Guide 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants," the FERC / US Army Corp of Engineers dam inspections and maintenance or the Structures Monitoring Program will indicate the presence of increase in porosity and permeability, loss of strength/ leaching of calcium hydroxide.*** ***Inaccessible Areas:*** ***An aging management program is not necessary, even if reinforced concrete is exposed to flowing water, if there is documented evidence that confirms the in-place concrete was constructed in accordance with the recommendations in ACI 201.2R-77.***		

III STRUCTURES AND COMPONENT SUPPORTS
A2 Group *2* Structures (Water-Control Structures)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
~~III.A6-7~~ ***III.A2-7*** (T-20)	III.A6.1-h	Concrete: Exterior above and below grade; foundation; interior slab	Reinforced concrete	Water – flowing	Loss of material/ abrasion; cavitation	Chapter XI.S7, "Regulatory Guide 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants" or the FERC / US Army Corp of Engineers dam inspections and maintenance ***and/or Chapter XI.S6 "Structures Monitoring Program".***	No	Change recognizes that SMP is also commonly used for aging management of concrete in water control structures.
~~III.A6-8~~ ***III.A2-8*** (T-09)	III.A6.1-g	Concrete: Foundation; subfoundation	Reinforced concrete; porous concrete	Water - flowing under foundation	Reduction in foundation strength, cracking, differential settlement/ erosion of porous concrete subfoundation	Chapter XI.S6, "Structures Monitoring Program" Erosion of cement from porous concrete subfoundations beneath containment basemats is described in IN 97-11. IN 98-26 proposes Maintenance Rule Structures Monitoring for managing this aging effect, if applicable. If a de-watering system is relied upon for control of erosion of cement from porous concrete subfoundations, then the licensee is to ensure proper functioning of the de-watering system through the period of extended operation.	No, if within the scope of the applicant's structures monitoring program	

III STRUCTURES AND COMPONENT SUPPORTS
A2 Group *2* Structures (Water-Control Structures)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
~~III.A6-9~~ ***III.A2-9*** (T-22)	III.A6.4-a	Earthen water-control structures: Dams, embankments, reservoirs, channels, canals and ponds	Various	Water – flowing Water – standing	Loss of material, loss of form/ erosion, settlement, sedimentation, frost action, waves, currents, surface runoff, seepage	Chapter XI.S7, "Regulatory Guide 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants" or the FERC / US Army Corp of Engineers dam inspections and maintenance ***and/or Chapter XI.S6 "Structures Monitoring Program".***	No	Change recognizes that SMP is also commonly used for aging management of concrete in water control structures.
~~III.A6-10~~ ***III.A2-10*** (T-12)	III.A6.3-a	Masonry walls: All	Concrete block	Air – indoor uncontrolled or air - outdoor	Cracking due to restraint shrinkage, creep, and aggressive environment	Chapter XI.S5, "Masonry Wall Program" ***Note: May be part of XI.S6, "Structures Monitoring Program".***	No	Many plants use SMP for Masonry Walls.

III A2	STRUCTURES AND COMPONENT SUPPORTS Group **2** Structures (Water-Control Structures)							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
~~III.A6-11~~ ***III.A2-11*** (T-21)	III.A6.2-a	Metal components: All structural members	Steel; Copper alloys	Air – indoor uncontrolled or air – outdoor ***Water-flowing, water - standing***	Loss of material/ General (steel only), pitting and crevice corrosion	Chapter XI.S7, "Regulatory Guide 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants" or the FERC / US Army Corp of Engineers dam inspections and maintenance ***and/or Chapter XI.S6 "Structures Monitoring Program".*** If protective coatings are relied upon to manage the effects of aging, this AMP is to include provisions to address protective coating monitoring and maintenance.	No	Added environment for carbon steel in flowing or standing water for items such as sheet piles and gates. Change recognizes that SMP is also commonly used for aging management of concrete in water control structures.

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A*3*. GROUP *3* STRUCTURES (~~CONCRETE~~ TANKS AND MISSILE BARRIERS

Systems, Structures, and Components

In-scope Class 1 ***and Class 2*** structures are organized into ***three*** groups and are discussed separately under subheadings A1 through A*3*. This section addresses the elements of tanks ***(concrete and steel),*** missile barriers ***for tanks and BWR unit vent stack***. For this group, the applicable structural elements are identified: concrete and steel. The aging management review is presented for each applicable combination of structural element and aging effect.

System Interfaces

Physical interfaces exist with any system or component that either penetrates the structure wall or is supported by the structure wall, floor, and roof. The direct interface is through the system or component supports that are anchored to the structure. Structures also protect housed systems and components from internal and external design basis events. In the case of tanks, there is a functional interface with the associated system. Water-control structures are integral parts of the systems that provide plant cooling water and residual heat removal.

III STRUCTURES AND COMPONENT SUPPORTS
A*3* Group *3* Structures (~~Concrete~~ Tanks and Missile Barriers)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
~~III.A7-1~~ *III.A3-1* (T-03)	III.A7.1-c; *III.A8.1-c;*	Concrete: All	Reinforced concrete	Any	Expansion and cracking/ reaction with aggregates	Chapter XI.S6, "Structures Monitoring Program" Accessible Areas: Inspections/evaluations performed in accordance with "Structures Monitoring Program" will indicate the presence of expansion and cracking due to reaction with aggregates. Inaccessible Areas: Evaluation is needed if testing and petrographic examinations of aggregates performed in accordance with ASTM C295-54, ASTM C227-50, or ACI 201.2R-77 (NUREG-1557) demonstrate that the aggregates are reactive.	No, if within the scope of the applicant's structures monitoring program and stated conditions are satisfied for inaccessible areas	

III	STRUCTURES AND COMPONENT SUPPORTS
A*3*	Group *3* Structures (~~Concrete~~ Tanks and Missile Barriers)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
~~III.A7-2~~ ***III.A3-2*** (T-08)	III.A7.1-h; ***III.A8.1-f;***	Concrete: All	Reinforced concrete	Soil	Cracks and distortion due to increased stress levels from settlement	Chapter XI.S6, "Structures Monitoring Program" The initial licensing basis for some plants included a program to monitor settlement. If no settlement was evident during the first decade or so, the NRC may have given the licensee approval to discontinue the program. However, if a de-watering system is relied upon for control of settlement, then the licensee is to ensure proper functioning of the de-watering system through the period of extended operation.	No, if within the scope of the applicant's structures monitoring program	

III STRUCTURES AND COMPONENT SUPPORTS
A*3* Group *3* Structures (~~Concrete~~ Tanks and Missile Barriers)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
~~III.A7-3~~ ***III.A3-3*** (T-05)	III.A7.1-e; ***III.A8.1-d;***	Concrete: Below-grade exterior; foundation	Reinforced concrete	Aggressive environment	Cracking, loss of bond, and loss of material (spalling, scaling)/ corrosion of embedded steel	Chapter XI.S~~2~~*6*, "***Structures Monitoring Program (SMP)***~~ASME Section XI, Subsection IWL~~". Accessible Areas: Inspections performed in accordance with ***SMP***~~IWL~~ will indicate the presence of ~~increase in porosity and permeability,~~ cracking, ***loss of bond, and*** ~~or~~ loss of material (spalling, scaling) due to ***corrosion of embedded steel*** ~~aggressive chemical attack~~ Inaccessible Areas: Examination of representative samples of below-grade concrete, when excavated for any reason, is to be performed, if the below-grade environment is aggressive (pH < 5.5, chlorides > 500ppm, or sulfates > 1,500 ppm). Periodic monitoring of below-grade water chemistry (including consideration of potential seasonal variations) is an acceptable approach to demonstrate that the below-grade environment is aggressive or non-aggressive.	Yes, if environment is aggressive	Incorrect AMP specified. IWL is for inspection of the reinforced concrete Containment and is not used to inspect other structures. The SMP is utilized for the inspection of these other structures in the sections listed, which is consistent with previous versions of the GALL. Incorrect aging mechanism listed under the AMP for the aging mechanism specified in the previous column.

III A3	STRUCTURES AND COMPONENT SUPPORTS Group *3* Structures (~~Concrete~~ Tanks and Missile Barriers)							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
~~III.A7-4~~ ***III.A3-4*** (T-07)	III.A7.1-g; ***III.A8.1-e;***	Concrete: Below-grade exterior; foundation	Reinforced concrete	Aggressive environment	Increase in porosity and permeability, cracking, loss of material (spalling, scaling)/ aggressive chemical attack	Inaccessible Areas: Examination of representative samples of below-grade concrete, when excavated for any reason, is to be performed, if the below-grade environment is aggressive (pH < 5.5, chlorides > 500ppm, or sulfates > 1,500 ppm). Periodic monitoring of below-grade water chemistry (including consideration of potential seasonal variations) is an acceptable approach to demonstrate that the below-grade environment is aggressive or non-aggressive.	Yes, if environment is aggressive	

III STRUCTURES AND COMPONENT SUPPORTS
A3 Group ***3*** Structures (~~Concrete~~ Tanks and Missile Barriers)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
~~III.A7-5~~ ***III.A3-5*** (T-01)	III.A7.1-a; ***III.A8.1-a***;	Concrete: Exterior above and below grade; foundation	Reinforced concrete	Air – outdoor	Loss of material (spalling, scaling) and cracking/ freeze-thaw	Chapter XI.S6, "Structures Monitoring Program" Accessible Areas: Inspections performed in accordance with "Structures Monitoring Program" will indicate the presence of loss of material (spalling, scaling) and cracking due to freeze-thaw. Inaccessible Areas: Evaluation is needed for plants that are located in moderate to severe weathering conditions (weathering index > 100 day-inch/yr) (NUREG-1557). Documented evidence to confirm that existing concrete has air content of 3% to 6% and subsequent inspections did not exhibit degradation related to freeze-thaw, should be considered a part of the evaluation. The weathering index for the continental US is shown in ASTM C33-90, Fig.1.	No, if within the scope of the applicant's structures monitoring program and stated conditions are satisfied for inaccessible areas	

III A3	STRUCTURES AND COMPONENT SUPPORTS Group *3* Structures (~~Concrete~~ Tanks and Missile Barriers)							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
~~III.A7-6~~ ***III.A3-6*** (T-02)	III.A7.1-b; ***III.A8.1-b;***	Concrete: Exterior above and below grade; foundation	Reinforced concrete	Water – flowing	Increase in porosity and permeability, loss of strength/ leaching of calcium hydroxide	Chapter XI.S~~2~~*6*, "***Structures Monitoring Program (SMP)***~~ASME Section XI, Subsection IWL~~". Accessible areas: Inspections performed in accordance with ~~IWL~~ ***SMP*** will indicate the presence of increase in porosity, and permeability due to leaching of calcium hydroxide. Inaccessible Areas: ~~-~~ An aging management program *is not necessary*, even if reinforced concrete is exposed to flowing water, if there is documented evidence that confirms the in-place concrete was constructed in accordance with the recommendations in ACI 201.2R-77.	No, if concrete was constructed as stated for inaccessible *areas*	Incorrect AMP specified. IWL is for inspection of the *reinforced concrete* Containment and is not used to inspect other structures. The SMP is utilized for the inspection of these other structures in the sections listed, which is consistent with previous versions of the GALL.

III STRUCTURES AND COMPONENT SUPPORTS

A*3* Group *3* Structures (~~Concrete~~ Tanks and Missile Barriers)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
~~III.A7-7~~ ***III.A3-7*** (T-09)	III.A7.1-i; ***III.A8.1-g***;	Concrete: Foundation; subfoundation	Reinforced concrete; porous concrete	Water - flowing under foundation	Reduction in foundation strength, cracking, differential settlement/ erosion of porous concrete subfoundation	Chapter XI.S6, "Structures Monitoring Program" Erosion of cement from porous concrete subfoundations beneath containment basemats is described in IN 97-11. IN 98-26 proposes Maintenance Rule Structures Monitoring for managing this aging effect, if applicable. If a de-watering system is relied upon for control of erosion of cement from porous concrete subfoundations, then the licensee is to ensure proper functioning of the de-watering system through the period of extended operation.	No, if within the scope of the applicant's structures monitoring program	
~~III.A7-8~~ ***III.A3-8*** (T-06)	III.A7.1-f;	Concrete: Interior and above-grade exterior	Reinforced concrete	Aggressive environment	Increase in porosity and permeability, cracking, loss of material (spalling, scaling)/ aggressive chemical attack	Chapter XI.S6, "Structures Monitoring Program" Accessible Areas: Inspections performed in accordance with "Structures Monitoring Program" will indicate the presence of increase in porosity and permeability, cracking, or loss of material (spalling, scaling) due to aggressive chemical attack.	No, if within the scope of the applicant's structures monitoring program	

III A*3*	STRUCTURES AND COMPONENT SUPPORTS Group *3* Structures (~~Concrete~~ Tanks and Missile Barriers)							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
~~III.A7-9~~ ***III.A3-9*** (T-04)	III.A7.1-d;	Concrete: Interior and above-grade exterior	Reinforced concrete	Air – indoor uncontrolled or air - outdoor	Cracking, loss of bond, and loss of material (spalling, scaling)/ corrosion of embedded steel	Chapter XI.S6, "Structures Monitoring Program" Accessible areas: Inspections performed in accordance with "Structures Monitoring Program" will indicate the presence of cracking, loss of bond, and loss of material (spalling, scaling) due to corrosion of embedded steel.	No, if within the scope of the applicant's structures monitoring program	
~~III.A7-10~~ ***III.A3-10*** (T-11)	III.A7.2-a; ***III.A8.2-a***	Steel components: All structural steel	Steel	Air – indoor uncontrolled or air - outdoor	Loss of material/ corrosion	Chapter XI.S6, "Structures Monitoring Program" If protective coatings are relied upon to manage the effects of aging, the structures monitoring program is to include provisions to address protective coating monitoring and maintenance.	No, if within the scope of the applicant's structures monitoring program	
~~III.A7-11~~ ***II.A3-11*** (T-23)	III.A7.2-b; ***III.A8.2-b***	Steel components: Tank liner	Stainless steel	Water – standing	Cracking/ stress corrosion cracking Loss of material/pitting and crevice corrosion	A plant-specific aging management program is to be evaluated.	Yes	

COMPONENT SUPPORTS

B1. Supports for ASME Piping and Components ***and Class MC (BWR Containment Supports)***

B2. ***Other Supports – All other in-scope supports except as stated in B1***

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B1. SUPPORTS FOR ASME PIPING AND COMPONENTS *AND CLASS MC (BWR CONTAINMENT SUPPORTS)*

Systems, Structures, and Components

This section addresses supports and anchorage for ASME piping systems and components. It ***includes*** Class 1, Class 2 and 3, and Class MC ***(BWR Containment Supports).*** Applicable aging effects are identified and the aging management review is presented for each applicable combination of support component and aging effect.

System Interfaces

Physical interfaces exist with the structure, system, or component being supported and with the building structural element to which the support is anchored. A primary function of supports is to provide anchorage of the supported element for internal and external design basis events, so that the supported element can perform its intended function.

III	STRUCTURES AND COMPONENT SUPPORTS
B1	***Supports for ASME Piping and Components and Class MC (BWR Containment Supports)***

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
~~III.B1.1-1~~ ***III.B1-1*** (T-29)	III.B1.1.4-a; ***III.B1.2.3-a;*** ***III.B1.3.3-a;***	Building concrete at locations of expansion and grouted anchors; grout pads for support base plates	Reinforced concrete; grout	Air – indoor uncontrolled or air - outdoor	Reduction in concrete anchor capacity due to local concrete degradation/ service-induced cracking or other concrete aging mechanisms	Chapter XI.S6, "Structures Monitoring Program"	No, if within the scope of the applicant's structures monitoring program	
~~III.B1.1-2~~ ***III.B1-2*** (T-28)	III.B1.1.3-a; ***III.B1.2.2-a;*** ***III.B1.3.2-a***	Constant and variable load spring hangers; guides; stops; sliding surfaces; design clearances; vibration isolators	Steel and non-steel materials (e.g., ~~lubrite plates,~~ vibration isolators, etc.)	Air – indoor uncontrolled or air - outdoor	Loss of mechanical function/ corrosion, distortion, dirt, overload, fatigue due to vibratory and cyclic thermal loads; elastomer hardening	Chapter XI.S3, "ASME Section XI, Subsection IWF"	No	Industry experience and EPRI Civil Tools indicates this aging effect is not applicable to graphite plate (lubrite). This is consistent with past approved applications. See NUREG-1759, Turkey Point SER, NUREG-1769, Peach Bottom SER, NUREG-1785, H.B. Robinson SER NUREG-1766, North Anna/Surry SER
~~III.B1.1-3~~ ***III.B1-3*** (T-27)	III.B1.1.2-a	High strength bolting for NSSS component supports	Low alloy steel, yield strength >150 ksi	Air – indoor uncontrolled (External)	Cracking/ stress corrosion cracking ***Loss of material/general corrosion***	Chapter XI.M18, "Bolting Integrity" ***or Chapter XI.S3, "ASME Section XI, Subsection IWF"***	No	Loss of material is an applicable aging effect for this component. IWF is also an adequate program for inspection of high strength bolting for the

III	STRUCTURES AND COMPONENT SUPPORTS
B1	***Supports for ASME Piping and Components and Class MC (BWR Containment Supports)***

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
								identified aging effects.
~~III.B1.1-4~~ ***III.B1-4*** (TP-8)	III.B1.1; ***III.B1.2;*** ***III.B1.3***	Support members; welds; bolted connections; support anchorage to building structure	Galvanized steel, aluminum	Air – indoor uncontrolled	~~Loss of material/ galvanic corrosion~~ ***None***	~~Chapter XI.S6, "Structures Monitoring Program"~~ ***None***	No	Galvanic corrosion requires an electrolyte present between differential noble metals in contact to occur. Air – indoor uncontrolled environment is defined as normally dry in GALL chapter IX. Structural support members are generally within the less noble galvanic series and are not typically in contact with high noble metals, sacrificial shims are typically installed between different metals. Per NUREG basis document for GALL it states "Difficult to justify loss of materials in this environment. All issued SERs have accepted "no aging effects." Therefore aging effect and AMP should be changed to

III STRUCTURES AND COMPONENT SUPPORTS
B1 ***Supports for ASME Piping and Components and Class MC (BWR Containment Supports)***

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
								"None". Also applies to III.B1.2-3, III.B1.3-3, III.B2-4, III.B3-2, III.B4-4 & III.B5-2 if consolidation is not accepted.
~~III.B1.1-5~~ ***III.B1-5*** (TP-3)	III.B1.1; ***III.B1.2; III.B1.3***	Support members; welds; bolted connections; support anchorage to building structure	Galvanized steel, aluminum	Air with borated water leakage	Loss of material/ boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No	
~~III.B1.1-6~~ ***III.B1-6*** (TP-5)	III.B1.1; ***III.B1.2; III.B1.3***	Support members; welds; bolted connections; support anchorage to building structure	Stainless steel	Air – indoor uncontrolled	None	None	No	

III STRUCTURES AND COMPONENT SUPPORTS
B1 ***Supports for ASME Piping and Components and Class MC (BWR Containment Supports)***

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
~~III.B1.1-7~~ ***III.B1-7*** (TP-4)	III.B1.1; ***III.B1.2; III.B1.3***	Support members; welds; bolted connections; support anchorage to building structure	Stainless steel	Air with borated water leakage	None	None	No	
~~III.B1.1-8~~ ***III.B1-8*** (T-26)	III.B1.1.1-c; ***III.B1.2.1-c; III.B1.3.1-b;***	Support members; welds; bolted connections; support anchorage to building structure	Steel	Air – indoor uncontrolled	Cumulative fatigue damage/ fatigue (Only if CLB fatigue analysis exists)	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA	
~~III.B1.1-9~~ ***III.B1-9*** (T-24)	III.B1.1.1-a; ***III.B1.2.1-a; III.B1.3.1-a***	Support members; welds; bolted connections; support anchorage to building structure	Steel	Air – indoor uncontrolled or air outdoor	Loss of material/ general and pitting corrosion	Chapter XI.S3, "ASME Section XI, Subsection IWF"	No	

III STRUCTURES AND COMPONENT SUPPORTS
B1 ***Supports for ASME Piping and Components and Class MC (BWR Containment Supports)***

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
~~III.B1.1-10~~ *III.B1-10* (T-25)	III.B1.1.1-b; ***III.B1.2.1-b;***	Support members; welds; bolted connections; support anchorage to building structure	Steel	Air with borated water leakage	Loss of material/ boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No	.
New III.B1(1)		***BWR support components***	***Carbon steel; stainless steel***	***Treated Water < 60C (<140^0 F)***	***Loss of Material/ general, crevice ,pitting corrosion***	***Chapter XI.M2 Water Chemistry***	***No***	*Added new item to cover BWR support components in treated water.*

B2. *OTHER SUPPORTS – ALL OTHER SUPPORTS EXCEPT THOSE STATED IN B1*

Systems, Structures, and Components

This section addresses supports and anchorage for ***all other in-scope supports except supports for ASME piping and components (e.g.;*** cable trays, conduit, HVAC ducts, tube track, instrument tubing, and non-ASME piping and components, ***racks, panels, cabinets, and enclosures for electrical equipment and instrumentation, miscellaneous mechanical equipment and miscellaneous structures, etc.*** Applicable aging effects are identified and the aging management review is presented for each applicable combination of support component and aging effect.

System Interfaces

Physical interfaces exist with the structure, system, or component being supported and with the building structural element to which the support is anchored. A primary function of supports is to provide anchorage of the supported element for internal and external design basis events, so that the supported element can perform its intended function.

III STRUCTURES AND COMPONENT SUPPORTS
B2 Supports for Cable Trays, Conduit, HVAC Ducts, TubeTrack, Instrument Tubing, Non-ASME Piping and Components

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
III.B2-1 (T-29)	III.B2.2-a; ***III.B3.2-a; III.B4.3-a; III.B5.2-a***	Building concrete at locations of expansion and grouted anchors; grout pads for support base plates	Reinforced concrete; grout	Air – indoor uncontrolled or air - outdoor	Reduction in concrete anchor capacity due to local concrete degradation/ service-induced cracking or other concrete aging mechanisms	Chapter XI.S6, "Structures Monitoring Program"	No, if within the scope of the applicant's structures monitoring program	
III.B2-2 (TP-2)	III.B2; ***III.B4.***	Sliding support bearings and sliding support surfaces	Lubrite	Air – outdoor	Loss of mechanical function/ corrosion, distortion, dirt, overload, fatigue due to vibratory and cyclic thermal loads; elastomer hardening	Chapter XI.S6, "Structures Monitoring Program"	No	

III STRUCTURES AND COMPONENT SUPPORTS
B2 Supports for Cable Trays, Conduit, HVAC Ducts, TubeTrack, Instrument Tubing, Non-ASME Piping and Components

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
~~III.B2-3~~ ~~(TP-1)~~	~~III.B2;~~ ~~*III.B4.*~~	~~Steel components:~~ ~~Radial beam seats in BWR drywell; RPV support shoes for PWR with nozzle supports; other supports~~	~~Lubrite~~	~~Air – indoor uncontrolled~~	~~Loss of mechanical function/ corrosion, distortion, dirt, overload, fatigue due to vibratory and cyclic thermal loads; elastomer hardening~~	~~Chapter XI.S6, "Structures Monitoring Program"~~	~~No~~	Industry experience and EPRI Civil Tools indicates this aging effect is not applicable to graphite plate (lubrite). This is consistent with past approved applications. See NUREG-1759, Turkey Point SER, NUREG-1769, Peach Bottom SER, NUREG-1785, H.B. Robinson SER NUREG-1766, North Anna/Surry SER
III.B2-4 (TP-8)	III.B2; ***III.B3; III.B4; III.B5.***	Support members; welds; bolted connections; support anchorage to building structure	Galvanized steel, aluminum	Air – indoor uncontrolled	~~Loss of material/ galvanic corrosion~~ ***None***	~~Chapter XI.S6, "Structures Monitoring Program"~~ ***None***	No	Galvanic corrosion requires an electrolyte present between differential noble metals in contact to occur. Air – indoor uncontrolled environment is defined as normally dry in GALL chapter IX. Structural support members are generally within the less noble galvanic series and are not typically in contact with high noble metals,

III STRUCTURES AND COMPONENT SUPPORTS
B2 Supports for Cable Trays, Conduit, HVAC Ducts, TubeTrack, Instrument Tubing, Non-ASME Piping and Components

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
								sacrificial shims are typically installed between different metals. Per NUREG basis document for GALL it states "Difficult to justify loss of materials in this environment. All issued SERs have accepted "no aging effects." Therefore aging effect and AMP should be changed to "None".
III.B2-5 (TP-3)	III.B2; ***III.B3; III.B4; III.B5.***	Support members; welds; bolted connections; support anchorage to building structure	Galvanized steel, aluminum	Air with borated water leakage	Loss of material/ boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No	

III	STRUCTURES AND COMPONENT SUPPORTS							
B2	Supports for Cable Trays, Conduit, HVAC Ducts, TubeTrack, Instrument Tubing, Non-ASME Piping and Components							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
III.B2-6 (TP-6)	III.B2; ***III.B4***.	Support members; welds; bolted connections; support anchorage to building structure	~~Galvanized steel,~~ aluminum, stainless steel	Air – outdoor	Loss of material/ ~~pitting and crevice~~ corrosion	Chapter XI.S6, "Structures Monitoring Program"	No	Past precedence – Robinson SER (NUREG 1785). See new item III.B2-11
III.B2-7 (TP-5)	III.B2; ***III.B3; III.B4; III.B5.***	Support members; welds; bolted connections; support anchorage to building structure	***Galvanized steel,*** Stainless steel	Air – indoor uncontrolled	None	None	No	Past precedence – Robinson SER (NUREG 1785). Also, see new item III.B2-11
III.B2-8 (TP-4)	III.B2; ***III.B3; III.B4; III.B5.***	Support members; welds; bolted connections; support anchorage to building structure	Stainless steel	Air with borated water leakage	None	None	No	

III STRUCTURES AND COMPONENT SUPPORTS
B2 Supports for Cable Trays, Conduit, HVAC Ducts, TubeTrack, Instrument Tubing, Non-ASME Piping and Components

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
III.B2-9 (T-30)	III.B2.1-a; ***III.B3.1-a; III.B4.1-a; III.B5.1-a***	Support members; welds; bolted connections; support anchorage to building structure	Steel	Air – indoor uncontrolled (External), ***or air-outdoor***	Loss of material/ general and pitting corrosion	Chapter XI.S6, "Structures Monitoring Program"	No, if within the scope of the applicant's structures monitoring program	Add Air- outdoor to account for components located outdoors such as platforms, instrument racks, and electrical enclosures.
III.B2-10 (T-25)	III.B2.1-b; ***III.B3.1-b; III.B4.1-b; III.B5.1-b***	Support members; welds; bolted connections; support anchorage to building structure	Steel	Air with borated water leakage	Loss of material/ boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No	
~~III.B4-11~~ ***II.B2-11*** (T-31)	III.B4.2-a	Vibration isolation elements	Non-metallic (e.g., Rubber)	Air – indoor uncontrolled or air - outdoor	Reduction or loss of isolation function/ radiation hardening, temperature, humidity, sustained vibratory loading	Chapter XI.S6, "Structures Monitoring Program"	No, if within the scope of the applicant's structures monitoring program	

III B2	STRUCTURES AND COMPONENT SUPPORTS Supports for Cable Trays, Conduit, HVAC Ducts, TubeTrack, Instrument Tubing, Non-ASME Piping and Components							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
New III.B2(1)		***Support members; welds; bolted connections; support anchorage to building structure***	***Galvanized steel***	***Air – outdoor***	***Loss of material/ general corrosion***	***Chapter XI.S6, "Structures Monitoring Program"***	***No***	Past precedence – Robinson SER (NUREG 1785).
New III.B2(2)		***Any***	***Wood***	***Any***	***Loss of Material, Change in material properties***	***Chapter XI.S6, Structures Monitoring Program, Chapter XI.S7 RG 1.127, Inspection of Water-Controlled Structures, or Plant Specific AMP***	***No***	This material is not in GALL but used for component supports & water control structures, small dams and ponds, and power poles

Chapter IV Reactor Vessel, Internals, and Reactor Coolant System

General Comments

1. New Line Items

Proposed new lines items are listed at the end of each table. The new lines are designated "New IV.X(Y) where X is the table identifier and Y is a sequential number for new lines in that table. Where the same line is proposed in multiple tables, the other tables are listed below the designation.

2. Stainless Steel in Treated Borated Water

The GALL does not currently address the aging effect of loss of material for stainless steel in a treated borated water environment. Some sections of GALL note that the effect is minor and specifically not mentioned in GALL. For example, the introductory text to table IV.C2 says:

> The effects of pitting and crevice corrosion on stainless steel components are not significant in treated borated water and, therefore, are not included in this section.

Operating experience has shown loss of material to be a negligible effect because the water chemistry requirements minimize contaminants that would lead to loss of material. In other words, the water chemistry programs for PWRs manage the aging effect of loss of material. Although the effect is minor, past applications have listed (and future applications will list) loss of material as an aging effect for stainless steel in a treated borated water environment with water chemistry as the aging management program. Because the aging effect is minor, the water chemistry program by itself is sufficient to manage this aging effect, as evidenced by past operating experience. This was acknowledged in previous SERs including the following excerpt from Farley SER Section 3.1.2.3.1.2,

> Pitting corrosion and crevice corrosion may occur in ASME Code, Class 1, stainless steel or NiCrFe components under exposure to aggressive, oxidizing environments. Normally, the presence of elevated dissolved oxygen and/or aggressive ionic impurity concentrations is necessary to create these oxidizing environments in the RCS. The applicant's response to RAI 3.1.3.1.1-1, Part b, provides an acceptable explanation for citing the Water Chemistry Control Program as a basis for minimizing the dissolved oxygen and ionic impurity concentrations that could otherwise, if left present in high concentrations, lead to an aggressive, oxidizing RCS coolant environment. The GALL Report does not indicate that the loss of material due to pitting corrosion or crevice corrosion is an aging effect of concern for stainless steel or NiCrFe ASME Code Class 1 components. Since the applicant has conservatively assumed that the loss of material due to pitting corrosion or crevice corrosion is an applicable aging effect for these RV components, the staff concludes that the Water Chemistry Control Program provides a sufficient mitigative strategy for managing this aging effect relative to the recommendations of the GALL Report.

New lines are proposed for systems where this MEAP is appropriate. The introductory text indicating the effect is not addressed is deleted.

3. **Water Chemistry Reference**

The reference to the specific EPRI document need not be included in the Aging Management Program column. This information is identified in the AMP description in Chapter XI of GALL.

4. **Alloy 600 to Nickel Alloys**

Aging issues commonly associated with Alloy 600 also affect other nickel alloys such as Alloy 690/52/152. Where appropriate, the term "Nickel Alloys" should replace Alloy 600 to include the broader range of materials (this term is defined in GALL Section IX).

5. **Add Nickel Alloys**

Materials for certain items may include nickel alloy in addition to stainless steel; e.g., PWR RVI baffle former bolts. Similarly, some items currently listed as nickel alloy can also be steel clad with nickel alloy, e.g., SG primary side components. The added materials are subject to the same aging effects as those currently listed.

6. **Bolting**

Consistent with Chapter IX comment, high strength low alloy steel should be 'Low alloy steel.' Also, bolt strength does not impact the fatigue or wear aging mechanisms.

Although some utilities have conservatively applied loss of preload as aging effect for bolting, the industry does not consider loss of preload as an aging effect requiring management. In accordance with EPRI 1003056, "Mechanical Tool," Appendix F, loss of preload is a design driven effect and not an aging effect requiring management. The bolting at most facilities is standard grade B7 carbon steel, or similar material, except in rare specialized applications. Loss of preload due to stress relaxation (creep) for this material can only be a concern in very high temperature applications (> 700°F) as stated in the ASME Code Section II Part D Table 4 Note 4. However, there is no bolting used in BWRs and PWRs that operate at 700°F, with the exception of unique applications, such as the emergency diesel generator exhaust. Therefore, loss of preload due to stress relaxation (creep) is not a valid aging effect.

In addition, the industry has taken actions to address NUREG –1339, "Resolution to Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants." Licensees have implemented good bolting practices in accordance with those referenced in EPRI NP-5769, EPRI NP-5067 and EPRI TR-104213 in normal maintenance and design activities. Normal maintenance and design activities thus address the potential for loss of preload such that it is not a concern for the current or extended operating term. Proper joint preparation and make-up in accordance with industry standards precludes loss of preload. Even other design factors that could contribute to a loss of preload in closure bolting applications, such as vibration, should not result in loosening in a properly designed and assembled bolted joint.

The impact to the GALL tables is that, with elimination of loss of preload as an aging effect, closure bolting has the same MEAP as external surfaces and the lines could be combined.

7. **Integration of CASS with Stainless Steel**

Cast austenitic stainless steel (CASS) is currently treated as a separate material in GALL. However, with the exception of the loss of fracture toughness due to thermal and neutron irradiation embrittlement, CASS and stainless steel share the same aging effects/mechanisms in GALL.

To simplify GALL, CASS should be treated as a subset of stainless steel. CASS would only be listed as a material when loss of fracture toughness due to thermal (or thermal and irradiation) embrittlement is at issue, or where unique AMP requirements are given. This would provide consistency with GALL's treatment of other material groups, e.g., gray cast iron as a subset of steel, and copper alloy >15% zinc as a subset of copper alloy. Gray cast iron and copper alloy >15% zinc are both susceptible to selective leaching and are only listed as materials when selective leaching is addressed.

This change will have the added benefit of eliminating the need for new MEAP combinations to address CASS in non-Class 1 systems where stainless steel is adequately evaluated but CASS, if it is to be considered a separate material, is not. CASS is currently listed in only a few lines in Chapters V and VII and not at all in Chapter VIII.

8. Composite Component Descriptions

The combination of some lines to produce generic lines resulted in structure/component descriptions that included all the components previously (GALL 2001) listed in the individual lines. These comprehensive lists include components that do not apply to all system/structure tables. For example, in table IV.A1 for BWR reactor vessels, line R-04 addresses fatigue for "Piping, piping components, and piping elements; flanges; heater sheaths and sleeves; penetrations; pressure housings; pump casing/cover; spray head; thermal sleeves; vessel shell heads and welds." Some of these components are not applicable to BWR reactor vessels and may lead to confusion when comparing plant AMR results to this line of GALL.

The September 2004 version of GALL used generic descriptions, such as "Piping, piping components, and piping elements." The reason for the switch from the generic name used in the September version to the compound name in the January version is unclear. The compound names will create confusion because specific component types are listed in systems that don't contain those components.

if it is important to maintain the list of component types, component names in the rolled up lines should be split to their original configuration so that the list of component types matches the system. Alternatively, the component names should be restored to the generic names used in September. As a general rule, a generic line (rolled up line) should not be created unless the component description can be simplified to be applicable to all the systems that use it.

9. Commitments

The Aging Management Program entries for many lines have been changed. In many lines, AMPs described in GALL Chapter XI were replaced by commitments to be specified in the FSAR supplement. These commitments are less well defined than the programs they replaced. Four different commitment statements were used in the GALL lines. They are discussed further below, identified by key phrases in each.

<u>"provide a commitment . . . to implement applicable. . . (2) staff-accepted industry guidelines."</u>

Most of the lines that now list this commitment previously listed a GALL described AMP such as XI.M11, "Ni-alloy Nozzles and Penetrations or ISI." The Bases Document does not explain why these programs were replaced by an open-ended commitment. The commitment itself is unclear.

<u>"submit a plant-specific AMP delineating commitments to Orders, Bulletins, or Generic Letters that inspect stipulated components for cracking of wetted surfaces"</u>

Clarification is requested regarding the meaning of "submit a plant-specific AMP delineating commitments to Orders, Bulletins, or Generic Letters that inspect stipulated components for cracking of wetted surfaces." Is a plant-specific program required only if there are applicable Generic Communications?

<u>upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan</u>

This commitment statement appears in a large number of lines associated with PWR internals. The AMP XI.M16, "PWR Vessel Internals" was formerly used for these lines. This GALL described program was credited in prior approved applications. The reason for the change is unclear and the proposed commitment is also unclear. If the industry develops a staff accepted program, then submittal of the program would seem unnecessary.

What is the intent of submittal 24 months prior to the POEO? Would submittal 24 months before any proposed inspection be an acceptable reword?

<u>"submit, for NRC review and approval, an inspection plan for tube support lattice bars"</u>

This commitment occurs in only one line. The line originally referred to a plant specific program.

Highlighting

Yellow Highlight emphasizes portions of text that are neither added nor deleted but that still relate to a comment or question.

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.A1-1 (R-68)	IV.A1.4-a	Nozzle safe ends (and associated welds) High pressure core spray Low pressure core spray Control rod drive return line Recirculating water Low pressure coolant injection or RHR injection mode	Stainless steel, nickel alloy	Reactor coolant	Cracking/ stress corrosion cracking and intergranular stress corrosion cracking	Chapter XI.M7, "BWR Stress Corrosion Cracking," and Chapter XI.M2, "Water Chemistry," for BWR water ~~in BWRVIP-29 (EPRI TR-103515)~~	No	See General Comment 3
IV.A1-5 (R-69)	IV.A1.5-a	Penetrations Control rod drive stub tubes Instrumentation Jet pump instrument Standby liquid control Flux monitor Drain line	Stainless steel, nickel alloy	Reactor coolant	Cracking/ stress corrosion cracking, intergranular stress corrosion cracking, cyclic loading	Chapter XI.M8, "BWR Penetrations," and Chapter XI.M2, "Water Chemistry," for BWR water ~~in BWRVIP-29 (EPRI TR-103515)~~	No	See General Comment 3

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.A1-6 (R-04)	IV.A1.2-b IV.A1.4-b IV.A1.1-b IV.A1.2-a IV.A1.3-d IV.A1.6-a IV.A1.5-b IV.A1.3-a	~~Piping, piping components, and piping elements;~~ ~~f~~Flanges; ***nozzles;*** ~~heater sheaths and sleeves~~; penetrations; ~~pressure housings; pump casing/cover; spray head~~; ~~thermal sleeves~~; ***safe ends;*** vessel shell**s,** heads and welds Or Piping, piping components, and piping elements~~;~~ ~~flanges; heater sheaths and sleeves; penetrations; pressure housings; pump casing/cover; spray head; thermal sleeves; vessel shell heads and welds~~ (with GIX	Steel, stainless steel, ~~cast austenitic stainless steel, carbon~~ steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage/ fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA	See General Comment 8 See General Comment 7

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
		definition updated to include pentetrations, pressure housings, vessel shells heads and welds.)						
IV.A1-8 (R-60)	IV.A1.1-c	Top head enclosure Closure studs and nuts	~~High-strength~~ ~~l~~Low alloy steel ~~Maximum tensile strength < 1172 MPa (<170 Ksi)~~	Air with reactor coolant leakage	Cracking/ stress corrosion cracking and intergranular stress corrosion cracking	Chapter XI.M3, "Reactor Head Closure Studs"	No	See related comment to GIX (high strength low alloy steel definition). Comment also applies to the following GIV items: IV.A2-1; IV.C1-12; IV.C2-8, IV.C2-9. See also related comment for Item IV.A2-2 below.
IV.A1-9 (R-61)	IV.A1.1-d	Top head enclosure Vessel flange leak detection line	Stainless steel, nickel alloy	Air with reactor coolant leakage ***(Internal)*** Or ~~Air with~~ ~~r~~Reactor ~~c~~Coolant ~~leakage~~	Cracking/ stress corrosion cracking and intergranular stress corrosion cracking	A plant-specific aging management program is to be evaluated because existing programs may not be capable of mitigating or detecting crack initiation and growth due to SCC in the vessel flange leak detection line.	Yes, plant-specific	GIX defines this environment as leakage onto a surface, which implies an external surface, and germane to BWRs. Of the 13 GIV items that list this environment, 10 items are for effects on bolting. This item addresses the collection of reactor coolant leakage inside the component (e.g, extending pressure boundary). As such, clarification of the environment is warranted. Comment is also applicable to the following GIV items: IV.A2-4, IV.A2-13.

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.A1-10 (R-59)	IV.A1.1-a	Top head enclosure (without cladding) Top head Nozzles (vent, top head spray or RCIC, and spare)	Steel	Reactor coolant	Loss of material/ general, pitting, and crevice corrosion	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry," for BWR water ~~in BWRVIP-29 (EPRI TR-103515)~~	No	See General Comment 3
IV.A1-11 (R-64)	A1.2-e	Vessel shell Attachment welds	Stainless steel, nickel alloy	Reactor coolant	Cracking/ stress corrosion cracking and intergranular stress corrosion cracking	Chapter XI.M4, "BWR Vessel ID Attachment Welds," and Chapter XI.M2, "Water Chemistry," for BWR water ~~in BWRVIP-29 (EPRI TR-103515)~~	No	See General Comment 3
New IV.A1(1)	***N/A***	***Flanges, nozzles; penetrations; pressure housings; safe ends; vessel shells, heads and welds***	***Stainless steel; steel with nickel-alloy or stainless steel cladding; nickel-alloy***	***Reactor Coolant***	***Loss of material /pitting and crevice corrosion***	***Chapter XI.M2, "Water chemistry," for BWR water***	***No***	Based on New IV.A2(1), this line should also be included for consistency.

IV.A2 Reactor Vessel (Pressurized Water Reactor)

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.A2-1 (R-71)	IV.A2.1-c	Closure head Stud assembly	~~High-strength /~~Low alloy steel ~~Maximum tensile strength < 1172 MPa (<170 Ksi)~~	Air with reactor coolant leakage	Cracking/ stress corrosion cracking	Chapter XI.M3, "Reactor Head Closure Studs"	No	See related comment to Item IV.A1-8 above (high strength low alloy steel clarification).
IV.A2-2 (R-73)	IV.A2.1-e	Closure head Stud assembly	~~High-strength /~~Low alloy steel ~~Maximum tensile strength < 1172 MPa (<170 Ksi)~~	Air with reactor coolant leakage	Cumulative fatigue damage/ fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," *for acceptable methods for* meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA	The 170 ksi tensile strength limitation is understandable for SCC considerations, but what is the issue relative to fatigue? This limitation is not applicable for this item and should be *removed from the material* description. Comment is also applicable to the following GIV items: IV.A2-3, IV.C1-12, and IV.C2-9. See also related comment for Chapter IX (high strength low *alloy steel definition*).
IV.A2-3	IV.A2.1-d	Closure head Stud assembly	~~High-strength /~~Low alloy steel ~~Maximum tensile strength < 1172 MPa (<170 Ksi)~~	Air with reactor coolant leakage	Loss of material/ wear	Chapter XI.M3, "Reactor Head Closure Studs"	No	See related comment for Item IV.A2-2 above (high strength low alloy steel clarification).

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.A2-4 (R-074)	IV.A2.1-f	Closure head Vessel flange leak detection line	Stainless steel	Air with reactor coolant leakage ***(Internal)*** Or ~~Air with~~ ~~r~~*R*eactor ~~c~~*C*oolant ~~leakage~~	Cracking/ stress corrosion cracking	A plant-specific aging management program is to be evaluated because existing programs may not be capable of mitigating or detecting crack initiation and growth due to SCC in the vessel flange leak detection line.	Yes, plant-specific	See related comment for Item IV.A1-9 above (Environment clarification).
IV.A2-8 (R-75)	IV.A2.2-a	Control rod drive head penetration Nozzle and welds	Nickel alloy	Reactor coolant	Cracking/ primary water stress corrosion cracking	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry," for PWR primary water ~~in EPRI TR-105714~~ and, for ~~Alloy 600~~***Nickel Alloys***, provide a commitment in the FSAR supplement to implement applicable (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff-accepted industry guidelines.	No, but licensee commitments to be confirmed	See General Comment 3 See General Comment 4 See General Comment 9

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.A2-10	IV.A2.2-b	Control rod drive head penetration Pressure housing	Stainless steel, ~~cast austenitic stainless steel,~~ nickel alloy	Reactor coolant	Cracking/ stress corrosion cracking, primary water stress corrosion cracking	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry," for PWR primary water ~~in EPRI TR-105714~~ and, for ***Nickel Alloys*** ~~Alloy 600~~, provide a commitment in the FSAR supplement to implement applicable (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff-accepted industry guidelines.	No, but licensee commitments to be confirmed	See General Comment 7 See General Comment 3 See General Comment 4 See General Comment 9
IV.A2-11 (R-88)	IV.A2.6-a	Core support pads/core guide lugs	Nickel alloy	Reactor coolant	Cracking/ primary water stress *corrosion* cracking	Chapter XI.M2, "Water Chemistry," and Chapter XI.M32 "One-Time Inspection" or Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," and provide a commitment in the FSAR supplement to submit a plant-specific AMP delineating commitments to Orders, Bulletins, or Generic Letters that inspect stipulated components for cracking of wetted surfaces.	No, unless licensee commitments need to be confirmed	AMP logic is unclear (punctuation and spacing around "and" and "or" is vague) .Is the intent that 'Water Chemistry and One-Time Inspection' is an acceptable option while 'ISI and FSAR commitment' is another acceptable option; or is ISI and FSAR commitment an alternative to OTI, or is the intent something else? The feasibility of a one-time inspection of the core support pads/core guide lugs is also questionable. See General Comment 9

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.A2-13 (RP-13)	IV.A2	Instrument penetration Bottom-mounted guide tube	Stainless steel	Air with reactor coolant leakage ***(Internal)*** Or ~~Air with~~ ~~r~~Reactor ~~c~~Coolant ~~leakage~~	Cracking/ stress corrosion cracking	A plant-specific aging management program is to be evaluated.	Yes, plant-specific	See related comment for Item IV.A1-9 above (Environment clarification). Section 3.1.2.4.5 of the VCSNS SER more correctly describes the environment for this item (Basis document lists an incorrect section of the RNP SER for the change from Rev. 0 to this item).
IV.A2-14	IV.A2.4-b	Nozzle safe ends and welds Inlet Outlet Safety injection	Stainless steel~~, cast austenitic stainless steel~~ (nickel alloy welds and/or buttering)	Reactor coolant	Cracking/ stress corrosion cracking, primary water stress corrosion cracking	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry," for PWR primary water ~~in EPRI TR-105714~~	No	See General Comment 7 See General Comment 3

IV A2	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Vessel (Pressurized Water Reactor)							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
IV.A2-17	IV.A2.7-b	Penetrations Head vent pipe (top head) Instrument tubes (top head)	Nickel alloy	Reactor coolant	Cracking/ primary water stress corrosion cracking	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry," for PWR primary water ~~in EPRI TR-105714~~ and, for ***Nickel Alloys***~~Alloy 600~~, provide a commitment in the FSAR supplement to implement applicable (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff-accepted industry guidelines.	No, but licensee commitments to be confirmed	See General Comment 3 See General Comment 4 See General Comment 9
IV.A2-18 (R-89)	IV.A2.7-a	Penetrations Instrument tubes (bottom head)	Nickel alloy	Reactor coolant	Cracking/ primary water stress corrosion cracking	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry," for PWR primary water ~~in EPRI TR-105714~~ and, for ~~Alloy 600~~***Nickel Alloys***, provide a commitment in the FSAR supplement to implement applicable (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff-accepted industry guidelines.	No, but licensee commitments to be confirmed	See General Comment 3 See General Comment 4 See General Comment 9

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.A2-19 (R-04)	IV.A2.5-d IV.A2.4-a IV.A2.1-b IV.A2.3-c IV.A2.2-c	~~Piping, piping components, and piping elements;~~ ~~f~~Flanges; ~~heater sheaths and sleeves;~~ ***nozzles;*** penetrations; pressure housings; ***safe ends;*** ~~pump casing/cover; spray head; thermal sleeves;~~ vessel shell***s***, heads and welds	Steel, stainless steel, ~~cast austenitic stainless steel, carbon~~ steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage/ fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA	See General Comment 8 See General Comment 7
New IV.A2(1)	***N/A***	***Flanges, nozzles; penetrations; pressure housings; safe ends; vessel shells, heads and welds***	***Stainless steel; steel with nickel-alloy or stainless steel cladding; nickel-alloy***	***Reactor Coolant***	***Loss of material /pitting and crevice corrosion***	***Chapter XI.M2, "Water chemistry," for PWR primary water***	***No***	See General Comment 2

IV.B1 Reactor Vessel Internals (Boiling Water Reactor)

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.B1-1 (R-92)	IV.B1.1-a	Core shroud (including repairs) and core plate Core shroud (upper, central, lower)	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, intergranular stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M9, "BWR Vessel Internals," for core shroud and Chapter XI.M2, "Water Chemistry" for BWR water ~~in BWRVIP-29 (EPRI TR-103515)~~	No	See General Comment 3
IV.B1-2 (R-96)	IV.B1.1-f	Core shroud (including repairs) and core plate Shroud support structure (shroud support cylinder, shroud support plate, shroud support legs)	Nickel alloy	Reactor coolant	Cracking/ stress corrosion cracking, intergranular stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M9, "BWR Vessel Internals," for core shroud and Chapter XI.M2, "Water Chemistry" for BWR water ~~in BWRVIP-29 (EPRI TR-103515)~~	No	See General Comment 3
IV.B1-3 (R-97)	IV.B1.1-g	Core shroud and core plate LPCI coupling	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, intergranular stress corrosion cracking, irradiation-assisted stress corrosion	Chapter XI.M9, "BWR Vessel Internals," for core shroud and Chapter XI.M2, "Water Chemistry" for BWR water ~~in BWRVIP-29 (EPRI TR-103515)~~	No	See General Comment 3

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
					cracking			
IV.B1-4 (R-95)	IV.B1.1-e	Core shroud and core plate Access hole cover (mechanical covers)	Nickel alloy	Reactor coolant	Cracking/ stress corrosion cracking, intergranular stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry," for BWR water ~~in BWRVIP-29 (EPRI TR-103515)~~	No	See General Comment 3
IV.B1-5 (R-94)	IV.B1.1-d	Core shroud and core plate Access hole cover (welded covers)	Nickel alloy	Reactor coolant	Cracking/ stress corrosion cracking, intergranular stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry," for BWR water ~~in BWRVIP-29 (EPRI TR-103515)~~ Because cracking initiated in crevice regions is not amenable to visual inspection, for BWRs with a crevice in the access hole covers, an augmented inspection is to include ultrasonic testing (UT) or other demonstrated acceptable inspection of the access hole cover welds.	No	See General Comment 3

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.B1-6 (R-93)	IV.B1.1-b	Core shroud and core plate Core plate Core plate bolts (used in early BWRs)	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, intergranular stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M9, "BWR Vessel Internals," for core shroud and Chapter XI.M2, "Water Chemistry" for BWR water ~~in BWRVIP-29 (EPRI TR-103515)~~	No	See General Comment 3
IV.B1-7 (R-99)	IV.B1.3-a	Core spray lines and spargers Core spray lines (headers) Spray rings Spray nozzles Thermal sleeves	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, intergranular stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M9, "BWR Vessel Internals," for core shroud and Chapter XI.M2, "Water Chemistry" for BWR water ~~in BWRVIP-29 (EPRI TR-103515)~~	No	See General Comment 3
IV.B1-8 (R-104)	IV.B1.5-c	Fuel supports and control rod drive assemblies Control rod drive housing	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking and intergranular stress corrosion cracking	Chapter XI.M9, "BWR Vessel Internals," for core shroud and Chapter XI.M2, "Water Chemistry" for BWR water ~~in BWRVIP-29 (EPRI TR-103515)~~	No	See General Comment 3
IV.B1-10 (R-105)	IV.B1.6-a	Instrumentation Intermediate range monitor (IRM) dry tubes Source range monitor	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, intergranular stress corrosion cracking, irradiation-	Chapter XI.M9, "BWR Vessel Internals," for core shroud and Chapter XI.M2, "Water Chemistry" for BWR water ~~in BWRVIP-29 (EPRI TR-~~	No	See General Comment 3

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
		(SRM) dry tubes Incore neutron flux monitor guide tubes			assisted stress corrosion cracking	~~103515)~~		
IV.B1-13 (R-100)	IV.B1.4-a	Jet pump assemblies Thermal sleeve Inlet header Riser brace arm Holddown beams Inlet elbow Mixing assembly Diffuser Castings	Nickel alloy, ~~cast austenitic stainless steel,~~ stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, intergranular stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M9, "BWR Vessel Internals," for core shroud and Chapter XI.M2, "Water Chemistry" for BWR water ~~in BWRVIP-29 (EPRI TR-103515)~~	No	See General Comment 7 See General Comment 3
IV.B1-14 (R-53)	B1.1-c B1.2-b B1.3-b B1.4-b B1.5-b B1.6-b	Reactor vessel internals components	Stainless steel~~, cast austenitic stainless steel~~, nickel alloy	Reactor coolant	Cumulative fatigue damage/ fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA	See General Comment 7

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.B1-16 (R-98)	IV.B1.2-a	Top guide	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, intergranular stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M9, "BWR Vessel Internals," for core shroud and Chapter XI.M2, "Water Chemistry" for BWR water ~~in BWRVIP-29 (EPRI TR-103515)~~	No	See General Comment 3
New IV.B1(1)	***N/A***	***Reactor vessel internals components***	***Stainless steel, nickel alloy***	***Reactor coolant***	***Loss of material/pitting and crevice corrosion***	***Chapter XI.M2, "Water chemistry," for BWR water***	***No***	Based on New IV.A2(1), this line should also be included for consistency.

IV.B2 Reactor Vessel Internals (PWR) - Westinghouse

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.B2-1 (R-124)	IV.B2.4-b	Baffle/former assembly Baffle and former plates	Stainless steel	Reactor coolant	Changes in dimensions/Void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed	See General Comment 9
IV.B2-2 (R-123)	IV.B2.4-a	Baffle/former assembly Baffle and former plates	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water chemistry" for PWR primary water~~, as described in EPRI TR-105714~~. No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating	No, but licensee commitment to be confirmed.	See General Comment 3 See General Comment 9

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.		
IV.B2-3 (R-127)	IV.B2.4-e	Baffle/former assembly Baffle and former plates	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/ neutron irradiation embrittlement, void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 9

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.B2-4 (R-126)	IV.B2.4-d	Baffle/former assembly Baffle/former bolts	Stainless steel, **nickel alloy**	Reactor coolant	Changes in dimensions/Void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed	See General Comment 5 See General Comment 9
IV.B2-5 (R-128)	IV.B2.4-f	Baffle/former assembly Baffle/former bolts	Stainless steel; ***nickel alloy***	Reactor coolant and neutron flux	Loss of fracture toughness/ neutron irradiation embrittlement	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before	No, but licensee commitment to be confirmed	See General Comment 5 See General Comment 9

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B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.		
IV.B2-6 (R-129)	IV.B2.4-h	Baffle/former assembly Baffle/former bolts	Stainless steel, nickel alloy	Reactor Coolant	Loss of preload/ stress relaxation	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 9

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.B2-7 (R-121)	IV.B2.3-b	Core barrel Core barrel (CB) CB flange (upper) CB outlet nozzles Thermal shield	Stainless steel	Reactor coolant	Changes in dimensions/Void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed	See General Comment 9

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.B2-8 (R-120)	IV.B2.3-a	Core barrel Core barrel (CB) CB flange (upper) CB outlet nozzles Thermal shield	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water chemistry" for PWR primary water~~, as described in EPRI TR-105714~~. No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 3 See General Comment 9

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B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.B2-9 (R-122)	IV.B2.3-c	Core barrel Core barrel (CB) CB flange (upper) CB outlet nozzles Thermal shield	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/ neutron irradiation embrittlement, void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed	See General Comment 9
IV.B2-10 (R-125)	IV.B2.4-c	***Baffle/former assembly***~~Core barrel assembly~~ ~~Baffle/former assembly~~ Baffle/former bolts and screws	Stainless steel	Reactor coolant ~~and high fluence (>1 x 1021 n/cm2, E >0.1 MeV)~~	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	***Chapter XI.M2, "Water Chemistry," for PWR primary water and*** No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to	No, but licensee commitment to be confirmed	The GALL Link is for B/F Bolts, not Core Barrel Assy. Looks like a typo that should be returned to the original format. Environment should be "Reactor Coolant" and AMP should be 'Chemistry and FSAR commitment' for consistency with the twenty-five (25) other Items in Chapter IV for this EAP. Item R-125 is the only one that uses the specified environment and the only one for this Reactor Internals MEAP that does not list Chemistry and FSAR

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B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.		commitment as the AMP. In GALL Rev. 0, a plant-specific AMP was used. See General Comment 9
IV.B2-11 (R-144)	IV.B2.6-b	Instrumentation support structures Flux thimble guide tubes	Stainless steel	Reactor coolant	Changes in dimensions/Void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed	See General Comment 9

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.B2-12 (R-143)	IV.B2.6-a	Instrumentation support structures Flux thimble guide tubes	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water chemistry" for PWR primary water~~, as described in EPRI TR-105714~~. No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 3 See General Comment 9
IV.B2-13 (R-145)	IV.B2.6-c	Instrumentation support structures Flux thimble tubes	Stainless steel	Reactor coolant	Loss of material/ wear	***Chapter XI.MXX "Flux Thimble Tube Inspection Program"*** ~~Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and recommendations of NRC I&E Bulletin 88-09 "Thimble Tube Thinning in Westinghouse Reactors,"~~	No	Replace existing AMP discussion with reference to new AMP proposed for flux thimble tube inspections. Flux thimble tubes are not within the scope of ASME Section XI. Flux thimble tubes are less than 1" in diameter. Per sections IWB-1220(b), IWC-1222(a)(1), and IWD-1220(a), the flux thimble tubes are exempt from Section

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Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						~~described below: In response to I&E Bulletin 88-09, an inspection program, with technical justification, is to be established and is to include (a) an appropriate thimble tube wear acceptance criterion, e.g., percent through-wall loss, and includes allowances for inspection methodology and wear scar geometry uncertainty, (b) an appropriate inspection frequency, e.g., every refueling outage, and (c) inspection methodology such as eddy current technique that is capable of adequately detecting wear of the thimble tubes. In addition, corrective actions include isolation or replacement if a thimble tube fails to meet the above acceptance criteria. Inspection schedule is in accordance with the guidelines of I&E Bulletin 88-09.~~		XI examination requirements. Furthermore, IEB 88-09 stated: "There are currently no inservice inspection or testing requirements for thimble tubes." IEB 88-09 does not include any guidelines concerning an inspection schedule beyond the initial inspections which were performed fifteen years ago at each utility. What IEB 88-09 says about schedule is already addressed in part (b) of the AMP description.
IV.B2-14 (R-134)	IV.B2.5-f	Lower internal assembly Fuel alignment pins Lower support plate column	Stainless steel, nickel alloy	Reactor coolant	Changes in dimensions/Void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating	No, but licensee commitment to be confirmed	See General Comment 9

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
		bolts Clevis insert bolts				and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.		
IV.B2-15 B	IV.B2.5-f	Lower internal assembly Fuel alignment pins Lower support plate column bolts Clevis insert bolts	Stainless steel, nickel alloy	Reactor coolant	Changes in dimensions/Void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed	See General Comment 9

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B2	Reactor Vessel Internals (PWR) - Westinghouse							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
IV.B2-16 (R-133)	IV.B2.5-e	Lower internal assembly Fuel alignment pins Lower support plate column bolts Clevis insert bolts	Stainless steel, nickel alloy	Reactor coolant	Cracking/ stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water chemistry" for PWR primary water~~, as described in EPRI TR-105714~~. No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 3 See General Comment 9
IV.B2-17 (R-135)	IV.B2.5-g	Lower internal assembly Fuel alignment pins Lower support plate column bolts	Stainless steel, nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness/ neutron irradiation embrittlement, void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects	No, but licensee commitment to be confirmed	See General Comment 9

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Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
		Clevis insert bolts				on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.		
IV.B2-18 (R-132)	IV.B2.5-c	Lower internal assembly Lower core plate	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/ neutron irradiation embrittlement, void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed	See General Comment 9

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B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.B2-19 (R-131)	IV.B2.5-b	Lower internal assembly Lower core plate Radial keys and Clevis inserts	Stainless steel, ***nickel alloy***	Reactor coolant	Changes in dimensions/Void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed	See General Comment 5 See General Comment 9
IV.B2-20 (R-130)	IV.B2.5-a	Lower internal assembly Lower core plate Radial keys and clevis inserts	Stainless steel, ***nickel alloys***	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water chemistry" for PWR primary water~~, as described in EPRI TR-105714~~. No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects	No, but licensee commitment to be confirmed.	See General Comment 5 See General Comment 3 See General Comment 9

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Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.		
IV.B2-22 (R-141)	IV.B2.5-n	Lower grid assembly Lower support forging Lower support plate columns	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/ neutron irradiation embrittlement, void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed	See General Comment 9

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Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.B2-23 (R-139)	IV.B2.5-l	Lower internal assembly Lower support forging or casting Lower support plate columns	Stainless steel~~, cast austenitic stainless steel~~	Reactor coolant	Changes in dimensions/Void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed	See General Comment 7 See General Comment 9
IV.B2-24 (R-138)	IV.B2.5-k	Lower internal assembly Lower support forging or casting Lower support plate columns	Stainless steel~~, cast austenitic stainless steel~~	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water chemistry" for PWR primary water~~, as described in EPRI TR-105714~~. No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects	No, but licensee commitment to be confirmed.	See General Comment 7 See General Comment 3 See General Comment 9

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.		
IV.B2-25 (R-136)	IV.B2.5-h	Lower internal assembly Lower support plate column bolts	Stainless steel, nickel alloy	Reactor coolant	Loss of preload/ stress relaxation	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed	See General Comment 9

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Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.B2-27 (R-119)	IV.B2.2-e	RCCA guide tube assemblies RCCA guide tube bolts RCCA guide tube support pins	Stainless steel, nickel alloy	Reactor coolant	Changes in dimensions/Void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed	See General Comment 9
IV.B2-28 (R-118)	IV.B2.2-d	RCCA guide tube assemblies RCCA guide tube bolts RCCA guide tube support pins	Stainless steel, nickel alloy	Reactor coolant	Cracking/ stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water chemistry" for PWR primary water~~, as described in EPRI TR-105714~~. No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects	No, but licensee commitment to be confirmed.	See General Comment 3 See General Comment 9

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Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.		
IV.B2-29 (R-117)	IV.B2.2-b	RCCA guide tube assemblies RCCA guide tubes	Stainless steel	Reactor coolant	Changes in dimensions/Void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed	See General Comment 9

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Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.B2-30 (R-116)	IV.B2.2-a	RCCA guide tube assemblies RCCA guide tubes	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water chemistry" for PWR primary water~~, as described in EPRI TR-105714~~. No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 3 See General Comment 9
IV.B2-31 (R-53)	IV.B2.1-c IV.B2.1-h IV.B2.1-m IV.B2.2-c IV.B2.2-f IV.B2.3-d IV.B2.4-g IV.B2.5-d	Reactor vessel internals components	Stainless steel~~, cast austenitic stainless steel~~, nickel alloy	Reactor coolant	Cumulative fatigue damage/ fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements	Yes, TLAA	See General Comment 7

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Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
	IV.B2.5-j IV.B2.5-p					of 10 CFR 54.21(c)(1).		
IV.B2-32 (R-108)	IV.B2.1-d	Upper internals assembly Hold-down spring	Stainless steel	Reactor coolant	Loss of preload/ stress relaxation	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed	See General Comment 9
IV.B2-34 (R-110)	IV.B2.1-f	Upper internals assembly Upper support column	Stainless steel~~, cast austenitic stainless steel~~	Reactor coolant	Changes in dimensions/Void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and	No, but licensee commitment to be confirmed	See General Comment 7 See General Comment 9

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						(3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.		
IV.B2-35 (R-109)	IV.B2.1-e	Upper internals assembly Upper support column	Stainless steel~~, cast austenitic stainless steel~~	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water chemistry" for PWR primary water~~, as described in EPRI TR-105714~~. No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 7 See General Comment 3 See General Comment 9

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Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.B2-37 (R-114)	IV.B2.1-k	Upper internals assembly Upper support column bolts	Stainless steel, nickel alloy	Reactor coolant	Loss of preload/ stress relaxation	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed	See General Comment 9
IV.B2-38 (R-113)	IV.B2.1-j	Upper internals assembly Upper support column bolts Upper core plate alignment pins Fuel alignment pins	Stainless steel, nickel alloy	Reactor coolant	Changes in dimensions/Void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and	No, but licensee commitment to be confirmed	See General Comment 9

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						(3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.		
IV.B2-39 (R-112)	IV.B2.1-i	Upper internals assembly Upper support column bolts Upper core plate alignment pins Fuel alignment pins	Stainless steel, nickel alloy	Reactor coolant	Cracking/ stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water chemistry" for PWR primary water~~, as described in EPRI TR-105714~~. No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 3 See General Comment 9

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.B2-40	IV.B2.1-b	Upper internals assembly Upper support plate Upper core plate Hold-down spring	Stainless steel	Reactor coolant	Changes in dimensions/Void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed	See General Comment 9

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B2 Reactor Vessel Internals (PWR) - Westinghouse

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.B2-41 (R-106)	IV.B2.1-a	Upper internals assembly Upper support plate Upper core plate Hold-down spring	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water chemistry" for PWR primary water~~, as described in EPRI TR-105714~~. No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 3 See General Comment 9
New IV.B2(1) ***IV.B#*** ***IV.B4***	***N/A***	***Reactor vessel internals components***	***Stainless steel, nickel alloy***	***Reactor coolant***	***Loss of material/ pitting and crevice corrosion***	***Chapter XI.M2, "Water chemistry," for PWR primary water***	***No***	See General Comment 2

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

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B3 Reactor Vessel Internals (PWR) - Combustion Engineering

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.B3-2 (R-149)	IV.B3.2-a	CEA shroud assemblies	Stainless steel~~, cast austenitic stainless steel~~	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water chemistry" for PWR primary water~~, as described in EPRI TR-105714~~. No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 7 See General Comment 3 See General Comment 9
IV.B3-4 (R-151)	B3.2-c	CEA shroud assemblies CEA shrouds bolts	Stainless steel, ~~cast austenitic stainless steel,~~ nickel alloy	Reactor coolant	Changes in dimensions/Void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating	No, but licensee commitment to be confirmed.	See General Comment 7 See General Comment 9

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B3 Reactor Vessel Internals (PWR) - Combustion Engineering

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.		
IV.B3-5 (R-150)	IV.B3.2-b	CEA shroud assemblies CEA shrouds bolts	Stainless steel, nickel alloy	Reactor coolant	Cracking/ stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water chemistry" for PWR primary water~~, as described in EPRI TR-105714~~. No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit	No, but licensee commitment to be confirmed.	See General Comment 3 See General Comment 9

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B3 Reactor Vessel Internals (PWR) - Combustion Engineering

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						an inspection plan for reactor internals to the NRC for review and approval.		
IV.B3-6 (R-154)	IV.B3.2-g	CEA shroud assemblies CEA shrouds bolts	Stainless steel, nickel alloy	Reactor coolant	Loss of preload/ stress relaxation	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 9
IV.B3-7 (R-164)	IV.B2.5-n	Lower internal assembly Lower support forging Lower support plate columns	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/ neutron irradiation embrittlement, void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to	No, but licensee commitment to be confirmed.	See General Comment 9

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B3 Reactor Vessel Internals (PWR) - Combustion Engineering

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.		
IV.B3-8 (R-165)	IV.B3.4-h	Core shroud assembly Core shroud assembly bolts Core shroud tie rods	Stainless steel, nickel alloy	Reactor coolant	Loss of preload/ stress relaxation	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 9
IV.B3-9 (R-161)	IV.B3.3-a	Core support barrel Core support barrel upper flange	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/ neutron irradiation embrittlement, void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry	No, but licensee commitment to be confirmed.	See General Comment 9

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.		
IV.B3-10 (R-160)	IV.B3.4-b	Core shroud assembly Core shroud tie rods (core support plate attached by welds in later plants)	Stainless steel, ~~cast austenitic stainless steel,~~ nickel alloy	Reactor coolant	Changes in dimensions/Void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 7 See General Comment 9

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B3 Reactor Vessel Internals (PWR) - Combustion Engineering

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.B3-11 (R-163)	IV.B3.4-f	Core shroud assembly Core shroud assembly bolts (later plants are welded)	Stainless steel, nickel alloy	Reactor coolant	Changes in dimensions/Void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 9
IV.B3-12 (R-162)	IV.B3.4-e	Core shroud assembly Core shroud assembly bolts (later plants are welded)	Stainless steel, nickel alloy	Reactor coolant	Cracking/ stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water chemistry" for PWR primary water~~, as described in EPRI TR-105714~~. No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry	No, but licensee commitment to be confirmed.	See General Comment 3 See General Comment 9

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B3 Reactor Vessel Internals (PWR) - Combustion Engineering

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.		
IV.B3-13 (R-159)	IV.B3.4-a	Core shroud assembly Core shroud tie rods (core support plate attached by welds in later plants)	Stainless steel~~, cast austenitic stainless steel~~	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water chemistry" for PWR primary water~~, as described in EPRI TR-105714~~. No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 7 See General Comment 3 See General Comment 9

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.B3-14 (R-158)	IV.B3.3-b	Core support barrel Core support barrel upper flange	Stainless steel	Reactor coolant	Changes in dimensions/Void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 9
IV.B3-15 (R-155)	IV.B3.3-a	Core support barrel Core support barrel upper flange	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water chemistry" for PWR primary water~~, as described in EPRI TR-105714~~. No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry	No, but licensee commitment to be confirmed.	See General Comment 3 See General Comment 9

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B3 Reactor Vessel Internals (PWR) - Combustion Engineering

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.		
IV.B3-16 (R-157)	IV.B3.4-c	Core shroud assembly Core shroud tie rods (core support plate attached by welds in later plants)	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/ neutron irradiation embrittlement, void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 9

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B3 Reactor Vessel Internals (PWR) - Combustion Engineering

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.B3-19 (R-168)	IV.B3.5-c	Lower internal assembly Core support plate Fuel alignment pins Lower support structure beam assemblies Core support column bolts Core support barrel snubber assemblies	Stainless steel, ~~cast austenitic stainless steel,~~ nickel alloy	Reactor coolant	Changes in dimensions/Void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 7 See General Comment 9
IV.B3-20 (R-169)	IV.B3.5-d	Lower internal assembly Core support plate Fuel alignment pins Lower support structure beam assemblies Core support column bolts Core support barrel snubber assemblies	Stainless steel, nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness/ neutron irradiation embrittlement, void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before	No, but licensee commitment to be confirmed.	See General Comment 9

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B3 Reactor Vessel Internals (PWR) - Combustion Engineering

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.		
IV.B3-21 (R-166)	IV.B3.5-a	Lower internal assembly Core support plate Lower support structure beam assemblies Core support column Core support barrel snubber assemblies	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water chemistry" for PWR primary water~~, as described in EPRI TR-105714~~. No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 7 See General Comment 9

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B3 Reactor Vessel Internals (PWR) - Combustion Engineering

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.B3-23 (R-167)	IV.B3.5-b	Lower internal Assembly Fuel alignment pins Core support column bolts	Stainless steel, nickel alloy	Reactor coolant	Cracking/ stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water chemistry" for PWR primary water~~, as described in EPRI TR-105714~~. No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 7 See General Comment 9
IV.B3-24 (R-54)	IV.B3.2-f IV.B3.4-d IV.B3.5-g	Reactor vessel internals components	Stainless steel~~, cast austenitic stainless steel~~, nickel alloy	Reactor coolant	Cumulative fatigue damage/ fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA	See General Comment 7

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.B3-26 (R-147)	IV.B3.1-b	Upper Internals Assembly Upper guide structure support plate Fuel alignment plate Fuel alignment plate guide lugs and guide lug inserts	Stainless steel	Reactor coolant	Changes in dimensions/Void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 9
IV.B3-27 (R-146)	IV.B3.1-a	Upper Internals Assembly Upper guide structure support plate Fuel alignment plate Fuel alignment plate guide lugs and guide lug inserts	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water chemistry" for PWR primary water~~, as described in EPRI TR-105714~~. No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry	No, but licensee commitment to be confirmed.	See General Comment 3 See General Comment 9

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B3 Reactor Vessel Internals (PWR) - Combustion Engineering

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.		
New IV.B3(1) *IV.B2 IV.B4*	*N/A*	*Reactor vessel internals components*	*Stainless steel, nickel alloy*	*Reactor coolant*	*Loss of material/ pitting and crevice corrosion*	*Chapter XI.M2, "Water chemistry," for PWR primary water*	*No*	See General Comment 2

IV.B4 Reactor Vessel Internals (PWR) - Babcock and Wilcox

IV	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B4	Reactor Vessel Internals (PWR) - Babcock and Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.B4-1 (R-180)	IV.B4.3-a	Control rod guide tube (CRGT) assembly CRGT pipe and flange CRGT spacer casting CRGT rod guide tubes CRGT rod guide sectors	Stainless steel~~, cast austenitic stainless steel~~	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water chemistry" for PWR primary water~~, as described in EPRI TR-105714~~. No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 7 See General Comment 3 See General Comment 9
IV.B4-3 (R-181)	IV.B4.3-b	Control rod guide tube (CRGT) assembly CRGT spacer screws	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water chemistry" for PWR primary water~~, as described in EPRI TR-105714~~. No further aging management review is	No, but licensee commitment to be confirmed.	See General Comment 3 See General Comment 9

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
		Flange-to-upper grid screws				necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.		
IV.B4-4 (R-182)	IV.B4.3-c	Control rod guide tube (CRGT) assembly CRGT pipe and flange CRGT spacer casting CRGT spacer screws Flange-to-upper grid screws CRGT rod guide tubes CRGT rod guide sectors	Stainless steel~~, cast austenitic stainless steel~~	Reactor coolant	Changes in dimensions/Void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit	No, but licensee commitment to be confirmed.	See General Comment 7 See General Comment 9

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B4 Reactor Vessel Internals (PWR) - Babcock and Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						an inspection plan for reactor internals to the NRC for review and approval.		
IV.B4-5 (R-184)	IV.B4.3-e	Control rod guide tube (CRGT) assembly Flange-to-upper grid screws	Stainless steel	Reactor coolant	Loss of preload/ stress relaxation	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 9
IV.B4-6 (R-125)	IV.B4.5-g	Core barrel assembly Baffle/former bolts and screws	Stainless steel	Reactor Coolant ~~and high fluence (>1 x 1021 n/cm2, E >0.1 MeV)~~	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	***Chapter XI.M2, "Water Chemistry" for PWR primary water.*** No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects	No, but licensee commitment to be confirmed	See related comment for Item IV.B2-10 above (Environment clarification and AMP consistency). See General Comment 9

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.		
IV.B4-7 (R-199)	IV.B4.5-h	Core barrel assembly Baffle/former bolts and screws	Stainless steel	Reactor coolant	Changes in dimensions/Void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 9

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B4 Reactor Vessel Internals (PWR) - Babcock and Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.B4-8 (R-201)	IV.B4.5-j	Core barrel assembly Baffle/former bolts and screws	Stainless steel	Reactor coolant	Loss of preload/ stress relaxation	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 9
IV.B4-9 (R-200)	IV.B2.5-c	Lower internal assembly Lower core plate	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/ neutron irradiation embrittlement, void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before	No, but licensee commitment to be confirmed.	See General Comment 9

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B4 Reactor Vessel Internals (PWR) - Babcock and Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.		
IV.B4-10 (R-195)	IV.B4.5-c	Core barrel assembly Core barrel cylinder (top and bottom flange) Lower internals assembly-to-core barrel bolts Core barrel-to-thermal shield bolts Baffle plates and formers	Stainless steel, nickel alloy	Reactor coolant	Changes in dimensions/Void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 9
IV.B4-11 (R-196)	IV.B2.5-g	Lower internal assembly Fuel alignment pins Lower support plate column bolts Clevis insert bolts	Stainless steel, nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness/ neutron irradiation embrittlement, void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the	No, but licensee commitment to be confirmed.	See General Comment 9

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.		
IV.B4-12 (R-194)	IV.B4.5-b	Core barrel assembly Lower internals assembly-to-core barrel bolts Core barrel-to-thermal shield bolts	Stainless steel, nickel alloy	Reactor coolant	Cracking/ stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water chemistry" for PWR primary water~~, as described in EPRI TR-105714~~. No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 3 See General Comment 9

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B4 Reactor Vessel Internals (PWR) - Babcock and Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.B4-13 (R-197)	IV.B4.5-e	Core barrel assembly Lower internals assembly-to-core barrel bolts Core barrel-to-thermal shield bolts	Stainless steel, nickel alloy	Reactor coolant	Loss of preload/ stress relaxation	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 9
IV.B4-14 (R-193)	IV.B4.5-a	Core barrel assembly Core barrel cylinder (top and bottom flange) Baffle plates and formers	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water chemistry" for PWR primary water~~, as described in EPRI TR-105714~~. No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry	No, but licensee commitment to be confirmed.	See General Comment 3 See General Comment 9

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.		
IV.B4-16 (R-188)	IV.B4.4-d	Core support shield assembly Core support shield cylinder (top and bottom flange) Core support shield-to-core barrel bolts Outlet and vent valve (VV) nozzles VV assembly locking device	Stainless steel, nickel alloy~~, PH Stainless Steel forging~~	Reactor coolant and neutron flux	Loss of fracture toughness/ neutron irradiation embrittlement, void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	Consistent with the related comment for Chapter IX, PH stainless steel is a subset of SS with respect to aging. See General Comment 9
IV.B4-17 (R-187)	IV.B4.4-c	Core support shield assembly Core support shield cylinder	Stainless steel, nickel alloy~~, PH Stainless Steel forging~~	Reactor coolant	Changes in dimensions/Void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1)	No, but licensee commitment to be confirmed.	See related comment for IV.B4-16 above (PH SS forging a subset of SS). See General Comment 9

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
		(top and bottom flange) Core support shield-to-core barrel bolts VV retaining ring VV assembly locking device				participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.		
IV.B4-18 (R-185)	IV.B4.4-a	Core support shield assembly Core support shield cylinder (top and bottom flange) Outlet and vent valve (VV) nozzles VV body and retaining ring	Stainless steel~~, PH stainless steel forging, CASS~~	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water chemistry" for PWR primary water~~, as described in EPRI TR-105714~~. No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before	No, but licensee commitment to be confirmed.	See related comment for IV.B4-16 above (PH SS forging a subset of SS). See General Comment 7 See General Comment 3 See General Comment 9

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B4 Reactor Vessel Internals (PWR) - Babcock and Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.		
IV.B4-19 (R-192)	IV.B4.4-h	Core support shield assembly Core support shield-to-core barrel bolts	Stainless steel, nickel alloy	Reactor coolant	Loss of preload/ stress relaxation	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 9
IV.B4-20 (R-186)	IV.B4.4-b	Core support shield assembly Core support shield-to-core barrel bolts VV assembly locking device	Stainless steel, nickel alloy	Reactor coolant	Cracking/ stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water chemistry" for PWR primary water~~, as described in EPRI TR-105714~~. No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1)	No, but licensee commitment to be confirmed.	See General Comment 3 See General Comment 9

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B4 Reactor Vessel Internals (PWR) - Babcock and Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.		
IV.B4-22 (R-209)	IV.B4.7-a	Flow distributor assembly Flow distributor head and flange Incore guide support plate Clamping ring	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water chemistry" for PWR primary water~~, as described in EPRI TR-105714~~. No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before	No, but licensee commitment to be confirmed.	See General Comment 3 See General Comment 9

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B4 Reactor Vessel Internals (PWR) - Babcock and Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.		
IV.B4-23 (R-211)	IV.B4.7-c	Flow distributor assembly Flow distributor head and flange Shell forging-to-flow distributor bolts Incore guide support plate Clamping ring	Stainless steel, nickel alloy	Reactor coolant	Changes in dimensions/Void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 9
IV.B4-24 (R-212)	IV.B4.7-d	Flow distributor assembly Flow distributor head and flange Shell forging-to-flow distributor	Stainless steel, nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness/ neutron irradiation embrittlement, void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the	No, but licensee commitment to be confirmed.	See General Comment 9

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
		bolts Incore guide support plate Clamping ring				results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.		
IV.B4-25 (R-210)	IV.B4.7-b	Flow distributor assembly Shell forging-to-flow distributor bolts	Stainless steel, nickel alloy	Reactor coolant	Cracking/ stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water chemistry" for PWR primary water~~, as described in EPRI TR-105714~~. No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 3 See General Comment 9

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B4 Reactor Vessel Internals (PWR) - Babcock and Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.B4-26 (R-213)	IV.B4.7-e	Flow distributor assembly Shell forging-to-flow distributor bolts	Stainless steel, nickel alloy	Reactor coolant	Loss of preload/ stress relaxation	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 9
IV.B4-29 (R-202)	IV.B4.6-a	Lower grid assembly Lower grid rib section Fuel assembly support pads Lower grid flow dist. plate Orifice plugs Lower grid and shell forgings Guide blocks Shock pads Support post pipes	Stainless steel~~, cast austenitic stainless steel~~	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water chemistry" for PWR primary water~~, as described in EPRI TR-105714~~. No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry	No, but licensee commitment to be confirmed.	See General Comment 7 See General Comment 3 See General Comment 9

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B4 Reactor Vessel Internals (PWR) - Babcock and Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
		Incore guide tube spider castings				programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.		
IV.B4-30 (R-204)	IV.B4.6-c	Lower grid assembly Lower grid rib section Fuel assembly support pads Lower grid rib-to-shell forging screws Lower grid flow dist. plate Orifice plugs Lower grid and shell forgings Lower internals assembly-to-thermal shield bolts Guide blocks and bolts Shock pads and bolts Support post pipes Incore guide	Stainless steel~~, cast austenitic stainless steel~~, nickel alloy	Reactor coolant	Changes in dimensions/Void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 7 See General Comment 9

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
		tube spider castings						
IV.B4-31 (R-205)	IV.B4.2-e	Upper grid assembly Upper grid rib section Upper grid ring forging Fuel assembly support pads Plenum rib pads Rib-to-ring screws	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/ neutron irradiation embrittlement, void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 9
IV.B4-32 (R-203)	IV.B4.6-b	Lower grid assembly Lower grid rib-to-shell forging screws Lower internals assembly-to-thermal shield bolts Guide blocks and bolts	Stainless steel, nickel alloy	Reactor coolant	Cracking/ stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water chemistry" for PWR primary water~~, as described in EPRI TR-105714~~. No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects	No, but licensee commitment to be confirmed.	See General Comment 3 See General Comment 9

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B4 Reactor Vessel Internals (PWR) - Babcock and Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
		Shock pads and bolts				on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.		
IV.B4-33 (R-207)	IV.B4.6-g	Lower grid assembly Lower grid rib-to-shell forging screws Lower internals assembly-to-thermal shield bolts	Stainless steel, nickel alloy	Reactor coolant	Loss of preload/ stress relaxation	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 9

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B4 Reactor Vessel Internals (PWR) - Babcock and Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.B4-34 (R-172)	IV.B4.1-a	Plenum cover and plenum cylinder Plenum cover assembly Plenum cylinder Reinforcing plates	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water chemistry" for PWR primary water~~, as described in EPRI TR-105714~~. No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 3 See General Comment 9
IV.B4-35 (R-174)	IV.B4.1-c	Plenum cover and plenum cylinder Plenum cover assembly Plenum cylinder Reinforcing plates Top flange-to-	Stainless steel	Reactor coolant	Changes in dimensions/Void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry	No, but licensee commitment to be confirmed.	See General Comment 9

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
		cover bolts Bottom flange-to-upper grid screws				programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.		
IV.B4-36 (R-173)	IV.B4.1-b	Plenum cover and plenum cylinder Top flange-to-cover bolts Bottom flange-to-upper grid screws	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water chemistry" for PWR primary water~~, as described in EPRI TR-105714~~. No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 3 See General Comment 9

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B4 Reactor Vessel Internals (PWR) - Babcock and Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.B4-37 (R-54)	IV.B4.1-d IV.B4.2-d IV.B4.3-f ***IV.B4.4-e*** IV.B4.5-f IV.B4.6-f	Reactor vessel internals components	Stainless steel~~, cast austenitic stainless steel~~, nickel alloy	Reactor coolant	Cumulative fatigue damage/ fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA	See comment Item IV.B4-38. See General Comment 7
~~IV.B4-38~~ ~~(R-189)~~	~~IV.B4.4-e~~	~~Reactor vessel internals components~~	~~Stainless steel, cast austenitic stainless steel, nickel alloy, PH Stainless Steel forging~~	~~Reactor coolant~~	~~Cumulative fatigue damage/ fatigue~~	~~Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).~~	~~Yes, TLAA~~	Suggest deleting this Item and revising Item IV.B4-37 to include a link to IV.B4.4-e. The difference between Items IV.B4-37 and IV.B4-38 include: • "PH Stainless Steel forging" is listed as an additional material in IV.B4-38, whereas both list SS, CASS, and nickel alloy. • IV.B4-38 is the only RV Internals (IV.B1, B2, B3, B4) item which includes environmental effects in the AMP discussion.
IV.B4-39 (R-215)	IV.B4.8-b	Thermal shield	Stainless steel	Reactor coolant	Changes in dimensions/Void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry	No, but licensee commitment to be confirmed.	See General Comment 9

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B4 Reactor Vessel Internals (PWR) - Babcock and Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.		
IV.B4-40 (R-214)	IV.B4.8-a	Thermal shield	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water chemistry" for PWR primary water~~, as described in EPRI TR-105714~~. No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 3 See General Comment 9

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B4 Reactor Vessel Internals (PWR) - Babcock and Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.B4-41 (R-216)	IV.B4.8-c	Thermal shield	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/ neutron irradiation embrittlement, void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 9
IV.B4-43 (R-176)	IV.B4.2-b	Upper grid assembly Rib- to-ring screws	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water chemistry" for PWR primary water~~, as described in EPRI TR-105714~~. No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry	No, but licensee commitment to be confirmed.	See General Comment 3 See General Comment 9

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.		
IV.B4-44 (R-175)	IV.B4.2-a	Upper grid assembly Upper grid rib section Upper grid ring forging Fuel assembly support pads Plenum rib pads	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water chemistry" for PWR primary water~~, as described in EPRI TR-105714~~. No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 3 See General Comment 9

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B4 Reactor Vessel Internals (PWR) - Babcock and Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.B4-45 (R-177)	IV.B4.2-c	Upper grid assembly Upper grid rib section Upper grid ring forging Fuel assembly support pads Plenum rib pads Rib-to-ring screws	Stainless steel	Reactor coolant	Changes in dimensions/Void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	No, but licensee commitment to be confirmed.	See General Comment 9
IV.B4-46 (R-178)	IV.B4.2-e	Upper grid assembly Upper grid rib section Upper grid ring forging Fuel assembly support pads Plenum rib pads Rib-to-ring screws	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness/ neutron irradiation embrittlement, void swelling	No further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before	No, but licensee commitment to be confirmed.	See General Comment 9

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
B4 Reactor Vessel Internals (PWR) - Babcock and Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.		
New IV.B4(1) *IV.B2 IV.B3*	*N/A*	*Reactor vessel internals components*	*Stainless steel, nickel alloy*	*Reactor coolant*	*Loss of material/ pitting and crevice corrosion*	*Chapter XI.M2, "Water chemistry," for PWR primary water*	*No*	See General Comment 2

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

IV	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM							
C1	Reactor Coolant Pressure Boundary (Boiling Water Reactor)							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
C1 Reactor Coolant Pressure Boundary (Boiling Water Reactor)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.C1-1 (R-03)	IV.C1.1-i	Class 1 piping, fittings and branch connections < NPS 4	Stainless steel; steel	Reactor coolant	Cracking/ stress corrosion cracking, and intergranular stress corrosion cracking	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry," for BWR water ~~in BWRVIP-29 (EPRI TR-103515)~~ Inspection in accordance with ASME Section XI does not require volumetric examination of pipes less than NPS 4. A plant-specific destructive examination or a nondestructive examination (NDE) that permits inspection of the inside surfaces of the ***small-bore*** piping ***less than NPS 4, including pipe, fittings and branch connections,*** is to be conducted to ensure that cracking has not occurred and the component intended function will be maintained during the extended period of operation. ***Chapter XI.M32, "One-Time Inspection" is an acceptable verification method.***	Yes, parameters monitored/ inspected and detection of aging effects are to be evaluated	See General Comment 3 One-time Inspection is an acceptable method of verification for SCC per GALL Rev 0 Item IV.V1.1-i. As presently listed, GALL specifies a plant specific destructive or non-destructive examination for SCC and a **separate** plant-specific inspection (One-time Inspection is acceptable) for cracking of the same components due to thermal and mechanical loading. When rolled-up to the SRP-LR (Items 9 {R-03) and 21 {R-55}), the explicit indication is that examination for SSC must be separate from the thermal/ mechanical loading inspection and that a one-time inspection is not adequate for SCC but is for thermal/ mechanical loading. See SRP-LR Sections 3.1.2.2.12 and 3.1.2.2.4.1. Revise wording of IV.C1-1 to be consistent with IV.C1-2 wording and to clarify that a One-time inspection is an acceptable method for either mechanism. Also need to revise wording of SRP-LR Section 3.1.2.2.4.1 accordingly.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
C1 Reactor Coolant Pressure Boundary (Boiling Water Reactor)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.C1-5 (R-15)	IV.C1.4-a	Isolation condenser tube side components	Stainless steel; steel	Reactor coolant	Cracking/ stress corrosion cracking and intergranular stress corrosion cracking	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry," for BWR water ~~in BWRVIP-29 (EPRI TR-103515)~~ The AMP in Chapter XI.M1 is to be augmented to detect cracking due to stress corrosion cracking ~~and cyclic loading or loss of material due to pitting and crevice corrosion,~~ and verification of the effectiveness of the program is necessary to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period of operation. An acceptable verification program is to include temperature and radioactivity monitoring of the shell side water, and eddy current testing of tubes.	Yes, detection of aging effects is to be evaluated	See General Comment 3 . AMP discussion is in error since cyclic loading is not an aging mechanism listed for this item, or for BWR items other than for jet-pump sensing lines (IV.B1-12) and Nozzles (IV. A1-3 & IV.A1-2) and loss of material for this component is addressed in Item IV.C1-6. Item IV.C1-5 corresponds to SRP-LR (and GALL Vol 1) Item 11 and Section 3.1.2.2.4.3 whereas Item IV.C1-6 corresponds to SRP-LR Item 5 and Section 3.1.2.2.2. SRP-LR Section 3.1.2.2.2 discussion is accurate as it addresses augmenting the AMP for loss of material only. SRP-LR Section 3.1.2.2.4.3 discussion is in error as it reflects the extra mechanisms for this item and should be corrected.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
C1 Reactor Coolant Pressure Boundary (Boiling Water Reactor)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.C1-6 (R-16)	IV.C1.4-b	Isolation condenser tube side components	Stainless steel; steel	Reactor coolant	Loss of material/ general (steel only), pitting and crevice corrosion	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry," for BWR water ~~in BWRVIP-29 (EPRI TR-103515)~~ The AMP in Chapter XI.M1 is to be augmented to detect ~~cracking due to stress corrosion cracking and cyclic loading or~~ loss of material due to pitting and crevice corrosion, and verification of the effectiveness of the program is necessary to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period of operation. An acceptable verification program is to include temperature and radioactivity monitoring of the shell side water, and eddy current testing of tubes.	Yes, detection of aging effects is to be evaluated	See General Comment 3 . See related comment to Item IV.C1-5 above. Aging effect/mechanism in the AMP discussion is not consistent with the AE/AM listed in that column.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
C1 Reactor Coolant Pressure Boundary (Boiling Water Reactor)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.C1-8 (R-21)	IV.C1.1-f	Piping, piping components, and piping elements greater than or equal to 4 NPS	Nickel alloy	Reactor coolant	Cracking/ stress corrosion cracking and intergranular stress corrosion cracking	Chapter XI.M7, "BWR Stress Corrosion Cracking," and Chapter XI.M2, "Water Chemistry," for BWR water ~~in BWRVIP-29 (EPRI TR-103515)~~	No	See General Comment 3
IV.C1-9 (R-22)	IV.C1.1-f IV.C1.3-c	Piping, piping components, and piping elements greater than or equal to 4 NPS	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking and intergranular stress corrosion cracking	Chapter XI.M7, "BWR Stress Corrosion Cracking," and Chapter XI.M2, "Water Chemistry," for BWR water ~~in BWRVIP-29 (EPRI TR-103515)~~	No	See General Comment 3
IV.C1-10 (R-20)	IV.C1.1-f IV.C1.2-b IV.C1.3-c	Piping, piping components, and piping elements greater than or equal to 4 NPS	Stainless steel~~, cast austenitic stainless steel~~	Reactor coolant	Cracking/ stress corrosion cracking and intergranular stress corrosion cracking	Chapter XI.M7, "BWR Stress Corrosion Cracking," and Chapter XI.M2, "Water Chemistry," for BWR water ~~in BWRVIP-29 (EPRI TR-103515)~~	No	See General Comment 7 See General Comment 3
IV.C1-11 (R-04)	IV.C1.1-b IV.C1.1-d IV.C1.1-e IV.C1.1-h IV.C1.2-a IV.C1.3-d	Piping, piping components, and piping elements; flanges;-heater sheaths and sleeves; penetrations; ~~pressure housings; pump casing/cover;~~	Steel, stainless steel, ~~cast austenitic stainless steel, carbon~~ steel with nickel-alloy or stainless steel cladding,	Reactor coolant	Cumulative fatigue damage/ fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for	Yes, TLAA	See General Comment 8 See General Comment 7

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
		~~spray head;~~ thermal sleeves; vessel shell heads and welds	nickel-alloy			meeting the requirements of 10 CFR 54.21(c)(1).		
~~IV.C1-12~~ ~~(R-27)~~	~~IV.C1.2-e~~ ~~IV.C1.3-f~~	~~Pump and valve closure bolting~~	~~High-strength low-alloy steel SA 193 Gr. B7~~	~~System temperature up to 288°C (550°F)~~	~~Loss of preload/ stress relaxation~~	~~Chapter XI.M18, "Bolting Integrity"~~	~~No~~	Loss of preload is not an aging effect. See General Comment 6.
IV.C1-15 (R-029)	IV.C1.2-d IV.C1.3-e	Pump and valve seal flange closure bolting	Stainless steel; steel	~~Air with metal~~ ***System*** temperature up to 288°C (550°F)	Loss of material/ wear	Chapter XI.M18, "Bolting Integrity"	No	Environment should be 'System temperature up to 288°C (550°F)'. As defined in Chapter IX, specified environment is synonymous with the suggested revision but intended to describe the environment experienced by a PWR Pressurizer support skirt.
New IV.C1(1)	***N/A***	***Reactor coolant pressure boundary components***	***Steel with stainless steel or nickel alloy cladding; stainless steel; nickel alloy***	***Reactor coolant***	***Loss of material/ pitting and crevice corrosion***	***Chapter XI.M2, "Water chemistry," for BWR water***	***No***	Based on New IV.A2(1), this line should also be included for consistency.

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

Introductory Text

Delete the last sentence of the first paragraph related to pitting and crevice corrosion of stainless steel in a borated water environment.

IV C2	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Reactor Coolant System and Connected Lines (Pressurized Water Reactor)							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
IV.C2-1 (R-02)	IV.C2.2-h IV.C2.1-g	Class 1 piping, fittings and branch connections < NPS 4	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry," for PWR primary water ~~in EPRI TR-105714~~ Inspection in accordance with ASME Section XI does not require volumetric examination of pipes less than NPS 4. A plant-specific destructive examination or a nondestructive examination (NDE) that permits inspection of the inside surfaces of the ***small-bore*** piping ***less than NPS 4, including pipe, fittings, and branch connections,*** is to be conducted to ensure that cracking has not occurred and the component intended function will be maintained	Yes, parameters monitored/ inspected and detection of aging effects are to be evaluated	See General Comment 3 See related comment for Item IV.C1-1 (OTI). The option for utilizing XI.M32 in combination with XI.M1 and XI.M2 has been removed and should be reinstated. SRP-LR Section 3.1.2.2.7.3 also requires revision, accordingly, to specify that OTI is an acceptable verification method.

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						during the extended period of operation. ***See Chapter XI.M32, "One-Time Inspection" for an acceptable verification method.***		
IV.C2-2 (R-57)	IV.C2.1-g IV.C2.2-h	Class 1 piping, fittings and branch connections < NPS 4	Stainless steel~~; steel with stainless steel cladding~~	Reactor coolant	Cracking/ thermal and mechanical loading	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components Inspection in accordance with ASME Section XI does not require volumetric examination of pipes less than NPS 4. A plant-specific destructive examination or a nondestructive examination (NDE) that permits inspection of the inside surfaces of the piping is to be conducted to ensure that cracking has not occurred and the component intended function will be maintained during the extended period of operation. The AMP~~s are~~ **is** to be augmented by verifying that ~~*service-induced weld*~~ cracking ***due to thermal and mechanical loading*** is not occurring in the small-	Yes, parameters monitored/ inspected and detection of aging effects are to be evaluated	Small bore piping is not clad with stainless steel, but small bore RCS piping is stainless steel. The AMP description still includes attributes that are specific to SCC (but are not applicable to thermal and mechanical loading).

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
C2 Reactor Coolant System and Connected Lines (Pressurized Water Reactor)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						bore piping less than NPS < 4, including pipe, fittings, and branch connections. See Chapter XI.M32, "One-Time Inspection" for an acceptable verification method.		
IV.C2-3 (R-07)	IV.C2.5-h IV.C2.5-m IV.C2.2-f	Class 1 piping, fittings and primary nozzles, safe ends, manways, and flanges	Stainless steel; steel with stainless steel cladding	Reactor coolant	Cracking/ stress corrosion cracking	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry~~,~~" for PWR primary water ~~in EPRI TR-105714~~	No	See General Comment 3
IV.C2-4 (R-05)	IV.C2.5-i IV.C2.2-g IV.C2.1-e	Class 1 piping, piping components, and piping elements	~~Cast austenitic stainless steel~~ ***Stainless steel***	Reactor coolant	Cracking/ stress corrosion cracking	~~Monitoring and control of primary water chemistry in accordance with the guidelines in EPRI TR-105714 (Rev. 3 or later revisions or update) minimize the potential of SCC, and material selection according to the NUREG-0313, Rev. 2 guidelines of ≤0.035% C and ≥7.5% ferrite reduces susceptibility to SCC.~~ ~~For CASS components that do not meet either one of the above guidelines, a plant-specific aging management program is to~~	~~Yes, plant-specific~~ ***No***	See General Comment 7 The existing AMP entry requires a plant specific program if water chemistry is not maintained <u>or</u> the material is susceptible to SCC. The proposed change to the entry assumes the material is susceptible, but lists water chemistry as the aging management program which, by the current logic, eliminates the need for a plant specific program. However, ISI has been conservatively added as an AMP. The mention of the thermal embrittlement effect is

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
C2 Reactor Coolant System and Connected Lines (Pressurized Water Reactor)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						~~be evaluated. The program is to 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.~~***Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and*** ***Chapter XI.M2, "Water Chemistry," for PWR primary water***		unnecessary in this line. Thermal embrittlement in CASS is covered by line R-52 for this general component group. With these changes, this line could be combined with R-07.
IV.C2-6 (R-09)	IV.C2.4-b IV.C2.3-b	Class 1 pump casings and valve bodies	~~CASS~~ ***Stainless steel***, carbon steel with stainless steel cladding	Reactor coolant	Cracking/ stress corrosion cracking	~~Monitoring and control of primary water chemistry in accordance with the guidelines in EPRI TR-105714 (Rev. 3 or later revisions or update) minimize the potential of SCC, and material selection according to the NUREG-0313, Rev. 2 guidelines of ≤0.035% C and ≥7.5% ferrite reduces susceptibility to SCC.~~ ~~For CASS components that do not meet either one of the above guidelines, an aging management~~	No	See General Comment 7 As with line R-05, the change to the AMP column effectively requires both the first provision (water chemistry) of the existing entry, and conservatively applies the conditional AMP (ISI) making the question of material composition moot.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
C2 Reactor Coolant System and Connected Lines (Pressurized Water Reactor)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						~~program should conform to Chapter XI.M1, "ASME Section XI, Subsections IWB, IWC, and IWD."~~ ***Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and*** ***Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714***		
IV.C2-8 (R-11)	IV.C2.3-e IV.C2.4-e IV.C2.5-n	Closure bolting	~~High-strength l~~***L***ow-alloy steel, stainless steel	Air with reactor coolant leakage	Cracking/ stress corrosion cracking	Chapter XI.M18, "Bolting Integrity"	No	See related comment to Item IV.A1-8 above (high strength low alloy steel clarification).
IV.C2-9 (R-12)	IV.C2.3-g IV.C2.5-p IV.C2.4-g	Closure bolting	~~High-strength low-alloy steel, s~~***S***tainless steel	~~Air with reactor coolant leakage~~***System temperature up to 340°C (644°F)***	Loss of preload/ stress relaxation	Chapter XI.M18, "Bolting Integrity"	No	Loss of preload is not an aging effect for low alloy steel bolting. See General Comment 6. Environment should be 'System temperature up to 340°C (644°F) per the links to Rev. 0.
IV.C2-15 (R-04)	IV.C2.1-a IV.C2.1-b IV.C2.2-a IV.C2.2-b IV.C2.2-c IV.C2.3-a IV.C2.4-a	Piping, piping components, and piping elements; flanges; heater sheaths and sleeves;	Steel, stainless steel, ~~cast austenitic stainless steel, carbon~~ steel with	Reactor coolant	Cumulative fatigue damage/ fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be	Yes, TLAA	See General Comment 8 See General Comment 7

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
C2 Reactor Coolant System and Connected Lines (Pressurized Water Reactor)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
	IV.C2.5-a IV.C2.5-d IV.C2.5-e IV.C2.5-f IV.C2.5-q	penetrations~~;~~ ~~pressure housings; pump casing/cover; spray head;~~ thermal sleeves; vessel shell heads and welds	nickel-alloy or stainless steel cladding, nickel-alloy			addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).		
IV.C2-17 (R-24)	IV.C2.5-j	Pressurizer Spray head	Nickel alloy, ~~cast austenitic stainless steel,~~ stainless steel	Reactor coolant	Cracking/ primary water stress corrosion cracking	Chapter XI.M2, "Water Chemistry" and Chapter XI.M32 "One-Time Inspection" or Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," and provide a commitment in the FSAR supplement to submit a plant-specific AMP delineating commitments to Orders, Bulletins, or Generic Letters that inspect stipulated components for cracking of wetted surfaces.	No, unless licensee commitments need to be confirmed	See General Comment 7 See General Comment 9 See related comment for IV.A2-11 above (Clarify AMP Intent). Also, note that the ISI programs do not inspect the spray head. So, a commitment to implement the ISI program won't provide any basis that the component intended function will be maintained.
IV.C2-18 (R-58)	IV.C2.5-c IV.C2.5-g	Pressurizer components	Steel with stainless steel or nickel alloy cladding; or stainless steel	Reactor coolant	Cracking/ cyclic loading	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry," for PWR primary water ~~in EPRI TR-105714~~ Cracks in the pressurizer	No	See General Comment 3

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						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.		
IV.C2-19 (R-25)	IV.C2.5-c IV.C2.5-g	Pressurizer components	Steel with stainless steel or nickel alloy cladding; or stainless steel	Reactor coolant	Cracking/ stress corrosion cracking, primary water stress corrosion cracking	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry~~,~~" for PWR primary water~~ in EPRI TR-105714~~ and, for ~~Alloy 600~~***Nickel Alloys***, provide a commitment in the FSAR supplement to implement applicable (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff-accepted industry guidelines.	No, but licensee commitments to be confirmed	See General Comment 3 See General Comment 4 See General Comment 9

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
C2 Reactor Coolant System and Connected Lines (Pressurized Water Reactor)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.C2-20 (R-217)	IV.C2.5-r	Pressurizer heater sheaths and sleeves, and heater bundle diaphragm plate	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry~~,~~" for PWR primary water ~~in EPRI TR-105714~~	No	See General Comment 3
IV.C2-21 (R-06)	IV.C2.5-k IV.C2.5-s IV.C2.5-m	Pressurizer instrumentation penetrations, heater sheaths and sleeves, heater bundle diaphram plate, and manways and flanges	Nickel alloy or nickel alloy cladding	Reactor coolant	Cracking/ primary water stress corrosion cracking	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry~~,~~" for PWR primary water ~~in EPRI TR-105714~~ and, for ~~Alloy 600~~***Nickel Alloys***, provide a commitment in the FSAR supplement to implement applicable (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff-accepted industry guidelines.	No, but licensee commitments to be confirmed	See General Comment 3 See General Comment 4 See General Comment 9

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
C2 Reactor Coolant System and Connected Lines (Pressurized Water Reactor)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.C2-22 (R-14)	IV.C2.6-c	Pressurizer relief tank Tank shell and heads Flanges and nozzles	Stainless steel; steel with stainless steel cladding	Treated borated water >60°C (>140°F)	Cracking/ stress corrosion cracking	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 2 components and Chapter XI.M2, "Water Chemistry," for PWR primary water ~~in EPRI TR-105714~~	No	See General Comment 3
IV.C2-24 (RP-22)	IV.C2.	Pressurizer steam space nozzles and welds	Nickel alloy	Reactor coolant/ steam	Cracking/ Primary water stress corrosion cracking	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry~~,~~" for PWR primary water ~~in EPRI TR-105714~~ and, for ~~Alloy 600~~***Nickel Alloys***, provide a commitment in the FSAR supplement to implement applicable (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff-accepted industry guidelines.	No, but licensee commitments to be confirmed	See General Comment 3 See General Comment 4 See General Comment 9

IV	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
C2	Reactor Coolant System and Connected Lines (Pressurized Water Reactor)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.C2-26 (R-30)	IV.C2.1-c	Reactor coolant system piping and fittings Cold leg Hot leg Surge line Spray line	Stainless steel; steel with stainless steel cladding	Reactor coolant	Cracking/ stress corrosion cracking	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry," for PWR primary water~~ in EPRI TR-105714~~	No	See General Comment 3
New IV.C2(1)	***N/A***	***Piping, piping components, and piping elements; flanges; heater sheaths and sleeves; penetrations; thermal sleeves; vessel shell heads and welds***	***Steel with stainless steel or nickel alloy cladding; stainless steel; nickel alloy***	***Reactor coolant***	***Loss of material/ pitting and crevice corrosion***	***Chapter XI.M2, "Water chemistry," for PWR primary water***	***No***	See General Comment 2

IV.D1 Steam Generator (Recirculating)

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D1 Steam Generator (Recirculating)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.D1-1 (R-07)	IV.D1.1-i	~~Class 1 piping, fittings and~~ ~~p~~Primary nozzles, safe ends, manways, and flanges	Stainless steel; steel with stainless steel cladding	Reactor coolant	Cracking/ stress corrosion cracking	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry," for PWR primary water ~~in EPRI TR-105714~~	No	See General Comment 8 Draft NUREG/CR-XX (basis document) does not address this change. Manways and flanges are reasonable additions for recirculating steam generators due to being primary side forgings, but class 1 piping and fittings create confusion since piping and fittings are addressed in Section IV.C2. Which class 1 pipe and fittings is this item referring to that are not already addressed in IV.C2? Please clarify components for which this item is applicable. See General Comment 3

IV	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM							
D1	Steam Generator (Recirculating)							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
IV.D1-4 (R-01)	***IV.D1.1-i*** ~~IV.D1.1-j~~	Instrument penetrations***;*** and primary side nozzles and welds	Nickel alloy; ***steel with nickel alloy cladding***	Reactor coolant	Cracking/ primary water stress corrosion cracking	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry," for PWR primary water ~~in EPRI TR-105714~~ and, for ***Nickel Alloys***~~Alloy 600~~, provide a commitment in the FSAR supplement to implement applicable (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff-accepted industry guidelines.	No, but licensee commitments to be confirmed	Primary side nozzles and welds component is duplicated in IV.D-6 (R-218), for the same environment, aging effect/mechanism and AMP. See General Comment 5 See General Comment 3 See General Comment 4 See General Comment 9

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D1 Steam Generator (Recirculating)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.D1-5 (R-04)	IV.D1.1-h	~~Piping, piping components, and piping elements;~~ ~~f~~Flanges; ~~heater sheaths and sleeves;~~ penetrations; ~~pressure housings; pump casing/cover; spray head~~; ~~thermal sleeves~~; ***nozzles; safe ends;*** ~~vessel shell~~ ***lower*** heads and welds	Steel, stainless steel, ~~cast austenitic stainless steel, carbon~~ steel with nickel-alloy or stainless steel cladding, nickel-alloy	Reactor coolant	Cumulative fatigue damage/ fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA	See General Comment 8 See General Comment 7
~~IV.D1-6~~ ~~(R-218)~~	~~D1.1-i~~	~~Pressure boundary and structural~~ ~~Primary nozzles, safe ends, and welds~~	~~Nickel alloy or nickel alloy cladding~~	~~Reactor coolant~~	~~Cracking/ primary water stress corrosion cracking~~	~~Chapter XI.M1, "ASMeE Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry,"~~ for PWR primary water ~~in EPRI TR-105714 and, for Alloy 600, provide a commitment in the FSAR supplement to implement applicable (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff-accepted industry guidelines.~~	~~No, but licensee commitments to be confirmed~~	Suggest deletion of this Item since component is duplicated in Item IV.D1-4 (R-01) with only a slight clarification: "IV.D1.1-i" added to **Link** and "safe ends," added to **Structure and/or Component**, and "steel with nickel alloy cladding" added to **Material** per the related comment to Item IV.D1-4 above. SRP-LR Table 3.1-1 Item 22 should also be updated to remove R-218 from the 'Related Item' field.

IV	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM							
D1	Steam Generator (Recirculating)							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
IV.D1-8 (RP-16)	IV.D1	Steam generator structural tube bundle wrapper	Steel	Secondary feedwater/ steam	Loss of material/ erosion, general, pitting, and crevice corrosion	Chapter XI.M19, "Steam Generator Tubing Integrity" and Chapter XI.M2, "Water Chemistry," for PWR secondary water ~~in EPRI TR-102134~~	No	See General Comment 3
IV.D1-9 (RP-14)	IV.D1.	Steam generator ***structural*** anti-vibration bars	Chrome plated ***steel;*** ~~Nickel alloy,~~ stainless steel; ~~N~~*n*ickel alloy	Secondary feedwater/ steam	Cracking/ stress corrosion cracking	Chapter XI.M19, "Steam Generator Tubing Integrity" and Chapter XI.M2, "Water Chemistry," for PWR secondary water ~~in EPRI TR-102134~~	No	Suggest clarification of 'Structure and/or Component' for item grouping purposes (alphabetical component listing). Chrome plated nickel alloy is incorrect for a material. Chrome plated steel is a possible material for AV bars. See General Comment 3
IV.D1-10 (RP-15)	IV.D1.	Steam generator ***structural*** anti-vibration bars	Chrome plated ***steel;*** ~~Nickel alloy,~~ stainless steel; *n*~~,N~~ickel alloy	Secondary feedwater/ steam	Loss of material/ crevice corrosion and fretting	Chapter XI.M19, "Steam Generator Tubing Integrity" and Chapter XI.M2, "Water Chemistry," for PWR secondary water ~~in EPRI TR-102134~~	No	See related comment for Item IV.D1-9 above (alphabetical component listing). Chrome plated nickel alloy is incorrect for a material. Chrome plated steel is a possible material for AV bars. See General Comment 3
IV.D1-11 (RP-21)	IV.D1.	~~Steam Generator~~***Primary side*** Divider Plate	Nickel alloy***; steel with nickel alloy cladding***	Reactor coolant	Cracking/ Primary water stress corrosion cracking	~~Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for~~	No, but licensee commitments to be	See related comment for Item IV.D1-9 above (alphabetical component listing).

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D1 Steam Generator (Recirculating)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						~~Class 1 components and~~ Chapter XI.M2, "Water Chemistry," for PWR primary water ~~in EPRI TR-105714~~ and, for ~~Alloy 600~~***Nickel alloys***, provide a commitment in the FSAR supplement to implement applicable (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff-accepted industry guidelines.	confirmed	See General Comment 5 This component is not an RCPB component. The ISI Program does not directly examine this component. Previous applicants have credited only the Water Chemistry Program. This is a location of lower failure and lower consequences. See General Comment 3 See General Comment 4 See General Comment 9
IV.D1-12 (RP-17)	IV.D1.	~~Steam Generator~~***Primary side*** Divider Plate	Stainless steel	Reactor coolant	Cracking/ stress corrosion cracking	~~Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and~~ Chapter XI.M2, "Water Chemistry," for PWR primary water ~~in EPRI TR-105714~~	No	See related comment for Item IV.D1-9 above (alphabetical component listing). See related comment for Item IV.D1-11 (ISI is incorrect AMP for component). See General Comment 3
~~IV.D1-13~~ ~~(R-32)~~	~~IV.D1.1-f~~	~~Steam generator closure bolting~~	~~Steel~~	~~System temperature up to 340°C (644°F)~~	~~Loss of preload/ stress relaxation~~	~~Chapter XI.M18, "Bolting Integrity"~~	~~No~~	Loss of preload is not an aging effect. See General Comment 6.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D1 Steam Generator (Recirculating)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.D1-14 (R-33)	IV.D1.1-a IV.D1.1-b	Steam generator components ***Top head; steam nozzle and safe end; Upper and lower shell; FW nozzle and safe end; FW impingement plate and support***	Steel	Secondary feedwater/ steam	Cumulative fatigue damage/ fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA	See General Comment 8 Steam generator components" is too broad a 'Structure and/or Component' identification for this item. Item IV.D1-5 addresses the same aging effect for primary side steam generator components. Rev 0 structure/component description was adequate and should be returned. Comment also applies to item IV.D2-9 below.
IV.D1-16 (R-34)	IV.D1.1-c	Steam generator ***components***~~shell assembly (for OTSG),~~ upper and lower shell, and transition cone ~~(for recirculating steam generator)~~	Steel	Secondary feedwater/ steam	Loss of material/ general, pitting, and crevice corrosion	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 2 components and Chapter XI.M2, "Water Chemistry," for PWR secondary water~~in EPRI TR-102134~~ As noted in NRC Information Notice IN 90-04, ***If*** general and pitting corrosion of the shell ***is known to*** exist~~s~~, the AMP guidelines in Chapter XI.M1 may not be sufficient to detect general and pitting corrosion ***(and the resulting corrosion-fatigue cracking)***, and additional	Yes, detection of aging effects is to be evaluated	See General Comment 8 See General Comment 3 The AMP description as written does not clearly convey the nature of the degradation mechanism, which is related to a very specific set of conditions, not to most SGs. IN 90-04 indicates that pitting corrosion on the surface served as corrosion fatigue crack initiation sites, not that pitting corrosion resulted in sufficient degradation to cause loss of component function. Further, this degradation mode has been limited to isolated cases of weld-zone cracking in Westinghouse Model 44 and 51 SGs, where a high stress region exists in the area of the shell to transition cone weld.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D1 Steam Generator (Recirculating)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						inspection procedures are to be developed. ***This issue is limited to Westinghouse Model 44 and 51 Steam Generators where a high stress region exists at the shell to transition cone weld.***		
IV.D1-17 (R-040)	IV.D1.2-i IV.D1.2-j	Tube plugs	Nickel alloy	Reactor coolant	Cracking/ primary water stress corrosion cracking	Chapter XI.M19, "Steam Generator Tubing Integrity" and Chapter XI.M2, "Water Chemistry," for PWR primary water ~~in EPRI TR-105714~~.	No	See General Comment 3
IV.D1-18 (R-41)	IV.D1.2-h	***Steam generator structural*** Tube support lattice bars	Steel	Secondary feedwater/ steam	Wall thinning/ flow-accelerated corrosion	Applicant must provide a commitment in the FSAR supplement to submit, for NRC review and approval, an inspection plan for tube support lattice bars as based upon staff approved NEI 97-06 guidelines, or other alternative regulatory basis for steam generator degradation management, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed	See related comment for Item IV.D1-9 above (alphabetical component listing). See General Comment 9

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D1 Steam Generator (Recirculating)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.D1-19 (R-42)	IV.D1.2-k	***Steam generator structural*** Tube support plates	Steel	Secondary feedwater/ steam	Ligament cracking/ corrosion	Chapter XI.M19, "Steam Generator Tubing Integrity" and Chapter XI.M2, "Water Chemistry," for PWR secondary water ~~in EPRI TR-102134~~	No	See related comment for Item IV.D1-9 above (alphabetical component listing). It's not clear whether the aging effect for "Ligament cracking/ corrosion" is Cracking, LOM, or both. Entry should be clarified. SRP-LR Table 3.1-1 Item 65 should also be clarified. See General Comment 3
IV.D1-20 (R-043)	IV.D1.2-g	Tubes	Nickel alloy	Secondary feedwater/ steam	Denting/ corrosion of carbon steel tube support plate	Chapter XI.M19, "Steam Generator Tubing Integrity" and Chapter XI.M2, "Water Chemistry," for PWR secondary water ~~in EPRI TR-102134~~. *For plants that could experience denting at the upper support plates, the applicant should evaluate potential for rapidly propagating cracks and then develop and take corrective actions consistent with Bulletin 88-02, "Rapidly Propagating Cracks in SG Tubes."*	No	See General Comment 3

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D1 Steam Generator (Recirculating)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.D1-21 (R-044)	IV.D1.2-a	Tubes and sleeves	Nickel alloy	Reactor coolant	Cracking/ primary water stress corrosion cracking	Chapter XI.M19, "Steam Generator Tubing Integrity" and Chapter XI.M2, "Water Chemistry," for PWR primary water ~~in EPRI TR-105714~~.	No	See General Comment 3
IV.D1-23 (R-048)	IV.D1.2-c	Tubes and sleeves	Nickel alloy	Secondary feedwater/ steam	Cracking/ intergranular attack	Chapter XI.M19, "Steam Generator Tubing Integrity" and Chapter XI.M2, "Water Chemistry," for PWR secondary water ~~in EPRI TR-102134~~.	No	See General Comment 3
IV.D1-24 (R-047)	IV.D1.2-b	Tubes and sleeves	Nickel alloy	Secondary feedwater/ steam	Cracking/ outer diameter stress corrosion cracking	Chapter XI.M19, "Steam Generator Tubing Integrity" and Chapter XI.M2, "Water Chemistry," for PWR secondary water ~~in EPRI TR-102134~~	No	See General Comment 3
IV.D1-25 (R-049)	IV.D1.2-e	Tubes and sleeves	Nickel alloy	Secondary feedwater/ steam	Loss of material/ fretting and wear	Chapter XI.M19, "Steam Generator Tubing Integrity" and Chapter XI.M2, "Water Chemistry," for PWR secondary water ~~in EPRI TR-102134~~	No	See General Comment 3

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D1 Steam Generator (Recirculating)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.D1-26 (R-50)	IV.D1.2-f	Tubes and sleeves (exposed to phosphate chemistry)	Nickel alloy	Secondary feedwater/ steam	Loss of material/ wastage and pitting corrosion	Chapter XI.M19, "Steam Generator Tubing Integrity" and Chapter XI.M2, "Water Chemistry," for PWR secondary water ~~in EPRI TR-102134~~	No	See General Comment 3
New IV.D1(1) *IV.D2*	*N/A*	*Primary side steam generator components*	*Steel with stainless steel or nickel alloy cladding; stainless steel; nickel alloy*	*Reactor coolant*	*Loss of material/ pitting and crevice corrosion*	*Chapter XI.M2, "Water chemistry," for PWR primary water*	*No*	See General Comment 2
New IV.D1(2)	*N/A*	*Tubesheet*	*Steel with stainless steel or nickel alloy cladding; stainless steel; nickel alloy*	*Reactor Coolant*	*Cracking/ stress corrosion cracking*	*Chapter XI.M2, "Water chemistry," for PWR primary water*	*No*	This line addresses the primary side of the tubesheet. The line is similar to RP-17 with added materials.
New IV.D1(3)	*N/A*	*Tubesheet*	*Steel with stainless steel or nickel alloy cladding; stainless steel; nickel alloy*	*Secondary feedwater/ steam*	*Cracking/ stress corrosion cracking*	*Chapter XI.M2, "Water chemistry," for PWR secondary water, and* *Plant specific inspection program*	*Yes, plant specific*	This line addresses the secondary side of the tubesheet. The line is similar to lines RP-14 and R-36 except that a plant specific inspection program is used since neither ISI nor SG Tube Integrity is applicable.

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D1 Steam Generator (Recirculating)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
New IV.D1(4) *IV.D2*	*N/A*	*Steam generator components*	*Steel with stainless steel or nickel alloy cladding; stainless steel; nickel alloy*	*Secondary feedwater/ steam*	*Loss of material/ pitting and crevice corrosion*	*Chapter XI.M2, "Water chemistry," for PWR secondary water, and* *Plant specific inspection program*	*Yes, plant specific*	This line addresses stainless steel and nickel alloy secondary side components. The line is similar to lines SP-16 and S-22 except that a plant specific inspection program is used and nickel alloy is included.

IV.D2 Steam Generator (Once-Through)

IV	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM							
D2	Steam Generator (Once-Through)							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
IV.D2-2 (R-01)	IV.D2.1-h	Instrument penetrations and primary side nozzles and welds	Nickel alloy	Reactor coolant	Cracking/ primary water stress corrosion cracking	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry," for PWR primary water ~~in EPRI TR-105714~~ and, for ***Nickel Alloys*** ~~*Alloy 600*~~, provide a commitment in the FSAR supplement to implement applicable (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff-accepted industry guidelines.	No, but licensee commitments to be verified.	See General Comment 3 See General Comment 4 See General Comment 9
IV.D2-3 (R-04)	IV.D2.1-c	***Primary side*** nozzles***; safe ends***~~Piping, piping components, and piping elements; flanges; heater sheaths and sleeves; penetrations; pressure housings; pump casing/cover; spray head;~~	Steel, stainless steel;~~, cast austenitic stainless steel, carbon~~ steel with nickel-alloy or stainless steel cladding~~,~~***;*** nickel-alloy	Reactor coolant	Cumulative fatigue damage/ fatigue	Fatigue is a time-limited aging analysis (TLAA) to be performed for the period of extended operation, and, for Class 1 components, environmental effects on fatigue are to be addressed. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA	See General Comment GALL Rev 0 link is for primary nozzles only, but item was changed to be more broad and specify piping, piping components, and piping elements. Multiple structures and/or components that are not part of a SG fit into this component as defined in Chapter IX. This listing creates confusion as to which SG components are included and could lead to a wrong

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D2 Steam Generator (Once-Through)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
		~~thermal sleeves; vessel shell heads~~ and welds						interpretation for which components are to be evaluated for fatigue TLAA. See General Comment 7
IV.D2-4 (R-38)	IV.D2.1-f	~~Pressure boundary and structural~~*Steam generator components* FW and AFW nozzles and safe ends Steam nozzles and safe ends	Steel	Secondary feedwater/ steam	Wall thinning/ flow-accelerated corrosion	Chapter XI.M17, "Flow-Accelerated Corrosion"		See related comment for Item IV.D1-9 above (alphabetical component listing).
~~IV.D2-6~~ ~~(R-032)~~	~~IV.D2.1-k~~	~~Steam generator closure bolting~~	~~Steel~~	~~System temperature up to 340°C (644°F)~~	~~Loss of preload/ stress relaxation~~	~~Chapter XI.M18, "Bolting Integrity"~~	~~No~~	Loss of preload is not an aging effect. See General Comment 6.
IV.D2-7 (R-033)	IV.D2.1-d IV.D2.1-g	Steam generator components ***Shell assembly; Feedwater (FW) and Auxiliary Feedwater (AFW) nozzles and safe ends; Steam nozzles and safe ends***	Steel	Secondary feedwater/ steam	Cumulative fatigue damage/ fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA	See related comment for Item IV.A1-6 above ('Structure and/or Component' not consistent with specified item). See related comment for Item GIV.D1-14 above (Rev 0 Structure/Component discussion should be restored).

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D2 Steam Generator (Once-Through)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.D2-9 (R-35)	IV.D2.1-a	~~Steam generator~~ ***Primary side*** components Upper and lower heads Tube sheets	Steel with stainless steel or nickel-alloy cladding	Reactor coolant	Cracking/ stress corrosion cracking, primary water stress corrosion cracking	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 1 components and Chapter XI.M2, "Water Chemistry," for PWR primary water ~~in EPRI TR-105714~~ and, for ***Nickel Alloys*** ~~*Alloy 600*~~, provide a commitment in the FSAR supplement to implement applicable (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff-accepted industry guidelines.	No, but licensee commitments to be confirmed.	See related comment for Item IV.D1-9 above (alphabetical component listing). See General Comment 3 See General Comment 4 See General Comment 9
IV.D2-10 (R-34)	IV.D2.1-e	Steam generator ***components*** shell assembly ~~(for OTSG), upper and lower shell, and transition cone (for recirculating steam generator)~~	Steel	Secondary feedwater/ steam	Loss of material/ general, pitting, and crevice corrosion	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," for Class 2 components and Chapter XI.M2, "Water Chemistry," for PWR secondary water ~~in EPRI TR-102134~~ As noted in NRC Information Notice IN 90-04, *if* general and pitting	Yes, detection of aging effects is to be evaluated	See General Comment 8 See General Comment 3 See related comment for Item IV.D1-16 above (AMP discussion clarification).

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D2 Steam Generator (Once-Through)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
						corrosion of the shell ***is known to*** ~~exists~~, the AMP guidelines in Chapter XI.M1 may not be sufficient to detect general and pitting corrosion ***(and the resulting corrosion-fatigue cracking)***, and additional inspection procedures are to be developed. ***This issue is limited to Westinghouse Model 44 and 51 Steam Generators where a high stress region exists at the shell to transition cone weld.***		
IV.D2-11 (R-040)	IV.D2.2-f IV.D2.2-g	Tube plugs	Nickel alloy	Reactor coolant	Cracking/ primary water stress corrosion cracking	Chapter XI.M19, "Steam Generator Tubing Integrity" and Chapter XI.M2, "Water Chemistry," for PWR primary water ~~in EPRI TR-105714.~~	No	See General Comment 3
IV.D2-12 (R-044)	IV.D2.2-a	Tubes and sleeves	Nickel alloy	Reactor coolant	Cracking/ primary water stress corrosion cracking	Chapter XI.M19, "Steam Generator Tubing Integrity" and Chapter XI.M2, "Water Chemistry," for PWR primary water ~~in EPRI TR-105714.~~	No	See General Comment 3

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D2 Steam Generator (Once-Through)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
IV.D2-14 (R-048)	IV.D2.2-c	Tubes and sleeves	Nickel alloy	Secondary feedwater/ steam	Cracking/ intergranular attack	Chapter XI.M19, "Steam Generator Tubing Integrity" and Chapter XI.M2, "Water Chemistry," for PWR secondary water ~~in EPRI TR-102134~~	No	See General Comment 3
IV.D2-15 (R-047)	IV.D2.2-b	Tubes and sleeves	Nickel alloy	Secondary feedwater/ steam	Cracking/ outer diameter stress corrosion cracking	Chapter XI.M19, "Steam Generator Tubing Integrity" and Chapter XI.M2, "Water Chemistry," for PWR secondary water ~~in EPRI TR-102134~~	No	See General Comment 3
IV.D2-16 (R-49)	IV.D2.2-d	Tubes and sleeves	Nickel alloy	Secondary feedwater/ steam	Loss of material/ fretting and wear	Chapter XI.M19, "Steam Generator Tubing Integrity" and Chapter XI.M2, "Water Chemistry," for PWR secondary ~~water in EPRI TR-102134~~	No	See General Comment 3
New IV.D2(1)	***N/A***	***Primary side steam generator components***	***Steel with stainless steel or nickel alloy cladding; stainless steel; nickel alloy***	***Reactor coolant***	***Loss of material/ pitting and crevice corrosion***	***Chapter XI.M2, "Water chemistry," for PWR primary water***	***No***	See General Comment 2

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
D2 Steam Generator (Once-Through)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
New IV.D2(2) *IV.D1*	*N/A*	*Steam generator components*	*Steel with stainless steel or nickel alloy cladding; stainless steel; nickel alloy*	*Secondary feedwater/ steam*	*Loss of material/ pitting and crevice corrosion*	*Chapter XI.M2, "Water chemistry," for PWR secondary water, and* *Plant specific inspection program*	*Yes, plant specific*	This line addresses stainless steel and nickel alloy secondary side components. The line is similar to lines SP-16 and S-22 except that a plant specific inspection program is used and nickel alloy is included.

IV E.	REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM Common Miscellaneous Material Environment Combinations							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
~~IV.E-1~~ ~~(RP-02)~~	~~E.~~	~~Piping, piping components, and piping elements~~	~~Cast austenitic stainless steel~~	~~Air – indoor uncontrolled (External)~~	~~None~~	~~None~~	~~No~~	CASS covered by RP-04 entry for stainless steel
New IV.E(1)		***Piping, piping components, and piping elements external surfaces***	***Steel***	***Containment environment (inert)***	***None***	***None***	***No***	***See Chapter IX comment on Containment environment (inert).***

Chapter V Engineered Safety Features

General Comments

1. New Line Items

Proposed new lines items are listed at the end of each table. The new lines are designated "New V.X(Y) where X is the table identifier and Y is a sequential number for new lines in that table. Where the same line is proposed in multiple tables, the other tables are listed below the designation.

2. Stainless Steel in Treated Borated Water

The GALL does not currently address the aging effect of loss of material for stainless steel in a treated borated water environment. Some sections of GALL note that the effect is minor and specifically not mentioned in GALL. For example, the introductory text to table V.D1 says:

> The effects of pitting and crevice corrosion on stainless steel components are not significant in treated borated water and, therefore, are not included in this section.

Operating experience has shown loss of material to be a negligible effect because the water chemistry requirements minimize contaminants that would lead to loss of material. In other words, the water chemistry programs for PWRs manage the aging effect of loss of material. Although the effect is minor, past applications have listed (and future applications will list) loss of material as an aging effect for stainless steel in a treated borated water environment with water chemistry as the aging management program. Because the aging effect is minor, the water chemistry program by itself is sufficient to manage this aging effect, as evidenced by past operating experience. This was acknowledged in previous SERs including the following excerpt from Farley SER Section 3.1.2.3.1.2,

> Pitting corrosion and crevice corrosion may occur in ASME Code, Class 1, stainless steel or NiCrFe components under exposure to aggressive, oxidizing environments. Normally, the presence of elevated dissolved oxygen and/or aggressive ionic impurity concentrations is necessary to create these oxidizing environments in the RCS. The applicant's response to RAI 3.1.3.1.1-1, Part b, provides an acceptable explanation for citing the Water Chemistry Control Program as a basis for minimizing the dissolved oxygen and ionic impurity concentrations that could otherwise, if left present in high concentrations, lead to an aggressive, oxidizing RCS coolant environment. The GALL Report does not indicate that the loss of material due to pitting corrosion or crevice corrosion is an aging effect of concern for stainless steel or NiCrFe ASME Code Class 1 components. Since the applicant has conservatively assumed that the loss of material due to pitting corrosion or crevice corrosion is an applicable aging effect for these RV components, the staff concludes that the Water Chemistry Control Program provides a sufficient mitigative strategy for managing this aging effect relative to the recommendations of the GALL Report.

New lines are proposed for systems where this MEAP is appropriate. The introductory text indicating the effect is not addressed is deleted.

3. **Water Chemistry Reference**

The reference to the specific EPRI document need not be included in the Aging Management Program column. This information is identified in the AMP description in Chapter XI of GALL.

4. **External Environments**

Introductory text to Chapter V systems tables refers to an external surfaces table E at the end of the chapter for aging management programs for the degradation of external surfaces of components and miscellaneous bolting. However, most external surfaces and many external bolting entries are still within the individual systems tables.

The changes proposed below move most external surface and external bolting lines to the external table. Any lines which refer to a unique aging management program (e.g., Fire Protection Program to monitor fire doors) will remain with the system table.

5. **Heat Exchanger Components Description**

Many GALL items referring to heat exchanger components indicate whether the line applies to the tube side or the shell side. However, the designation of the tube side or shell side of a heat exchanger unnecessarily limits the applicability of the GALL line item. Small heat exchangers in particular can be configured with the cooled fluid on either the shell or tube side. For all heat exchangers with a given set of materials and environments, the configuration of the heat exchanger (tube side vs. shell side) will not alter the aging effects or the aging management programs. Consequently, the component descriptions for these lines should be changed to delete the tube side or shell side designation. Similarly, some lines list heat exchanger components including tubes. For all aging effects other than the reduction of heat transfer, tubes may be considered with all other heat exchanger components. Thus, for aging effects other than reduction of heat transfer, the component description should be "Heat exchanger components." The descriptor "Heat exchanger tubes" should be used when addressing reduction of heat transfer.

6. **Bolting**

Although some utilities have conservatively applied loss of preload as aging effect for bolting, the industry does not consider loss of preload as an aging effect requiring management. In accordance with EPRI 1003056, "Mechanical Tool," Appendix F, loss of preload is a design driven effect and not an aging effect requiring management. The bolting at most facilities is standard grade B7 carbon steel, or similar material, except in rare specialized applications. Loss of preload due to stress relaxation (creep) for this material can only be a concern in very high temperature applications (> 700°F) as stated in the ASME Code Section II Part D Table 4 Note 4. However, there is no bolting used in BWRs and PWRs that operate at 700°F, with the exception of unique applications, such as the emergency diesel generator exhaust. Therefore, loss of preload due to stress relaxation (creep) is not a valid aging effect.

In addition, the industry has taken actions to address NUREG –1339, "Resolution to Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants." Licensees have implemented good bolting practices in accordance with those referenced in EPRI NP-5769, EPRI NP-5067 and EPRI TR-104213 in normal maintenance and design activities. Normal maintenance and design activities thus address the potential for loss of preload such that it is not a concern for the current or extended operating term. Proper joint preparation and make-up in accordance with industry standards precludes loss of preload. Even other design factors that could contribute to a loss of preload in closure bolting applications, such as vibration, should not result in loosening in a properly designed and assembled bolted joint.

The impact to the GALL tables is that, with elimination of loss of preload as an aging effect, closure bolting has the same MEAP as external surfaces and the lines could be combined.

7. **Integration of CASS with Stainless Steel**

Cast austenitic stainless steel (CASS) is currently treated as a separate material in GALL. However, with the exception of the loss of fracture toughness due to thermal and neutron irradiation embrittlement, CASS and stainless steel share the same aging effects/mechanisms in GALL.

To simplify GALL, CASS should be treated as a subset of stainless steel. CASS would only be listed as a material when loss of fracture toughness due to thermal (or thermal and irradiation) embrittlement is at issue, or where unique AMP requirements are given. This would provide consistency with GALL's treatment of other material groups, e.g., gray cast iron as a subset of steel, and copper alloy >15% zinc as a subset of copper alloy. Gray cast iron and copper alloy >15% zinc are both susceptible to selective leaching and are only listed as materials when selective leaching is addressed.

This change will have the added benefit of eliminating the need for new MEAP combinations to address CASS in non-Class 1 systems where stainless steel is adequately evaluated but CASS, if it is to be considered a separate material, is not. CASS is currently listed in only a few lines in Chapters V and VII and not at all in Chapter VIII.

One change (Table F) is required in GALL Chapter V as listed below. For this line, that lists CASS alone, an identical line exists in the system table for stainless steel so the CASS line may be deleted.

V.A Containment Spray System (PWR)

Introductory Text

In introductory comments for Containment Spray (Page VA-1), delete the sentence following the third paragraph; "Aging management for the degradation of external surfaces of carbon steel components are included in V.E." The same sentence leads off the next paragraph.

V ENGINEERED SAFETY FEATURES
A Containment Spray System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
~~V.A-1~~ ~~(E-26)~~	~~V.A.5-a~~ ~~V.A.2-a~~	~~Ducting, piping and components external surfaces~~	~~Steel~~	~~Air – indoor uncontrolled (External)~~	~~Loss of material/ general corrosion~~	~~A plant-specific aging management program is to be evaluated.~~	~~Yes, plant-specific~~	See General Comment 4
~~V.A-2~~ ~~(E-28)~~	~~V.A.3-b~~ ~~V.A.1-b~~ ~~V.A.4-b~~ ~~V.A.6-d~~ ~~V.A.5-b~~	~~External surfaces~~	~~Steel~~	~~Air with borated water leakage~~	~~Loss of material/ boric acid corrosion~~	~~Chapter XI.M10, "Boric Acid Corrosion"~~	~~No~~	See General Comment 4
V.A-3 (E-17)	V.A.5-a	Heat exchanger ~~shell side~~ components	Steel	Closed cycle cooling water	Loss of material/ pitting and crevice corrosion	Chapter XI.M21, "Closed-Cycle Cooling Water System"	No	See General Comment 5
V.A-4 (E-19)	V.A.6-c	Heat exchanger ~~shell side~~ components ~~including tubes~~	Stainless steel	Closed cycle cooling water	Loss of material/ pitting and crevice corrosion	Chapter XI.M21, "Closed-Cycle Cooling Water System"	No	See General Comment 5
V.A-5 (E-20)	V.A.6-a	Heat exchanger ~~shell side~~ components ~~including tubes~~	Stainless steel	Raw water	Loss of material/ pitting, crevice, and microbiologically influenced corrosion, and fouling	Chapter XI.M20, "Open-Cycle Cooling Water System"	No	See General Comment 5
V.A-6 (E-18)	V.A.6-a	Heat exchanger ~~shell side~~ components ~~including tubes~~	Steel	Raw water	Loss of material/ general, pitting, crevice, and microbiologically influenced corrosion, and fouling	Chapter XI.M20, "Open-Cycle Cooling Water System"	No	See General Comment 5

V ENGINEERED SAFETY FEATURES
A Containment Spray System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
V.A-7 (EP-13)	V.A.	Heat exchanger ~~tubes~~ ***components***	Copper alloy	Closed cycle cooling water	Loss of material/ pitting, crevice, and galvanic corrosion	Chapter XI.M21, "Closed-Cycle Cooling Water System"	No	See General Comment 5
V.A-8 (EP-37)	V.A.	Heat exchanger ~~tubes~~ ***components***	Copper alloy >15% Zn	Closed cycle cooling water	Loss of material/ selective leaching	Chapter XI.M33, "Selective Leaching of Materials"	No	See General Comment 5
V.A-12 (E-43)	V.A.	~~Motor Cooler~~ ***Heat exchanger components***	Gray cast iron	Treated water	Loss of material/ Selective leaching	Chapter XI.M33, "Selective Leaching of Materials"	No	See General Comment 5 The motor cooler is the same as a heat exchanger.
V.A-17 (E-12)	V.A.1-a V.A.3-a V.A.4-a V.A.1-c	Piping, piping components, piping elements, and tanks	Stainless steel	Treated borated water >60°C (>140°F)	Cracking/ stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water ~~in EPRI TR-105714~~	No	See General Comment 3
New V.A(1)		***Heat exchanger tubes***	***Copper alloy***	***Closed cycle cooling water***	***Reduction of heat transfer/ fouling***	***Chapter XI.M21, "Closed-Cycle Cooling Water System"***	***No***	This provides for fouling of copper-alloy tubes. EP-35 is the same except for the material.
New V.A(2)		***Piping, piping components, piping elements, and tanks***	***Stainless steel***	***Treated borated water***	***Loss of material/ pitting and crevice corrosion***	***Chapter XI.M2, "Water Chemistry" for PWR primary water***	***No***	See General Comment 2
New V.A(3)		***Piping, piping components, piping elements, and tanks***	***Stainless steel***	***Condensation (internal)***	***Loss of material/ pitting and crevice corrosion***	***A plant-specific aging management program is to be evaluated.***	***Yes, plant specific***	This line is the same as E-14 with the inclusion of tanks.
New V.A(4)		***Encapsulation Components***	***Steel***	***Air – indoor uncontrolled (internal)***	***Loss of material/ general, pitting, and crevice corrosion***	***A plant-specific aging management program is to be evaluated.***	***Yes, plant-specific***	Encapsulation components are not currently GALL item, but they are included in the Farley and Millstone applications. See conforming change in Chapter IX.

V ENGINEERED SAFETY FEATURES
A Containment Spray System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
New V.A(5)		*Encapsulation Components*	*Steel*	*Air with borated water leakage (internal)*	*Loss of material/ general, pitting, and crevice corrosion*	*A plant-specific aging management program is to be evaluated.*	*Yes, plant-specific*	Note: this leakage is on the inside of the encapsulation components. Encapsulation components are currently not included in the GALL, but are included in the Farley and Millstone applications. See conforming change in Chapter IX.
New V.A(6) *V.C* *V.D1* *V.D2*		*Piping, piping components, and piping elements*	*Stainless steel*	*Closed cycle cooling water >60°C (>140°F)*	*Cracking/ stress corrosion cracking*	*Chapter XI.M21, "Closed-Cycle Cooling Water System"*	*Yes, plant-specific*	The basis for AP-60 is applicable to components in ESF systems where closed cooling water temperatures exceed 140°F in supporting systems such as bearing and lube oil coolers and associated piping that are considered part of the ESF systems. EP-33 addresses a different aging mechanism for this same group of components.
New V.A(7) *V.D1* *V.D2*		*Piping, piping components, and piping elements*	*Copper alloy*	*Lubricating oil*	*Loss of material/ pitting and crevice corrosion*	*A plant-specific aging management program is to be evaluated.*	*Yes, plant-specific*	The bases for AP-47 and SP-32 are applicable to components such as pump and motor lubricating oil systems that are included as part of the ESF systems.
New V.A(8) *V.D1* *V.D2*	8	*Piping, piping components, and piping elements*	*Steel*	*Lubricating oil*	*Loss of material/ general, pitting, and crevice corrosion*	*A plant-specific aging management program is to be evaluated.*	*Yes, plant-specific*	The bases for AP-30 and SP-25 are applicable to components such as pump and motor lubricating oil systems that are included as part of the ESF systems.

V ENGINEERED SAFETY FEATURES
A Containment Spray System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
New V.A(9) *V.D1* *V.D2*		*Heat exchanger tubes*	*Copper alloy*	*Lubricating oil*	*Reduction of heat transfer/ fouling*	*A plant-specific aging management program is to be evaluated.*	*Yes, plant-specific*	CNP SER Section 3.2.2.3.3 acknowledged the potential for fouling of copper alloy components in a lubricating oil environment. The SER addressed components in A PWR ECCS system, but these conditions would be equally applicable to components of other lube oil systems. The reduction of heat transfer due to fouling of heat exchanger tubes will be addressed by a plant specific program. (Oil analysis for CNP as shown in Table 3.2.2-3 of the CNP LRA)
New V.A(10) *V.D1* *V.D2*		*Heat exchanger tubes*	*Steel*	*Lubricating oil*	*Reduction of heat transfer/ fouling*	*A plant-specific aging management program is to be evaluated.*	*Yes, plant-specific*	This line is similar to the preceding line except for a different material. Heat exchanger tubes of steel would also be subject to fouling in a lubricating oil environment.
New V.A(11) *V.D1* *V.D2*		*Heat exchanger tubes*	*Stainless steel*	*Lubricating oil*	*Reduction of heat transfer/ fouling*	*A plant-specific aging management program is to be evaluated.*	*Yes, plant-specific*	This line is similar to the preceding line except for a different material. Heat exchanger tubes of stainless steel would also be subject to fouling in a lubricating oil environment.

V.B Standby Gas Treatment System (BWR)

V ENGINEERED SAFETY FEATURES B Standby Gas Treatment System (BWR)								
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
~~V.B-1~~ ~~(E-40)~~	~~V.B.1-a~~	~~Ducting closure bolting~~	~~Steel~~	~~Air – indoor uncontrolled (External)~~	~~Loss of material/ general, pitting, and crevice corrosion~~	~~A plant-specific aging management program is to be evaluated.~~	~~Yes, plant-specific~~	See General Comment 4
~~V.B-2~~ ~~(E-26)~~	~~V.B.2-a~~ ~~V.B.1-a~~	~~Ducting, piping and components external surfaces~~	~~Steel~~	~~Air – indoor uncontrolled (External)~~	~~Loss of material/ general corrosion~~	~~A plant-specific aging management program is to be evaluated.~~	~~Yes, plant-specific~~	See General Comment 4
~~V.B-8~~ ~~(E-42)~~	~~V.B.~~	~~Piping, piping components, and piping elements~~	~~Steel~~	~~Soil~~	~~Loss of material/ general, pitting, and crevice corrosion~~	~~Chapter XI.M28, "Buried Piping and Tanks Surveillance," or~~ ~~Chapter XI.M34, "Buried Piping and Tanks Inspection"~~	~~No~~ ~~Yes, detection of aging effects and operating experience are to be further evaluated~~	See General Comment 4

V.C Containment Isolation Components

V C	ENGINEERED SAFETY FEATURES Containment Isolation Components							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
~~V.C-1~~ ~~(E-35)~~	~~V.C.1-a~~	~~Containment isolation piping and components external surfaces~~	~~Steel~~	~~Air – indoor uncontrolled (External)~~	~~Loss of material/ general corrosion~~	~~A plant-specific aging management program is to be evaluated.~~	~~Yes, plant-specific~~	See General Comment 4
~~V.C-2~~ ~~(E-30)~~	~~V.C.1-a~~	~~Containment isolation piping and components external surfaces~~	~~Steel~~	~~Condensation (External)~~	~~Loss of material/ general corrosion~~	~~A plant-specific aging management program is to be evaluated.~~	~~Yes, plant-specific~~	See General Comment 4
V.C-5 (E-34)	V.C.1-b	Containment isolation piping and components internal surfaces	Stainless steel	~~Untreated water~~ Raw Water	Loss of material/ pitting, crevice, and microbiologically influenced corrosion, and fouling	A plant-specific aging management program is to be evaluated. See IN 85-30 for evidence of microbiologically influenced corrosion.	Yes, plant-specific	Change to address conforming change in Chapter IX.
V.C-8 (E-32)	V.C.1-a	Containment isolation piping and components internal surfaces	Steel	~~Untreated water~~ Raw Water	Loss of material/ general, pitting, crevice, and microbiologically influenced corrosion	A plant-specific aging management program is to be evaluated. See IN 85-30 for evidence of microbiologically influenced corrosion.	Yes, plant-specific	Change to address conforming change in Chapter IX.
V.C-9 (EP-33)	V.C.	Piping, piping components, and piping elements ***internal surfaces***	Stainless steel	Closed cycle cooling water	Loss of material/ pitting and crevice corrosion	Chapter XI.M21, "Closed-Cycle Cooling Water System"	No	Add ***internal surfaces*** to be consistent with the rest of the chapter.
New V.C(1)	***V.C.***	***Piping, piping components, and piping elements internal surfaces***	***Steel***	***Closed cycle cooling water***	***Loss of material/ general, pitting and crevice corrosion***	***Chapter XI.M21, "Closed-Cycle Cooling Water System"***	***No***	Add carbon steel material. This is the same as lines RP-10 and A-25.

V ENGINEERED SAFETY FEATURES
C Containment Isolation Components

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
New V.C(2) *V.A* *V.D1* *V.D2*		*Piping, piping components, and piping elements*	*Stainless steel*	*Closed cycle cooling water >60°C (>140°F)*	*Cracking/ stress corrosion cracking*	*Chapter XI.M21, "Closed-Cycle Cooling Water System"*	*Yes, plant-specific*	The basis for AP-60 is applicable to components in ESF systems where closed cooling water temperatures exceed 140°F in supporting systems such as bearing and lube oil coolers and associated piping that are considered part of the ESF systems. EP-33 addresses a different aging mechanism for this same group of components.

V.D1 Emergency Core Cooling System (PWR)

Introductory Text

Delete the last sentence of the first paragraph related to pitting and crevice corrosion of stainless steel in a borated water environment.

V D1	ENGINEERED SAFETY FEATURES Emergency Core Cooling System (PWR)							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
~~V.D1-1~~ ~~(E-28)~~	~~V.D1.8-b~~ ~~V.D1.6-d~~ ~~V.D1.1-d~~ ~~V.D1.5-b~~ ~~V.D1.4-c~~ ~~V.D1.2-b~~ ~~V.D1.3-a~~ ~~V.D1.7-a~~	~~External surfaces~~	~~Steel~~	~~Air with borated water leakage~~	~~Loss of material/ boric acid corrosion~~	~~Chapter XI.M10, "Boric Acid Corrosion"~~	~~No~~	See General Comment 4
V.D1-3 (E-19)	V.D1.5-a V.D1.6-a	Heat exchanger components ~~including tubes~~	Stainless steel	Closed cycle cooling water	Loss of material/ pitting and crevice corrosion	Chapter XI.M21, "Closed-Cycle Cooling Water System"	No	See General Comment 5
V.D1-4 (E-20)	V.D1.6-b	Heat exchanger ~~shell side~~ components ~~including tubes~~	Stainless steel	Raw water	Loss of material/ pitting, crevice, and microbiologically influenced corrosion, and fouling	Chapter XI.M20, "Open-Cycle Cooling Water System"	No	See General Comment 5
V.D1-5 (E-18)	V.D1.6-b	Heat exchanger ~~shell side~~ components ~~including tubes~~	Steel	Raw water	Loss of material/ general, pitting, crevice, and microbiologically influenced corrosion, and fouling	Chapter XI.M20, "Open-Cycle Cooling Water System"	No	See General Comment 5
V.D1-6 (EP-13)	V.D1.	Heat exchanger ~~tubes~~ ***components***	Copper alloy	Closed cycle cooling water	Loss of material/ pitting, crevice, and galvanic corrosion	Chapter XI.M21, "Closed-Cycle Cooling Water System"	No	See General Comment 5
V.D1-7 (EP-37)	V.D1.	Heat exchanger ~~tube~~ ***components***	Copper alloy >15% Zn	Closed cycle cooling water	Loss of material/ selective leaching	Chapter XI.M33, "Selective Leaching of Materials"	No	See General Comment 5

V ENGINEERED SAFETY FEATURES
D1 Emergency Core Cooling System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
V.D1-9 (E-21)	V.D1.6-c	Heat exchanger tubes ~~(serviced by open-cycle cooling water)~~	Stainless steel	Raw water	Reduction of heat transfer/ fouling	Chapter XI.M20, "Open-Cycle Cooling Water System"	No	See General Comment 5
V.D1-10 (E-43)	V.D1.	~~Motor Cooler~~ ***Heat exchanger components***	Gray cast iron	Treated water	Loss of material/ Selective leaching	Chapter XI.M33, "Selective Leaching of Materials"	No	See General Comment 5 The motor cooler is the same as a heat exchanger. The change is for consistency.
V.D1-12 (E-01)	V.D1.8-c	Partially encased tanks with breached moisture barrier	Stainless steel	~~Untreated water or~~ Raw water	Loss of material/ pitting and crevice corrosion	A plant-specific aging management program is to be evaluated for pitting and crevice corrosion of tank bottom because moisture and water can egress under the tank due to cracking of the perimeter seal from weathering.	Yes, plant-specific	Change to address conforming change in Chapter IX.
~~V.D1-17~~ ~~(EP-31)~~	~~V.D1.~~	~~Piping, piping components, and piping elements~~	~~Stainless steel~~	~~Soil~~	~~Loss of material/ pitting and crevice corrosion~~	~~A plant-specific aging management program is to be evaluated.~~	~~Yes, plant-specific~~	See General Comment 4
V.D1-19 (E-12)	V.D1.7-b V.D1.8-a V.D1.1-a V.D1.4-b V.D1.2-a	Piping, piping components, piping elements, and tanks	Stainless steel	Treated borated water >60°C (>140°F)	Cracking/ stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR primary water ~~in EPRI TR-105714~~	No	See General Comment 3
~~V.D1-20~~ ~~(E-38)~~	~~V.D1.7-b~~	~~Safety injection tank (accumulator)~~	~~Steel with stainless steel cladding~~	~~Treated borated water >60°C (>140°F)~~	~~Cracking/ stress corrosion cracking~~	~~Chapter XI.M2, "Water Chemistry," for PWR primary water in EPRI TR-105714~~	~~No~~	Is included in Line Item V.D1-19.

V ENGINEERED SAFETY FEATURES
D1 Emergency Core Cooling System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
New V.D1(1)	*V.D1-2a*	*Pump Casings*	*Carbon steel with stainless steel clad*	*Treated borated water*	*Cracking/ under-clad cracking*	*A plant-specific aging management program is to be evaluated.*	*Yes- verify plant specific program addresses clad cracking.*	Unique problems with stainless cladding have been identified for HHSI pumps. Reference NRC Information Notice 94-63, "Boric Acid Corrosion of Charging Pump Casings Caused by Cladding Cracks".
New V.D1(2)		*Piping, piping components, and piping elements*	*Stainless steel*	*Lubricating Oil*	*Loss of material/ pitting and crevice corrosion*	*A plant-specific aging management program is to be evaluated.*	*Yes, plant specific*	Since GALL assumes water pooling for the lubricating oil environment, loss of material is a plausible aging effect that should be managed.
New V.D1(3)		*Heat exchanger components*	*Gray cast iron*	*Closed cycle cooling water*	*Loss of material/ Selective leaching*	*Chapter XI.M33, "Selective Leaching of Materials."*	*No*	
New V.D1(4)		*Piping, piping components, piping elements, and tanks*	*Stainless steel*	*Treated borated water*	*Loss of material/ pitting and crevice corrosion*	*Chapter XI.M2, "Water Chemistry" for PWR primary water*	*No*	This component/ material/environment combination was not found in GALL Table V.D1 The chemistry program is used to manage "Loss of Material" for stainless steel and treated water borated water in at least the Farley, Millstone, North Anna and Surry applications.
New V.D1(5)		*Piping, piping components, piping elements, and tanks*	*Stainless steel*	*Condensation*	*Loss of material/ pitting and crevice corrosion*	*A plant-specific aging management program is to be evaluated.*	*Yes, plant specific*	This is an internal environment. The RWST vent line in Farley application is an example of this material/ environment/aging effect combination.

V ENGINEERED SAFETY FEATURES
D1 Emergency Core Cooling System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
New V.D1(6) *V.A* *V.C* *V.D2*		*Piping, piping components, and piping elements*	*Stainless steel*	*Closed cycle cooling water >60°C (>140°F)*	*Cracking/ stress corrosion cracking*	*Chapter XI.M21, "Closed-Cycle Cooling Water System"*	*Yes, plant-specific*	The basis for AP-60 is applicable to components in ESF systems where closed cooling water temperatures exceed 140°F in supporting systems such as bearing and lube oil coolers and associated piping that are considered part of the ESF systems. EP-33 addresses a different aging mechanism for this same group of components.
New V.D1(7) *V.A* *V.D2*		*Piping, piping components, and piping elements*	*Copper alloy*	*Lubricating oil*	*Loss of material/ pitting and crevice corrosion*	*A plant-specific aging management program is to be evaluated.*	*Yes, plant-specific*	The bases for AP-47 and SP-32 are applicable to components such as pump and motor lubricating oil systems that are included as part of the ESF systems.
New V.D1(8) *V.A* *V.D2*	8	*Piping, piping components, and piping elements*	*Steel*	*Lubricating oil*	*Loss of material/ general, pitting, and crevice corrosion*	*A plant-specific aging management program is to be evaluated.*	*Yes, plant-specific*	The bases for AP-30 and SP-25 are applicable to components such as pump and motor lubricating oil systems that are included as part of the ESF systems.

V ENGINEERED SAFETY FEATURES
D1 Emergency Core Cooling System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
New V.D1(9) *V.A* *V.D2*		*Heat exchanger tubes*	*Copper alloy*	*Lubricating oil*	*Reduction of heat transfer/ fouling*	*A plant-specific aging management program is to be evaluated.*	*Yes, plant-specific*	CNP SER Section 3.2.2.3.3 acknowledged the potential for fouling of copper alloy components in a lubricating oil environment. The SER addressed components in A PWR ECCS system, but these conditions would be equally applicable to components of other lube oil systems. The reduction of heat transfer due to fouling of heat exchanger tubes will be addressed by a plant specific program. (Oil analysis for CNP as shown in Table 3.2.2-3 of the CNP LRA)
New V.D1(10) *V.A* *V.D2*		*Heat exchanger tubes*	*Steel*	*Lubricating oil*	*Reduction of heat transfer/ fouling*	*A plant-specific aging management program is to be evaluated.*	*Yes, plant-specific*	This line is similar to the preceding line except for a different material. Heat exchanger tubes of steel would also be subject to fouling in a lubricating oil environment.
New V.D1(11) *V.A* *V.D2*		*Heat exchanger tubes*	*Stainless steel*	*Lubricating oil*	*Reduction of heat transfer/ fouling*	*A plant-specific aging management program is to be evaluated.*	*Yes, plant-specific*	This line is similar to the preceding line except for a different material. Heat exchanger tubes of stainless steel would also be subject to fouling in a lubricating oil environment.

V ENGINEERED SAFETY FEATURES
D1 Emergency Core Cooling System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
New V.D1(12)		*Piping, piping components, and piping elements*	*Stainless steel*	*Raw water*	*Loss of material/ pitting, crevice, and microbiologically influenced corrosion.*	*A plant-specific aging management program is to be evaluated. See IN 85-30 for evidence of microbiologically influenced corrosion.*	*Yes, plant-specific*	Line E-34 addresses the same material, environment, aging effect and plant specific program for containment isolation piping and components. The proposed line extends the components addressed by line E-34 to include piping and components inside and outside containment, such as ECCS piping that is also used as part of the normal containment sump piping.

V.D2 Emergency Core Cooling System (BWR)

V	ENGINEERED SAFETY FEATURES							
D2	Emergency Core Cooling System (BWR)							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
~~V.D2-2~~ ~~(E-26)~~	~~V.D2.1-e~~ ~~V.D2.5-a~~	~~Ducting, piping and components external surfaces~~	~~Steel~~	~~Air—indoor uncontrolled (External)~~	~~Loss of material/ general corrosion~~	~~A plant-specific aging management program is to be evaluated.~~	~~Yes, plant-specific~~	See General Comment 4
~~V.D2-15~~ ~~(EP-2)~~	~~V.D2.~~	~~Piping, piping components, and piping elements~~	~~Aluminum~~	~~Air with borated water leakage~~	~~Loss of material/ boric acid corrosion~~	~~Chapter XI.M10, "Boric Acid Corrosion"~~	~~No~~	See General Comment 4
~~V.D2-21~~ ~~(EP-31)~~	~~V.D2.~~	~~Piping, piping components, and piping elements~~	~~Stainless steel~~	~~Soil~~	~~Loss of material/ pitting and crevice corrosion~~	~~A plant-specific aging management program is to be evaluated.~~	~~Yes, plant-specific~~	See General Comment 4
V.D2-23 (EP-32)	V.D2.	Piping, piping components, and piping elements	Stainless steel	Treated water	Loss of material/ pitting and crevice corrosion	~~A plant-specific aging management program is to be evaluated.~~ ***Chapter XI.M2, "Water Chemistry" for BWR water.*** ***The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.***	Yes, ~~plant-specific~~ ***detection of aging effects is to be evaluated***	See NUREG 1801 lines VII.E4.1-a and VIII.E.5-b for consistency.

V ENGINEERED SAFETY FEATURES
D2 Emergency Core Cooling System (BWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
V.D2-24 (E-37)	V.D2.3-c V.D2.1-c	Piping, piping components, and piping elements	Stainless steel	Treated water >60°C (>140°F)	Cracking/ stress corrosion cracking and intergranular stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for BWR water, ~~in BWRVIP-29 (EPRI TR-103515)~~ and Chapter XI.M7, "BWR Stress Corrosion Cracking" or ***Chapter XI.M32, "One-Time Inspection***	No	GALL Aging Management Program XI.M7, "BWR Stress Corrosion Cracking" is essentially the applicant's response to Generic Letter 88-01, which required increased ISI inspections and piping material replacement within the reactor coolant pressure boundary. The GALL assigns this Aging Management Program for Chapter V -ECCS systems, specifically in GALL lines V.D2.1-c and V.D2.3-c. Typically the ECCS Systems do not fall under the guidance of GL 88-01. The reactor coolant pressure boundary of the ECCS systems may be covered under GL-88-01, however those portions of the ECCS systems are included under GALL Chapter IV (Reactor Coolant Pressure Boundary), lines IV.C1.1-f and IV.C1.3-c. Therefore, a One-Time Inspection AMP should be available as an alternate choice to verify effectiveness of the water chemistry to prevent cracking. See General Comment 3

V ENGINEERED SAFETY FEATURES
D2 Emergency Core Cooling System (BWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
V.D2-25 (E-07)	V.D2.1-f	Piping, piping components, and piping elements	Steel	~~Air and~~ steam	Wall thinning/ flow-accelerated corrosion	Chapter XI.M17, "Flow-Accelerated Corrosion"	No	Change the environment from "Air and steam" to "Steam". Typically the steam lines to the HPCI and RCIC Turbine, for which this NUREG 1801 line was meant for, is rarely used since these steam lines are used less than 2% of the time and are therefore not susceptible to flow accelerated corrosion. The steam drain lines however, see constant steam flow and are therefore susceptible to flow accelerated corrosion.
V.D2-27 (E-08)	V.D2.2-a V.D2.3-b V.D2.1-a	Piping, piping components, and piping elements	Steel	Treated water	Loss of material/ general, pitting, and crevice corrosion	Chapter XI.M2, "Water Chemistry," ~~for BWR water in BWRVIP-29 (EPRI TR-103515).~~ The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated	See General Comment 3

V ENGINEERED SAFETY FEATURES
D2 Emergency Core Cooling System (BWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
New V.D2(1) *V.A* *V.C* *V.D1*		*Piping, piping components, and piping elements*	*Stainless steel*	*Closed cycle cooling water >60°C (>140°F)*	*Cracking/ stress corrosion cracking*	*Chapter XI.M21, "Closed-Cycle Cooling Water System"*	*Yes, plant-specific*	The basis for AP-60 is applicable to components in ESF systems where closed cooling water temperatures exceed 140°F in supporting systems such as bearing and lube oil coolers and associated piping that are considered part of the ESF systems. EP-33 addresses a different aging mechanism for this same group of components.
New V.D2(2) *V.A* *V.D1*		*Piping, piping components, and piping elements*	*Copper alloy*	*Lubricating oil*	*Loss of material/ pitting and crevice corrosion*	*A plant-specific aging management program is to be evaluated.*	*Yes, plant-specific*	The bases for AP-47 and SP-32 are applicable to components such as pump and motor lubricating oil systems that are included as part of the ESF systems.
New V.D2(3) *V.A* *V.D1*	8	*Piping, piping components, and piping elements*	*Steel*	*Lubricating oil*	*Loss of material/ general, pitting, and crevice corrosion*	*A plant-specific aging management program is to be evaluated.*	*Yes, plant-specific*	The bases for AP-30 and SP-25 are applicable to components such as pump and motor lubricating oil systems that are included as part of the ESF systems.

V ENGINEERED SAFETY FEATURES
D2 Emergency Core Cooling System (BWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
New V.D2(4) *V.A* *V.D1*		*Heat exchanger tubes*	*Copper alloy*	*Lubricating oil*	*Reduction of heat transfer/ fouling*	*A plant-specific aging management program is to be evaluated.*	*Yes, plant-specific*	CNP SER Section 3.2.2.3.3 acknowledged the potential for fouling of copper alloy components in a lubricating oil environment. The SER addressed components in A PWR ECCS system, but these conditions would be equally applicable to components of other lube oil systems. The reduction of heat transfer due to fouling of heat exchanger tubes will be addressed by a plant specific program. (Oil analysis for CNP as shown in Table 3.2.2-3 of the CNP LRA)
New V.D2(5) *V.A* *V.D1*		*Heat exchanger tubes*	*Steel*	*Lubricating oil*	*Reduction of heat transfer/ fouling*	*A plant-specific aging management program is to be evaluated.*	*Yes, plant-specific*	This line is similar to the preceding line except for a different material. Heat exchanger tubes of steel would also be subject to fouling in a lubricating oil environment.
New V.D2(6) *V.A* *V.D1*		*Heat exchanger tubes*	*Stainless steel*	*Lubricating oil*	*Reduction of heat transfer/ fouling*	*A plant-specific aging management program is to be evaluated.*	*Yes, plant-specific*	This line is similar to the preceding line except for a different material. Heat exchanger tubes of stainless steel would also be subject to fouling in a lubricating oil environment.

V.E External Surfaces of Components and Miscellaneous Bolting

Introductory Text

Revise VE Page VE-1 as follows: This section includes the aging management programs for the degradation of external surfaces of ~~of all carbon steel~~ structures and components including closure boltings in the engineered safety features systems of ~~in~~-pressurized water reactors (PWRs) and boiling water reactors (BWRs). ~~For the carbon steel components in PWRs, this section addresses only boric acid corrosion of external surfaces as a result of the dripping borated water that is leaking from an adjacent PWR component. Boric acid corrosion can also occur for carbon steel components containing borated water due to leakage; such components and the related aging management program are covered in the appropriate major plant sections in V.~~

V ENGINEERED SAFETY FEATURES
E External Surfaces of Components and Miscellaneous Bolting

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
V.E-1 (EP-1) ***(E-45)***	V.E.	***External Surfaces including closure*** bolting	Steel	Air – outdoor (External)	Loss of material/ general, pitting, and crevice corrosion	~~Chapter XI.M18, "Bolting Integrity"~~ ***"Chapter XI.M##, External Surfaces Monitoring***	No	Credit previously accepted "External Surfaces Monitoring" AMP. Consolidate listings by including E-45 (V.E-8)
V.E-2 (E-41) (E-28)	V.E.1-a ***V.A.3-b*** ***V.A.1-b*** ***V.A.4-b*** ***V.A.6-d*** ***V.A.5-b*** ***V.D1.8-b*** ***V.D1.6-d*** ***V.D1.1-d*** ***V.D1.5-b*** ***V.D1.4-c*** ***V.D1.2-b*** ***V.D1.3-a*** ***V.D1.7-a***	***External Surfaces Including closure*** bolting	Steel	Air with borated water leakage	Loss of material/ boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No	Consolidate listings by including E-28 (V.E-9). Relocated V.A-2, V.D1-1
V.E-3 (E-03) ***(E-02)***	V.E.2-b ***V.E.2-a***	Closure bolting	~~High-strength s~~ Steel > 150 ksi	Air with steam or water leakage	Cracking/ cyclic loading, stress corrosion cracking	"Chapter XI.M18, "Bolting Integrity"	No	Credit previously accepted "External Surfaces Monitoring" AMP. Consolidate listings by including E-02 (V.E-6)?

V ENGINEERED SAFETY FEATURES
E External Surfaces of Components and Miscellaneous Bolting

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
V.E-4 (EP-25) *(EP-24)* *(E-44)* *(E-26)* *(E-40)* *(E-35)*	V.E. ***V.A.5-a*** ***V.A.2-a*** ***V.B.2-a*** ***V.B.1-a*** ***V.C.1-a*** ***V.D2.1-e*** ***V.D2.5-a***	***External Surfaces including closure*** bolting	Steel	Air – indoor uncontrolled (External)	Loss of material/ general, pitting, and crevice corrosion	~~"Chapter XI.M18, "Bolting Integrity"~~ ***"Chapter XI.M##, External Surfaces Monitoring"***	No	Credit previously accepted "External Surfaces Monitoring" AMP. Consolidate listings by including E-24 (V.E-5), E-44 (V.E-7). Relocated V.A-1, V.B-1, V.B-2, V.C-1, V.D2-2
~~V.E-5~~ ~~(EP-24)~~	~~V.E.~~	~~Closure bolting~~	~~Steel~~	~~Air – indoor uncontrolled (External)~~	~~Loss of preload/ stress relaxation~~	~~Chapter XI.M18, "Bolting Integrity"~~	~~No~~	Loss of preload is not an aging effect. See General Comment 6
V.E-10 (E-46) (E-30)	V.E. ***V.C.1-a***	External surfaces	Steel	Condensation (External)	Loss of material/ General corrosion	~~A plant-specific aging management program is to be evaluated.~~ ***"Chapter XI.M##, External Surfaces Monitoring***	~~Yes, plant-specific~~ ***No***	Credit previously accepted "External Surfaces Monitoring" AMP. Relocated V.C-2
New V.E(1) (E-42)	***V.B.***	***Piping, piping components, and piping elements***	***Steel***	***Soil***	***Loss of material/ general, pitting, and crevice corrosion***	***Chapter XI.M28, "Buried Piping and Tanks Surveillance," or*** ***Chapter XI.M34, "Buried Piping and Tanks Inspection"***	***No*** ***Yes, detection of aging effects and operating experience are to be further evaluated***	Relocated V.B-8
New V.E(2) (EP-31)	***V.D1.***	***Piping, piping components, and piping elements external surfaces***	***Stainless steel***	***Soil***	***Loss of material/ pitting and crevice corrosion***	***A plant-specific aging management program is to be evaluated.***	***Yes, plant-specific***	Relocated V.D1-17, V.D2-21
New V.E(3) (EP-2)	***V.D2.***	***Piping, piping components, and piping elements external surfaces***	***Aluminum***	***Air with borated water leakage***	***Loss of material/ boric acid corrosion***	***Chapter XI.M10, "Boric Acid Corrosion"***	***No***	Relocated V.D2-15

V ENGINEERED SAFETY FEATURES
E External Surfaces of Components and Miscellaneous Bolting

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
New V.E(4) *(E-06)*		*Elastomer seals*	*Elastomers*	*Air – indoor uncontrolled*	*Hardening and loss of strength/ elastomer degradation*	*A plant-specific aging management program is to be evaluated.*	*Yes, plant-specific*	Relocated V.B-4
New V.E(5)x		*Piping, piping components, and piping elements*	*Gray cast iron*	*Soil*	*Loss of material/ selective leaching*	*Chapter XI.M33, "Selective Leaching of Materials"*	*No*	The bases for AP-42 and SP-26 are applicable for buried gray cast iron components of the standby gas treatment system. E-42 added steel in soil for standby gas treatment system components. This proposed line addresses gray cast iron as a subset of those steel components.

V.F Common Miscellaneous Material Environment Combinations

V F	ENGINEERED SAFETY FEATURES Common Miscellaneous Material Environment Combinations							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
~~V.F-3~~ ~~(EP-8)~~	~~F.~~	~~Piping, piping components, and piping elements~~	~~Cast austenitic stainless steel~~	~~Air – indoor uncontrolled (External)~~	~~None~~	~~None~~	~~No~~	CASS covered by EP-18 entry for stainless steel
V.F-18 (EP-21)	V.F.	Piping, piping components, and piping elements	Stainless steel	Lubricating oil ***(No water pooling)***	None	None	No	GALL assumes water pooling for the lubricating oil environment. The industry assumes that if water pooling is present, loss of material is a plausible aging effect that should be managed. To support a finding of no aging effect, the environment should be annotated to exclude water pooling.
New V.F(1)		***Piping, piping components, piping elements,***	***Steel***	***Containment environment (inert)***	***None***	***None***	***No***	***See Chapter IX comment on Containment environment (inert).***

ELECTRICAL COMMENTS (VI, IX, XI)

The comments in this section cover the following areas:

- Draft NUREG-1801, Volume1, Table 6
- Draft NUREG-1801, Volume 2, Chapter VI
- Draft NUREG-1801, Volume 2, Chapter IX
- Draft NUREG-1801, Volume 2, Chapter XI

The purpose of GALL is to give credit for programs that have already been previously approved.

Comments made regarding GALL are not duplicated for the SRP but GALL changes should be reflected in the SRP where applicable.

Draft NUREG-1801, Rev. 1, Vol. 1

Table 6. Summary of Aging management Programs for the Electrical Components Evaluated in Chapter VI of the GALL Report

ID 11 – High voltage insulators: "Degradation of insulation quality due to presence of any salt deposits and surface contamination" is a condition that can be caused in a matter of hours under the right environmental condition rather than an aging effect that changes the long-term physical properties of high voltage insulators. Also, "loss of material caused by mechanical wear due to wind blowing on transmission conductors" has not been indicated by operating experience.

ID 12 – Transmission conductors and connections. Switchyard bus and connections: "Loss of material due to wind induced abrasion and fatigue" has not been indicated by operating experience. An aging management program is not needed for this component based on previously approved staff positions along with operating experience and testing that has demonstrated greater than an 80-year service life.

ID 13 – Cable Connections Metallic Parts: This is a proposed AMP that has not been published previously, has not been proposed in a draft or final ISG, and has not been required for any previously approved license renewal application. This AMP should be eliminated as these items either have no aging effects requiring management or are adequately covered by other AMPs. Operating experience does not indicate a need for this proposed program.

Draft NUREG-1801, Rev. 1, Vol. 2

New Aging Management Programs

ISG-17 and XI.E4 Periodic Inspection of Bus Ducts Program

A new inspection program is proposed to monitor bus duct, more properly called metal-enclosed bus (MEB). The staff's proposed program was issued as ISG-17 in

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the Federal Register on December 23, 2004, and also as XI.E4 in the GALL, each with a different comment period. An alternate to the proposed GALL AMP is provided with this document.

Since no MEB aging management program has previously been required for nuclear plants with renewed licenses, there should be provisions for a licensee to show that the materials and environment for MEB do not produce the stressors that cause aging effects requiring management and an AMP is therefore not required.

XI.E5 Aging Management Program for Fuse Holders

A new aging management program is proposed to monitor fuse holders as a special type of terminal block. Fuse holders and terminal blocks are already covered in XI.E1, "Electrical Cables and Connections Not Subject To 10 CFR 50.49 Environmental Qualification Requirements," and listed in NUREG-1800, Table 2.1.5, Item 77, page 2.1-22 as part of the commodity group "Cables and Connections, Bus, Electrical portions of Electrical and I&C Penetration Assemblies." There should not need to be a special program for a specific type of a subcomponent of another AMP, if that program adequately covers the subcomponent. Most fuse holders are a subcomponent of active equipment and not subject to aging management review.

Since no fuse holder aging management program has previously been required for nuclear plants with renewed licenses, there should be provisions for a licensee to show that the materials and environments for fuse holders do not produce stressors that cause aging effects requiring management and an AMP is therefore not required. One example of these conditions would be a fuse panel installed to the manufacturer's specifications in a dry environment without sources of water in the vicinity, with both normal and peak loadings less than the rated amperage for the fuse at the design voltage and the fuses are not routinely removed for circuit isolation purposes.

Visual inspection of fuse holders is sufficient to manage the effects of aging on fuse holders that are not subject to mechanical abuse. Testing should not be needed for fuse holders designed and constructed to UL standards that do not show signs of excessive temperature (*discoloration, scorching, burned surfaces*).

There is no technical or operating experience basis for the statement that "... failures of a deteriorated cable system (cables, connections including fuse holders, and penetrations) might be induced during event conditions." In addition, fuse holders are included in the EQ programs of many plants, similar to terminal blocks and other non-cable electrical components within the scope of 10 CFR 50.49.

Operating experience has not shown significant failures due to fuse holder aging, but rather failures due to removal and reinsertion of fuses for circuit isolation purposes. This use of fuses is a design, procedural, or operational deficiency, for fuse holders that are not designed for this type of use.

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ELECTRICAL COMMENTS (VI, IX, XI)

XI.E6 Aging Management Program for Electrical Cable Connections

The GALL revision proposes a new aging management program for cable connections, including splices (butt or bolted), crimp-type ring lugs, and terminal blocks. Cable connections, connectors, and terminal blocks are already covered in XI.E1, "Electrical Cable Connections Not Subject To 10 CFR 50.49 Environmental Qualification Requirements," and listed in NUREG-1800, Table 2.1.5, Item 77, page 2.1-22 as part of the commodity group "Cables and Connections, Bus, Electrical portions of Electrical and I&C Penetration Assemblies." There should not be a special program for this subcomponent that is already included in another AMP.

Most connections are essentially inaccessible as they are taped, heat shrink covered, or enclosed in an engineered splice kit; or they exist as a part of, or internal to, other active equipment. Therefore, if a program is required, an exemption should clearly state that connections to or part of an active component should not subject to this AMP, similar to the exemption for fuse holders in an active component or panel.

Operating experience has not shown that there are a significant failures or a high failure rate due to connection aging. "Available failure data for connectors show that the number of connector failures is very low in proportion to the large number of connectors found in nuclear plants.[1]" Generic failure rates for Mil Spec[2] (comparable to nuclear QA) crimp, weld, clip, and terminal block connections are 0.000015 to 0.062 per million hours of service in a benign, ground-based environment. Two-thirds of the aging-related failures reported in NPRDS for 1977 to 1994 were in low-voltage or neutron monitor connectors. The XI.E2 AMP already manages the aging of these connectors. At low voltages, the XI.E1 AMP manages the aging of terminal blocks and splice/insulation connections. Low-voltage compression/fusion fittings are typically used at the end connection to active components in instrumentation and control circuits where a failure is readily apparent. As subcomponents of an active assembly, such connections are not subject to aging management review.

References:

1. "Electrical Connectors Application Guidelines," EPRI Technical Report 1003471, December 2002.
2. "Reliability Prediction of Electronic Equipment," MIL-HDBK-217F, December 1991.

Chapter VI, Electrical Components, Section A

The listing of medium-voltage cables in the range of 2kV to 15kV is misleading. While this range adequately covers most plants, several plants have extruded insulation cables with applied voltages above 15kV that are considered medium-voltage cables. Systems with applied voltages of 13.8kV and 15kV typically use cables constructed and rated at 25kV or higher. Applied medium-voltage levels may be as high as 34.9kV using cables rated above that value and of the same basic construction and materials as cables used at the lower end of the medium-voltage range. Therefore, the statement that "High voltage (>15kV) power cables and connections have unique, specialized

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constructions and must be evaluated on an application specific basis." is not appropriate based on actual cable construction and power industry use.

Table VI-A (also Draft NUREG-1800, Rev. 1, Section 3.6.2.2.5)

Item VI.A-9 (LP-07) (also Draft NUREG-1800, Rev. 1, Sections 3.6.2.2.5 and 3.6.3.2.5)
Thirteen (13) SERs covering 26 reactors have been issued without a single program required for managing aging effects for high voltage insulators. Another 7 applications in review covering 16 reactors do not propose a program for managing aging effects for high voltage insulators. GALL should not state aging effects or request "plant specific" programs when the industry has not identified aging effects that require management for high voltage insulators. The operating experience for this electrical commodity goes back as much as 80 years, without indications that there is an aging affect requiring management for license renewal.

Salt spray deposits can be caused in a matter of hours under the right temporary environmental conditions. Such deposits are the result of environmental conditions rather than aging effects, since they occur in the short term and do not degrade the electrical or mechanical properties of the porcelain insulating material or its support structure. Specific environmental conditions create the deposits and any significant moisture, such as fog or rain, eliminates the problem. Other external substances could cause a temporary electrical path to be formed at an insulator, including dust, rain, or an animal, none of which are aging related.

Item VI.A-10 (LP-11) (also Draft NUREG-1800, Rev. 1, Sections 3.6.2.2.5 and 3.6.3.2.5)
A separate item is not needed for high voltage insulators since there are no aging effects requiring management. There is no operating experience to support "loss of material/mechanical wear due to wind blowing on transmission conductors" as an aging effect for high-voltage insulators.

Item VI.A-11 through VI.A-14 (LP-04, LP-10, LP-06, LP-05)
The proper electrical and industry standard (IEEE and ANSI) designation for this equipment is "Metal-Enclosed Bus" and should be used in any reference or regulation. Bus duct typically refers only to the outside enclosure around the electrically conducting bus bar, its supports, and insulating assemblies. The aging effect for phase bus should be either "increased contact resistance" or "loss of electrical continuity". See the proposed industry alternate metal-enclosed bus aging management program.

Item VI.A-15 (LP-09) (also Draft NUREG-1800, Rev. 1, Sections 3.6.2.2.6 and 3.6.3.2.6)
The industry has not identified aging effects requiring management for switchyard bus and connectors. The operating experience for this electrical commodity goes back as much as 80 years in the utility industry, without indications that there is an aging effect requiring management for the period of extended operation. Thirteen (13) SERs covering 26 reactors have been issued without a single program required for managing aging effects for switchyard bus and connectors. Another seven (7) applications under review covering 16 reactors do not propose a program for managing aging effects for

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switchyard bus and connectors. GALL should state that aging effects are minimal, there is no need for a plant-specific AMP, and no further evaluation is required. License renewal experience indicates that future applications will not be consistent with GALL for this item.

Specific Aging Effect/Mechanism:
There is no operating experience to support "Loss of material/wind induced abrasion and fatigue" as an aging effect/mechanism for switchyard bus and connections. Switchyard bus and connections are rigidly supported conductors that interconnect inherently rugged electrical components that are rigidly supported on concrete foundations or connected by flexible conductors to equipment that does not normally vibrate. The buses are supported by insulators and ultimately by static, structural components such as structural steel and concrete footings.

Switchyard bus is typically constructed of high quality aluminum pipe or tubing and fittings that are swaged, welded, or bolted together and to the electrical components they interconnect. Aluminum in an "Air – outdoor" environment quickly forms a protective oxidation layer that resists further corrosion. Industry operating experience does not support "Loss of conductor strength/corrosion" as an aging effect/mechanism for switchyard bus and connections that would result in a loss of intended function during the period of extended operation.

Similarly, industry operating experience does not support "Increased resistance of connection/oxidation or loss of pre-load" as an aging effect/mechanism for switchyard bus and connections that would result in a loss of intended function during the period of extended operation. Pre-load of bolted connections is maintained by appropriate design features such as Belleville washers.

Item VI.A-16 (LP-08)
Thirteen (13) SERs covering 26 reactors have been issued without a single program required for managing aging effects for transmission conductors. Another seven (7) applications under review covering 16 reactors do not propose a program for managing aging effects for transmission conductors.

The most prevalent mechanism contributing to loss of conductor strength of an ACSR (aluminum conductor steel reinforced) transmission conductor is corrosion, which includes corrosion of the steel core and aluminum strand pitting. For ACSR conductors, degradation begins as a loss of zinc from the galvanized steel core wires. Corrosion rates depend largely on air quality, which includes suspended particles chemistry, SO_2 concentration in air, precipitation, fog chemistry and meteorological conditions [IEEE Transactions on Power Delivery, "Aged ACSR Conductors, Part I – Testing Procedures for Conductors and Line Elements", Vol. 7, No. 2, April 1992, D. G. Harvard, G. Bellamy, P. G. Buchan, H. A. Ewing, et. al., The Institute of Electrical and Electronic Engineers, Inc., pages 581, 584]. Tests performed by Ontario Hydroelectric showed a 30% loss of composite conductor strength of an 80-year-old ACSR conductor due to corrosion.

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ELECTRICAL COMMENTS (VI, IX, XI)

There is set percentage of composite conductor strength established at which a transmission conductor is replaced. As illustrated below, there is ample strength margin to maintain the transmission conductor intended function through the extended period of operation.

The National Electrical Safety Code (NESC) requires that tension on installed conductors be a maximum of 60% of the ultimate conductor strength. The NESC also sets the maximum tension a conductor must be designed to withstand under heavy load requirements, which includes consideration of ice, wind and temperature. These requirements were reviewed concerning the specific conductors included in a typical plant aging management review. The conductors with the smallest ultimate strength margin are 4/0 ACSR, which will be used for illustration.

The ultimate strength and the NESC heavy load tension limit of 4/0 ACSR are 8350 lbs. and 2761 lbs., respectively. The margin between the NESC heavy load limit and the ultimate strength is 5589 lb. (i.e., there is a 67% of ultimate strength margin). The Ontario Hydroelectric study showed a 30% loss of composite conductor strength in an 80-year-old conductor. In the case of the 4/0 ACSR transmission conductors, a 30% loss of ultimate strength would mean that there would still be a 37% ultimate strength margin between what is required by the NESC and the actual conductor strength. The 4/0 ACSR conductors have the lowest initial design margin of any transmission conductors included in the aging management review. This illustrates with reasonable assurance the transmission conductors will have ample strength through the period of extended operation.

Corrosion of ACSR conductors is a very slow acting mechanism that is even slower in rural areas typical of nuclear plant sites, with generally fewer suspended particles and lower SO_2 concentrations in the air than in urban areas. Therefore there are no applicable aging effects due to corrosion that could affect the intended function of the transmission conductors for the period of extended operation.

Specific Aging Effect/Mechanism:
There is no operating experience to support "Loss of material/wind induced abrasion and fatigue" as an aging effect for transmission conductors and connections. Loss of material due to mechanical wear could be an aging effect for transmission conductors and connections if they are subject to significant movement. Movement of connections can be caused by wind blowing the connected transmission conductor, causing it to swing from side to side. Although this mechanism is possible, industry experience has shown that transmission conductors do not normally swing and that when they do, due to a substantial wind, they do not continue to swing for very long once the wind has subsided. Industry operating experience does not support "Loss of material/wind induced abrasion and fatigue" as an aging effect/mechanism for transmission conductors and connections that would result in a loss of intended function during the period of extended operation.

Similarly, industry operating experience does not support "Increased resistance of connection/oxidation or loss of pre-load" as an aging effect/mechanism for switchyard

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bus and connections that would result in a loss of intended function during the period of extended operation. Pre-load of bolted connections is maintained by appropriate design features such as Belleville washers.

Chapter IX, Selected Definitions of Terms Used for Standardizing Structures, Components, Materials, Environments, Aging Effects, and Aging Mechanisms

B. Structures and Components
Table IX.B, page IX-3
High voltage insulators: The definition of "High-voltage insulators" for license renewal should include the insulator used to support the bus bar in metal-enclosed bus at low and medium-voltages.

C. Materials
Draft Bases Document for NUREG-1801 Rev. 1: page 64 (LP-07) & page 66 (LP-11) shows "Malleable iron" under materials. This material should be included in Table IX.C (page IX-7) under "Steel" of Draft NUREG-1801 Rev. 1, Vol. 2 so that this material is defined under steel. Otherwise, malleable iron is defined nowhere else in the NUREGs.

Table IX.C, page IX-5
Glass: The description of glass should include porcelain.

D. Environments
Temperature threshold of 95°F (35°C) for thermal stresses in elastomers, Page IX-8: When applied to the elastomers used in cable insulation it should be noted that most cable insulation is manufactured as either 75°C (167°F) or 90°C (194°F) rated material.

E. Aging Effects
Table IX.E, page IX-14
Degradation of Insulator Quality: The description of Degradation of Insulator Quality should note that a decrease in insulating capacity resulting from the presence of salt deposits or surface contamination is not a permanent change to the material properties of an insulator, but the result of temporary, transient environmental conditions, and therefore not an aging mechanism or aging effect.

Table IX.E, page IX-15
Fatigue: The description of Fatigue in copper fuse holder clamps should note that electrical transients beyond the rating of the fuse holder are events or design-driven issues and are not part of the normal environment evaluated in the aging management review. If a transient is within the rating of the fuse holder, then it does not contribute to degradation of the fuse holder. The designs of fuse holders to meet the ANSI/UL 512 standard require them to have the ability to withstand thousands of fuse insertions and removals without failure. Therefore, when a fuse and fuse holder are used solely for circuit protection, there is no credible aging mechanism due to infrequent fuse manipulation.

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ELECTRICAL COMMENTS (VI, IX, XI)

Loosening of bolted connections: Bolted connections for bus bar in metal-enclosed bus are designed for thermal cycling caused by ohmic heating up to the rated loading of the bus, and thus thermal cycling during service up to the rated loading is not an aging mechanism. Ohmic heating due to bus overload beyond the design rating is a beyond design basis event rather than a normal stressor evaluated in the aging management review.

Table IX.E, page IX-16
Loss of Material: The description of Loss of Material should not include high-voltage insulators, loss of material attributed to mechanical wear or wind-induced abrasion and fatigue due to wind blowing on transmission conductors. There are no high-voltage insulator failures attributed to this effect or operating experience to support loss of material from high-voltage insulators.

F. Aging Mechanisms
Table IX.F, page IX-20
Electrical Transients: The description of Electrical Transients used here and in Table IX.E, page IX-15, does not quantify the transient or provide a threshold for this as an aging mechanism. If the transient does not result in sufficient ohmic heating to affect the physical properties of the fuse clamp metal, no aging effect will occur.

Table IX.F, page IX-23
Fatigue is not a subset of Mechanical Wear in any practical or theoretical manner. Wear is a physical removal of material at the surface of a metal and fatigue is a change in the internal physical properties of the metal at the grain boundary level. This is not a proper description of an aging mechanism for fuse holders or electrical lines due to wind blowing on transmission conductors.

Ohmic heating is induced by current flow through a conductor can be calculated using first principles of electricity and heat transfer. Ohmic heating is not restricted only to conductors passing through electrical penetrations.

Table IX.F, page IX-24
The description of Presence of any Salt Deposits should not include degradation of insulator quality since surface contamination is a short-term transient condition that does not change the material properties of an insulator. The effects of such deposits are relevant to the current license term and as such, should be addressed under 10 CFR 50.

Table IX.F, page IX-26
Based on analysis of medium-voltage underground cable failures, electric trees that may result from water trees that penetrate the insulation are not the predominant cause of failures in underground polymeric-jacketed cables. This issue is being addressed by an NEI task force on medium-voltage underground cables as a separate concern under 10 CFR 50 in response to the February 5, 2004, letter from Jose Calvo (NRC).

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ELECTRICAL COMMENTS (VI, IX, XI)

Page XI E-5
Program XI.E2, Basis 4: A proven system test for this AMP only need be committed to and generally described to meet the GALL requirement. The test does not have to be either specifically chosen or "... justified in the application." However, an appropriate test must be chosen and initially performed prior to the period extended operation. Ten years is not a frequency rather it should be stated as the interval between tests. Frequency is the number of occurrences per unit of time, not the unit of time.

Other Items
The NRC has become overly prescriptive regarding the specific content of the "Corrective Actions" aging management program (AMP) element, such as requiring an engineering evaluation and listing specific steps that may not be applicable to the as-found condition, when the standard requirements of 10 CFR 50, Appendix B, have already been deemed acceptable by the staff to address corrective actions in prior LR applications.

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Chapter VIII Steam and Power Conversion Systems

General Comments

1. **New Line Items**

   Proposed new lines items are listed at the end of each table. The new lines are designated "New VIII.X(Y) where X is the table identifier and Y is a sequential number for lines in that table. Where the same line is proposed in multiple tables, the other tables are listed below the designation.

2. **Not Used**

3. **Water Chemistry Reference**

   The reference to the specific EPRI document need not be included in the Aging Management Program column. This information is identified in the AMP description in Chapter XI of GALL.

4. **External Environments**

   Introductory text to Chapter VIII systems tables refers to an external surfaces table H at the end of the chapter for aging management programs for the degradation of external surfaces of components and miscellaneous bolting. However, many external surfaces and external bolting entries are still within the individual systems tables.

   The changes proposed below move most external surface and external bolting lines to the external table. Any lines which refer to a unique aging management program (e.g., Fire Protection Program to monitor fire doors) will remain with the system table.

5. **Heat Exchanger Components Description**

   Many GALL items referring to heat exchanger components indicate whether the line applies to the tube side or the shell side. However, the designation of the tube side or shell side of a heat exchanger unnecessarily limits the applicability of the GALL line item. Small heat exchangers in particular can be configured with the cooled fluid on either the shell or tube side. For all heat exchangers with a given set of materials and environments, the configuration of the heat exchanger (tube side vs. shell side) will not alter the aging effects or the aging management programs. Consequently, the component descriptions for these lines should be changed to delete the tube side or shell side designation. Similarly, some lines list heat exchanger components including tubes. For all aging effects other than the reduction of heat transfer, tubes may be considered with all other heat exchanger components. Thus, for aging effects other than reduction of heat transfer, the component description should be "Heat exchanger components." The descriptor "Heat exchanger tubes" should be used when addressing reduction of heat transfer.

6. Bolting

Although some utilities have conservatively applied loss of preload as aging effect for bolting, the industry does not consider loss of preload as an aging effect requiring management. In accordance with EPRI 1003056, "Mechanical Tool," Appendix F, loss of preload is a design driven effect and not an aging effect requiring management. The bolting at most facilities is standard grade B7 carbon steel, or similar material, except in rare specialized applications. Loss of preload due to stress relaxation (creep) for this material can only be a concern in very high temperature applications (> 700°F) as stated in the ASME Code Section II Part D Table 4 Note 4. However, there is no bolting used in BWRs and PWRs that operate at 700°F, with the exception of unique applications, such as the emergency diesel generator exhaust. Therefore, loss of preload due to stress relaxation (creep) is not a valid aging effect.

In addition, the industry has taken actions to address NUREG –1339, "Resolution to Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants." Licensees have implemented good bolting practices in accordance with those referenced in EPRI NP-5769, EPRI NP-5067 and EPRI TR-104213 in normal maintenance and design activities. Normal maintenance and design activities thus address the potential for loss of preload such that it is not a concern for the current or extended operating term. Proper joint preparation and make-up in accordance with industry standards precludes loss of preload. Even other design factors that could contribute to a loss of preload in closure bolting applications, such as vibration, should not result in loosening in a properly designed and assembled bolted joint.

The impact to the GALL tables is that, with elimination of loss of preload as an aging effect, closure bolting has the same MEAP as external surfaces and the lines could be combined.

VII A	AUXILIARY SYSTEMS Steam Turbine System							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
VIII.A-4 (S-04)	VIII.A.2-b VIII.A.1-b	Piping, piping components, and piping elements	Steel	Steam	Loss of material/ general, pitting, and crevice corrosion	Chapter XI.M2, "Water Chemistry," for BWR water ~~in BWRVIP-29 (EPRI TR-103515).~~ The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated	See General Comment 3
VIII.A-5 (S-06)	VIII.A.1-b VIII.A.2-b	Piping, piping components, and piping elements	Steel	Steam	Loss of material/ general, pitting, and crevice corrosion	Chapter XI.M2, "Water Chemistry," for PWR secondary water ~~in EPRI TR-102134~~ The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated	See General Comment 3

VII AUXILIARY SYSTEMS
A Steam Turbine System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
New VIII.A(1) *(SP-45)*	*VIII.A1.*	*Piping, piping components, and piping elements*	*Stainless steel*	*Steam*	*Cracking/ stress corrosion cracking*	*Chapter XI.M2, "Water Chemistry," for BWR water* *The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.*	*No*	Stainless steel should be added to the steam turbine system, just as it has been added to the main steam system. This line is the same as SP-45.
New VIII.A(2) *(SP-44)*	*VIII.A1.*	*Piping, piping components, and piping elements*	*Stainless steel*	*Steam*	*Cracking/ stress corrosion cracking*	*Chapter XI.M2, "Water Chemistry," for PWR secondary water*	*No*	Stainless steel should be added to the steam turbine system, just as it has been added to the main steam system. This line is the same as SP-44.
New VIII.A(3) *(SP-46)*	*VIII.A1.*	*Piping, piping components, and piping elements*	*Stainless steel*	*Steam*	*Loss of material/ pitting and crevice corrosion*	*Chapter XI.M2, "Water Chemistry," for BWR water*	*No*	Stainless steel should be added to the steam turbine system, just as it has been added to the main steam system. This line is the same as SP-46.
New VIII.A(4) *(SP-43)*	*VIII.A1.*	*Piping, piping components, and piping elements*	*Stainless steel*	*Steam*	*Loss of material/ pitting and crevice corrosion*	*Chapter XI.M2, "Water Chemistry," for PWR secondary water*	*No*	Stainless steel should be added to the steam turbine system, just as it has been added to the main steam system. This line is the same as SP-43.
New VIII.A(5) *(SP-31)*		*Piping, piping components, and piping elements*	*Copper alloy*	*Raw water*	*Loss of material/ pitting, crevice, and microbiologicall y influenced corrosion*	*Chapter XI.M20, "Open-Cycle Cooling Water System"*	*No*	Addresses the exciter and isophase bus coolers for the turbines. This line is the same as SP-31.
New VIII.A(6) *(SP-30)*		*Piping, piping components, and piping elements*	*Copper alloy >15% Zn*	*Raw water*	*Loss of material/ selective leaching*	*Chapter XI.M33, "Selective Leaching of Materials"*	*No*	Models the exciter and isophase bus cooler for the turbines. This line is the same as SP-30.

VII AUXILIARY SYSTEMS
A Steam Turbine System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
New VIII.A(7)-X *(SP-27)*		*Piping, piping components, and piping elements*	*Gray cast iron*	*Treated water*	*Loss of material/ selective leaching*	*Chapter XI.M33, "Selective Leaching of Materials"*	*No*	Need to address loss of material due to selective leaching for gray cast iron. This line is the same as SP-27
New VIII.A(8) *(SP-28)*		*Piping, piping components, and piping elements*	*Gray cast iron*	*Untreated water*	*Loss of material/ selective leaching*	*Chapter XI.M33, "Selective Leaching of Materials"*	*No*	Need to address loss of material due to selective leaching for gray cast iron. This line is the same as SP-28.
New VIII.A(9) *(S-23)*		*Heat exchanger Components*	*Steel*	*Closed cycle cooling water*	*Loss of material/ general, pitting, and crevice corrosion*	*Chapter XI.M21, "Closed-Cycle Cooling Water System"*	*No*	This addresses the stator coolers. This line is the same as S-23.
New VIII.A(10) *VIII.F*		*Piping, piping components, and piping elements*	*Copper alloy*	*Treated water*	*Loss of material/general, pitting, and crevice corrosion*	*Chapter XI.M2, "Water Chemistry"*	*No*	This line compliments new line VIII.F(1). This addresses loss of material other than selective leaching. This effect can apply to all copper alloys.

VIII.B Main Steam System (PWR)

VII AUXILIARY SYSTEMS
B1 Main Steam System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
VIII.B1-1 (SP-18)	VIII.B1.	Piping, piping components, and piping elements	Nickel-based alloys	Steam	Loss of material/ pitting and crevice corrosion	Chapter XI.M2, "Water Chemistry," for PWR secondary water ~~in EPRI TR-102134~~	No	See General Comment 3
VIII.B1-2 (SP-44)	VIII.B1.	Piping, piping components, and piping elements	Stainless steel	Steam	Cracking/ stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR secondary water ~~in EPRI TR-102134~~	No	See General Comment 3
VIII.B1-3 (SP-43)	VIII.B1.	Piping, piping components, and piping elements	Stainless steel	Steam	Loss of material/ pitting and crevice corrosion	Chapter XI.M2, "Water Chemistry," for PWR secondary water ~~in EPRI TR-102134~~	No	See General Comment 3
VIII.B1-4 (S-08)	VIII.B1.1-b	Piping, piping components, and piping elements	Steel	Steam ***or treated water***	Cumulative fatigue damage/ fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA	Fatigue may not be limited to piping with an internal steam environment. Treated water added to extend applicability; however, an environment of "Any" might be more appropriate.
VIII.B1-5 (S-07)	VIII.B1.2-a VIII.B1.1-a	Piping, piping components, and piping elements	Steel	Steam	Loss of material/ pitting and crevice corrosion	Chapter XI.M2, "Water Chemistry," for PWR secondary water ~~in EPRI TR-102134~~	No	See General Comment 3
New VIII.B1(1) ***VIII.G***		***Piping, piping components and piping elements***	***Steel***	***Condensation (Internal)***	***Loss of material/ general, pitting, and crevice corrosion***	***A plant-specific aging management program is to be evaluated.***	***Yes, plant-specific***	This item covers steam piping that is empty in standby condition, such as AFW turbine steam lines, during normal operations. MIC cannot occur in these lines as the water has condensed from temperatures in excess of 212 °F. This line is the same as AP-71.

VII AUXILIARY SYSTEMS
B1 Main Steam System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
New VIII.B1(2) *(SP-17)*		*Piping, piping components, and piping elements*	*Stainless steel*	*Treated water >60°C (>140°F)*	*Cracking/ stress corrosion cracking*	*Chapter XI.M2, "Water Chemistry," for PWR secondary water* *The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.*	*Yes, detection of aging effects is to be evaluated*	Treated Water is added due to steam condensation from drain lines/drain pots, etc.. This line is the same as SP-17.
New VIII.B1(3) *(SP-16)* *VIII.C*		*Piping, piping components, and piping elements*	*Stainless steel*	*Treated water*	*Loss of material/ pitting and crevice corrosion*	*Chapter XI.M2, "Water Chemistry," for PWR secondary water* *The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.*	*Yes, detection of aging effects is to be evaluated*	Treated Water is added due to steam condensation from drain lines/drain pots, etc.. This line is the same as SP-16.
New VIII.B1(4) *(S-10)* *VIII.C*		*Piping, piping components, and piping elements*	*Steel*	*Treated water*	*Loss of material/ general, pitting, and crevice corrosion*	*Chapter XI.M2, "Water Chemistry," for PWR secondary water* *The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.*	*Yes, detection of aging effects is to be evaluated*	Treated Water is added due to steam condensation from drain lines/drain pots, etc.. This line is the same as S-10.

VII AUXILIARY SYSTEMS
B1 Main Steam System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
New VIII.B1(5) *(S-16)*		*Piping, piping components, and piping elements*	*Steel*	*Treated water*	*Wall thinning/ flow-accelerated corrosion*	*Chapter XI.M17, "Flow-Accelerated Corrosion"*	*No*	Treated Water is added due to steam condensation from drain lines/drain pots, etc. This line is the same as S-16.
VIII.B1-X		*Piping, piping components, and piping elements*	*Steel*	*Air – outdoor (internal)*	*Loss of material/general, pitting, and crevice corrosion*	*Chapter XI.M32, "One-Time Inspection."*	*No*	This component is added to cover the interior surfaces of the main steam safety vents to atmosphere downstream of safety valves. This line is similar to other lines with the combination of steel, outdoor air and loss of material, such as line A-24. Where line A-24 refers to a plant specific program this line uses a one time inspection

VII AUXILIARY SYSTEMS B2 Main Steam System (BWR)								
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
VIII.B2-1 (SP-45)	VIII.B2.	Piping, piping components, and piping elements	Stainless steel	Steam	Cracking/ stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for BWR water ~~in BWRVIP-29 (EPRI TR-103515).~~ The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated	See General Comment 3
VIII.B2-2 (SP-46)	VIII.B2.	Piping, piping components, and piping elements	Stainless steel	Steam	Loss of material/ pitting and crevice corrosion	Chapter XI.M2, "Water Chemistry," for BWR water ~~in BWRVIP-29 (EPRI TR-103515).~~	No	See General Comment 3
VIII.B2-3 (S-08)	VIII.B2.1-c	Piping, piping components, and piping elements	Steel	Steam ***or treated water***	Cumulative fatigue damage/ fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA	Fatigue may not be limited to piping with an internal steam environment. Treated water added to extend applicability; however, an environment of "Any" might be more appropriate.
VIII.B2-4 (S-05)	VIII.B2.2-b VIII.B2.1-a	Piping, piping components, and piping elements	Steel	Steam	Loss of material/ pitting and crevice corrosion	Chapter XI.M2, "Water Chemistry," for BWR water ~~in BWRVIP-29 (EPRI TR-103515).~~	No	See General Comment 3

VII AUXILIARY SYSTEMS
B2 Main Steam System (BWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
New VIII.B2(1) *(S-09)* *VIII.C*		*Piping, piping components, and piping elements*	*Steel*	*Treated water*	*Loss of material/ general, pitting, and crevice corrosion*	*Chapter XI.M2, "Water Chemistry," for BWR water* *The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.*	*Yes, detection of aging effects is to be evaluated*	Treated Water is added due to steam condensation. This line is the same as S-09.
New VIII.B2(2) *VIII.C*		*Piping, piping components, and piping elements*	*Stainless steel*	*Treated water*	*Loss of material/ pitting and crevice corrosion*	*Chapter XI.M2, "Water Chemistry," for BWR water* *The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.*	*Yes, detection of aging effects is to be evaluated*	Treated Water is added due to steam condensation. This line is the same as A-58 and similar to S-21.

VIII.C Extraction Steam System

VII C	AUXILIARY SYSTEMS Extraction Steam System							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
VIII.C-1 (S-04)	VIII.C.1-b VIII.C.2-b	Piping, piping components, and piping elements	Steel	Steam	Loss of material/ general, pitting, and crevice corrosion	Chapter XI.M2, "Water Chemistry," for BWR water ~~in BWRVIP-29 (EPRI TR-103515).~~ The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated	See General Comment 3
VIII.C-2 (S-06)	VIII.C.1-b VIII.C.2-b	Piping, piping components, and piping elements	Steel	Steam	Loss of material/ general, pitting, and crevice corrosion	Chapter XI.M2, "Water Chemistry," for PWR secondary water ~~in EPRI TR-102134~~ The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated	See General Comment 3

VII AUXILIARY SYSTEMS
C Extraction Steam System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
New VIII.C(1) *(S-09)* *VIII.B2*		*Piping, piping components, and piping elements*	*Steel*	*Treated water*	*Loss of material/ general, pitting, and crevice corrosion*	*Chapter XI.M2, "Water Chemistry," for BWR water* *The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.*	*Yes, detection of aging effects is to be evaluated*	Treated Water is added due to steam condensation. This line is the same as S-09.
New VIII.C(2) *VIII.B2*		*Piping, piping components, and piping elements*	*Stainless steel*	*Treated water*	*Loss of material/ pitting and crevice corrosion*	*Chapter XI.M2, "Water Chemistry," for BWR water* *The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.*	*Yes, detection of aging effects is to be evaluated*	Treated Water is added due to steam condensation. This line is the same as A-58 and similar to S-21.
New VIII.C(3) *(SP-16)* *VIII.B1*		*Piping, piping components, and piping elements*	*Stainless steel*	*Treated water*	*Loss of material/ pitting and crevice corrosion*	*Chapter XI.M2, "Water Chemistry," for PWR secondary water* *The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.*	*Yes, detection of aging effects is to be evaluated*	Treated Water is added due to steam condensation from drain lines/drain pots, etc.. This line is the same as SP-16.

VII AUXILIARY SYSTEMS
C Extraction Steam System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
New VIII.C(4) *(S-10)* *VIII.B1*		*Piping, piping components, and piping elements*	*Steel*	*Treated water*	*Loss of material/ general, pitting, and crevice corrosion*	*Chapter XI.M2, "Water Chemistry," for PWR secondary water* *The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.*	*Yes, detection of aging effects is to be evaluated*	Treated Water is added due to steam condensation from drain lines/drain pots, etc.. This line is the same as S-10.

VIII.D1 Feedwater System (PWR)

VII AUXILIARY SYSTEMS D1 Feedwater System (PWR)								
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
VIII.D1-4 (SP-16)	VIII.D1.	Piping, piping components, and piping elements	Stainless steel	Treated water	Loss of material/ pitting and crevice corrosion	Chapter XI.M2, "Water Chemistry," for PWR secondary water ~~in EPRI TR-102134~~ The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated	See General Comment 3
VIII.D1-5 (SP-17)	VIII.D1.	Piping, piping components, and piping elements	Stainless steel	Treated water >60°C (>140°F)	Cracking/ stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR secondary water ~~in EPRI TR-102134~~ The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated	See General Comment 3

VII AUXILIARY SYSTEMS
D1 Feedwater System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
VIII.D1-8 (S-10)	VIII.D1.2-b VIII.D1.3-a VIII.D1.1-c	Piping, piping components, and piping elements	Steel	Treated water	Loss of material/ general, pitting, and crevice corrosion	Chapter XI.M2, "Water Chemistry," for PWR secondary water ~~in EPRI TR-102134~~ The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated	See General Comment 3

VIII.D2 Feedwater System (BWR)

VII AUXILIARY SYSTEMS D2 Feedwater System (BWR)								
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
VIII.D2-4 (SP-16)	VIII.D2.	Piping, piping components, and piping elements	Stainless steel	Treated water	Loss of material/ pitting and crevice corrosion	Chapter XI.M2, "Water Chemistry," for PWR secondary water ~~in EPRI TR-102134~~ The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated	See General Comment 3
VIII.D2-7 (S-09)	VIII.D2.1-b VIII.D2.2-b VIII.D2.3-b	Piping, piping components, and piping elements	Steel	Treated water	Loss of material/ general, pitting, and crevice corrosion	Chapter XI.M2, "Water Chemistry," for BWR water ~~in BWRVIP-29 (EPRI TR-103515).~~ The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated	See General Comment 3

VIII.E Condensate System

VII E	AUXILIARY SYSTEMS Condensate System							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
~~VIII.E-1~~ ~~(S-01)~~	~~VIII.E.5-d~~	~~Buried piping, piping components, piping elements, and tanks~~	~~Steel (with or without coating or wrapping)~~	~~Soil~~	~~Loss of material/ general, pitting, crevice, and microbiologically influenced corrosion~~	~~Chapter XI.M28, "Buried Piping and Tanks Surveillance," or~~ ~~Chapter XI.M34, "Buried Piping and Tanks Inspection"~~	~~No~~ ~~Yes, detection of aging effects and operating experience are to be further evaluated~~	See General Comment 4
~~VIII.E-17~~ ~~(SP-26)~~	~~VIII.E.~~	~~Piping, piping components, and piping elements~~	~~Gray cast iron~~	~~Soil~~	~~Loss of material/ selective leaching~~	~~Chapter XI.M33, "Selective Leaching of Materials"~~	~~No~~	See General Comment 4
VIII.E-2 (S-21)	VIII.E.4-a VIII.E.4-d	~~BWR h~~Heat exchanger ~~shell side~~ components	Stainless steel	Treated water	Loss of material/ pitting and crevice corrosion	Chapter XI.M2, "Water Chemistry," for BWR water ~~in BWRVIP-29 (EPRI TR-103515).~~ The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated	See General Comment 3

VII AUXILIARY SYSTEMS
E Condensate System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
~~VIII.E-22~~ ~~(SP-37)~~	~~VIII.E.~~	~~Piping, piping components, and piping elements~~	~~Stainless steel~~	~~Soil~~	~~Loss of material/ pitting and crevice corrosion~~	~~A plant-specific aging management program is to be evaluated.~~	~~Yes, plant-specific~~	See General Comment 4
VIII.E-23 (SP-16)	VIII.E.	Piping, piping components, and piping elements	Stainless steel	Treated water	Loss of material/ pitting and crevice corrosion	Chapter XI.M2, "Water Chemistry," for PWR secondary water ~~in EPRI TR-102134~~ The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated	See General Comment 3
VIII.E-24 (SP-17)	VIII.E.	Piping, piping components, and piping elements	Stainless steel	Treated water >60°C (>140°F)	Cracking/ stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR secondary water ~~in EPRI TR-102134~~ The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated	See General Comment 3

VII AUXILIARY SYSTEMS
E Condensate System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
VIII.E-25 (SP-19)	VIII.E.	Piping, piping components, and piping elements	Stainless steel	Treated water >60°C (>140°F)	Cracking/ stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for BWR water ~~in BWRVIP-29 (EPRI TR-103515).~~ The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated	See General Comment 3
VIII.E-27 (S-09)	VIII.E.1-b VIII.E.6-a VIII.E.3-a VIII.E.5-a VIII.E.2-b	Piping, piping components, and piping elements	Steel	Treated water	Loss of material/ general, pitting, and crevice corrosion	Chapter XI.M2, "Water Chemistry," for BWR water ~~in BWRVIP-29 (EPRI TR-103515).~~ The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated	See General Comment 3

VII AUXILIARY SYSTEMS
E Condensate System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
VIII.E-28 (S-10)	VIII.E.1-b VIII.E.2-b VIII.E.3-a VIII.E.5-a VIII.E.6-a	Piping, piping components, and piping elements	Steel	Treated water	Loss of material/ general, pitting, and crevice corrosion	Chapter XI.M2, "Water Chemistry," for PWR secondary water ~~in EPRI TR-102134~~ The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated	See General Comment 3
VIII.E-3 (S-18)	VIII.E.4-d VIII.E.4-a	~~BWR h~~Heat exchanger ~~shell side~~ components	Steel	Treated water	Loss of material/ general, pitting, and crevice corrosion	Chapter XI.M2, "Water Chemistry," for BWR water ~~in BWRVIP-29 (EPRI TR-103515).~~ The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated	See General Comment 5 See General Comment 3

VII AUXILIARY SYSTEMS
E Condensate System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
VIII.E-30 (S-22)	VIII.E.4-d VIII.E.4-a	~~PWR h~~*H*eat exchanger ~~*shell side*~~ components	Stainless steel	Treated water	Loss of material/ pitting and crevice corrosion	Chapter XI.M2, "Water Chemistry," for PWR primary water ~~*in EPRI TR-105714*~~ The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated	See General Comment 5 See General Comment 3
VIII.E-31 (S-19)	VIII.E.4-a VIII.E.4-d	~~PWR h~~*H*eat exchanger ~~*shell side*~~ components	Steel	Treated water	Loss of material/ general, pitting, and crevice corrosion	Chapter XI.M2, "Water Chemistry," for PWR secondary water ~~*in EPRI TR-102134*~~ The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated	See General Comment 5 See General Comment 3

VII AUXILIARY SYSTEMS
E Condensate System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
~~VIII.E-32~~ ~~(S-13)~~	~~VIII.E.5-b~~	~~Tanks~~	~~*Stainless steel*~~	~~*Treated water*~~	~~*Loss of material/ pitting and crevice corrosion*~~	~~*Chapter XI.M2, "Water Chemistry," for BWR water in BWRVIP-29 (EPRI TR-103515).*~~ ~~*The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.*~~	~~*Yes, detection of aging effects is to be evaluated*~~	Tanks are included in the Piping Components.
~~VIII.E-33~~ ~~(S-14)~~	~~VIII.E.5-b~~	~~Tanks~~	~~*Stainless steel*~~	~~*Treated water*~~	~~*Loss of material/ pitting and crevice corrosion*~~	~~*Chapter XI.M2, "Water Chemistry," for PWR secondary water in EPRI TR-102134*~~ ~~*The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.*~~	~~*Yes, detection of aging effects is to be evaluated*~~	Tanks are included in the Piping Components.
~~VIII.E-34~~ ~~(SP-42)~~	~~VIII.E.~~	~~Tanks~~	~~*Stainless steel*~~	~~*Treated water >60°C (>140°F)*~~	~~*Cracking/ stress corrosion cracking*~~	~~*Chapter XI.M21, "Closed Cycle Cooling Water System"*~~ ~~*The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.*~~	~~*Yes, detection of aging effects is to be evaluated*~~	Tanks are included in the Piping Components.

VII AUXILIARY SYSTEMS
E Condensate System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
~~VIII.E-35~~ ~~(S-31)~~	~~VIII.E.5-c~~	~~Tanks~~	~~Steel~~	~~Air – outdoor (External)~~	~~Loss of material/ general corrosion~~	~~Chapter XI.M29, "Aboveground Carbon Steel Tanks"~~	~~No~~	See General Comment 4
VIII.E-4 (S-25)	VIII.E.4-e	Heat exchanger ~~tube-side~~ components	Stainless steel	Closed cycle cooling water	Loss of material/ pitting and crevice corrosion	Chapter XI.M21, "Closed-Cycle Cooling Water System"	No	See General Comment 5
VIII.E-5 (S-26)	VIII.E.4-b	Heat exchanger ~~tube-side~~ components	Stainless steel	Raw water	Loss of material/ pitting, crevice, and microbiologically influenced corrosion, *and fouling*	Chapter XI.M20, "Open-Cycle Cooling Water System"	No	See General Comment 5
VIII.E-6 (S-23)	VIII.E.4-e	Heat exchanger ~~tube-side~~ components	Steel	Closed cycle cooling water	Loss of material/ general, pitting, and crevice corrosion	Chapter XI.M21, "Closed-Cycle Cooling Water System"	No	See General Comment 5
VIII.E-7 (S-24)	VIII.E.4-b	Heat exchanger ~~tube-side~~ components	Steel	Raw water	Loss of material/ general, pitting, crevice, and microbiologically influenced corrosion, *and fouling*	Chapter XI.M20, "Open-Cycle Cooling Water System"	No	See General Comment 5
New VIII.E(1)		***Heat exchanger tubes***	***Copper alloy***	***Closed cycle cooling water***	***Reduction of heat transfer/ fouling***	***Chapter XI.M21, "Closed Cycle Cooling Water System"***	***No***	This applies to heat exchangers in any system with CCCW on either side of the tubes. This is similar to GALL line AP-63 but for a different material.

VII AUXILIARY SYSTEMS
E Condensate System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
New VIII.E(2) *VIII.F VIII.G*		*Heat exchanger tubes*	*Copper alloy*	*Raw water*	*Reduction of heat transfer/ fouling*	*Chapter XI.M20, "Open-Cycle Cooling Water System"*	*No*	This applies to heat exchangers in any system with raw water on either side of the tubes. This line is the same as A-72.
New VIII.E(3) *VIII.F VIII.G*		*Heat exchanger tubes*	*Copper alloy*	*Treated water*	*Reduction in heat transfer/ fouling*	*Chapter XI.M2, "Water Chemistry"*	*No*	This applies to heat exchangers in any system with treated water on either side of the tubes. This is similar to GALL line SP-40 but for a different material.
New VIII.E(4) *VIII.F VIII.G*		*Piping, piping components, and piping elements*	*Copper alloy >15% Zn*	*Treated water*	*Loss of material/ selective leaching*	*Chapter XI.M33, "Selective Leaching of Materials"*	*No*	This applies to heat exchanger and other supporting equipment. This line is the same as AP-32
New VIII.E(5) *VIII.F VIII.G*		*Piping, piping components, and piping elements*	*Stainless steel*	*Closed cycle cooling water >60°C (>140°F)*	*Cracking/ stress corrosion cracking*	*Chapter XI.M21, "Closed-Cycle Cooling Water System"*	*Yes, plant-specific*	The basis for AP-60 is applicable to components in S&PC systems where closed cooling water temperatures exceed 140°F in supporting systems such as bearing and lube oil coolers and associated piping that are considered part of the S&PC systems. SP-39 addresses a different aging mechanism for this same group of components.

VIII.F Steam Generator Blowdown System (PWR)

VII F	AUXILIARY SYSTEMS Steam Generator Blowdown System (PWR)							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
VIII.F-1 (S-25)	VIII.F.4-e	Heat exchanger ~~tube-side~~ components	Stainless steel	Closed cycle cooling water	Loss of material/ pitting and crevice corrosion	Chapter XI.M21, "Closed-Cycle Cooling Water System"	No	See General Comment 5
VIII.F-17 (SP-16)	VIII.F.	Piping, piping components, and piping elements	Stainless steel	Treated water	Loss of material/ pitting and crevice corrosion	Chapter XI.M2, "Water Chemistry," for PWR secondary water ~~*in EPRI TR-102134*~~ The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated	See General Comment 3
VIII.F-18 (SP-17)	VIII.F.	Piping, piping components, and piping elements	Stainless steel	Treated water >60°C (>140°F)	Cracking/ stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR secondary water ~~*in EPRI TR-102134*~~ The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated	See General Comment 3

VII AUXILIARY SYSTEMS
F Steam Generator Blowdown System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
VIII.F-19 (S-10)	VIII.F.2-b VIII.F.1-b VIII.F.3-a	Piping, piping components, and piping elements	Steel	Treated water	Loss of material/ general, pitting, and crevice corrosion	Chapter XI.M2, "Water Chemistry," for PWR secondary water ~~in EPRI TR-102134~~ The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated	See General Comment 3
VIII.F-2 (S-26)	VIII.F.4-b	Heat exchanger ~~tube side~~ components	Stainless steel	Raw water	Loss of material/ pitting, crevice, and microbiologically influenced corrosion, and fouling	Chapter XI.M20, "Open-Cycle Cooling Water System"	No	See General Comment 5
VIII.F-21 (S-22)	VIII.F.4-a VIII.F.4-d	~~PWR h~~*H*eat exchanger ~~*shell side*~~ components	Stainless steel	Treated water	Loss of material/ pitting and crevice corrosion	Chapter XI.M2, "Water Chemistry," for PWR primary water ~~in EPRI TR-105714~~ The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated	See General Comment 5 See General Comment 3

VII AUXILIARY SYSTEMS
F Steam Generator Blowdown System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
VIII.F-22 (S-19)	VIII.F.4-d VIII.F.4-a	~~PWR h~~*H*eat exchanger ~~*shell side*~~ components	Steel	Treated water	Loss of material/ general, pitting, and crevice corrosion	Chapter XI.M2, "Water Chemistry," for PWR secondary water ~~*in EPRI TR-102134*~~ The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated	See General Comment 5 See General Comment 3
VIII.F-3 (S-23)	VIII.F.4-e	Heat exchanger ~~*tube side*~~ components	Steel	Closed cycle cooling water	Loss of material/ general, pitting, and crevice corrosion	Chapter XI.M21, "Closed-Cycle Cooling Water System"	No	See General Comment 5
VIII.F-4 (S-24)	VIII.F.4-b	Heat exchanger ~~*tube side*~~ components	Steel	Raw water	Loss of material/ general, pitting, crevice, and microbiologically influenced corrosion, and fouling	Chapter XI.M20, "Open-Cycle Cooling Water System"	No	See General Comment 5
VIII.F-5 (S-39)	VIII.F.4-a	Heat exchanger ~~*tube side*~~ components ~~*including tubes*~~	Stainless steel	Treated water >60°C (>140°F)	Cracking/ stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR secondary water ~~*in EPRI TR-102134*~~ The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated	See General Comment 5 See General Comment 3

VII AUXILIARY SYSTEMS
F Steam Generator Blowdown System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
New VIII.F(1) *VIII.E* *VIII.G*		*Piping, piping components, and piping elements*	*Copper alloy >15% Zn*	*Treated water*	*Loss of material/ selective leaching*	*Chapter XI.M33, "Selective Leaching of Materials"*	*No*	This applies to heat exchanger and other supporting equipment. This line is the same as AP-32
New VIII.F(2) *VIII.E* *VIII.G*		*Heat exchanger tubes*	*Copper alloy*	*Treated water*	*Reduction in heat transfer/ fouling*	*Chapter XI.M2, "Water Chemistry"*	*No*	This applies to heat exchangers in any system with treated water on either side of the tubes. This is similar to GALL line SP-40 but for a different material.
New VIII.F(3) *VIII.E* *VIII.G*		*Heat exchanger tubes*	*Copper alloy*	*Raw water*	*Reduction of heat transfer/ fouling*	*Chapter XI.M20, "Open-Cycle Cooling Water System"*	*No*	This applies to heat exchangers in any system with raw water on either side of the tubes. This line is the same as A-72.
New VIII.F(4) *VIII.E* *VIII.G*		*Piping, piping components, and piping elements*	*Stainless steel*	*Closed cycle cooling water >60°C (>140°F)*	*Cracking/ stress corrosion cracking*	*Chapter XI.M21, "Closed-Cycle Cooling Water System"*	*Yes, plant-specific*	The basis for AP-60 is applicable to components in S&PC systems where closed cooling water temperatures exceed 140°F in supporting systems such as bearing and lube oil coolers and associated piping that are considered part of the S&PC systems. SP-39 addresses a different aging mechanism for this same group of components.
VIII.F(5) *VIII.A*		*Piping, piping components, and piping elements*	*Copper alloy*	*Treated water*	*Loss of material/general, pitting, and crevice corrosion*	*Chapter XI.M2, "Water Chemistry"*	*No*	This line compliments new line VIII.F(1) above. This addresses loss of material other than selective leaching. This effect can apply to all copper alloys.

VIII.G Auxiliary Feedwater (AFW) System (PWR)

VII G	AUXILIARY SYSTEMS Auxiliary Feedwater (AFW) System (PWR)							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
~~VIII.G-1~~ ~~(S-01)~~	~~VIII.G.4-d~~ ~~VIII.G.1-e~~	~~Buried piping, piping components, piping elements, and tanks~~	~~Steel (with or without coating or wrapping)~~	~~Soil~~	~~Loss of material/ general, pitting, crevice, and microbiologically influenced corrosion~~	~~Chapter XI.M28, "Buried Piping and Tanks Surveillance," or~~ ~~Chapter XI.M34, "Buried Piping and Tanks Inspection"~~	~~No~~ ~~Yes, detection of aging effects and operating experience are to be further evaluated~~	See General Comment 4
~~VIII.G-17~~ ~~(SP-26)~~	~~VIII.G.~~	~~Piping, piping components, and piping elements~~	~~Gray cast iron~~	~~Soil~~	~~Loss of material/ selective leaching~~	~~Chapter XI.M33, "Selective Leaching of Materials"~~	~~No~~	See General Comment 4
VIII.G-2 (S-20)	VIII.G.5-d	Heat exchanger ~~shell side~~ components	Stainless steel	Lubricating oil	Loss of material/ pitting, crevice, and microbiologically influenced corrosion	A plant-specific aging management program is to be evaluated.	Yes, plant-specific	See General Comment 5

VII AUXILIARY SYSTEMS
G Auxiliary Feedwater (AFW) System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
~~VIII.G-23~~ ~~(SP-37)~~	~~VIII.G.~~	~~Piping, piping components, and piping elements~~	~~Stainless steel~~	~~Soil~~	~~Loss of material/ pitting and crevice corrosion~~	~~A plant-specific aging management program is to be evaluated.~~	~~Yes, plant-specific~~	See General Comment 4
VIII.G-24 (SP-16)	VIII.G.	Piping, piping components, and piping elements	Stainless steel	Treated water	Loss of material/ pitting and crevice corrosion	Chapter XI.M2, "Water Chemistry," for PWR secondary water ~~in EPRI TR-102134~~ The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated	See General Comment 3
VIII.G-25 (SP-17)	VIII.G.	Piping, piping components, and piping elements	Stainless steel	Treated water >60°C (>140°F)	Cracking/ stress corrosion cracking	Chapter XI.M2, "Water Chemistry," for PWR secondary water ~~in EPRI TR-102134~~ The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated	See General Comment 3

VII AUXILIARY SYSTEMS
G Auxiliary Feedwater (AFW) System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
VIII.G-28 (S-10)	VIII.G.3-a VIII.G.2-a VIII.G.1-c VIII.G.4-a	Piping, piping components, and piping elements	Steel	Treated water	Loss of material/ general, pitting, and crevice corrosion	Chapter XI.M2, "Water Chemistry," for PWR secondary water ~~in EPRI TR-102134~~ The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated	See General Comment 3
VIII.G-3 (S-17)	VIII.G.5-d	Heat exchanger ~~shell side~~ components	Steel	Lubricating oil	Loss of material/ general, pitting, crevice, and microbiologically influenced corrosion	A plant-specific aging management program is to be evaluated.	Yes, plant-specific	See General Comment 5
VIII.G-31 (S-14)	VIII.G.4-b	Tanks	Stainless steel	Treated water	Loss of material/ pitting and crevice corrosion	Chapter XI.M2, "Water Chemistry," for PWR secondary water ~~in EPRI TR-102134~~ The AMP is to be augmented by verifying the effectiveness of water chemistry control. See Chapter XI.M32, "One-Time Inspection," for an acceptable verification program.	Yes, detection of aging effects is to be evaluated	See General Comment 3

VII AUXILIARY SYSTEMS
G Auxiliary Feedwater (AFW) System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
~~VIII.G-32~~ ~~(S-31)~~	~~VIII.G.4-c~~	~~Tanks~~	~~Steel~~	~~Air – outdoor (External)~~	~~Loss of material/ general corrosion~~	~~Chapter XI.M29, "Aboveground Carbon Steel Tanks"~~	~~No~~	See General Comment 4
VIII.G-4 (S-25)	VIII.G.5-c	Heat exchanger ~~shell side~~ components	Stainless steel	Closed cycle cooling water	Loss of material/ pitting and crevice corrosion	Chapter XI.M21, "Closed-Cycle Cooling Water System"	No	See General Comment 5
VIII.G-5 (S-26)	VIII.G.5-a	Heat exchanger ~~shell side~~ components	Stainless steel	Raw water	Loss of material/ pitting, crevice, and microbiologically influenced corrosion, *and fouling*	Chapter XI.M20, "Open-Cycle Cooling Water System"	No	See General Comment 5 See General Comment 3
VIII.G-6 (S-23)	VIII.G.5-c	Heat exchanger ~~shell side~~ components	Steel	Closed cycle cooling water	Loss of material/ general, pitting, and crevice corrosion	Chapter XI.M21, "Closed-Cycle Cooling Water System"	No	See General Comment 5

VII AUXILIARY SYSTEMS
G Auxiliary Feedwater (AFW) System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
VIII.G-7 (S-24)	VIII.G.5-a	Heat exchanger ~~shell side~~ components	Steel	Raw water	Loss of material/ general, pitting, crevice, and microbiologically influenced corrosion~~,~~ *and fouling*	Chapter XI.M20, "Open-Cycle Cooling Water System"	No	See General Comment 5
New VIII.G(1) ***VIII.E*** ***VIII.F***		***Heat exchanger tubes***	***Copper alloy***	***Treated water***	***Reduction in heat transfer/ fouling***	***Chapter XI.M2, "Water Chemistry"***	***No***	This applies to heat exchangers in any system with treated water on either side of the tubes. This is similar to GALL line SP-40 but for a different material.
New VIII.G(2) ***VIII.E*** ***VIII.F***		***Heat exchanger tubes***	***Copper alloy***	***Raw water***	***Reduction of heat transfer/ fouling***	***Chapter XI.M20, "Open-Cycle Cooling Water System"***	***No***	This applies to heat exchangers in any system with raw water on either side of the tubes. This line is the same as A-72.
New VIII.G(3) ***VIII.E*** ***VIII.F***		***Piping, piping components, and piping elements***	***Copper alloy >15% Zn***	***Treated water***	***Loss of material/ selective leaching***	***Chapter XI.M33, "Selective Leaching of Materials"***	***No***	This applies to heat exchanger and other supporting equipment. This line is the same as AP-32

VII AUXILIARY SYSTEMS
G Auxiliary Feedwater (AFW) System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation	Basis for Change
New VIII.G(4)		*Heat exchanger tubes*	*Copper alloy*	*Lubricating oil*	*Reduction of heat transfer/ fouling*	*A plant-specific aging management program is to be evaluated.*	*Yes, plant-specific*	CNP SER Section 3.2.2.3.3 acknowledged the potential for fouling of copper alloy components in a lubricating oil environment. The SER addressed components in A PWR ECCS system, but these conditions would be equally applicable to components of other lube oil systems. The reduction of heat transfer due to fouling of heat exchanger tubes will be addressed by a plant specific program. (Oil analysis for CNP as shown in Table 3.2.2-3 of the CNP LRA)
New VIII.G(5)		*Heat exchanger tubes*	*Steel*	*Lubricating oil*	*Reduction of heat transfer/ fouling*	*A plant-specific aging management program is to be evaluated.*	*Yes, plant-specific*	This line is similar to the preceding line except for a different material. Heat exchanger tubes of steel would also be subject to fouling in a lubricating oil environment.
New VIII.G(6)		*Heat exchanger tubes*	*Stainless steel*	*Lubricating oil*	*Reduction of heat transfer/ fouling*	*A plant-specific aging management program is to be evaluated.*	*Yes, plant-specific*	This line is similar to the preceding line except for a different material. Heat exchanger tubes of stainless steel would also be subject to fouling in a lubricating oil environment.

VIII.H External Surfaces of Components and Miscellaneous Bolting

VII H	AUXILIARY SYSTEMS External Surfaces of Components and Miscellaneous Bolting							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
VIII.H-1 (S-32) ***(S-41)***	VIII.H. ***VIII.H.1-b***	***External Surfaces including closure b***~~B~~olting	Steel	Air – outdoor (External)	Loss of material/ general, pitting, and crevice corrosion	~~"Chapter XI.M18, "Bolting Integrity"~~ ***"Chapter XI.M##, External Surfaces Monitoring"***	No	Credit previously accepted "External Surfaces Monitoring" AMP. Consolidate listings by including S-41 (VIII.H-8).
VIII.H-10 (S-42)	VIII.H.1-b	External surfaces ***Including closure bolting***	Steel	Condensation (External)	Loss of material/ General corrosion	~~A plant-specific aging management program is to be evaluated.~~ ***"Chapter XI.M##, External Surfaces Monitoring"***	~~Yes, plant-specific~~ ***No***	Credit previously accepted "External Surfaces Monitoring" AMP. Consolidate listings by including S-41 (VIII.H-8). Relocated
VIII.H-2 (S-40) ***(S-30)***	VIII.H. ***VIII.H.1-b***	***External Surfaces including closure b***~~B~~olting	Steel	Air with borated water leakage	Loss of material/ boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No	Consolidate listings by including S-30 (VIII.H-9).
VIII.H-3 (S-03) ***(S-02)***	VIII.H.2-b ***VIII.H.2-a***	Closure bolting	~~High-strength~~ ***Low alloy*** steel > 150 ksi	Air with steam or water leakage	Cracking/ cyclic loading, stress corrosion cracking	~~"Chapter XI.M18, "Bolting Integrity"~~ ***"Chapter XI.M##, External Surfaces Monitoring"***	No	Credit previously accepted "External Surfaces Monitoring" AMP. Consolidate listings by including S-02 (VIII.H-6).
VIII.H-4 (S-34) ***(S-29)***	VIII.H. ***VIII.H.1-b***	External surfaces including c~~C~~losure bolting	Steel	Air – indoor uncontrolled (External)	Loss of material/ general, pitting, and crevice corrosion	~~"Chapter XI.M18, "Bolting Integrity"~~ ***"Chapter XI.M##, External Surfaces Monitoring"***	No	Credit previously accepted "External Surfaces Monitoring" AMP. Consolidate listings by including S-29 (VIII.H-7).
~~VIII.H-5~~ ~~(S-33)~~	~~VIII.H.~~	~~Closure bolting~~	~~Steel~~	~~Air – indoor uncontrolled (External)~~	~~Loss of preload/ stress relaxation~~	~~Chapter XI.M18, "Bolting Integrity"~~	~~No~~	Loss of preload is not an aging effect. See General Comment 6

VIII.I Common Miscellaneous Material Environment Combinations

VIII I	STEAM AND POWER CONVERSION SYSTEM Common Miscellaneous Material Environment Combinations							
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**	**Basis for Change**
New VIII.I(1)		*Piping, piping components, and piping elements external surfaces*	*Steel*	*Containment environment (inert)*	*None*	*None*	*No*	*See Chapter IX comment on Containment environment (inert).*

CHAPTER IX SELECTED DEFINITIONS OF TERMS USED FOR DESCRIBING AND STANDARDIZING STRUCTURES, COMPONENTS, MATERIALS, ENVIRONMENTS, AGING EFFECTS, AND AGING MECHANISMS

IX Selected Definitions of Terms Used for Describing and Standardizing Structures, Components, Materials, Environments, Aging Effects, and Aging Mechanisms **Definition of Selected Terms**		
Standardized Expression	**Description of Change**	**Basis**
Add new component: Encapsulation Components/ Valve Chambers	Add definition: These are airtight enclosures that function as a secondary containment boundary to completely enclose containment sump lines and isolation valves.	These are common components used throughout the nuclear industry.
Piping, piping components, and piping elements	This general category includes various features of the piping system that are within the scope of license renewal. Examples include piping, fittings, tubing, flow elements/indicators, demineralizer, nozzles, orifices, flex hoses, pump casing and bowl, safe ends, sight glasses, spray head, strainers, ***tanks*** thermowells, and valve body and bonnet.	This component category was intended to encompass all components when evaluating material and environment combinations where the configuration of the component, and thus its specific type, have no impact on the aging effects. While the word "examples" indicates the list is only partial, the inclusion of at least one larger component type would further clarify the definition as all inclusive.
Cast austenitic stainless steel (CASS)	Cast austenitic stainless steel (CASS) alloys such as CF-3, CF-8, CF-3M, and CF-8M have been widely used in LWRs. These CASS are similar to wrought grades Type 304L, Type 304, Type 316L, and Type 316, except CASS typically contains 5 to 25% ferrite. ***CASS is susceptible to loss of fracture toughness due to thermal and neutron irradiation embrittlement.***	These added words supplement the changes made to the entry for stainless steel. Together they clarify that CASS is considered to be the same as stainless steel except where thermal and neutron irradiation embrittlement is considered as an aging effect.

IX Selected Definitions of Terms Used for Describing and Standardizing Structures, Components, Materials, Environments, Aging Effects, and Aging Mechanisms		
Definition of Selected Terms		
Standardized Expression	**Description of Change**	**Basis**
~~High strength~~ low alloy steel	This term should be changed to "low alloy steel." The definition should be rewritten to simply describe typical bolting used in nuclear applications. A note is then added that high strength bolting are those with actual measured yield strength, S_y, ≥ 150 ksi. Suggested wording is as follows: ~~High-strength~~ Fe-Cr-Ni-Mo low alloy steel bolting materials with maximum tensile strength <1172 MPa (<170 Ksi) ~~are~~ ***may be*** subject to stress corrosion cracking ***if the actual measured yield strength, S_y, ≥ 150 ksi***. Examples of ~~high-strength~~ alloy steel designations that were earlier referenced in NUREG-1801 that comprise this category include SA540-Gr. B23/24, SA193-Gr. B8, and Grade L43 (AISI4340). ~~High-strength l~~ Low-alloy steel SA 193 Gr. B7 is a ferritic low-alloy steel bolting material for high-temperature service. Includes AISI steels 4140, 4142, 4145, 4140H, 4142H, and 4145H (UNS#: G41400, G41420, G41450, H41400, H41420, H41450). Bolting fabricated from high-strength (***actual measured yield strength, S_y, ≥ 150 ksi***) low-alloy steel SA 193 Gr. B7 is susceptible to stress corrosion cracking.	The "High strength" modifier refers to susceptibility to the cracking aging effect. SA193 Grade B7 bolting is not "high strength" per the definition provided on page 12 of NUREG-1339. The categorization should be based on the **actual** measured yield strength, S_y, and not on the **specified minimum** yield strength. This change affects the following items: o III T-27 B1.1.2-a o IV R-71 A2.1-c o IV R-73 A2.1-e o IV R-72 A2.1-d o IV R-60 A1.1-c
Low-alloy steel, yield strength >150 ksi	This definition should be deleted based on the comments to "High strength low alloy steel." This refers to high strength bolting for NSSS component supports is fabricated from low-alloy steel, yield strength >150 ksi	See comments to "High strength low alloy steel."

IX Selected Definitions of Terms Used for Describing and Standardizing Structures, Components, Materials, Environments, Aging Effects, and Aging Mechanisms Definition of Selected Terms		
Standardized Expression	**Description of Change**	**Basis**
PH stainless steel forging	PH stainless steel forging should be subsumed by the definition of "stainless steel." Precipitation hardened (PH) martensitic stainless steel.	There is nothing distinct in the AMR line items associated with this material that warrants special treatment. This change affects the following items: o IV R-188 B4.4-d o IV R-187 B4.4-c o IV R-185 B4.4-a o IV R-189 B4.4-e
Stainless steel	Wrought or forged austenitic, ferritic, martensitic, or duplex stainless steel (Cr content >11%) ***and cast austenitic stainless steel are grouped for AMRs under the term stainless steel. These materials are susceptible to a variety of aging effects and mechanisms including loss of material due to pitting and crevice corrosion and cracking due to stress corrosion cracking. However, cast austenitic stainless steel is also susceptible to loss of fracture toughness due to thermal and neutron irradiation embrittlement. Therefore, when this aging effect is being considered, cast austenitic stainless steel is specifically called out. Steel with stainless steel cladding may also be considered stainless steel when the aging effect is associated with the stainless steel surface of the material rather than the composite volume of the material.*** Examples of stainless steel designations that were earlier referenced in NUREG-1801 that comprise this category include A-286, SA193-Gr. B8, SA193-Gr. B8M, Gr. 660 (A-286), SA193-6, SA193-Gr. B8 or B-8M, SA453, Type 304, Type 304NG, Type 308, Type 308L, Type 309, Type 309L, Type 316, Type 347, Type 403, Type 416, ***CF-3, CF-8, CF-3M, and CF-8M.***	The added words address two changes. The first regognizes CASS to be the same as stainless steel for aging effects other than thermal and neutron irradiation embrittlement. This permits CASS to be treated as a subset of stainless steel. CASS is then only listed as a material when loss of fracture toughness due to thermal (or thermal and irradiation) embrittlement is at issue, or where unique AMP requirements are given. This provides consistency with GALL's treatment of other material groups, e.g., gray cast iron as a subset of steel, and copper alloy >15% zinc as a subset of copper alloy. Gray cast iron and copper alloy >15% zinc are both susceptible to selective leaching and are only listed as materials when selective leaching is addressed. The second change recognizes the stainless steel surface of cladding on steel to be the same as stainless steel for most common aging effects. For volumetric aging effect such as fatigue, steel with stainless steel cladding should be listed as a separate material.

IX Selected Definitions of Terms Used for Describing and Standardizing Structures, Components, Materials, Environments, Aging Effects, and Aging Mechanisms **Definition of Selected Terms**		
Standardized Expression	**Description of Change**	**Basis**
Steel	Malleable iron is referenced in Chapter VI of the Draft 2005 GALL. However, it is an undefined material in Chapter IX. "Malleable Iron" should be included in the definition of "Steel."	Materials used in the GALL tables should be included in Chapter IX.
Air	This term is undefined in Chapter IX although it is associated with AMR line Items AP-48 and SP-33 for Glass in Chapters VII and VIII. This environment should be replace in the AMR line Items with "Air – indoor controlled/uncontrolled, Air - outdoor."	Environments used in the GALL tables should be included in Chapter IX. This change affects the following items: o VII AP-48 J. o VIII SP-33 I.
Air - indoor	This term is undefined in Chapter IX although it is associated with Items LP-01 in Chapter VI. This term should be deleted from the AMR tables. The AMR line Item should be replaced with "Air – indoor uncontrolled (Internal/External)."	Changing the environment to "Air – indoor uncontrolled (Internal/External)" makes AMR line Item LP-01 and LP-02 in agreement. This change affects the following items: o VI LP-01 A.
Air, moist	"Air, moist" is duplicative" of "Condensation" and should be deleted.	Removing duplicate environments removes confusion from the GALL. This change affects the following items: o VII A-23 H2.2-a H2.3-a

IX Selected Definitions of Terms Used for Describing and Standardizing Structures, Components, Materials, Environments, Aging Effects, and Aging Mechanisms Definition of Selected Terms		
Standardized Expression	**Description of Change**	**Basis**
Closed cycle cooling water (see consolidated definitions at the end)	Delete the examples of closed cycle cooling water. The suggested rewording is as follows: Treated water subject to the closed cycle cooling water chemistry program. Closed cycle cooling water >60°C (>140°F) allows the possibility of stainless steel SCC. ~~Examples of environment descriptors that comprise this category include:~~ • ~~Chemically treated borated water; and treated component cooling water~~ • ~~Demineralized water on one side; closed cycle cooling water (treated water) on the other side~~ • ~~Chemically treated borated water on tube side and closed cycle cooling water on shell side~~	The examples should be removed as it may be viewed as limiting in terms of consistency.
Condensation (internal/external)	The definition is more complicated than required. A suggested rewording is as follows: The environment to which the internal or external surface of the component or structure is exposed. ~~Air and c~~ Condensation on *the* surfaces of ~~indoor~~ systems with temperatures below the dew point~~—for exterior surfaces and interior surfaces in communication ambient indoor air, condensation~~ is considered ***raw*** ~~untreated~~ water ~~due to potential for surface contamination~~.	The definition is simplified such that condensation can be considered anywhere it occurs. Also, "raw water" is already defined as water that may contain contaminants. See comments on the consolidation of untreated water and raw water.

IX Selected Definitions of Terms Used for Describing and Standardizing Structures, Components, Materials, Environments, Aging Effects, and Aging Mechanisms Definition of Selected Terms		
Standardized Expression	**Description of Change**	**Basis**
Containment environment (inert)	Add new environment. The drywell is made inert with nitrogen to render the primary containment atmosphere non-flammable by maintaining the oxygen content below 4% by volume during normal operation.	There currently is not an environment defined for inside the containment of BWRs, where the normal environment is inerted with nitrogen to less than 4% oxygen, as required by plant technical specifications. The environments "Air – indoor uncontrolled" and "Air – indoor uncontrolled or air – outdoor" as defined in GALL, and as explained in the basis document, include both "inside and outside containment", but make no reference to the low oxygen content inside the containment of BWRs. When the environment of the external surface of a component is containment atmosphere, and the oxygen content is <4%, then loss of material will not be an aging effect for steel, carbon steel, and other metals that would be susceptible to loss of material in an air environment. The NRC agreed to this position in the SER for the Dresden Quad Cities license renewal application (NUREG 1796). The following is taken from NUREG 1796, section 3.1.2.4.5 on page 3-160: "The staff agreed with the applicant that there are no applicable aging effects for external surfaces of carbon components exposed to a containment nitrogen environment because the low oxygen level present in the primary containment atmosphere precludes loss of material due to corrosion as a credible aging effect for the external surface of carbon steel components exposed to the containment environment." Conforming line items should be created in Chapters IV, V, VII, and VII. For example: Piping, piping components, and piping elements/Steel/Containment environment/None/None

IX Selected Definitions of Terms Used for Describing and Standardizing Structures, Components, Materials, Environments, Aging Effects, and Aging Mechanisms

Definition of Selected Terms

Standardized Expression	Description of Change	Basis
Sodium pentaborate solution	The definition is more complicated than required. A suggested rewording is as follows: ***Treated water that contains a mixture of borax and boric acid.*** ~~BWR operators are required to upgrade their standby liquid control systems with sodium pentaborate solution. The use of higher fuel enrichments and the popularity of MOX fuels tax reactivity controls at BWRs and enriched sodium pentaborate provides an excellent solution for these new requirements.~~	The original definition did not describe the solution.
Steam (dry)	This environment should be deleted. The BWR main steam system is considered dry (usually ~99.9% dry) but the PWR main steam system can contain moisture and thus not fall into this category.	Section VIII.B2 does not reference this environment.
~~Untreated water or~~ raw water ~~Water, raw~~	These environments are redundant and should be consolidated. A suggested rewording is as follows: Raw untreated fresh, salt, or ground water. Floor drains and reactor buildings and auxiliary building sumps may be exposed to a variety of untreated water that is thus classified as raw water for the determination of aging effects. ***Raw*** Water ~~that~~ may contain contaminants including oil and boric acid depending on the location and including originally treated water that is not monitored by a chemistry program. ~~Untreated is a very broad term that overlaps with raw water in that leaking groundwater can be included. Untreated water can also include liquid radwaste systems.~~	The consolidation of environments will add to the consistency in the usage of the GALL. This change affects the following items: o V E-34 C.1-b o V E-32 C.1-a o V E-01 D1.8-c o VII AP-67 K. o VII AP-29 C1. C3. G. H2. o VIII SP-28 G. o VIII S-12 G.1-d

IX Selected Definitions of Terms Used for Describing and Standardizing Structures, Components, Materials, Environments, Aging Effects, and Aging Mechanisms Definition of Selected Terms		
Standardized Expression	**Description of Change**	**Basis**
Cladding degradation	The definition in Chapter IX is not in line with the usage of this term in Chapter VII. AMR line Item A-40 refers to the lining of piping and piping elements. The lining is a design feature and is not credited as part of the pressure boundary. Therefore, the definition provided is not accurate. The SRP refers to the cracking of cladding. Suggested rewording: This refers to the degradation of the ~~stainless steel~~ cladding ~~(~~via any applicable degradation process ~~for stainless steel/applicable environment described in NUREG 1801)~~.	The definition needs revision so that it conforms to the usage in the SRP and the GALL.
Corrosion of carbon steel tube support plate	This standardized expression should be deleted. Corrosion (as defined above) of the carbon steel tube support plates which are plate-type component providing tube-tube mechanical support for the tubes in the tube bundle of the steam generator (recirculating) system of a PWR. The tubes pass through drill holes in the plate. The secondary coolant flows through the tube supports via flow holes between the tubes. [13, 14]	The definition of "corrosion" is adequate to encompass this combination of aging mechanism and location.

IX Selected Definitions of Terms Used for Describing and Standardizing Structures, Components, Materials, Environments, Aging Effects, and Aging Mechanisms Definition of Selected Terms		
Standardized Expression	**Description of Change**	**Basis**
Corrosion of embedded steel	This standardized expression should be deleted. If pH of the concrete in which steel is embedded is reduced (pH< 11.5) by intrusion of aggressive ions (e.g., chlorides > 500 ppm) in the presence of oxygen, embedded steel corrosion may occur. A reduction in pH may be caused by the leaching of alkaline products through cracks, entry of acidic materials, or carbonation. Chlorides may also be present in the constituents of the original concrete mix. The severity of the corrosion is affected by the properties and types of cement, aggregates, and moisture content. [9]	The definition of "corrosion" is adequate to encompass this combination of aging mechanism and environment .
Flow-accelerated corrosion (FAC)	The definition should be tied to the susceptibility reviews that can be performed per Section 4.2 of NSAC-202L-R2. Suggested rewording: Also termed erosion-corrosion. A co-joint activity involving corrosion and erosion in the presence of a moving corrosive fluid, leading to the accelerated loss of material. ***Susceptibility may be determined using the review process outlined in Section 4.2 of NSAC-202L-R2.***	NSAC-202L-R2 is used as a basis for AMP XI.M17 Flow-Accelerated Corrosion.
Outer Diameter Stress Corrosion Cracking (ODSCC)	This standardized expression should be deleted. Stress corrosion cracking initiating in the outer diameter (secondary side) surface of steam generator tubes. This differs from PWSCC which describes inner diameter (primary side) initiated cracking. [14]	The definition of "corrosion" is adequate to encompass this combination of aging mechanism and location.

IX	Selected Definitions of Terms Used for Describing and Standardizing Structures, Components, Materials, Environments, Aging Effects, and Aging Mechanisms Definition of Selected Terms	
Standardized Expression	**Description of Change**	**Basis**
Relaxation	This standardized expression should be deleted. The description can be added to the "stress relaxation." Relaxation in structural steel anchorage components can be an aging mechanism contributing to the aging effect of loss of prestress.	Loss of prestress due to relaxation is redundant to Loss of preload due to stress relaxation.

X.M1 METAL FATIGUE OF REACTOR COOLANT PRESSURE BOUNDARY

Item	Locator	Comment	Justification
1	Program Description	Eliminate change which added "and the severity."	This is adequately covered by the existing wording under the Parameters Monitored/Inspected subsection, and could be viewed as requiring the program to monitor severity of transients (which is not done for those programs that only count cycles).
2	Program Description	Restore original text regarding critical components.	Changes to the "Program Description" and "Monitoring and Trending" elements of the AMP suggest scope of critical components goes beyond those identified in NUREG/CR-6260. The Bases Document does not provide a technical justification for this change. Suggest leaving the original wording.
3	Monitoring and Trending	Restore original text regarding critical components.	Changes to the "Program Description" and "Monitoring and Trending" elements of the AMP suggest scope of critical components goes beyond those identified in NUREG/CR-6260. The Bases Document does not provide a technical justification for this change. Suggest leaving the original wording.
4	Corrective Actions	Restructure sentence to make original meaning clear.	Original (GALL 2001) sentence structure was awkward. Addition of word "during" attempted to make sentence read better, but completely changed original meaning. Proposed change resolves original problem.

X.M1 METAL FATIGUE OF REACTOR COOLANT PRESSURE BOUNDARY

Program Description

In order not to exceed the design limit on fatigue usage, the aging management program (AMP) monitors and tracks the number ~~and the severity~~ of critical thermal and pressure transients for the selected reactor coolant system components.

The AMP addresses the effects of the coolant environment on component fatigue life by assessing the impact of the reactor coolant environment on a sample of critical components ~~for the plant. Examples of critical~~ ***that includes, as a minimum, those*** components ~~are identified~~ ***selected*** in NUREG/CR-6260. The sample of critical components can be evaluated by applying environmental correction factors to the existing ASME Code fatigue analyses. Formulae for calculating the environmental life correction factors are contained in NUREG/CR-6583 for carbon and low-alloy steels and in NUREG/CR-5704 for austenitic stainless steels.

As evaluated below, this is an acceptable option for managing metal fatigue for the reactor coolant pressure boundary, considering environmental effects. Thus, no further evaluation is recommended for license renewal if the applicant selects this option under 10 CFR 54.21(c)(1)(iii) to evaluate metal fatigue for the reactor coolant pressure boundary.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program includes preventive measures to mitigate fatigue cracking of metal components of the reactor coolant pressure boundary caused by anticipated cyclic strains in the material.

2. ***Preventive Actions:*** Maintaining the fatigue usage factor below the design code limit and considering the effect of the reactor water environment, as described under the program description, will provide adequate margin against fatigue cracking of reactor coolant system components due to anticipated cyclic strains.

3. ***Parameters Monitored/Inspected:*** The program monitors all plant transients that cause cyclic strains, which are significant contributors to the fatigue usage factor. The number of plant transients that cause significant fatigue usage for each critical reactor coolant pressure boundary component is to be monitored. Alternatively, more detailed local monitoring of the plant transient may be used to compute the actual fatigue usage for each transient.

4. ***Detection of Aging Effects:*** The program provides for periodic update of the fatigue usage calculations.

5. ***Monitoring and Trending:*** The program monitors a sample of high fatigue usage locations. ***As a minimum, this***~~This~~ sample is to include the locations identified in NUREG/CR-6260 ~~and any additional critical components in the plant~~.

6. ***Acceptance Criteria:*** The acceptance criteria involves maintaining the fatigue usage below the design code limit considering environmental fatigue effects as described under the program description.

7. ***Corrective Actions:*** The program provides for corrective actions to prevent the usage factor from exceeding the design code limit during the period of extended operation. Acceptable corrective actions include ***repair of the component, replacement of the component, and a*** more rigorous analysis of the component to demonstrate that the design code limit will not be exceeded~~ during repair or replacement of the component~~. For programs that monitor a sample of high fatigue usage locations, corrective actions include a review of additional affected reactor coolant pressure boundary locations.

8. ***Conformation Process:*** Site quality assurance procedures, review and approval processes and administrative controls are implemented in accordance with the requirements of Appendix B to 10 CFR Part 50. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the confirmation process and administrative controls.

9. ***Administrative Controls:*** See Item 8, above.

10. ***Operating Experience:*** The program reviews industry experience regarding fatigue cracking. Applicable experience with fatigue cracking is to be considered in selecting the monitored locations.

References

NUREG/CR-5704, *Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels,* U.S. Nuclear Regulatory Commission, April 1999.

NUREG/CR-6260, *Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components,* U.S. Nuclear Regulatory Commission, March 1995.

NUREG/CR-6583, *Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels,* U.S. Nuclear Regulatory Commission, March 1998.

CHAPTER XI

AGING MANAGEMENT PROGRAMS (AMPs)

XI.M1 ASME SECTION XI INSERVICE INSPECTION, SUBSECTIONS IWB, IWC, AND IWD

Item	Locator	Comment	Justification
1	Program Description Footnote (ISI Footnote - XI.M1 & M3 through M9)	The footnote is based on ASME Section XI. The note discusses that the NRC adopts the use of updated versions of ASME XI in 10 CFR 50.55a but does not state that an applicant may credit the updated versions. The Bases Document for the revision to the GALL Report states that the addition of the code used in the plant's ISI program, which is based on 10 CFR 50.55a, can be used as an AMP in a LRA. Revise footnote wording as shown for XI.M1 in each AMP using the footnote (XI.M3 through M9).	The footnote added to several AMP program descriptions acknowledges that the ASME code required under 10CFR50.55a changes periodically but it does not clearly state the applicant can credit whatever code version is applicable during the period of extended operation. The rewording proposed corrects the discrepancy and will eliminate exceptions that applicants must take given the current wording. The proposed wording in the footnote will allow applicants to credit the revision of ASME XI that is credited in their current ISI plan as an acceptable aging management program for license renewal

XI.M1 ASME SECTION XI INSERVICE INSPECTION, SUBSECTIONS IWB, IWC, AND IWD

Program Description

The Code of Federal Regulations, 10 CFR 50.55a, imposes the inservice inspection (ISI) requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, for Class 1, 2, and 3 pressure-retaining components and their integral attachments in light-water cooled power plants. Inspection, repair, and replacement of these components are covered in Subsections IWB, IWC, and IWD, respectively in the 2001 edition[1] including the 2002 and 2003 Addenda. The program generally includes periodic visual, surface, and/or volumetric examination and leakage test of all Class 1, 2, and 3 pressure-retaining components and their integral attachments.

The ASME Section XI inservice inspection program in accordance with Subsections IWB, IWC, or IWD has been shown to be generally effective in managing aging effects in Class 1, 2, or 3 components and their integral attachments in light-water cooled power plants. However, in certain cases, the ASME inservice inspection program is to be augmented to manage effects of aging for license renewal and is so identified in the GALL report.

Evaluation and Technical Basis

1. ***Scope of Program:*** The ASME Section XI program provides the requirements for ISI, repair, and replacement. The components within the scope of the program are specified in Subsections IWB-1100, IWC-1100, and IWD-1100 for Class 1, 2, and 3 components, respectively, and include all pressure-retaining components and their integral attachments in light-water cooled power plants. The components described in Subsections IWB-1220, IWC-1220, and IWD-1220 are exempt from the examination requirements of Subsections IWB-2500, IWC-2500, and IWD-2500.

2. ***Preventive Actions:*** Operation within the limits prescribed in the Technical Specifications.

3. ***Parameters Monitored/Inspected:*** The ASME Section XI ISI program detects degradation of components by using the examination and inspection requirements specified in ASME Section XI Tables IWB-2500-1, IWC-2500-1, or IWD-2500-1, respectively, for Class 1, 2, or 3 components.

4. ***Detection of Aging Effects:*** The extent and schedule of the inspection and test techniques prescribed by the program are designed to maintain structural integrity and ensure that

[1] 10 CFR 50.55a is revised periodically to adopt, by reference, new editions and addenda of the ASME Code. For each successive 120-month (10 year) inspection interval, applicants are required to revise the nuclear plant's ISI program to incorporate the requirements specified in the version of the ASME Code incorporated into 10 CFR 50.55a 12 months before the start of the inspection interval. ***Because a plant's 10 year ISI programs is based on the edition of the ASME Code when the ISI program is prepared, the ISI program based on any edition and addenda of the ASME Code adopted by the NRC in 10 CFR 50.55a is an acceptable aging management program that can be credited in a license renewal application as consistent with NUREG 1801 without justifying exceptions.*** ~~NRC statements of consideration (SOC) associated with the adoption of new editions and addenda of the ASME Code in 10 CFR 50.55a may discuss the adequacy of the newer edition and addendum as they relate to the GALL Report. The information contained in these SOCs may provide a reasonable basis for exceptions relating to use of editions or addenda of the ASME Code that are not the same as those identified in the GALL Report.~~

aging effects will be discovered and repaired before the loss of intended function of the component. Inspection can reveal crack initiation and growth; loss of material due to corrosion; leakage of coolant; and indications of degradation due to wear or stress relaxation, such as verification of clearances, settings, physical displacements, loose or missing parts, debris, wear, erosion, or loss of integrity at bolted or welded connections.

Components are examined and tested as specified in Tables IWB-2500-1, IWC-2500-1, and IWD-2500-1, respectively, for Class 1, 2, and 3 components. The tables specify the extent and schedule of the inspection and examination methods for the components of the pressure-retaining boundaries. Alternative approved methods that meet the requirements of IWA-2240 are also specified in these tables.

The program uses three types of examination — visual, surface, and volumetric — in accordance with the general requirements of Subsection IWA-2000. Visual VT-1 examination detects discontinuities and imperfections, such as cracks, corrosion, wear, or erosion, on the surface of components. Visual VT-2 examination detects evidence of leakage from pressure-retaining components, as required during the system pressure test. Visual VT-3 examination (a) determines the general mechanical and structural condition of components and their supports by verifying parameters, such as clearances, settings, and physical displacements; (b) detects discontinuities and imperfections, such as loss of integrity at bolted or welded connections, loose or missing parts, debris, corrosion, wear, or erosion; and (c) observes conditions that could affect operability or functional adequacy of constant-load and spring-type components and supports.

Surface examination uses magnetic particle, liquid penetrant, or eddy current examinations to indicate the presence of surface discontinuities and flaws.

Volumetric examination uses radiographic, ultrasonic, or eddy current examinations to indicate the presence of discontinuities or flaws throughout the volume of material included in the inspection program.

For BWRs, the nondestructive examination (NDE) techniques appropriate for inspection of vessel internals, including the uncertainties inherent in delivering and executing an NDE technique in a boiling water reactor (BWR), are included in the approved boiling water reactor vessel and internals project (BWRVIP)-03. Also, an applicant may use the guidelines of the approved BWRVIP-62 for inspection relief for vessel internal components with hydrogen water chemistry.

The ASME Section XI examination categories used in this report are given below.

Class 1 Components, Table IWB-2500-1

Examination category B-B for pressure-retaining welds in vessels other than reactor vessels: This category specifies volumetric examination of circumferential and longitudinal shell-to-head welds and circumferential and meridional head welds in pressurizers, and circumferential and meridional head welds and tubesheet-to-head welds in steam generators (primary side). The welds selected during the first inspection interval are reexamined during successive inspection intervals.

Examination category B-D, for full penetration welds of nozzles in reactor vessels, pressurizers, steam generators (primary side), and heat exchangers (primary side): This

category specifies volumetric examination of all nozzle-to-vessel welds and the nozzle inside surface.
Examination category B-E, for pressure-retaining partial penetration welds in vessels: This category specifies visual VT-2 examination of partial penetration welds in nozzles and penetrations in reactor vessels and pressurizers during the hydrostatic test. In the 1995 edition of the ASME Code, examination category B-E is covered under examination category B-P.

Examination category B-F, for pressure-retaining dissimilar metal welds in reactor vessels, pressurizers, steam generators, heat exchangers, and piping: This category specifies volumetric examination of the inside diameter (ID) region and surface examination of the outside diameter (OD) surface for all nozzle-to-safe end butt welds of nominal pipe size (NPS) 4 in. or larger. Only surface examination is conducted for all butt welds less than NPS 4 in. and for all nozzle-to-safe end socket welds. Examinations are required for each safe end weld in each loop and connecting branch of the reactor coolant system. In the 1995 edition of the ASME Code, examination category B-F for piping is covered under examination category B-J for all pressure-retaining welds in piping.

Examination category B-G-1 for pressure-retaining bolting greater than 2 in. in diameter, and category B-G-2 for pressure-retaining bolting less than 2 in. in diameter in reactor vessels, pressurizers, steam generators, heat exchangers, piping, pumps, and valves:
Category B-G-1 specifies volumetric examination of studs in place, from the top of the nut to the bottom of the flange hole; and surface and volumetric examination of studs when removed; volumetric examination of flange threads; and visual VT-1 examination of the surfaces of nuts, washers, and bushings. Category B-G-2 specifies visual VT-1 examination of the surfaces of nuts, washers, and bushings. For heat exchangers, piping, pumps, and valves, examinations are limited to components selected for examination under examination categories B-B, B-J, B-L-2, and B-M-2.

Examination category B-H for integral attachments for vessels: This category specifies volumetric or surface examination of essentially 100% of the length of the attachment weld at each attachment subject to examination.

Examination category B-J for pressure-retaining welds in piping: This category specifies volumetric examination of the ID region and surface examination of the OD for circumferential and longitudinal welds in each pipe or branch run NPS 4 in. or larger. Surface examination is conducted for circumferential and longitudinal welds in each pipe or branch run less than NPS 4 in. and for all socket welds. The pipe welds selected during the first inspection interval are reexamined during each successive inspection interval.

Examination category B-L-1, for pressure-retaining welds in pump casing, and category B-L-2, for pump casing: Category B-L-1 specifies volumetric examination of all welds, and category B-L-2 specifies visual VT-3 examination of internal surfaces of the pump casing. All welds from at least one pump in each group of pumps performing similar functions in the system (such as recirculating coolant pumps) are inspected during each inspection interval. Visual examination is required only when the pump is disassembled for maintenance, repair, or volumetric examination, but one pump in a particular group of pumps is visually examined at least once during the inspection interval.

Examination category B-M-1, for pressure-retaining welds in valve bodies and category B-M-2, for valve bodies: Category B-M-1 specifies volumetric examination for all welds in valve bodies NPS 4 in. or larger, and surface examination of OD surfaces for all welds in valve bodies less than NPS 4 in. Category B-M-2 specifies visual VT-3 examination of internal surfaces of valve bodies. All welds from at least one valve in each group of valves that are of the same size, construction design (such as globe, gate, or check valves), and manufacturing method, and that perform similar functions in the system (such as the containment isolation valve) are inspected during each inspection interval. Visual examination is required only when the valve is disassembled for maintenance, repair, or volumetric examination, but one valve in a particular group of valves is visually examined at least once during the inspection interval.

Examination category B-N-1, for the interior of reactor vessels: Category B-N-1 specifies visual VT-3 examination of interior surfaces that are made accessible for examination by removal of components during normal refueling outages.

Examination category B-N-2, for integrally welded core support structures and interior attachments to reactor vessels: Category B-N-2 specifies visual VT-1 examination of all accessible welds in interior attachments within the beltline region; visual VT-3 examination of all accessible welds in interior attachments beyond the beltline region; and, for BWRs, visual VT-3 examination of all accessible surfaces in the core support structure.

Examination category B-N-3, which is applicable to pressurized water reactors (PWRs), for removable core support structures: Category B-N-3 specifies visual VT-3 examination of all accessible surfaces of reactor core support structures that can be removed from the reactor vessel.

Examination category B-O, for pressure-retaining welds in control rod housing: This category specifies volumetric or surface examination of the control rod drive (CRD) housing welds, including the weld buttering.

Examination category B-P, for all pressure-retaining components: This category specifies visual VT-2 examination of all pressure-retaining boundary components during the system leakage test and hydrostatic test (IWA-5000 and IWB-5000). The pressure-retaining boundary during the system leakage test corresponds to the reactor coolant system boundary, with all valves in the normal position, which is required for normal reactor operation startup. However, VT-2 visual examination extends to and includes the second closed valve at the boundary extremity. The 1995 edition of the ASME Code eliminates the hydrostatic test because equivalent results are obtained from the leakage test. The pressure-retaining boundary for the hydrostatic test (1989 edition) and system leakage test (1995 edition) conducted at or near the end of each inspection interval extends to all Class 1 pressure-retaining components within the system boundary.

Class 2 Components, Table IWC-2500-1

Examination category C-A, for pressure-retaining welds in pressure vessels: This category specifies volumetric examination of circumferential welds at gross structural discontinuities, such as junctions between shells of different thickness or cylindrical shell-to-conical shell junctions, and head-to-shell, shell (or head)-to-flange, and tubesheet-to-shell welds.

Examination category C-F-1, for pressure-retaining welds in austenitic stainless steel or high-alloy piping: This category specifies, for circumferential and longitudinal welds in each pipe or branch run NPS 4 in. or larger, volumetric and surface examination of the ID region, and surface examination of the OD surface for piping welds ≥3/8 in. wall thickness for piping >NPS 4 in. or for piping welds >1/5 in. wall thickness for piping ≥NPS 2 in. and ≤NPS 4 in. Surface examination is conducted for circumferential and longitudinal welds in pipe branch connections of branch piping ≥NPS 2 in. and for socket welds.

Examination category C-G, for all pressure-retaining welds in pumps and valves: This category specifies surface examination of either the inside or outside surface of all welds in the pump casing and valve body. In a group of multiple pumps or valves of similar design, size, function, and service in a system, examination of only one pump or one valve among each group of multiple pumps or valves is required to detect the loss of intended function of the pump or valve.

Examination category C-H, for all pressure-retaining components: This category specifies visual VT-2 examination during system pressure tests (IWA-5000 and IWC-5000) of all pressure-retaining boundary components. The pressure-retaining boundary includes only those portions of the system required to operate or support the safety function, up to and including the first normally closed valve (including a safety or relief valve) or valve capable of automatic closure when the safety function is required. The 1995 edition of the ASME Code eliminates the hydrostatic test because equivalent results are obtained from the leakage test.

Class 3 Components, Table IWD-2500-1

Examination category D-A (1989 edition), for systems in support of reactor shutdown function, and category D-B (1989 edition), for systems in support of emergency core cooling, containment heat removal, atmosphere cleanup, and reactor residual heat removal: Categories D-A and D-B specify visual VT-2 examination during system pressure tests (IWA-5000 and IWD-5000) of all pressure-retaining boundary components. The pressure-retaining boundary extends up to and includes the first normally closed valve or valve capable of automatic closure as required to perform the safety-related system function. Examination categories D-A and D-B, from the 1989 edition of the ASME Code, have been combined into examination category D-B for all pressure-retaining components in the 1995 edition of the ASME Code.

5. ***Monitoring and Trending:*** For Class 1, 2, or 3 components, the inspection schedule of IWB-2400, IWC-2400, or IWD-2400, respectively, and the extent and frequency of IWB-2500-1, IWC-2500-1, or IWD-2500-1, respectively, provides for timely detection of degradation. The sequence of component examinations established during the first inspection interval is repeated during each successive inspection interval, to the extent practical. If flaw conditions or relevant conditions of degradation are evaluated in accordance with IWB-3100, IWC-3100, or IWD-3100 , and the component is qualified as acceptable for continued service, the areas containing such flaw indications and relevant conditions are reexamined during the next three inspection periods of IWB-2110 for Class 1 components, IWC-2410 for Class 2 components, and IWD-2410 for Class 3 components. Examinations that reveal indications that exceed the acceptance standards described below are extended to include additional examinations in accordance with IWB-2430, IWC-2430, or IWD-2430 (1995 edition) for Class 1, 2, or, 3 components, respectively.

6. ***Acceptance Criteria:*** Any indication or relevant conditions of degradation detected are evaluated in accordance with IWB-3000, IWC-3000, or IWD-3000, for Class 1, 2, or 3 components, respectively. Examination results are evaluated in accordance with IWB-3100,

or IWC-3100, or IWD-3100 by comparing the results with the acceptance standards of IWB-3400 and IWB3500, or IWC-3400 and IWC-3500, or IWD3400 and IWD3500, respectively for Class 1 or Class 2 and 3 components. Flaws that exceed the size of allowable flaws, as defined in IWB-3500 or IWC-3500 or IWD3500, are evaluated by using the analytical procedures of IWB-3600 or IWC-3600 or IWD-3600, respectively, for Class 1 or Class 2, and 3 components. Flaws that exceed the size of allowable flaws, as defined in IWB-3500 or IWC-3500, are evaluated by using the analytical procedures of IWB-3600 or IWC-3600, respectively, for Class 1 or Class 2 and 3 components. Approved BWRVIP-14, BWRVIP-59, and BWRVIP-60 documents provide guidelines for evaluation of crack growth in stainless steels, nickel alloys, and low-alloy steels, respectively.

7. ***Corrective Actions:*** For Class 1, 2, and 3, respectively, repair is in conformance with IWB-4000, IWC-4000, and IWD-4000, and replacement according to IWB-7000, IWC-7000, and IWD-7000. Approved BWRVIP-44 and BWRVIP-45 documents, respectively, provide guidelines for weld repair of nickel alloys and for weldability of irradiated structural components. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.

8. ***Confirmation Process:*** Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the confirmation process and administrative controls.

9. ***Administrative Controls:*** See Item 8, above.

10. ***Operating Experience:*** Because the ASME Code is a consensus document that has been widely used over a long period, it has been shown to be generally effective in managing aging effects in Class 1, 2, and 3 components and their integral attachments in light-water cooled power plants (see Chapter I of the GALL report, Vol. 2).

    Some specific examples of operating experience of component degradation are as follows:

    BWR: Cracking due to intergranular stress corrosion cracking (IGSCC) has occurred in small- and large-diameter BWR piping made of austenitic stainless steels and nickel alloys. The IGSCC has also occurred in a number of vessel internal components, such as core shrouds, access hole covers, top guides, and core spray spargers (NRC IE Bulletin 80-13, NRC Information Notice [IN] 95-17, NRC General Letter [GL] 94-03, and NUREG-1544). Crack initiation and growth due to thermal and mechanical loading have occurred in high-pressure coolant injection (HPCI) piping (NRC IN 89-80) and instrument lines NRC Licensee Event Report [LER] 50-249/99-003-1). Jet pump BWRs are designed with access holes in the shroud support plate at the bottom of the annulus between the core shroud and the reactor vessel wall. These holes are used for access during construction and are subsequently closed by welding a plate over the hole. Both circumferential (NRC IN 88-03) and radial cracking (NRC IN 92-57) have been observed in access hole covers. Failure of the isolation condenser tube bundles due to thermal fatigue and transgranular stress corrosion cracking (TGSCC) due to leaky valves has also occurred (NRC LER 50-219/98-014).

    PWR Primary System: Although the primary pressure boundary piping of PWRs has generally not been found to be affected by SCC because of low dissolved oxygen levels and control of primary water chemistry, SCC has occurred in safety injection lines (NRC IN 97-19 and 84-18), charging pump casing cladding (NRC IN 80-38 and 94-63), instrument nozzles in safety injection tanks (NRC IN 91-05), CRD seal housing (NRC Inspection Report 50-255/99012), and safety-related stainless steel (SS) piping systems that contain oxygenated, stagnant, or essentially stagnant borated coolant (NRC IN 97-19). Cracking has occurred in SS baffle former bolts in a number of foreign plants (NRC IN 98-11) and has now been observed in plants in the United States. Crack initiation and growth due to thermal and mechanical loading has occurred in high-pressure injection and safety injection piping (NRC IN 97-46 and NRC BL 88-08).

PWR Secondary System: Steam generator tubes have experienced outside diameter stress corrosion cracking (ODSCC), intergranular attack (IGA), wastage, and pitting (NRC IN 97-88). Carbon steel support plates in steam generators have experienced general corrosion. Steam generator shells have experienced pitting and stress corrosion cracking (NRC INs 82-37, 85-65, and 90-04).

References

10 CFR 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2000.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, ASME Boiler and Pressure Vessel Code, 2001 edition including the 2002 and 2003 Addenda, American Society of Mechanical Engineers, New York, NY.

BWRVIP-03, *BWR Vessel and Internals Project, Reactor Pressure Vessel and Internals Examination Guidelines, (EPRI TR-105696 R1, March 30, 1999),* Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation for BWRVIP-03, July 15, 1999.

BWRVIP-14, *Evaluation of Crack Growth in BWR Stainless Steel RPV Internals, (EPRI TR-105873, July 11, 2000),* Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation for BWRVIP-14, December 3, 1999.

BWRVIP-44, *Underwater Weld Repair of Nickel Alloy Reactor Vessel Internals, (EPRI TR-108708, April 3, 2000),* Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation for BWRVIP-44, June 9, 1999.

BWRVIP-45, *Weldability of Irradiated LWR Structural Components, (EPRI TR-108707),* BWRVIP and Electric Power Research Institute, Palo Alto, CA, June 14, 2000.

BWRVIP-59, *Evaluation of Crack Growth in BWR Nickel-Base Austenitic Alloys in RPV Internals, (EPRI TR-108710),* BWRVIP and Electric Power Research Institute, Palo Alto, CA, March 24, 2000.

BWRVIP-60, *BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Low Alloy Steel RPV Internals, (EPRI TR-108709, April 14, 2000),* Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation for BWRVIP-60, July 8, 1999.

BWRVIP-62, *BWR Vessel and Internals Project, Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection, (EPRI TR-108705),* BWRVIP and Electric Power Research Institute, Palo Alto, CA, March 7, 2000.

NRC Bulletin 88-08, *Thermal Stresses in Piping Connected to Reactor Coolant System,* U.S. Nuclear Regulatory Commission, June 22, 1988; Supplement 1, June 24, 1988; Supplement 2, September 4, 1988; Supplement 3, April 4, 1989.

NRC Generic Letter 94-03, *Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors*, U.S. Nuclear Regulatory Commission, July 25, 1994.

NRC IE Bulletin 80-13, *Cracking in Core Spray Spargers*, U.S. Nuclear Regulatory Commission, May 12, 1980.

NRC Information Notice 80-38, *Cracking in Charging Pump Casing Cladding*, U.S. Nuclear Regulatory Commission, October 31, 1980.

NRC Information Notice 82-37, *Cracking in the Upper Shell to Transition Cone Girth Weld of a Steam Generator at an Operating PWR*, U.S. Nuclear Regulatory Commission, September 16, 1982.

NRC Information Notice 84-18, *Stress Corrosion Cracking in PWR Systems*, U.S. Nuclear Regulatory Commission, March 7, 1984.

NRC Information Notice 85-65, *Crack Growth in Steam Generator Girth Welds*, U.S. Nuclear Regulatory Commission, July 31, 1985.

NRC Information Notice 88-03, *Cracks in Shroud Support Access Hole Cover Welds*, U.S. Nuclear Regulatory Commission, February 2, 1988.

NRC Information Notice 89-80, *Potential for Water Hammer, Thermal Stratification, and Steam Binding in High-Pressure Coolant Injection Piping*, U.S. Nuclear Regulatory Commission, December 1, 1989.

NRC Information Notice 90-04, *Cracking of the Upper Shell-to-Transition Cone Girth Welds in Steam Generators*, U.S. Nuclear Regulatory Commission, January 26, 1990.

NRC Information Notice 91-05, *Intergranular Stress Corrosion Cracking in Pressurized Water Reactor Safety Injection Accumulator Nozzles*, U.S. Nuclear Regulatory Commission, January 30, 1991.

NRC Information Notice 92-57, *Radial Cracking of Shroud Support Access Hole Cover Welds*, U.S. Nuclear Regulatory Commission, August 11, 1992.

NRC Information Notice 94-63, *Boric Acid Corrosion of Charging Pump Casing Caused by Cladding Cracks*, U.S. Nuclear Regulatory Commission, August 30, 1994.

NRC Information Notice 95-17, *Reactor Vessel Top Guide and Core Plate Cracking*, U.S. Nuclear Regulatory Commission, March 10, 1995.

NRC Information Notice 97-19, *Safety Injection System Weld Flaw at Sequoyah Nuclear Power Plant, Unit 2*, U.S. Nuclear Regulatory Commission, April 18, 1997.

NRC Information Notice 97-46, *Unisolable Crack in High-Pressure Injection Piping*, U.S. Nuclear

Regulatory Commission, July 9, 1997.

NRC Information Notice 97-88, *Experiences During Recent Steam Generator Inspections*, U.S. Nuclear Regulatory Commission, December 16, 1997.

NRC Information Notice 98-11, *Cracking of Reactor Vessel Internal Baffle Former Bolts in Foreign Plants*, U.S. Nuclear Regulatory Commission, March 25, 1998.

NRC Inspection Report 50-255/99012, *Palisades Inspection Report*, Item E8.2, Licensee Event Report 50-255/99-004, *Control Rod Drive Seal Housing Leaks and Crack Indications*, U.S. Nuclear Regulatory Commission, January 12, 2000.

NRC Licensee Event Report LER 50-219/98-014-00, *Failure of the Isolation Condenser Tube Bundles due to Thermal Stresses/Transgranular Stress Corrosion Cracking Caused by Leaky Valve*, U.S. Nuclear Regulatory Commission, October 29, 1998.

NRC Licensee Event Report LER 50-249/ 99-003-01, *Supplement to Reactor Recirculation B Loop, High Pressure Flow Element Venturi Instrument Line Steam Leakage Results in Unit 3 Shutdown Due to Fatigue Failure of Socket Welded Pipe Joint*, U.S. Nuclear Regulatory Commission, August 30, 1999.

XI.M2 WATER CHEMISTRY

Item	Locator	Comment	Justification
1	Program Description	Add words as shown.	EPRI guidelines change with experience. Plant chemistry programs generally adopt new guidance. Since we will clearly use later editions than that listed in GALL, we do not want to tie ourselves to a specific edition. GALL water chemistry program description already permits use of "later revisions or updates of these reports as approved by the staff." The proposed wording will permit an applicant to credit a revision of the EPRI guidelines that has been reviewed and accepted as part of a previous application.
2	Scope of Program, Preventive Actions, and Detection of Aging Effects	Changed "crack initiation and growth" to "cracking"	Consistent with current GALL terminology.

XI.M2 WATER CHEMISTRY

Program Description

The main objective of this program is to mitigate damage caused by corrosion and stress corrosion cracking (SCC). The water chemistry program for boiling water reactors (BWRs) relies on monitoring and control of reactor water chemistry based on guidelines in the boiling water reactor vessel and internals project (BWRVIP)-29 (Electric Power Research Institute [EPRI] TR-103515). The BWRVIP-29 has three sets of guidelines: one for primary water, one for condensate and feedwater, and one for control rod drive (CRD) mechanism cooling water. The water chemistry program for pressurized water reactors (PWRs) relies on monitoring and control of reactor water chemistry based on the EPRI guidelines in TR-105714 for primary water chemistry and TR-102134 for secondary water chemistry. ***Later versions of these chemistry program guidelines are developed from collective operating experience using sound technical judgment, and are approved by the electric utility industry in an effort to constantly improve water chemistry and thereby manage or prevent aging effects. The later versions of these guidelines when reviewed and approved for implementation by the staff in applicant Safety Evaluation Reports may be used in lieu of the revisions or versions specified above.***

The water chemistry programs are generally effective in removing impurities from intermediate and high flow areas. The Generic Aging Lessons Learned (GALL) report identifies those circumstances in which the water chemistry program is to be augmented to manage the effects of aging for license renewal. For example, the water chemistry program may not be effective in low flow or stagnant flow areas. Accordingly, in certain cases as identified in the GALL report, verification of the effectiveness of the chemistry control program is undertaken to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period of operation. As discussed in the GALL report for these specific cases, an acceptable verification program is a one-time inspection of selected components at susceptible locations in the system.

Evaluation and Technical Basis

1. *Scope of Program:* The program includes periodic monitoring and control of known detrimental contaminants such as chlorides, fluorides (PWRs only), dissolved oxygen, and sulfate concentrations below the levels known to result in loss of material or crack***ing*** ~~initiation and growth~~. Water chemistry control is in accordance with the guidelines in BWRVIP-29 (EPRI TR-103515) for water chemistry in BWRs; EPRI TR-105714, Rev. 3, for primary water chemistry in PWRs; EPRI TR-102134, Rev. 3, for secondary water chemistry in PWRs; or later revisions or updates of these reports as approved by the staff.

2. *Preventive Actions:* The program includes specifications for chemical species, sampling and analysis frequencies, and corrective actions for control of reactor water chemistry. System water chemistry is controlled to minimize contaminant concentration and mitigate loss of material due to general, crevice and pitting corrosion and crack***ing*** ~~initiation and growth~~ caused by SCC. For BWRs, maintaining high water purity reduces susceptibility to SCC.

3. *Parameters Monitored/Inspected:* The concentration of corrosive impurities listed in the EPRI guidelines discussed above, which include chlorides, fluorides (PWRs only), sulfates, dissolved oxygen, and hydrogen peroxide, are monitored to mitigate degradation of

structural materials. Water quality (pH and conductivity) is also maintained in accordance with the guidance. Chemical species and water quality are monitored by in process methods or through sampling. The chemistry integrity of the samples is maintained and verified to ensure that the method of sampling and storage will not cause a change in the concentration of the chemical species in the samples.

BWR Water Chemistry: The guidelines in BWRVIP-29 (EPRI TR-103515) for BWR reactor water recommend that the concentration of chlorides, sulfates, and dissolved oxygen are monitored and kept below the recommended levels to mitigate corrosion. The two impurities, chlorides and sulfates, determine the coolant conductivity; dissolved oxygen, hydrogen peroxide, and hydrogen determine electrochemical potential (ECP). The EPRI guidelines recommend that the coolant conductivity and ECP are also monitored and kept below the recommended levels to mitigate SCC and corrosion in BWR plants. The EPRI guidelines in BWRVIP-29 (TR-103515) for BWR feedwater, condensate, and control rod drive water recommends that conductivity, dissolved oxygen level, and concentrations of iron and copper (feedwater only) are monitored and kept below the recommended levels to mitigate SCC. The EPRI guidelines in BWRVIP-29 (TR-103515) also include recommendations for controlling water chemistry in auxiliary systems: torus/pressure suppression chamber, condensate storage tank, and spent fuel pool.

PWR Primary Water Chemistry: The EPRI guidelines (EPRI TR-105714) for PWR primary water chemistry recommend that the concentration of chlorides, fluorides, sulfates, lithium, and dissolved oxygen and hydrogen are monitored and kept below the recommended levels to mitigate SCC of austenitic stainless steel, Alloy 600, and Alloy 690 components. TR-105714 provides guidelines for chemistry control in PWR auxiliary systems such as boric acid storage tank, refueling water storage tank, spent fuel pool, letdown purification systems, and volume control tank.

PWR Secondary Water Chemistry: The EPRI guidelines (EPRI TR-102134) for PWR secondary water chemistry recommend monitoring and control of chemistry parameters (e.g., pH level, cation conductivity, sodium, chloride, sulfate, lead, dissolved oxygen, iron, copper, and hydrazine) to mitigate steam generator tube degradation caused by denting, intergranular attack (IGA), outer diameter stress corrosion cracking (ODSCC), or crevice and pitting corrosion. The monitoring and control of these parameters, especially the pH level, also mitigates general (carbon steel components), crevice, and pitting corrosion of the steam generator shell and the balance of plant materials of construction (e.g., carbon steel, stainless steel, and copper).

4. ***Detection of Aging Effects:*** This is a mitigation program and does not provide for detection of any aging effects, such as loss of material and crack*ing* ~~initiation and growth~~.

   In certain cases as identified in the GALL report, inspection of select components is to be undertaken to verify the effectiveness of the chemistry control program and to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period of operation.

5. ***Monitoring and Trending:*** The frequency of sampling water chemistry varies (e.g., continuous, daily, weekly, or as needed) based on plant operating conditions and the EPRI water chemistry guidelines. Whenever corrective actions are taken to address an abnormal chemistry condition, increased sampling is utilized to verify the effectiveness of these actions.

6. ***Acceptance Criteria:*** Maximum levels for various contaminants are maintained below the system specific limits as indicated by the limits specified in the corresponding EPRI water chemistry guidelines. Any evidence of the presence of aging effects or unacceptable water chemistry results is evaluated, the root cause identified, and the condition corrected.

7. ***Corrective Actions:*** When measured water chemistry parameters are outside the specified range, corrective actions are taken to bring the parameter back within the acceptable range and within the time period specified in the EPRI water chemistry guidelines. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.

8. ***Confirmation Process:*** Following corrective actions, additional samples are taken and analyzed to verify that the corrective actions were effective in returning the concentrations of contaminants such as chlorides, fluorides, sulfates, dissolved oxygen, and hydrogen peroxide to within the acceptable ranges. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the confirmation process.

9. ***Administrative Controls:*** Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing administrative controls.

10. ***Operating Experience:*** The EPRI guideline documents have been developed based on plant experience and have been shown to be effective over time with their widespread use. The specific examples of operating experience are as follows:

    BWR: Intergranular stress corrosion cracking (IGSCC) has occurred in small- and large-diameter BWR piping made of austenitic stainless steels and nickel-base alloys. Significant cracking has occurred in recirculation, core spray, residual heat removal (RHR) systems, and reactor water cleanup (RWCU) system piping welds. IGSCC has also occurred in a number of vessel internal components, including core shroud, access hole cover, top guide, and core spray spargers (Nuclear Regulatory Commission [NRC] IE Bulletin 80-13, NRC Information Notice [IN] 95-17, NRC General Letter [GL] 94-03, and NUREG-1544). No occurrence of SCC in piping and other components in standby liquid control systems exposed to sodium pentaborate solution has ever been reported (NUREG/CR-6001).

    PWR Primary System: The primary pressure boundary piping of PWRs has generally not been found to be affected by SCC because of low dissolved oxygen levels and control of primary water chemistry. However, the potential for SCC exists due to inadvertent introduction of contaminants into the primary coolant system from unacceptable levels of contaminants in the boric acid; introduction through the free surface of the spent fuel pool, which can be a natural collector of airborne contaminants; or introduction of oxygen during cooldown (NRC IN 84–18). Ingress of demineralizer resins into the primary system has caused IGSCC of Alloy 600 vessel head penetrations (NRC IN 96-11, NRC GL 97-01). Inadvertent introduction of sodium thiosulfate into the primary system has caused IGSCC of steam generator tubes. The SCC has occurred in safety injection lines (NRC INs 97-19 and 84-18), charging pump casing cladding (NRC INs 80-38 and 94-63), instrument nozzles in safety injection tanks (NRC IN 91-05), and safety-related SS piping systems that contain oxygenated, stagnant, or essentially stagnant borated coolant (NRC IN 97-19). Steam generator tubes and plugs and Alloy 600 penetrations have experienced primary water stress corrosion cracking (PWSCC) (NRC INs 89-33, 94-87, 97-88, 90-10, and 96-11; NRC Bulletin 89-01 and its two supplements).

PWR Secondary System: Steam generator tubes have experienced ODSCC, IGA, wastage, and pitting (NRC IN 97-88, NRC GL 95-05). Carbon steel support plates in steam generators have experienced general corrosion. The steam generator shell has experienced pitting and stress corrosion cracking (NRC INs 82-37, 85-65, and 90-04).

Such operating experience has provided feedback to revisions of the EPRI water chemistry guideline documents.

References

BWRVIP-29 (EPRI TR-103515), *BWR Water Chemistry Guidelines-1993 Revision, Normal and Hydrogen Water Chemistry*, Electric Power Research Institute, Palo Alto, CA, February 1994.

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

EPRI TR-105714, *PWR Primary Water Chemistry Guidelines–Revision 3*, Electric Power Research Institute, Palo Alto, CA, Nov. 1995.

NRC IE Bulletin 80-13, *Cracking in Core Spray Piping*, U.S. Nuclear Regulatory Commission, May 12, 1980.

NRC IE Bulletin 89-01, *Failure of Westinghouse Steam Generator Tube Mechanical Plugs*, U.S. Nuclear Regulatory Commission, May 15, 1989.

NRC IE Bulletin 89-01, Supplement 1, *Failure of Westinghouse Steam Generator Tube Mechanical Plugs*, U.S. Nuclear Regulatory Commission, November 14, 1989.

NRC IE Bulletin 89-01, Supplement 2, *Failure of Westinghouse Steam Generator Tube Mechanical Plugs*, U.S. Nuclear Regulatory Commission, June 28, 1991.

NRC Generic Letter 94-03, *Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors*, U.S. Nuclear Regulatory Commission, July 25, 1994.

NRC Generic Letter 95-05, *Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking*, U.S. Nuclear Regulatory Commission, August 3, 1995.

NRC Generic Letter 97-01, *Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations*, U.S. Nuclear Regulatory Commission, April 1,1997.

NRC Information Notice 80-38, *Cracking In Charging Pump Casing Cladding*, U.S. Nuclear Regulatory Commission, October 31, 1980.

NRC Information Notice 82-37, *Cracking in the Upper Shell to Transition Cone Girth Weld of a Steam Generator at an Operating PWR*, U.S. Nuclear Regulatory Commission, September 16,1982.

NRC Information Notice 84-18, *Stress Corrosion Cracking in Pressurized Water Reactor Systems*, U.S. Nuclear Regulatory Commission, March 7, 1984.

NRC Information Notice 85-65, *Crack Growth in Steam Generator Girth Welds*, U.S. Nuclear Regulatory Commission, July 31, 1985.

NRC Information Notice 89-33, *Potential Failure of Westinghouse Steam Generator Tube Mechanical Plugs*, U.S. Nuclear Regulatory Commission, March 23, 1989.

NRC Information Notice 90-04, *Cracking of the Upper Shell-to-Transition Cone Girth Welds in Steam Generators*, U.S. Nuclear Regulatory Commission, January 26, 1990.

NRC Information Notice 90-10, *Primary Water Stress Corrosion Cracking (PWSCC) of Inconel 600*, U.S. Nuclear Regulatory Commission, February 23, 1990.

NRC Information Notice 91-05, *Intergranular Stress Corrosion Cracking In Pressurized Water Reactor Safety Injection Accumulator Nozzles*, U.S. Nuclear Regulatory Commission, January 30, 1991.

NRC Information Notice 94-63, *Boric Acid Corrosion of Charging Pump Casing Caused by Cladding Cracks*, U.S. Nuclear Regulatory Commission, August 30, 1994.

NRC Information Notice 94-87, *Unanticipated Crack in a Particular Heat of Alloy 600 Used for Westinghouse Mechanical Plugs for Steam Generator Tubes*, U.S. Nuclear Regulatory Commission, December 22, 1994.

NRC Information Notice 95-17, *Reactor Vessel Top Guide and Core Plate Cracking*, U.S. Nuclear Regulatory Commission, March 10, 1995.

NRC Information Notice 96-11, *Ingress of Demineralizer Resins Increase Potential for Stress Corrosion Cracking of Control Rod Drive Mechanism Penetrations*, U.S. Nuclear Regulatory Commission, February 14, 1996.

NRC Information Notice 97-19, *Safety Injection System Weld Flaw at Sequoyah Nuclear Power Plant, Unit 2*, U.S. Nuclear Regulatory Commission, April 18, 1997.

NRC Information Notice 97-88, *Experiences during Recent Steam Generator Inspections*, U.S. Nuclear Regulatory Commission, December 16, 1997.

NUREG-1544, *Status Report: Intergranular Stress Corrosion Cracking of BWR Core Shrouds and Other Internal Components*, U.S. Nuclear Regulatory Commission, March 1996.

NUREG/CR-6001, *Aging Assessment of BWR Standby Liquid Control Systems*, G. D. Buckley, R. D. Orton, A. B. Johnson Jr., and L. L. Larson, 1992.

XI.M7 BWR STRESS CORROSION CRACKING

Item	Locator	Comment	Justification
1	Program Description	The program description limits the scope to stainless steel. This program should also manage nickel-based alloys.	Nickel-based alloys are used for joint welds as well as for weld overlays in BWR reactor coolant pressure boundary piping. Nickel-based alloys are susceptible to IGSCC per Chapter IV of NUREG-1801. Specific discussions of inspection frequencies for nickel-based alloy materials are contained in BWRVIP-75. In addition, Items R-68 and R-21 specifically align this program and material.
2	Program Description	The program description limits the scope to stainless steel. This program should also manage cast austenitic stainless steel.	No change is required; if CASS is included in the definition of stainless steel. Cast austenitic stainless steel is used pumps, valves, and fittings in BWR reactor coolant pressure boundary piping. Cast austenitic stainless steel is susceptible to IGSCC per Chapter IV of NUREG-1801. Items R-20 and A-101 specifically align this program and material.
3	Program Description	There is a typographical error in the last sentence. The close parenthesis should be after "75":	Editorial correction, no justification is required.
4	1. Scope of Program	The program description limits the scope to stainless steel. This program should also manage nickel-based alloys and cast austenitic stainless steel.	See Justifications to Comments 1 and 2.
5	2. Preventive Action	The last paragraph states: "Reactor coolant water chemistry is monitored and maintained in accordance with the guidelines in BWRVIP-29 (Electric Power Research Institute [EPRI] TR-103515). The program description, and evaluation and technical basis of monitoring and maintaining reactor water chemistry are presented in Section XI.M2, "Water Chemistry." The EPRI Water Chemistry Guidelines and their corresponding BWRVIP documents are living documents that capture industry operating experience and best	Recent applicants have received approval of their water chemistry programs based on later versions of the water chemistry guidelines. For example, NUREG-1769, Safety Evaluation Report Related to the License Renewal of Peach Bottom Atomic Power Station, Units 2 and 3, stated in Section 3.0.3.2.2 on page 3-9 that: "The staff finds the provisions of the 2000 revision of EPRI TR-103515 acceptable because the program is based on updated industry experience."

		practices. The reference to a specific version of the water chemistry guidelines would require each applicant to compare its water chemistry program to an old version of the guidelines with little chance of having a program that is consistent. The BWR SCC Program should only reference Section XI.M2, Water Chemistry.	In conclusion, it is important to maintain the flexibility to modify plant chemistry control procedures based on the best industry guidance developed from the collective operating experience of similar reactors.
6	6. Acceptance Criteria	The first paragraph states: "As recommended in NRC GL 88-01, any indication detected is evaluated in accordance with the ASME Section XI, Subsection IWB-3640 (2001 edition[6] including the 2002 and 2003 Addenda) and the guidelines of NUREG-0313." NRC GL 88-01 does not state that any indication detected is evaluated in accordance with the ASME Section XI, Subsection IWB-3640. This paragraph should be revised as follows: "As recommended in NRC GL 88-01, any indication detected is evaluated in accordance with the ASME Section XI, Subsection IWB-3600 (1986 edition) and the guidelines of NUREG-0313."	For inspections performed per Generic Letter 88-01 the following acceptance criterion is applicable. The original Generic Letter, when discussing the staff position on crack evaluations, states the following: "Methods and criteria for crack evaluation and repair should be in conformance with IWB-3600 of Section XI of the 1986 Edition of the ASME Boiler and Pressure Vessel Code." Supplement 1 to Generic Letter 88-01 did not revise this requirement.
7	6. Acceptance Criteria	Since Note 6 is not required, the note description at the bottom of page XI M-26 should be deleted.	See the Justification for Comment 6.
8	References	The 1986 Edition of ASME Section XI is not in the list of references. It should be added as follows: "ASME Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, ASME Boiler and Pressure Vessel Code, 1986 edition, American Society of Mechanical Engineers, New York, NY."	See the Justification for Comment 6.
9	References	Make conforming change to the reference.	See the Justification for Comment 5.

XI.M7 BWR STRESS CORROSION CRACKING

Program Description

The program to manage intergranular stress corrosion cracking (IGSCC) in boiling water reactor (BWR) coolant pressure boundary piping made of stainless steel (SS)*, **and nickel alloy*** is delineated in NUREG-0313, Rev. 2, and Nuclear Regulatory Commission (NRC) Generic Letter (GL) 88-01 and its Supplement 1. The program includes (a) preventive measures to mitigate IGSCC, and (b) inspection and flaw evaluation to monitor IGSCC and its effects. The staff-approved boiling water reactor vessel and internals project ~~(BWRVIP)-75~~ ***(BWRVIP-75)*** report allows for modifications of inspection scope in the GL 88-01 program.

Evaluation and Technical Basis

1. *Scope of Program:* The program focuses on (a) managing and implementing countermeasures to mitigate IGSCC and (b) performing inservice inspection (ISI) to monitor IGSCC and its effects on the intended function of BWR components. The program is applicable to all BWR piping made of austenitic SS ***and nickel alloy*** that is 4 in. or larger in nominal diameter and contains reactor coolant at a temperature above 93°C (200°F) during power operation, regardless of code classification. The program also applies to pump casings, valve bodies and reactor vessel attachments and appurtenances, such as head spray and vent components. NUREG-0313 and NRC GL 88-01, respectively, describe the technical basis and staff guidance regarding mitigation of IGSCC in BWRs. Attachment A of NRC GL 88-01 delineates the staff-approved positions regarding materials, processes, water chemistry, weld overlay reinforcement, partial replacement, stress improvement of cracked welds, clamping devices, crack characterization and repair criteria, inspection methods and personnel, inspection schedules, sample expansion, leakage detection, and reporting requirements.

2. *Preventive Actions:* The comprehensive program outlined in NUREG-0313 and NRC GL 88-01 addresses improvements in all three elements that, in combination, cause IGSCC. These elements consist of a susceptible (sensitized) material, a significant tensile stress, and an aggressive environment. Sensitization of nonstabilized austenitic SSs containing greater than 0.03 wt.% carbon involves precipitation of chromium carbides at the grain boundaries during certain fabrication or welding processes. The formation of carbides creates an envelope of chromium depleted region that, in certain environments, is susceptible to stress corrosion cracking (SCC). Residual tensile stresses are introduced from fabrication processes, such as welding, surface grinding, or forming. High levels of dissolved oxygen or aggressive contaminants, such as sulfates or chlorides, accelerate the SCC processes.

   The program delineated in NUREG-0313 and NRC GL 88-01 and in the staff-approved BWRVIP-75 report includes recommendations regarding selection of materials that are resistant to sensitization, use of special processes that reduce residual tensile stresses, and monitoring and maintenance of coolant chemistry. The resistant materials are used for new and replacement components and include low-carbon grades of austenitic SS and weld metal, with a maximum carbon of 0.035 wt.% and a minimum ferrite of 7.5% in weld metal and cast austenitic stainless steel (CASS). Inconel 82 is the only commonly used nickel-base weld metal considered to be resistant to SCC; other nickel-alloys, such as Alloy 600 are evaluated on an individual basis. Special processes are used for existing, new, and replacement components. These processes include solution heat treatment, heat sink welding, induction heating, and mechanical stress improvement.

The program delineated in NUREG-0313 and NRC GL 88-01 does not provide specific guidelines for controlling reactor water chemistry to mitigate IGSCC. Maintaining high water purity reduces susceptibility to SCC or IGSCC. Reactor coolant water chemistry is monitored and maintained in accordance with the guidelines in BWRVIP-29 ***or later revisions*** (Electric Power Research Institute [EPRI] TR-103515). The program description, and evaluation and technical basis of monitoring and maintaining reactor water chemistry are presented in Section XI.M2, "Water Chemistry."

3. ***Parameters Monitored/Inspected:*** The program detects and sizes cracks and detects leakage by using the examination and inspection guidelines delineated in NUREG 0313, Rev. 2, and NRC GL 88-01 or the referenced BWRVIP-75 guideline as approved by the NRC staff.

4. ***Detection of Aging Effects:*** The extent, method, and schedule of the inspection and test techniques delineated in NRC GL 88-01 or BWRVIP-75 are designed to maintain structural integrity and ensure that aging effects will be discovered and repaired before the loss of intended function of the component. The program uses volumetric examinations to detect IGSCC.

   The NRC GL 88-01 recommends that the detailed inspection procedure, equipment, and examination personnel be qualified by a formal program approved by the NRC. These inspection guidelines, updated in the approved BWRVIP-75 document, provide the technical basis for revisions to NRC GL 88-01 inspection schedules. Inspection can reveal crack initiation and growth and leakage of coolant. The extent and frequency of inspection recommended by the program are based on the condition of each weld (e.g., whether the weldments were made from IGSCC-resistant material, whether a stress improvement process was applied to a weldment to reduce residual stresses, and how the weld was repaired if it had been cracked). The inspection guidance in approved BWRVIP-75 replaces the extent and schedule of inspection in NRC GL 88-01.

5. ***Monitoring and Trending:*** The extent and schedule for inspection, in accordance with the recommendations of NRC GL 88-01 or approved BWRVIP-75 guidelines, provide timely detection of cracks and leakage of coolant. Based on inspection results, NRC GL 88-01 or approved BWRVIP-75 guidelines provide guidelines for additional samples of welds to be inspected when one or more cracked welds are found in a weld category.

Acceptance Criteria: As recommended in NRC GL 88-01, any indication detected is evaluated in accordance with the ASME Section XI, Subsection ~~IWB-3640 (2001 edition[2] including the~~

~~[2] 10 CFR 50.55a is revised periodically to adopt, by reference, new editions and addenda of the ASME Code. For each successive 120-month (10-year) inspection interval, applicants are required to revise the nuclear plant's ISI program to incorporate the requirements specified in the version of the ASME Code incorporated into 10 CFR 50.55a 12 months before the start of the inspection interval. NRC statements of consideration (SOC) associated with the adoption of new editions and addenda of the ASME Code in 10 CFR 50.55a may discuss the adequacy of the newer edition and addendum as they relate to the GALL Report. The information contained in these SOCs may provide a reasonable basis for exceptions relating to use of editions or addenda of the ASME Code that are not the same as those identified in the GALL Report.~~

~~2002 and 2003 Addenda)~~ ***IWB-3600 of Section XI of the 1986 Edition of the ASME Boiler and Pressure Vessel Code*** and the guidelines of NUREG-0313.

Applicable and approved BWRVIP-14, BWRVIP-59, BWRVIP-60, and BWRVIP-62 documents provide guidelines for evaluation of crack growth in SSs, nickel alloys, and low-alloy steels. An applicant may use BWRVIP-61 guidelines for BWR vessel and internals induction heating stress improvement effectiveness on crack growth in operating plants.

7. ***Corrective Actions***: The guidance for weld overlay repair and stress improvement or replacement is provided in NRC GL 88-01; ASME Section XI, Subsections IWB-4000 and IWB-7000, IWC-4000 and IWC-7000, or IWD-4000 and IWD-7000, respectively for Class 1, 2, or 3 components; and ASME Code Case N 504-1. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.

8. ***Confirmation Process:*** Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the confirmation process and administrative controls.

9. ***Administrative Controls:*** See Item 8, above.

10. ***Operating Experience:*** Intergranular stress corrosion cracking has occurred in small- and large-diameter BWR piping made of austenitic stainless steel and nickel-base alloys. Cracking has occurred in recirculation, core spray, residual heat removal (RHR), and reactor water cleanup (RWCU) system piping welds (NRC GL 88-01, NRC Information Notices [INs] 82-39 and 84-41). The comprehensive program outlined in NRC GL 88-01 and NUREG-0313 and in the staff-approved BWRVIP-75 report addresses mitigating measures for SCC or IGSCC (e.g., susceptible material, significant tensile stress, and an aggressive environment). The GL 88-01 program has been effective in managing IGSCC in BWR reactor coolant pressure-retaining components and the revision to the GL 88-01 program, according to the staff-approved BWRVIP-75 report, will adequately manage IGSCC degradation.

References

10 CFR 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2000.

ASME Code Case N-504-1, *Alternative Rules for Repair of Class 1, 2, and 3 Austenitic Stainless Steel Piping*, Section XI, Division 1, 2001 edition including the 2002 and 2003 Addenda, American Society of Mechanical Engineers, New York, NY.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, ASME Boiler and Pressure Vessel Code, ~~2001~~ ***1986*** edition ~~including the 2002 and 2003 Addenda~~, American Society of Mechanical Engineers, New York, NY.

BWRVIP-14, *Evaluation of Crack Growth in BWR Stainless Steel RPV Internals, (EPRI TR-105873, July 11, 2000)*, Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation for BWRVIP-14, December 3, 1999.

BWRVIP-29 ***or latest revision*** (EPRI TR-103515), *BWR Vessel and Internals Project, BWR Water Chemistry Guidelines~~-1993 Revision~~, Normal and Hydrogen Water Chemistry,* Electric Power Research Institute, Palo Alto, CA, February 1994.

BWRVIP-59, *Evaluation of Crack Growth in BWR Nickel-Base Austenitic Alloys in RPV Internals, (EPRI TR-108710),* BWRVIP and Electric Power Research Institute, Palo Alto, CA, March 24, 2000.

BWRVIP-60, *BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Low Alloy Steel RPV Internals, (EPRI TR-108709, April 14, 2000),* Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation for BWRVIP-60, July 8, 1999.

BWRVIP-61, *BWR Vessel and Internals Induction Heating Stress Improvement Effectiveness on Crack Growth in Operating Reactors, (EPRI TR-112076),* BWRVIP and Electric Power Research Institute, Palo Alto, CA, January 29, 1999.

BWRVIP-62, *BWR Vessel and Internals Project, Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection, (EPRI TR-108705),* BWRVIP and Electric Power Research Institute, Palo Alto, CA, March 7, 2000.

BWRVIP-75, *Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (NUREG-0313), (EPRI TR-113932, Feb. 29, 2000),* Initial Safety Evaluation Report by the Office of Nuclear Reactor Regulation for BWRVIP-75, September 15, 2000.

NRC Generic Letter 88-01, *NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping,* U.S. Nuclear Regulatory Commission, January 25, 1988; Supplement 1, February 4, 1992.

NRC Information Notice 82-39, *Service Degradation of Thick Wall Stainless Steel Recirculation System Piping at a BWR Plant,* U.S. Nuclear Regulatory Commission, September 21, 1982.

NRC Information Notice 84-41, *IGSCC in BWR Plants,* U.S. Nuclear Regulatory Commission, June 1, 1984.

NUREG-0313, Rev. 2, *Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping,* W. S. Hazelton and W. H. Koo, U.S. Nuclear Regulatory Commission, 1988.

XI.M17 FLOW-ACCELERATED CORROSION

Item	Locator	Comment	Justification
1	Monitoring and Trending	Modified criteria for sample expansion.	Plant FAC programs typically use a threshold limit for sample expansion. Some plants use 125% of time to next outage, some use 70% min wall thickness, some use minimum allowable, some use 50%, or less than 1 operating cycle, or significant unexpected wall thinning. Some plants use a combination of the above. But a strict threshold based on measured degradation being more than predicted would require sample expansion where it is unnecessary. In a real world application, if the FAC program owner predicts the wear to be 5 mils over an 18 month period and the actual as found wear is 6 mils then, per the current GALL, sample expansion is necessary. Clearly, the wear is one element, but the minimum wall thickness is a critical input to the need for sample expansion. In this particular example, if the pipe wall is 0.25 inches and has a min allowable wall thickness of 0.15 inches, then the pipe will last 16 refueling cycles. Sample expansion in this case is really not necessary

XI.M17 FLOW-ACCELERATED CORROSION

Program Description

The program relies on implementation of the Electric Power Research Institute (EPRI) guidelines in the Nuclear Safety Analysis Center (NSAC)-202L-R2 for an effective flow-accelerated corrosion (FAC) program. The program includes performing (a) an analysis to determine critical locations, (b) limited baseline inspections to determine the extent of thinning at these locations, and (c) follow-up inspections to confirm the predictions, or repairing or replacing components as necessary.

Evaluation and Technical Basis

1. ***Scope of Program:*** The FAC program, described by the EPRI guidelines in NSAC-202L-R2, includes procedures or administrative controls to assure that the structural integrity of all carbon steel lines containing high-energy fluids (two phase as well as single phase) is maintained. Valve bodies retaining pressure in these high-energy systems are also covered by the program. The FAC program was originally outlined in NUREG-1344 and was further described through the Nuclear Regulatory Commission (NRC) Generic Letter (GL) 89-08. A program implemented in accordance with the EPRI guidelines predicts, detects, and monitors FAC in plant piping and other components, such as valve bodies, elbows and expanders. Such a program includes the following recommendations: (a) conducting an analysis to determine critical locations; (b) performing limited baseline inspections to determine the extent of thinning at these locations; and (c) performing follow-up inspections to confirm the predictions, or repairing or replacing components as necessary. The NSAC-202L-R2 (April 1999) provides general guidelines for the FAC program. To ensure that all the aging effects caused by FAC are properly managed, the program includes the use of a predictive code, such as CHECWORKS, that uses the implementation guidance of NSAC-202L-R2 to satisfy the criteria specified in 10 CFR Part 50, Appendix B, criteria for development of procedures and control of special processes.

2. ***Preventive Actions:*** The FAC program is an analysis, inspection, and verification program; thus, there is no preventive action. However, it is noted that monitoring of water chemistry to control pH and dissolved oxygen content, and selection of appropriate piping material, geometry, and hydrodynamic conditions, are effective in reducing FAC.

3. ***Parameters Monitored/Inspected:*** The aging management program (AMP) monitors the effects of FAC on the intended function of piping and components by measuring wall thickness.

4. ***Detection of Aging Effects:*** Degradation of piping and components occurs by wall thinning. The inspection program delineated in NSAC-202L consists of identification of susceptible locations as indicated by operating conditions or special considerations. Ultrasonic and radiographic testing is used to detect wall thinning. The extent and schedule of the inspections assure detection of wall thinning before the loss of intended function.

5. ***Monitoring and Trending:*** CHECWORKS or a similar predictive code is used to predict component degradation in the systems conducive to FAC, as indicated by specific plant data, including material, hydrodynamic, and operating conditions. CHECWORKS is acceptable because it provides a bounding analysis for FAC. CHECWORKS was developed and benchmarked by using data obtained from many plants. The inspection schedule

developed by the licensee on the basis of the results of such a predictive code provides reasonable assurance that structural integrity will be maintained between inspections. ~~If degradation is detected such that the wall thickness is less than the predicted thickness, additional examinations are performed in adjacent areas to bound the thinning~~ ***Inspection results are evaluated to determine if additional inspections are needed to assure that the extent of wall thinning is adequately determined, identify corrective actions, and assure that intended function will not be lost.***

6. *Acceptance Criteria:* Inspection results are used as input to a predictive computer code, such as CHECWORKS, to calculate the number of refueling or operating cycles remaining before the component reaches the minimum allowable wall thickness. If calculations indicate that an area will reach the minimum allowed thickness before the next scheduled outage, the component is to be repaired, replaced, or reevaluated.

7. *Corrective Actions:* Prior to service, reevaluate, repair, or replace components for which the acceptance criteria are not satisfied. Longer term corrective actions could consist of adjustment of operating parameters or selection of materials resistant to FAC. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.

8. *Confirmation Process:* Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the confirmation process and administrative controls.

9. *Administrative Controls:* See Item 8, above.

10. *Operating Experience:* Wall-thinning problems in single-phase systems have occurred in feedwater and condensate systems (NRC IE Bulletin No. 87-01; NRC Information Notices [INs] 81-28, 92-35, 95-11) and in two-phase piping in extraction steam lines (NRC INs 89-53, 97-84) and moisture separation reheater and feedwater heater drains (NRC INs 89-53, 91-18, 93-21, 97-84). Operating experience shows that the present program, when properly implemented, is effective in managing FAC in high-energy carbon steel piping and components.

References

NRC Generic Letter 89-08, *Erosion/Corrosion-Induced Pipe Wall Thinning*, U.S. Nuclear Regulatory Commission, May 2, 1989.

NRC IE Bulletin 87-01, *Thinning of Pipe Walls in Nuclear Power Plants*, U.S. Nuclear Regulatory Commission, July 9, 1987.

NRC Information Notice 81-28, *Failure of Rockwell-Edward Main Steam Isolation Valves*, U.S. Nuclear Regulatory Commission, September 3, 1981.

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

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

NRC Information Notice 91-18, Supplement 1, *High-Energy Piping Failures Caused by Wall Thinning*, U.S. Nuclear Regulatory Commission, December 18, 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*, U.S. Nuclear Regulatory Commission, May 6, 1992.

NRC Information Notice 93-21, *Summary of NRC Staff Observations Compiled during Engineering Audits or Inspections of Licensee Erosion/Corrosion Programs*, U.S. Nuclear Regulatory Commission, March 25, 1993.

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

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

NSAC-202L-R2, *Recommendations for an Effective Flow Accelerated Corrosion Program*, Electric Power Research Institute, Palo Alto, CA, April 8, 1999.

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

XI.M18 BOLTING INTEGRITY

Item	Locator	Comment	Justification
1	Entire Description	The Bolting Integrity Program description has been extensively revised as shown below. Additions have been shown in bold italics. To improve clarity, deletions have not been shown although they can be viewed as deletions by viewing the final with markup version in WORD.	Basis for most changes explained in the revisions to the Program Description section.
2	Parameters Monitored/ Inspected	Explain that loss of preload is not an aging effect requiring management for non class 1 bolting	Loss of preload is not an aging effect requiring management for non class 1 bolting. In accordance with EPRI 1003056 Appendix F, loss of preload is a design driven effect and not an aging effect requiring management. Loss of preload due to stress relaxation (creep) for standard grade B7 carbon steel bolting is only a concern in very high temperatures (> 700°F) as stated in the ASME Code Section II Part D Table 4 Note 4. Non class 1 systems never exceed this temperature. The majority of non class 1 bolting at most facilities is this same grade or very similar except in rare specialized applications. As a result stress relaxation should not occur for any of the non class 1 systems. In addition, the resolution to GSI-29 for all licensees would have taken actions to address the potential for this effect such that it is not a concern for the current or extended operating term. This position has been previously approved by the staff in the VC Summer SER NUREG 1787 section 3.0.3.7.2.

XI.M18 BOLTING INTEGRITY

Program Description

Bolting integrity issues, including the effects of aging, have received considerable attention over several decades. Degraded bolting was identified by various maintenance and inspection actions and resulted in the establishment of GSI-29 in 1982. Since then, the staff has issued a number of bulletins, generic letters, and information notices on bolting events judged to be safety-significant (See NUREG-1339). These actions and continuing programs were considered by the staff in resolving GSI-29. The NRC resolved GSI-29 without developing any new requirements, based on licensees continuing to implement actions taken in response to previous NRC guidance and the industry's initiatives in this area.

The programs established by licensees to address bolting integrity include inspections specified by ASME Section XI as required by 10 CFR 50.55a. These programs ***also*** rely on ***(a)*** recommendations for a comprehensive bolting integrity program, as delineated in NUREG-1339, and industry ***guidelines such*** as Electric Power Research Institute (EPRI) NP-5769, with the exceptions noted in NUREG-1339 for safety related bolting ***and (b)*** industry ***guidelines*** for a comprehensive bolting maintenance ***program provided*** in EPRI TR-104213 for ***certain*** pressure retaining bolting and structural bolting. The program includes periodic inspection of closure bolting for indication of loss of preload, ***and*** cracking ***for high strength bolting (actual yield strength ≥ 150 ksi).***

The programs established by licensees to address bolting integrity include inspections specified by ASME Section XI as required by 10 CFR 50.55a. Much of the bolting within the scope of license renewal is subject to the requirements of Section XI of the ASME Code and/or the programs established in response to previous NRC guidance and the industry's initiatives in this area. The ASME Code requirements are reviewed and updated on a regular basis and the current regulatory process requires periodic update of the licensee's Section XI ISI program to incorporate applicable changes to the ASME Code and includes provisions for staff review and approval. In addition, other AMPs in this Chapter address the ASME Section XI programs and address particular aging effects. Therefore, this AMP will not replicate or duplicate what is addressed under other AMPs. The focus of this AMP will be on the age-related aspects of the program established in response to previous NRC guidance and the associated industry initiatives. This program covers bolting within the scope of license renewal but not subject to the requirements of ASME Section XI, Subsections IWB, IWC, IWD, IWE IWF, or IWL.

Evaluation and Technical Basis

1. *Scope of Program:* This program ***includes*** safety-related bolting, bolting for NSSS component supports, bolting for other pressure retaining components, and structural bolting. ***It covers the applicable aging effects for bolting within the scope of license renewal for which the generic letters and bulletins listed above require continuing programs, and which are not covered by other NUREG-1801 aging management programs.*** The program covers both greater than and smaller than 2-in. diameter bolting. The Nuclear Regulatory Commission (NRC) staff recommendations and guidelines for comprehensive bolting integrity programs that encompass all safety-related bolting are delineated in

NUREG-1339. The industry's technical basis for the program for safety related bolting and guidelines for inservice inspection (ISI), maintenance, and evaluation of the structural integrity of bolted joints, are outlined in EPRI NP-5769, with the exceptions noted in NUREG 1339. For other bolting, this information is set forth in EPRI TR-104213.

2. ***Preventive Actions:*** Selection of bolting material and the use of lubricants and sealants is in accordance with the guidelines of EPRI NP-5769 and the additional recommendations of NUREG-1339 to prevent or mitigate degradation and failure of safety-related bolting (see item 10, below). (NUREG-1339 takes exception to certain items in EPRI NP-5769, and recommends additional measures with regard to them.)

3. ***Parameters Monitored/Inspected:*** The aging management program (AMP) monitors the effects of aging on the intended function of closure bolting, including cracking, and loss of preload. ***Loss of preload is not an aging effect requiring management for the majority of non class 1 bolting. Most carbon steel bolting is not subject to loss of preload except at temperatures exceeding 700°F.*** High strength bolts (actual yield strength ≥ 150 ksi) are monitored for cracking. Bolting for pressure retaining components is inspected for signs of leakage ***that would indicate loss of preload in bolting subject to this aging effect.***

4. ***Detection of Aging Effects: Bolting is inspected by visual observation. Degradation of the closure bolting due to loss of preload, or crack initiation would result in leakage.*** If bolting is found corroded, a closer inspection is performed to assess extent of corrosion.

5. ***Monitoring and Trending: Torque values are monitored during the bolt torquing process.*** If bolting for pressure retaining components (not covered by ASME Section XI) is reported to be leaking, then ***inspection intervals should be established until a repair of the leak is performed.***

6. ***Acceptance Criteria:*** Indications of cracking in component support bolting warrant replacement of the cracked bolt. For other pressure retaining components, a leak from a joint ***should be*** repaired ***as soon as practical and monitored for degradation until repaired*** if it is a major leak and ***has the potential to*** cause adverse effects such as corrosion or contamination.

7. ***Corrective Actions:*** As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions. Repair and replacement of other bolting including structural bolting is in conformance with the guidelines and recommendations of EPRI TR-104213.

8. ***Confirmation Process:*** Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the confirmation process and administrative controls.

9. ***Administrative Controls:*** See item 8, above.

10. ***Operating Experience:*** Degradation of threaded fasteners in closures for the reactor coolant pressure boundary has occurred from boric acid corrosion, stress corrosion

cracking, and fatigue loading (NRC IE Bulletin 82-02, NRC Generic Letter [GL] 91-17). Stress corrosion cracking has occurred in high strength bolts used for NSSS component supports. The bolting integrity programs developed and implemented ***by the industry*** in accordance with ***previous*** commitments made in response to NRC communications on bolting events ensure ***continued*** bolting reliability. ***Guidelines for*** these programs are ***provided*** in EPRI NP-5769 and TR-104213.

References

10 CFR 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2000.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, ASME Boiler and Pressure Vessel Code, 2001 edition including the 2002 and 2003 Addenda, American Society of Mechanical Engineers, New York, NY.

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-104213, *Bolted Joint Maintenance & Application Guide*, Electric Power Research Institute, Palo Alto, CA, December 1995.

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

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

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.

NUREG-1787, Safety Evaluation Report Related to the License Renewal of the Virgil C. Summer Nuclear Station, U.S Nuclear Regulatory Commission, March 2004

XI.M21 CLOSED-CYCLE COOLING WATER SYSTEM

Item	Locator	Comment	Justification
1	Program Description	Clarify intent of guidance reference in EPRI TR-107396.	There are no specific standards for testing and inspections in the guideline. The EPRI document merely mentions testing and inspections that are commonly applied.
2	Program Description	Acknowledge acceptability of later versions of EPRI document.	The proposed wording will permit an applicant to credit a revision of the EPRI guidelines that has been reviewed and accepted as part of a previous application.
3	Parameters Monitored/ Inspected	Clarify intent of guidance reference in EPRI TR-107396.	There are no specific standards for testing and inspections in the guideline. The EPRI document merely mentions testing and inspections that are commonly applied.
4	Detection of Aging Effects	Clarify intent of guidance reference in EPRI TR-107396.	EPRI does not provide details on extent and schedule of inspections. It is a guidance document not a standard.
5	Detection of Aging Effects	Eliminate "pump wear characteristics"	Pump wear characteristics is an active function that is not in the scope of license renewal.
6	Monitoring and Trending	Clarify intent of guidance reference in EPRI TR-107396.	There is no testing interval specification in the EPRI guideline
7	Acceptance Criteria	Clarify intent of guidance reference in EPRI TR-107396.	EPRI does not provide guidance on performance test results
8	Corrective Actions	Clarify intent of guidance reference in EPRI TR-107396.	EPRI does not provide corrective actions for performance failures.

XI.M21 CLOSED-CYCLE COOLING WATER SYSTEM

Program Description

The program includes (a) preventive measures to minimize corrosion and (b) ~~surveillance~~ testing and inspection to monitor the effects of corrosion on the intended function of the component. The program relies on maintenance of system corrosion inhibitor concentrations within specified limits of Electric Power Research Institute [EPRI] TR-107396 to minimize corrosion. ***Non chemistry monitoring techniques such as*** ~~Surveillance~~ testing and inspection*s* in accordance with ***guidance*** ~~standards~~ in EPRI TR-107396 for closed-cycle cooling water (CCCW) systems ***provide one acceptable method*** ~~is performed~~ to evaluate system and component performance. These measures will ensure that the intended functions of the CCCW system and components serviced by the CCCW system are performing their functions acceptably and are not compromised by aging. ***Later versions of this EPRI closed cycle cooling water guideline are developed from collective operating experience using sound technical judgment, and are approved by the electric utility industry in an effort to constantly improve water chemistry and thereby manage or prevent aging effects. The later versions of these guidelines when reviewed and approved for implementation by the staff in applicant Safety Evaluation Reports may be used in lieu of the revision or version specified above.***

Evaluation and Technical Basis

1. *Scope of Program:* A CCCW system is defined as part of the service water system that is not subject to significant sources of contamination, in which water chemistry is controlled and in which heat is not directly rejected to a heat sink. The program described in this section applies only to such a system. If one or more of these conditions are not satisfied, the system is to be considered an open-cycle cooling water system. The staff notes that if the adequacy of cooling water chemistry control can not be confirmed, the system is treated as an open-cycle system as indicated in Action III of Generic Letter (GL) 89-13.

2. *Preventive Actions:* The program relies on the use of appropriate materials, lining, or coating to protect the underlying metal surfaces and maintenance of system corrosion inhibitor concentrations within specified limits of EPRI TR-107396 to minimize corrosion. The program includes monitoring and control of cooling water chemistry to minimize exposure to aggressive environments and application of corrosion inhibitor in the CCCW system to mitigate general, crevice, and pitting corrosion.

3. *Parameters Monitored/Inspected:* The aging management program (AMP) monitors the effects of corrosion by ~~surveillance~~ testing and inspection in accordance with ***guidance*** ~~standards~~ in EPRI TR-107396 to evaluate system and component ***condition*** ~~performance~~. ***Examples of techniques include visual and NDE inspections, heat transfer testing and performance trending.*** For pump~~s~~ ***performance, typical*** ~~the~~ parameters ***that could be*** monitored include flow and discharge and suction pressures. For heat exchangers, ***typical*** ~~the~~ parameters ***that could be*** monitored include flow, inlet and outlet temperatures, and differential pressure.

4. *Detection of Aging Effects:* Control of water chemistry does not preclude corrosion at locations of stagnant flow conditions or crevices. Degradation of a component due to corrosion would result in degradation of system or component performance. The extent and schedule of inspections and testing ***should*** ~~in accordance with EPRI TR-107396~~, assure

detection of corrosion before the loss of intended function of the component. Performance and functional testing ~~in accordance with EPRI TR-107396,~~ ensures acceptable functioning of the CCCW system or components serviced by the CCCW system. For systems and components in continuous operation, performance adequacy ***should be*** ~~is~~ ***verified*** ~~determined~~ by monitoring ***component performance through*** data trends for evaluation of heat transfer ***capability***~~fouling~~, ~~pump wear characteristics,~~and ***system*** branch flow changes, ***and chemistry data trends***. Components not ***normally*** in operation are periodically tested to ensure operability.

5. ***Monitoring and Trending:*** The frequency of sampling water chemistry varies and can occur on a continuous, daily, weekly, or as needed basis, as indicated by plant operating conditions ***and the type of chemical treatment.*** ~~Per~~In accordance with EPRI TR-107396, ***internal visual inspections and*** performance/ ~~and~~ functional tests are ***to be*** performed ~~at least every 18 months~~ periodically to demonstrate system operability ***and confirm the effectiveness of the program.***, ~~and include~~ ***T***ests to evaluate heat removal capability of the system and degradation of system components ***may also be used.*** ~~The testing intervals are performed every five years.~~ The testing intervals ***should be*** established based on plant-specific considerations such as system conditions, trending, and past operating experience, and may be adjusted on the basis of the results of ***a*** ~~the~~ reliability analysis, type of service, frequency of operation, or age of components and systems.

6. ***Acceptance Criteria:*** Corrosion inhibitor concentrations are maintained within the limits specified in the EPRI water chemistry guidelines for CCCW. System and component performance test results are evaluated in accordance ***with system and component design basis requirements.*** ~~with the guidelines of EPRI TR-107396. and the plant operating license and design basis~~ Acceptance criteria and tolerances are ~~also~~ ***to be*** based on system design parameters and functions.

7. ***Corrective Actions:*** Corrosion inhibitor concentrations outside the allowable limits are returned to the acceptable range within the time period specified in the EPRI water chemistry guidelines for CCCW. If the system or component fails to perform adequately, corrective actions are taken ~~in accordance with EPRI TR-107396~~. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.

8. ***Confirmation Process:*** Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the confirmation process and administrative controls.

9. ***Administrative Controls:*** See Item 8, above.

10. ***Operating Experience:*** Degradation of closed-cycle cooling water systems due to corrosion product buildup (NRC Licensee Event Report [LER] 50-327/93-029-00) or through-wall cracks in supply lines (NRC 50-280/91-019-00) has been observed in operating plants. Accordingly, operating experience demonstrates the need for this program.

References

EPRI TR-107396, *Closed Cooling Water Chemistry Guidelines*, Electric Power Research Institute, Palo Alto, CA, October 1997.

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

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

NRC Licensee Event Report LER 50-280/91-019-00, *Loss of Containment Integrity due to Crack in Component Cooling Water Piping*, October 26, 1991.

NRC Licensee Event Report LER 50-327/93-029-00, *Inoperable Check Valve in the Component Cooling System as a Result of a Build-Up of Corrosion Products between Valve Components*, December 13, 1993.

XI.M23 INSPECTION OF OVERHEAD HEAVY LOAD AND LIGHT LOAD (RELATED TO REFUELING) HANDLING SYSTEMS

Item	Locator	Comment	Justification
1	Parameters Monitored/ Inspected	Delete sentence related to number of lifts.	The line needs to be deleted. It has been proven in previous applications that this is not a parameter which is monitored. Calculated number of lifts is significantly lower than crane is designed for.
2	Detection of Aging Effect	Delete sentence related to functional testing.	Functional tests do not detect aging effects.

XI.M23 INSPECTION OF OVERHEAD HEAVY LOAD AND LIGHT LOAD (RELATED TO REFUELING) HANDLING SYSTEMS

Program Description

Most commercial nuclear facilities have between 50 and 100 cranes. Many are industrial grade cranes which meet the requirements of 29 CFR Volume XVII, Part 1910, and Section 1910.179. Most are not within the scope of 10 CFR 54.4, and therefore are not required to be part of the integrated plant assessment (IPA).

Normally, fewer than 10 cranes fall within the scope of 10 CFR 54.4. These cranes comply with the Maintenance Rule requirements provided in 10 CFR 50.65. The Nuclear Regulatory Commission Regulatory Guide (RG) 1.160 provides guidance for monitoring the effectiveness of maintenance at nuclear power plants.

The program demonstrates that testing and monitoring programs have been implemented and have ensured that the structures, systems, and components of these cranes are capable of sustaining their rated loads. This is their intended function during the period of extended operation. It is noted that many of the systems and components of these cranes perform an intended function with moving parts or with a change in configuration, or subject to replacement based on qualified life. In these instances, these types of crane systems and components are not within the scope of this aging management program (AMP). This program is primarily concerned with structural components that make up the bridge and trolley. NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," provides specific guidance on the control of overhead heavy load cranes.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program manages the effects of general corrosion on the crane and trolley structural components for those cranes that are within the scope of 10 CFR 54.4, and the effects of wear on the rails in the rail system.

2. ***Preventive Actions:*** No preventive actions are identified. The crane program is an inspection program.

3. ***Parameters Monitored/Inspected:*** The program evaluates the effectiveness of the maintenance monitoring program and the effects of past and future usage on the structural reliability of cranes. ~~The number and magnitude of lifts made by the crane are also reviewed.~~

4. ***Detection of Aging Effect:*** Crane rails and structural components are visually inspected on a routine basis for degradation. ~~Functional tests are also performed to assure their integrity~~.

5. ***Monitoring and Trending:*** Monitoring and trending are not required as part of the crane inspection program.

6. ***Acceptance Criteria:*** Any significant visual indication of loss of material due to corrosion or wear are evaluated according to applicable industry standards and good industry practice. The crane may also have been designed to a specific Service Class as defined in the EOCI Specification #61 (or later revisions), or CMAA Specification #70 (or later revisions), or

CMAA Specification #74 (or later revisions). The specification that was applicable at the time the crane was manufactured is used.

7. ***Corrective Actions:*** Site corrective actions program, quality assurance (QA) procedures, site review and approval process, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the corrective actions, confirmation process, and administrative controls.

8. ***Confirmation Process:*** See Item 7, above.

9. ***Administrative Controls:*** See Item 7, above.

10. ***Operating Experience:*** Because of the requirements for monitoring the effectiveness of maintenance at nuclear power plants provided in 10 CFR 50.65, there has been no history of corrosion-related degradation that has impaired cranes. Likewise, because cranes have not been operated beyond their design lifetime, there have been no significant fatigue-related structural failures.

References

10 CFR 50.65, *Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, January 1997.

Crane Manufactures Association of America, Inc., CMAA Specification No. 70, *Specifications for Electric Overhead Traveling Cranes*, 1970 (or later revisions).

Crane Manufactures Association of America, Inc., CMAA Specification No. 74, *Specifications for Top Running and Under Running Single Girder Electric Overhead Traveling Cranes*, 1974 (or later revisions).

Electric Overhead Crane Institute, Inc., EOCI Specification No. 61, *Specifications for Electric Overhead Traveling Cranes*, 1961 (or later revisions).

NUREG-0612, *Control of Heavy Loads at Nuclear Power Plants*, U.S. Nuclear Regulatory Commission, 1980.

NRC Regulatory Guide 1.160, Rev. 2, *Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*, U.S. Nuclear Regulatory Commission, March 1997.

XI.M26 FIRE PROTECTION

Item	Locator	Comment	Justification
1	Detection of Aging Effect	Replace prescriptive inspection requirements with general requirement that inspectors are qualified.	There are no regulatory requirements or any other type of requirements specifying that these inspections be performed to VT-1 or VT-3 standards.
2	Acceptance Criteria	Add words to recognize allowable defect indications.	As currently worded, no indications of defects are acceptable. The criteria need to recognize variations from these strict limits that have been approved for the seal configuration.

XI.M26 FIRE PROTECTION

Program Description

For operating plants, the fire protection aging management program (AMP) includes a fire barrier inspection program and a diesel-driven fire pump inspection program. The fire barrier inspection program requires periodic visual inspection of fire barrier penetration seals, fire barrier walls, ceilings, and floors, and periodic visual inspection and functional tests of fire rated doors to ensure that their operability is maintained. The diesel-driven fire pump inspection program requires that the pump be periodically tested to ensure that the fuel supply line can perform the intended function. The AMP also includes periodic inspection and test of halon/carbon dioxide fire suppression system.

Evaluation and Technical Basis

1. ***Scope of Program:*** For operating plants, the AMP manages the aging effects on the intended function of the penetration seals, fire barrier walls, ceilings, and floors, and all fire rated doors (automatic or manual) that perform a fire barrier function. It also manages the aging effects on the intended function of the fuel supply line. The AMP also includes management of the aging effects on the intended function of the halon/carbon dioxide fire suppression system.

2. ***Preventive Actions:*** For operating plants, the fire hazard analysis assesses the fire potential and fire hazard in all plant areas. It also specifies measures for fire prevention, fire detection, fire suppression, and fire containment and alternative shutdown capability for each fire area containing structures, systems, and components important to safety.

3. ***Parameters Monitored/Inspected:*** Visual inspection of approximately 10% of each type of penetration seal is performed during walkdowns carried out at least once every refueling outage. These inspections examine any sign of degradation such as cracking, seal separation from walls and components, separation of layers of material, seals rupture and puncture which are directly caused by increased hardness, and shrinkage of seal material due to weathering. Visual inspection of the fire barrier walls, ceilings, and floors examines any sign of degradation such as cracking, spalling, and loss of material caused by freeze-thaw, chemical attack, and reaction with aggregates. Hollow metal fire doors are visually inspected on a plant specific interval to verify the integrity of door surfaces and for clearances. The plant specific inspection intervals are to be determined by engineering evaluation to detect degradation of the fire doors prior to the loss of intended function.

   The diesel-driven fire pump is under observation during performance tests such as flow and discharge tests, sequential starting capability tests, and controller function tests for detecting any degradation of the fuel supply line.

   Periodic visual inspection and function test at least once every six months examines the signs of degradation of the halon/carbon dioxide fire suppression system. Material conditions that may affect the performance of the system, such as corrosion, mechanical damage, or damage to dampers, are observed during these tests.

4. ***Detection of Aging Effects:*** Visual inspection of penetration seals detects cracking, seal separation from walls and components, and rupture and puncture of seals. Visual inspection ~~(VT-1 or equivalent)~~ ***by qualified inspectors*** of approximately 10% of each type of seal in

walkdowns is performed at least once every refueling outage. If any sign of degradation is detected within that sample, the scope of the inspection and frequency is expanded to ensure timely detection of increased hardness and shrinkage of the penetration seal before the loss of the component function.. Visual inspection ~~(VT-1 or equivalent)~~ ***by qualified inspectors*** of the fire barrier walls, ceilings, and floors performed in walkdowns at least once every refueling outage ensures timely detection of concrete cracking, spalling, and loss of material. Visual inspection ~~(VT-3 or equivalent)~~ ***by qualified inspectors*** detects any sign of degradation of the fire door such as wear and missing parts. Periodic visual inspection and function tests detect degradation of the fire doors before there is a loss of intended function.

Periodic tests performed at least once every refueling outage, such as flow and discharge tests, sequential starting capability tests, and controller function tests performed on diesel-driven fire pump ensure fuel supply line performance. The performance tests detect degradation of the fuel supply lines before the loss of the component intended function.

Visual inspections of the halon/carbon dioxide fire suppression system detect any sign of degradation, such as corrosion, mechanical damage, or damage to dampers. The periodic function test and inspection performed at least once every six months detects degradation of the halon/carbon dioxide fire suppression system before the loss of the component intended function. The monthly inspection ensures that the extinguishing agent supply valves are open and the system is in automatic mode.

5. ***Monitoring and Trending:*** The aging effects of weathering on fire barrier penetration seals are detectable by visual inspection and, based on operating experience, visual inspections performed at least once every refueling outage to detect any sign of degradation of fire barrier penetration seals prior to loss of the intended function.

   Concrete cracking, spalling, and loss of material are detectable by visual inspection and, based on operating experience, visual inspection performed at least once every refueling outage detects any sign of degradation of the fire barrier walls, ceilings, and floors before there is a loss of the intended function. Based on operating experience, degraded integrity or clearances in the fire door are detectable by visual inspection performed on a plant specific frequency. The visual inspections detect degradation of the fire doors prior to loss of the intended function.

   The performance of the fire pump is monitored during the periodic test to detect any degradation in the fuel supply lines. Periodic testing provides data (e.g., pressure) for trending necessary.

   The performance of the halon/carbon dioxide fire suppression system is monitored during the periodic test to detect any degradation in the system. These periodic tests provide data necessary for trending.

6. ***Acceptance Criteria:*** Inspection results are acceptable if there are no visual indications ***outside those allowed by approved penetration seal configurations, for*** ~~of~~ cracking, separation of seals from walls and components, separation of layers of material, or ruptures or punctures of seals, no visual indications of concrete cracking, spalling and loss of material of fire barrier walls, ceilings, and floors, no visual indications of missing parts, holes, and wear and no deficiencies in the functional tests of fire doors. No corrosion is acceptable

in the fuel supply line for the diesel-driven fire pump. Also, any signs of corrosion and mechanical damage of the halon/carbon dioxide fire suppression system are not acceptable.

7. ***Corrective Actions:*** For fire protection structures and components identified within scope that are subject to an aging management review for license renewal, the applicant's 10 CFR Part 50, Appendix B, program is used for corrective actions, confirmation process, and administrative controls for aging management during the period of extended operation. This commitment is documented in the final safety analysis report (FSAR) supplement in accordance with 10 CFR 54.21(d). As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address corrective actions, confirmation process, and administrative controls.

8. ***Confirmation Process:*** See Item 7, above.

9. ***Administrative Controls:*** See Item 7, above.

10. ***Operating Experience:*** Silicone foam fire barrier penetration seals have experienced splits, shrinkage, voids, lack of fill, and other failure modes (IN 88-56, IN 94-28, and IN 97-70). Degradation of electrical racing way fire barrier such as small holes, cracking, and unfilled seals are found on routine walkdown (IN 91-47 and GL 92-08). Fire doors have experienced wear of the hinges and handles. Operating experience with the use of this AMP has shown that no corrosion-related problem has been reported for the fuel supply line, pump casing of the diesel-driven fire pump, and the halon/carbon dioxide suppression system. No significant aging related problems have been reported of fire protection systems, emergency breathing and auxiliary equipment, and communication equipment.

References

NRC Generic Letter 92-08, *Thermo-Lag 330-1 Fire Barrier*, December 17, 1992.

NRC Information Notice 88-56, *Potential Problems with Silicone Foam Fire Barrier Penetration Seals*, August 14, 1988.

NRC Information Notice 91-47, *Failure of Thermo-Lag Fire Barrier Material to Pass Fire Endurance Test*, August 6, 1991.

NRC Information Notice 94-28, *Potential problems with Fire-Barrier Penetration Seals*, April 5, 1994.

NRC Information Notice 97-70, *Potential problems with Fire Barrier Penetration Seals*, September 19, 1997.

XI.M27 FIRE WATER SYSTEM

Item	Locator	Comment	Justification
1	Program Description	Modify code references.	The proposed changes recognize a second acceptable version of the NFPA Code.
2	Program Description and Parameters Monitored/ Inspected	Modify sprinkler flow test requirements.	The proposed change recognizes practical flow test limitations.
3	Parameters Monitored/ Inspected	Delete specific code reference.	References do not correspond to the 2002 edition.
4	Monitoring and Trending	Link requirements to plant commitments to code.	Existing plant commitments to code should determine monitoring and trending requirements.

XI.M27 FIRE WATER SYSTEM

Program Description

This aging management program applies to water-based fire protection systems that consist of sprinklers, nozzles, fittings, valves, hydrants, hose stations, standpipes, water storage tanks, and aboveground and underground piping and components that are tested in accordance with the applicable National Fire Protection Association (NFPA) codes and standards. Such testing assures the minimum functionality of the systems. Also, these systems are normally maintained at required operating pressure and monitored such that loss of system pressure is immediately detected and corrective actions initiated.

A sample of sprinkler heads is to be inspected by using the guidance of NFPA 25 ***(1998)***, Section ***2-3.1.1*** ~~2.3.3.1~~, ***or NFPA (2002), Section 5.3.1.1.1.*** This NFPA section states that "where sprinklers have been in place for 50 years, they shall be replaced or representative samples from one or more sample areas shall be submitted to a recognized testing laboratory for field service testing." It also contains guidance to perform this sampling every 10 years after the initial field service testing.

The fire protection ~~sprinkler~~ system piping is to be subjected to full flow ***tests or*** tests at ***flows representative of those expected during a fire,*** ~~the maximum design flow and pressure~~ or evaluated for wall thickness (e.g., non-intrusive volumetric testing or plant maintenance visual inspections) to ensure that corrosion aging effects are managed and that wall thickness is within acceptable limits. These inspections are performed before the end of the current operating term and at plant specific intervals thereafter during the period of extended operation. The plant specific inspection intervals are to be determined by engineering evaluation of the fire protection piping to ensure that degradation will be detected before the loss of intended function. The purpose of the full flow testing and wall thickness evaluations is to ensure that corrosion, microbiological influenced corrosion (MIC), or biofouling is managed such that the system function is maintained.

Evaluation and Technical Basis

1. ***Scope of Program:*** The aging management program focuses on managing loss of material due to corrosion, MIC, or biofouling of carbon steel and cast-iron components in fire protection systems exposed to water. Hose stations and standpipes are considered as piping in the AMP.

2. ***Preventive Actions:*** To ensure no significant corrosion, MIC, or biofouling has occurred in water-based fire protection systems, periodic flushing, system performance testing, and inspections may be conducted.

3. ***Parameters Monitored/Inspected:*** Loss of material due to corrosion and biofouling could reduce wall thickness of the fire protection piping system and result in system failure. Therefore, the parameters monitored are the system's ability to maintain pressure and internal system corrosion conditions. Periodic flow testing of the fire water system is performed using the guidelines of NFPA 25, ~~Chapter 13, Annexes A & D~~ at the maximum ~~design~~ flow or perform wall thickness evaluations to ensure that the system maintains its intended function.

4. ***Detection of Aging Effects:*** Fire protection system testing is performed to assure that the system functions by maintaining required operating pressures. Wall thickness evaluations of fire protection piping are performed on system components using non-intrusive techniques (e.g., volumetric testing) to identify evidence of loss of material due to corrosion. These inspections are performed before the end of the current operating term and at plant specific intervals thereafter during the period of extended operation. As an alternative to non-intrusive testing, the plant maintenance process may include a visual inspection of the internal surface of the fire protection piping upon each entry to the system for routine or corrective maintenance, as long as it can be demonstrated that inspections are performed (based on past maintenance history) on a representative number of locations on a reasonable basis. These inspections must be capable of evaluating (1) wall thickness to ensure against catastrophic failure and (2) the inner diameter of the piping as it applies to the flow requirements of the fire protection system. If the environmental and material conditions that exist on the interior surface of the below grade fire protection piping are similar to the conditions that exist within the above grade fire protection piping, the results of the inspections of the above grade fire protection piping can be extrapolated to evaluate the condition of below grade fire protection piping. If not, additional inspection activities are needed to ensure that the intended function of below grade fire protection piping will be maintained consistent with the current licensing basis for the period of extended operation. Continuous system pressure monitoring, system flow testing, and wall thickness evaluations of piping are effective means to ensure that corrosion and biofouling are not occurring and the system's intended function is maintained.

   General requirements of existing fire protection programs include testing and maintenance of fire detection and protection systems and surveillance procedures to ensure that fire detectors, as well as fire protection systems and components are operable.

   Visual inspection of yard fire hydrants performed annually in accordance with NFPA 25 ensures timely detection of signs of degradation, such as corrosion. Fire hydrant hose hydrostatic tests, gasket inspections, and fire hydrant flow tests, performed annually, ensure that fire hydrants can perform their intended function and provide opportunities for degradation to be detected before a loss of intended function can occur.

   Sprinkler heads are inspected before the end of the 50-year sprinkler head service life and at 10 year intervals thereafter during the extended period of operation to ensure that signs of degradation, such as corrosion, are detected in a timely manner.

5. ***Monitoring and Trending:*** System discharge pressure is monitored continuously. Results of system performance testing are monitored and trended as specified by the ***associated plant commitments pertaining to*** NFPA codes and standards. Degradation identified by non-intrusive or internal inspection is evaluated.

6. ***Acceptance Criteria:*** The acceptance criteria are (a) the ability of a fire protection system to maintain required pressure, (b) no unacceptable signs of degradation observed during non-intrusive or visual assessment of internal system conditions, and (c) that no biofouling exists in the sprinkler systems that could cause corrosion in the sprinkler heads.

7. ***Corrective Actions:*** Repair and replacement actions are initiated as necessary. For fire water systems and components identified within scope that are subject to an aging management review for license renewal, the applicant's 10 CFR Part 50, Appendix B, program is used for corrective actions, confirmation process, and administrative controls for

aging management during the period of extended operation. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address corrective actions, confirmation process, and administrative controls.

8. **Confirmation Process:** See Item 7, above.

9. **Administrative Controls:** See Item 7, above.

10. ***Operating Experience:*** Water-based fire protection systems designed, inspected, tested and maintained in accordance with the NFPA minimum standards have demonstrated reliable performance.

References

NFPA 25: Inspection, Testing and Maintenance of Water-Based Fire Protection Systems, 1998 Edition.

NFPA 25: Inspection, Testing and Maintenance of Water-Based Fire Protection Systems, 2002 Edition.

XI.M31 REACTOR VESSEL SURVEILLANCE

Item	Locator	Comment	Justification
1	Program Description	Modify sentences delineating storage requirements for capsules.	The proposed revisions define a more appropriate set of storage guidelines that are consistent with vessel surveillance requirements.

XI.M31 REACTOR VESSEL SURVEILLANCE

Program Description

The Code of Federal Regulations, 10 CFR Part 50, Appendix H, requires that peak neutron fluence at the end of the design life of the vessel will not exceed 10^{17} n/cm^2 (E >1MeV), or that reactor vessel beltline materials be monitored by a surveillance program to meet the American Society for Testing and Materials (ASTM) E 185 Standard. However, the surveillance program in ASTM E 185 is based on plant operation during the current license term, and additional surveillance capsules may be needed for the period of extended operation. Alternatively, an integrated surveillance program for the period of extended operation may be considered for a set of reactors that have similar design and operating features in accordance with 10 CFR Part 50, Appendix H, Paragraph II.C. Additional surveillance capsules may also be needed for the period of extended operation for this alternative.

The existing reactor vessel material surveillance program provides sufficient material data and dosimetry to monitor irradiation embrittlement at the end of the period of extended operation, and to determine the need for operating restrictions on the inlet temperature, neutron spectrum, and neutron flux. If surveillance capsules are not withdrawn during the period of extended operation, operating restrictions are to be established to ensure that the plant is operated under the conditions to which the surveillance capsules were exposed.

All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of ASTM E 185-82 to the extent practicable for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation. ***Untested***~~All~~ capsules placed in storage must be maintained for future insertion ***if an applicant is to seek additional license renewals.*** ~~Any changes to s~~ Storage requirements ***for untested capsules*** must be approved by the NRC, as required by 10CFR 50, Appendix H.

An acceptable reactor vessel surveillance program consists of the following:

1. The extent of reactor vessel embrittlement for upper-shelf energy and pressure-temperature limits for 60 years is projected in accordance with the Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials." When using NRC RG 1.99, Rev. 2, an applicant has a choice of the following:

   a. Neutron Embrittlement Using Chemistry Tables

   An applicant may use the tables in NRC RG 1.99, Rev. 2, to project the extent of reactor vessel neutron embrittlement for the period of extended operation based on material chemistry and neutron fluence. This is described as Regulatory Position 1 in the RG.

   b. Neutron Embrittlement Using Surveillance Data

   When credible surveillance data are available, the extent of reactor vessel neutron embrittlement for the period of extended operation may be projected according to Regulatory Position 2 in NRC RG 1.99, Rev. 2, based on best fit of the surveillance data. The credible data could be collected during the current operating term. The applicant may have a plant-specific program or an integrated surveillance program during the period of extended operation to collect additional data.

2. An applicant that determines embrittlement by using the NRC RG 1.99, Rev. 2, tables (see item 1[a], above) uses the applicable limitations in Regulatory Position 1.3 of the RG. The limits are based on material properties, temperature, material chemistry, and fluence.

3. An applicant that determines embrittlement by using surveillance data (see item 1[b], above) defines the applicable bounds of the data, such as cold leg operating temperature and neutron fluence. These bounds are specific for the referenced surveillance data. For example, the plant-specific data could be collected within a smaller temperature range than that in the RG.

4. All pulled and tested capsules, unless discarded before August 31, 2000, are placed in storage. (Note: These specimens are saved for future reconstitution use, in case the surveillance program is reestablished.)

5. If an applicant has a surveillance program that consists of capsules with a projected fluence of less than the 60-year fluence at the end of 40 years, at least one capsule is to remain in the reactor vessel and is tested during the period of extended operation. The applicant may either delay withdrawal of the last capsule or withdraw a standby capsule during the period of extended operation to monitor the effects of long-term exposure to neutron irradiation.

6. If an applicant has a surveillance program that consists of capsules with a projected fluence exceeding the 60-year fluence at the end of 40 years, the applicant withdraws one capsule at an outage in which the capsule receives a neutron fluence equivalent to the 60-year fluence and tests the capsule in accordance with the requirements of ASTM E 185. Any capsules that are left in the reactor vessel provide meaningful metallurgical data (i.e., the capsule fluence does not significantly exceed the vessel fluence at an equivalent of 60 years). For example, in a reactor with a lead factor of three, after 20 years the capsule test specimens would have received a neutron exposure equivalent to what the reactor vessel would see in 60 years; thus, the capsule is to be removed since further exposure would not provide meaningful metallurgical data. Other standby capsules are removed and placed in storage. These standby capsules (and archived test specimens available for reconstitution) would be available for reinsertion into the reactor if additional license renewals are sought (e.g., 80 years of operation). If all surveillance capsules have been removed, operating restrictions are to be established to ensure that the plant is operated under conditions to which the surveillance capsules were exposed and the exposure conditions of the reactor vessel are monitored to ensure that they continue to be consistent with those used to project the effects of embrittlement to the end of license. If the reactor vessel exposure conditions (neutron flux, spectrum, irradiation temperature, etc.) are altered, then the basis for the projection to 60 years is reviewed; and, if deemed appropriate, an active surveillance program is re-instituted. Any changes to the reactor vessel exposure conditions and the potential need to re-institute a vessel surveillance program is discussed with the NRC staff prior to changing the plant's licensing basis.

7. Applicants without in-vessel capsules use alternative dosimetry to monitor neutron fluence during the period of extended operation, as part of the aging management program (AMP) for reactor vessel neutron embrittlement.

8. The applicant may choose to demonstrate that the materials in the inlet, outlet, and safety injection nozzles are not controlling, so that such materials need not be added to the material surveillance program for the license renewal term.

The reactor vessel monitoring program provides that, if future plant operations exceed the limitations or bounds specified in item 2 or 3, above (as applicable), such as operating at a lower cold leg temperature or higher fluence, the impact of plant operation changes on the extent of reactor vessel embrittlement will be evaluated and the NRC will be notified. An applicant without capsules in its reactor vessel is to propose reestablishing the reactor vessel surveillance program to assess the extent of embrittlement. This program will consist of (1) capsules from item 6, above; (2) reconstitution of specimens from item 4, above; and/or (3) capsules made from any available archival materials; or (4) some combination of the three previous options. This program could be a plant-specific program or an integrated surveillance program.

Evaluation and Technical Basis

Reactor vessel surveillance programs are plant specific, depending on matters such as the composition of limiting materials, availability of surveillance capsules, and projected fluence levels. In accordance with 10 CFR Part 50, Appendix H, an applicant submits its proposed withdrawal schedule for approval prior to implementation. Thus, further staff evaluation is required for license renewal.

References

10 CFR Part 50, Appendix H, *Reactor Vessel Material Surveillance Program Requirements*, Office of the Federal Register, National Archives and Records Administration, 2000.

ASTM E-185, *Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels*, American Society for Testing Materials, Philadelphia, PA.

NRC Regulatory Guide 1.99, Rev. 2, *Radiation Embrittlement of Reactor Vessel Materials*, U.S. Nuclear Regulatory Commission.

XI.M32 ONE-TIME INSPECTION

Item	Locator	Comment	Justification
1	Program Description	Move sentences related to timing of OTI to section 4.0	These sentences add no value since the timing is specified in section 4.0 and is not appropriate nor fit for the description.
2	Program Description	Editorial comments	
3	Scope of Program	Editorial comments	
4	Parameters Monitored/ Inspected	Clarify relevance of ASME Code to inspection requirements.	The majority of the components included in this program are not ASME Code components and as such should not be subjected to Code requirements. This new wording is consistent with wording in section 4.0.
5	Detection of Aging Effects	Editorial comments	
6	Detection of Aging Effects	Remove table of example parameters/inspections	The addition of this table is unnecessary and overly prescriptive. In the prior paragraph wording is provided that the program will rely on established NDE techniques consistent with the Code which is sufficient for determining techniques. The techniques specified in the proposed table are Code requirements that should not be applied to non code components. This will tie applicant's hands in the ability to use various approaches and other techniques that may be developed in the future and have to justify deviations.
7	Detection of Aging Effects	Move sentences related to timing of OTI to section 4.0	See comment 1
8	Operating Experience	Editorial comments	

XI.M32 ONE-TIME INSPECTION

Program Description

The program includes measures to verify the effectiveness of an aging management program (AMP) and confirm the absence of an aging effect. ~~As a plant will have accumulated at least 30 years of use before inspections under this program begin, sufficient time will have elapsed for aging effects, if any, to be manifest. In every case, more than half of the plant's total licensed lifespan will have elapsed. Nevertheless, there are~~ Situations in which additional confirmation is appropriate~~.~~ ~~These~~ include (a) an aging effect is not expected to occur but the data is insufficient to rule it out with reasonable confidence; (b) an aging effect is expected to progress very slowly in the specified environment, but the local environment may be more adverse than that generally expected; or (c) the characteristics of the aging effect include a long incubation period. For these cases, there is to be confirmation that either the aging effect is indeed not occurring, or the aging effect is occurring very slowly so as not to affect the component or structure intended function during the period of extended operation.

A one-time inspection may be used to provide additional assurance that aging that has not yet manifested itself is not occurring, ***or*** that the evidence of aging shows that the aging is so insignificant that an aging management program is not warranted. A one time inspection may also trigger development of a program necessary to assure component intended functions through the period of extended operation. For example, for structures and components, such as Class 1 piping with a diameter less than nominal pipe size (NPS) 4 inch that do not receive volumetric examination during inservice inspection, the ***one-time inspection*** program ***can be used to*** either confirm~~s~~ that crack initiation and growth due to stress corrosion cracking (SCC) or cyclic loading is not occurring and, therefore, there is no need for further inspections to monitor its progression for the period of extended operation or it detects cracking and triggers actions to characterize the nature and extent of the cracking and determine what subsequent monitoring is needed to assure intended functions are maintained through the period of extended operation.

One-time inspections may ***also*** be used to verify the system-wide effectiveness of an aging management program (AMP) that is designed to prevent or minimize aging to the extent that it will not cause the loss of intended function during the period of extended operation. For example, effective control of water chemistry can prevent some aging effects and minimize others. However, there may be locations that are isolated from the flow stream for extended periods and are susceptible to the gradual accumulation or concentration of agents that promote certain aging effects. This program provides inspections that either ~~verify~~ ***verifies*** that unacceptable degradation is not occurring or triggers additional actions that will assure the intended function of affected components will be maintained during the period of extended operation.

The elements of the program include (a) determination of the sample size based on an assessment of materials of fabrication, environment, plausible aging effects, and operating experience; (b) identification of the inspection locations in the system or component based on the aging effect; (c) determination of the examination technique, including acceptance criteria that would be effective in managing the aging effect for which the component is examined; and (d) evaluation of the need for follow-up examinations to monitor the progression of aging if age-related degradation is found that could jeopardize an intended function before the end of the period of extended operation .

When evidence of an aging effect is revealed by a one-time inspection, the routine evaluation of the inspection results would identify appropriate corrective actions.

As set forth below, an acceptable verification program may consist of a one-time inspection of selected components and susceptible locations in the system. An alternative acceptable program may include routine maintenance or a review of repair ***or inspection*** records to confirm that these components have been inspected for aging degradation and significant aging degradation has not occurred. One-time inspection, or any other action or program, is to be reviewed by the staff on a plant-specific basis.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program includes measures to verify that unacceptable degradation is not occurring, thereby validating the effectiveness of existing AMPs or confirming that there is no need to manage ag~~eing~~-related degradation for the period of extended operation. The structures and components for which one-time inspection is ***specified*** to verify the effectiveness of the AMPs (e.g., water chemistry control, etc.) have been identified in the Generic Aging Lessons Learned (GALL) report. Examples include small bore piping in the reactor coolant system or the feedwater system components in boiling water reactors (BWRs) and pressurized water reactors (PWRs).

2. ***Preventive Actions:*** One-time inspection is an inspection activity independent of methods to mitigate or prevent degradation.

3. ***Parameters Monitored/Inspected:*** The program monitors parameters directly related to the degradation of a component. Inspection is ***to be*** performed ***by qualified personnel following procedures consistent*** ~~in accordance~~ with the requirements of the American Society of Mechanical Engineers (ASME) Code and 10 CFR 50, Appendix B, ~~by~~ using a variety of nondestructive examination (NDE) methods, including visual, volumetric, and surface techniques.

4. ***Detection of Aging Effects:*** The inspection includes a representative sample of the system population, and, where practical, will focus on the bounding or lead components most susceptible to aging due to time in service, severity of operating conditions, and lowest design margin. For small-bore piping, actual inspection locations are based on physical accessibility, exposure levels, NDE techniques, and locations identified in Nuclear Regulatory Commission (NRC) Information Notice (IN) 97-46.

   The program will rely on established NDE techniques, including visual, ultrasonic, and surface techniques that are performed by qualified personnel following procedures consistent with the ASME Code and 10 CFR Part 50, Appendix B. For small-bore piping less than NPS 4 in., including pipe, fittings, and branch connections ***crediting this program***, a plant-specific destructive examination of piping replaced in the course of plant modifications or NDE that permits inspection of the inside surfaces of the piping is to be conducted to ensure that cracking has not occurred.

   The inspection and test techniques will have a demonstrated history of effectiveness in detecting the aging effect of concern. ~~Typically, the one time inspections should be performed as indicated in the following table.~~

~~**Examples of Parameters Monitored or Inspected and Aging Effect for Specific Structure or Component**~~[3]

~~**Aging Effect**~~	~~**Aging Mechanism**~~	~~**Parameter Monitored**~~	~~**Inspection Method**~~[4]
~~Loss of Material~~	~~Crevice Corrosion~~	~~Wall Thickness~~	~~Visual (VT-1) and/or Volumetric (RT or UT)~~
~~Loss of Material~~	~~Galvanic Corrosion~~	~~Wall Thickness~~	~~Visual (VT-3) and/or Volumetric (RT or UT)~~
~~Loss of Material~~	~~General Corrosion~~	~~Wall Thickness~~	~~Visual (VT-3) and/or Volumetric (RT or UT)~~
~~Loss of Material~~	~~MIC~~	~~Wall Thickness~~	~~Visual (VT-3) and/or Volumetric (RT or UT)~~
~~Loss of Material~~	~~Pitting Corrosion~~	~~Wall Thickness~~	~~Visual (VT-1) and/or Volumetric (RT or UT)~~
~~Loss of Material~~	~~Selective Leaching~~	~~Wall Thickness~~	~~Hardness test~~
~~Loss of Material~~	~~Erosion~~	~~Wall Thickness~~	~~Visual (VT-3) and/or Volumetric (RT or UT)~~
~~Loss of Heat Transfer~~	~~Fouling~~	~~Tube Fouling~~	~~Visual (VT-3) or remote visual) or Enhanced VT-1 for CASS~~
~~Cracking~~	~~SCC, thermal stratification and turbulent penetration~~	~~Cracks~~	~~Volumetric (RT or UT)~~
~~Loss of Preload~~	~~Stress Relaxation (gasket compression)~~	~~Dimension Changes~~	~~Visual (VT-3)~~

With respect to inspection timing, the population of components inspected before the end of the current operating term needs to be sufficient to provide reasonable assurance that the aging effect will not compromise any intended function at any time during the period of extended operation. Specifically, inspections need to be completed early enough to ensure that the aging effects that may affect intended functions early in the period of extended operation are appropriately managed. Conversely, inspections need to be timed to allow the inspected components to attain sufficient age to ensure that the aging effects with long incubation periods (i.e., those that may affect intended functions near the end of the period of extended operation) are identified. Within these constraints, the applicant ***should***~~may~~ schedule the inspection ***no earlier than 10 years prior to the period of extended operation, and*** in such a way as to minimize the impact on plant operations. ***As a plant will have accumulated at least 30 years of use before inspections under this program begin, sufficient time will have elapsed for aging effects, if any, to be manifest.***

5. ***Monitoring and Trending:*** The program provides for increasing of the inspection sample size and locations in the event that aging effects are detected. Unacceptable inspection findings are evaluated in accordance with the site corrective action process to determine the

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[3] The examples provided in the table may not be appropriate for all relevant situations. If the applicant chooses to use an alternative to the recommendations in ~~of~~ this table, a technical justification should be provided as an exception to this AMP. This exception should list the examination technique, acceptance criteria, evaluation standard and a description of the justification.
[4] Visual inspection may be used only when the inspection methodology examines the surface potentially experiencing the aging effect. Remote visual inspections may be used to perform the inspection.

need for subsequent (including periodic) inspections and for monitoring and trending the results.

6. ***Acceptance Criteria***: Any indication or relevant conditions of degradation detected are evaluated. The ultrasonic thickness measurements are to be compared to predetermined limits, such as design minimum wall thickness.

7. ***Corrective Actions:*** Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the corrective actions, confirmation process, and administrative controls.

8. ***Confirmation Process:*** See Item 7, above.

9. ***Administrative Controls:*** See Item 7, above.

10. ***Operating Experience:*** This program applies to potential aging effects for which there is currently no operating experience indicating the need for an aging management program. Nevertheless, the elements that comprise these inspections (e.g., the scope of the inspections and inspection techniques) are consistent with years of industry practice and staff experieeince.

References

10 CFR 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2000.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, ASME Boiler and Pressure Vessel Code, 2001 edition including the 2002 and 2003 Addenda, American Society of Mechanical Engineers, New York, NY.

NRC Information Notice 97-46, *Unisolable Crack in High-Pressure Injection Piping*, U.S. Nuclear Regulatory Commission, July 9, 1997.

New Aging Management Programs

Following are two proposed new aging management programs. The External Surfaces Monitoring Program is proposed for visual monitoring of system external surfaces. This program replaces "plant specific program" currently listed in numerous lines of the GALL tables. The comments on the external surfaces tables in Chapters V, VII and VIII incorporate this proposed program.

The Flux Thimble Tube Inspection Program is proposed to monitor thinning of the flux thimble tube walls. This program replaces aging management program elements currently listed in line IV.B2-13, (R-145) of the GALL tables. This proposed program is incorporated in the comments for Chapter IV.

XI.MXX EXTERNAL SURFACES MONITORING PROGRAM

Program Description

This program consists of periodic inspections of systems and components within the scope of license renewal and subject to aging management review in order to manage loss of material and cracking. The External Surfaces Monitoring Program manages these aging effects through visual inspection of external surfaces for leakage and evidence of material degradation, such as corrosion; cracking; degradation of coatings, sealants and caulking; and corrosion product buildup. This program confirms that the surfaces of components are not experiencing significant degradation such that component intended function would be affected. Visual inspections have proven effective throughout the industry in managing aging effects on plant equipment. The External Surfaces Monitoring Program provides reasonable assurance that effects of aging will be managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

Evaluation and Technical Basis

1. ***Scope of Program:*** This program entails inspections of external surfaces of components in systems within the scope of license renewal and subject to aging management review. The program may also be credited with managing loss of material from internal surfaces, for situations in which internal and external material and environment combinations are the same such that external surface condition is representative of internal surface condition.

   Inspections of component surfaces covered by insulation (e.g., anti-sweat and freeze protection) will be performed when insulation is removed during maintenance activities. Insulated surfaces that normally operate at elevated temperatures (> 212 °F) are outside the scope of this program due to the absence of aging effects in this environment.

2. ***Preventive Actions:*** The External Surfaces Monitoring Program is a visual monitoring program that does not include preventive actions.

3. ***Parameters Monitored/Inspected:*** System inspections should monitor for items which could affect system performance, safety, or reliability as well as general housekeeping, personnel safety hazards and radiological concerns. Examples of inspection parameters include:

   - excessive corrosion and material wastage (loss of material); cracking on equipment surfaces
   - leakage from or onto external surfaces, including evidence of boric acid
   - worn, flaking, or oxide-coated surfaces
   - loose or corroded fasteners , foundations, supports, and hangers
   - corrosion stains on thermal insulation
   - protective coating degradation (cracking and flaking)

4. ***Detection of Aging Effects:*** A general visual inspection is conducted in areas containing all system and component surfaces at least once per refueling cycle. This frequency is required in order to accommodate inspections of components that may be in locations that are normally only accessible during outages. Most areas will be inspected on a frequency that exceeds once per refueling cycle. The aging effects managed by this program typically

result from long-term degradation mechanisms such as general surface corrosion that can be effectively managed through inspections on this frequency. The frequency of inspections may be adjusted as necessary based on inspection results and industry experience.

This program is credited with managing the following aging effects.

- loss of material for external surfaces
- loss of material for internal surfaces exposed to the same environment as the external surface
- cracking of external surfaces

5. ***Monitoring and Trending:*** The External Surfaces Monitoring Program uses standardized monitoring and trending activities to track degradation. Deficiencies are documented using approved processes and procedures such that results can be trended. Degradation due to effects not associated with aging should not be used for trending activities associated with this program. The program does not include formal trending. However, the Corrective Action Program applies providing reasonable assurance that trends entailing repeat failures to meet acceptance criteria will be identified and addressed with appropriate corrective actions.

6. ***Acceptance Criteria:*** No visual indications of cracking, loss of material, leakage, or other aging mechanisms that would impact component intended function .

7. ***Corrective Actions:*** Deficiencies identified during inspections should be documented for review and resolution through approved site processes and procedures. Corrective actions including engineering evaluations should be accomplished in accordance with the site Corrective Action Program. Corrective actions should include repair, replacement, rework, increased monitoring and preventive measures as appropriate.

8. **Confirmation Process:** QA procedures, review and approval processes, and administrative controls are to be implemented in accordance with the requirements of 10 CFR 50, Appendix B. If degradation that requires repair is identified during monitoring activities, corrective actions should be implemented. Additionally, inspection results from reviews by outside organizations are used to help confirm the maintenance of plant integrity and materiel condition.

9. **Administrative Controls:** Administrative and implementing procedures are reviewed, approved, and maintained as controlled documents in accordance with the procedure control process and the site Quality Assurance Program.

10. ***Operating Experience:*** External surfaces inspections via system walkdowns have been in effect at many utilities since the mid 1990's in support of the Maintenance Rule (10CFR50.65) and have proven effective in maintaining the material condition of plant systems. This operating experience provides assurance that the program will be effective in managing effects of aging so that components crediting this program can perform their intended function consistent with the current licensing basis during the period of extended operation.

References

NUREG-1785, *Safety Evaluation Report Related to the License Renewal of the H.B. Robinson Steam Electric Plant,* U.S Nuclear Regulatory Commission, March 2004

NUREG-1786, *Safety Evaluation Report Related to the License Renewal of the R.E. Ginna Nuclear Power Plant,* U.S Nuclear Regulatory Commission, May 2004

XI.MXX FLUX THIMBLE TUBE INSPECTION PROGRAM

Program Description

The Flux Thimble Tube Inspection Program is an inspection program used to monitor for thinning of the flux thimble tube walls, which provide a path for the incore neutron flux monitoring system detectors and which form part of the RCS pressure boundary. Flux thimble tubes are subject to loss of material at certain locations in the reactor vessel where flow-induced fretting causes wear at discontinuities in the path from the reactor vessel instrument nozzle to the fuel assembly instrument guide tube. An NDE methodology, such as eddy current testing (ECT), is used to monitor for wear of the flux thimble tubes. This program implements the recommendations of NRC IE Bulletin 88-09, "Thimble Tube Thinning in Westinghouse Reactors," as described below.

Evaluation and Technical Basis

1. ***Scope of Program:*** The Flux Thimble Tube Inspection Program encompasses all of the flux thimble tubes that form part of the reactor coolant system pressure boundary. The flux thimble guide tubes are not in the scope of this program.

2. ***Preventive Actions:*** The program consists of inspection and evaluation and provides no guidance on methods to mitigate wear of the flux thimble tubes.

3. ***Parameters Monitored/Inspected:*** Flux thimble tube wall thickness will be monitored to detect loss of material from the flux thimble tubes during the period of extended operation.

4. ***Detection of Aging Effects:*** An inspection methodology (such as ECT) that has been demonstrated to be capable of adequately detecting wear of the flux thimble tubes will be employed to detect loss of material during the period of extended operation. The inspection results will be evaluated and compared with the acceptance criteria established as discussed below.

5. ***Monitoring and Trending:*** The wall thickness measurements will be trended and wear rates will be calculated. Examination frequency will be based upon wear predictions which have been technically justified as providing conservative estimates of flux thimble tube wear. The interval between inspections will be established such that no flux thimble tube is predicted to incur wear which exceeds the established acceptance criteria before the next inspection. The examination frequency may be adjusted based on plant-specific wear projections.

6. ***Acceptance Criteria:*** Appropriate acceptance criteria such as percent through-wall wear will be established. The acceptance criteria will be technically justified to provide an adequate margin of safety to ensure that the integrity of the reactor coolant system pressure boundary is maintained. The acceptance criteria will include allowances for factors such as instrument uncertainty, uncertainties in wear scar geometry, and other potential inaccuracies as applicable to the inspection methodology chosen for use in the program.

7. ***Corrective Actions:*** Flux thimble tubes which do not meet the established acceptance criteria must be isolated, capped, plugged, withdrawn, replaced, or otherwise removed from service in a manner that ensures the integrity of the reactor coolant system pressure boundary is maintained. Analyses may allow repositioning of flux thimble tubes that are approaching the acceptance criteria limit. Repositioning of a tube exposes a different portion of the tube to the discontinuity that is causing the wear.

   Flux thimble tubes that cannot be inspected over the tube length that is subject to wear due to restriction or other defect, and that can not be shown by analysis to be satisfactory for continued service, must be removed from service to ensure the integrity of the reactor coolant system pressure boundary.

   The site corrective actions program, quality assurance (QA) procedures, site review and approval process, and administrative controls are implemented in accordance with 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the corrective actions, confirmation process, and administrative controls.

8. ***Confirmation Process:*** See Item 7, above.

9. ***Administrative Controls:*** See Item 7, above.

10. ***Operating Experience:*** In IE Bulletin 88-09, "Thimble Tube Thinning in Westinghouse Reactors," the NRC requested that licensees implement a flux thimble tube inspection program due to several instances of leaks, and due to licensees identifying wear. Utilities established inspection programs in accordance with IE Bulletin 88-09 which have shown excellent results in identifying and managing wear of flux thimble tubes.

    As discussed in IE Bulletin 88-09, the amount of vibration the thimble tubes experience is determined by many plant-specific factors. Therefore, the only effective method for determining thimble tube integrity is through inspections which are adjusted to account for plant-specific wear patterns and history.

    Operating experience indicates that flux thimble tubes that are chrome-plated over the tube length that is subject to wear are more resistant to the onset of wear than non-plated tubes.

References

NRC IE Bulletin 88-09, "Thimble Tube Thinning in Westinghouse Reactors," dated July 26, 1988

NRC Information Notice No. 87-44, Supplement 1, "Thimble Tube Thinning in Westinghouse Reactors," dated March 28, 1988

NRC Information Notice No. 87-44, "Thimble Tube Thinning in Westinghouse Reactors," dated September 16, 1987

XI.S1 ASME SECTION XI, SUBSECTION IWE

Item	Locator	Comment	Justification
1	Program Description Footnote	The footnote is based on ASME Section XI. The note discusses that the NRC adopts the use of updated versions of ASME XI in 10 CFR 50.55a but does not state that an applicant may credit the updated versions. The Bases Document for the revision to the GALL Report states that the addition of the code used in the plant's ISI program, which is based on 10 CFR 50.55a, can be used as an AMP in a LRA. Revise footnote wording as shown for XI.S1 in XI.S2 also.	The footnote added to several AMP program descriptions acknowledges that the ASME code required under 10CFR50.55a changes periodically but it does not clearly state the applicant can credit whatever code version is applicable during the period of extended operation. The rewording proposed corrects the discrepancy and will eliminate exceptions that applicants must take given the current wording. The proposed wording in the footnote will allow applicants to credit the revision of ASME XI that is credited in their current ISI plan as an acceptable aging management program for license renewal

XI.S1 ASME SECTION XI, SUBSECTION IWE

Program Description

10 CFR 50.55a imposes the inservice inspection (ISI) requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, Subsection IWE for steel containments (Class MC) and steel liners for concrete containments (Class CC). The full scope of IWE includes steel containment shells and their integral attachments; steel liners for concrete containments and their integral attachments; containment hatches and airlocks; seals, gaskets and moisture barriers; and pressure-retaining bolting. This evaluation is based on the 2001 edition[1] including the 2002 and 2003 Addenda, as approved in 10 CFR 50.55a. ASME Code Section XI, Subsection IWE and the additional requirements specified in 10 CFR 50.55a(b)(2) constitute an existing mandated program applicable to managing aging of steel containments, steel liners of concrete containments, and other containment components for license renewal.

The primary ISI method specified in IWE is visual examination (general visual, VT-3, VT-1). Limited volumetric examination (ultrasonic thickness measurement) and surface examination (e.g., liquid penetrant) may also be necessary in some instances. Bolt preload is checked by either a torque or tension test. IWE specifies acceptance criteria, corrective actions, and expansion of the inspection scope when degradation exceeding the acceptance criteria is found.

The evaluation of 10 CFR 50.55a and Subsection IWE as an aging management program (AMP) for license renewal is provided below.

Evaluation and Technical Basis

1. ***Scope of Program:*** Subsection IWE-1000 specifies the components of steel containments and steel liners of concrete containments within its scope. The components within the scope of Subsection IWE are Class MC pressure-retaining components (steel containments) and their integral attachments; metallic shell and penetration liners of Class CC containments and their integral attachments; containment seals and gaskets; containment pressure-retaining bolting; and metal containment surface areas, including welds and base metal. The concrete portions of containments are inspected in accordance with Subsection IWL.

   Subsection IWE exempts the following from examination:

[1] 10 CFR 50.55a is revised periodically to adopt, by reference, new editions and addenda of the ASME Code. For each successive 120-month (10 year) inspection interval, applicants are required to revise the nuclear plant's ISI program to incorporate the requirements specified in the version of the ASME Code incorporated into 10 CFR 50.55a 12 months before the start of the inspection interval. ***Because a plant's 10 year ISI programs is based on the edition of the ASME Code when the ISI program is prepared, the ISI program based on any edition and addenda of the ASME Code adopted by the NRC in 10 CFR 50.55a is an acceptable aging management program that can be credited in a license renewal application as consistent with NUREG 1801 without justifying exceptions.*** ~~NRC statements of consideration (SOC) associated with the adoption of new editions and addenda of the ASME Code in 10 CFR 50.55a may discuss the adequacy of the newer edition and addendum as they relate to the GALL Report. The information contained in these SOCs may provide a reasonable basis for exceptions relating to use of editions or addenda of the ASME Code that are not the same as those identified in the GALL Report.~~

(1) Components that are outside the boundaries of the containment as defined in the plant-specific design specification;
(2) Embedded or inaccessible portions of containment components that met the requirements of the original construction code of record;
(3) Components that become embedded or inaccessible as a result of vessel repair or replacement, provided IWE-1232 and IWE-5220 are met; and
(4) Piping, pumps, and valves that are part of the containment system or that penetrate or are attached to the containment vessel (governed by IWB or IWC).

10 CFR 50.55a(b)(2)(ix) specifies additional requirements for inaccessible areas. It states that the licensee is to evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas. Examination requirements for containment supports are not within the scope of Subsection IWE.

2. ***Preventive Action:*** No preventive actions are specified; Subsection IWE is a monitoring program.

3. ***Parameters Monitored or Inspected:*** Table IWE-2500-1 specifies seven categories for examination. The categories, parts examined, and examination methods are presented in the following table. The first six examination categories (E-A through E-G) constitute the ISI requirements of IWE. Examination category E-P references 10 CFR Part 50, Appendix J leak rate testing. Appendix J leak rate testing is evaluated as a separate AMP for license renewal in XI.S4.

CATEGORY	PARTS EXAMINED	EXAMINATION METHOD[a]
E-A	Containment surfaces	General visual, visual VT-3
E-B[b]	Pressure retaining welds	Visual VT-1
E-C	Containment surfaces requiring augmented examination	Visual VT-1, volumetric
E-D	Seals, gaskets, and moisture barriers	Visual VT-3
E-F[b]	Pressure retaining dissimilar metal welds	Surface
E-G	Pressure retaining bolting	Visual VT-1, bolt torque or tension test
E-P	All pressure-retaining components (pressure retaining boundary, penetration bellows, airlocks, seals, and gaskets)	10 CFR Part 50, Appendix J (containment leak rate testing)

[a] The applicable examination method (where multiple methods are listed) depends on the particular subcategory within each category.

[b] These two categories are optional, in accordance with 10 CFR 50.55a(b)(2)(ix)(C).

Table IWE-2500-1 references the applicable section in IWE-3500 that identifies the aging effects that are evaluated. The parameters monitored or inspected depend on the particular examination category. For Examination Category E-A, as an example, metallic surfaces (without coatings) are examined for evidence of cracking, discoloration, wear, pitting, excessive corrosion, arc strikes, gouges, surface discontinuities, dents, and other signs of surface irregularities. For Examination Category E-D, seals, gaskets, and moisture barriers are examined for wear, damage, erosion, tear, surface cracks, or other defects that may violate the leak-tight integrity.

4. ***Detection of Aging Effects:*** The frequency and scope of examination specified in 10 CFR 50.55a and Subsection IWE ensure that aging effects would be detected before they would compromise the design-basis requirements. As indicated in IWE-2400, inservice examinations and pressure tests are performed in accordance with one of two inspection programs, A or B, on a specified schedule. Under Inspection Program A, there are four inspection intervals (at 3, 10, 23, and 40 years) for which 100% of the required examinations must be completed. Within each interval, there are various inspection periods for which a certain percentage of the examinations are to be performed to reach 100% at the end of that interval. In addition, a general visual examination is performed once each inspection period. After 40 years of operation, any future examinations will be performed in accordance with Inspection Program B. Under Inspection Program B, starting with the time the plant is placed into service, there is an initial inspection interval of 10 years and successive inspection intervals of 10 years each, during which 100% of the required examinations are to be completed. An expedited examination of containment is required by 10 CFR 50.55a in which an inservice (baseline) examination specified for the first period of the first inspection interval for containment is to be performed by September 9, 2001. Thereafter, subsequent examinations are performed every 10 years from the baseline examination. Regarding the extent of examination, all accessible surfaces receive a visual examination such as General Visual, VT-1, or VT-3 (see table in item 3 above). IWE-1240 requires augmented examinations (Examination Category E-C) of containment surface areas subject to degradation. A VT-1 visual examination is performed for areas accessible from both sides, and volumetric (ultrasonic thickness measurement) examination is performed for areas accessible from only one side.

5. ***Monitoring and Trending:*** With the exception of inaccessible areas, all surfaces are monitored by virtue of the examination requirements on a scheduled basis. When component examination results require evaluation of flaws, evaluation of areas of degradation, or repairs, and the component is found to be acceptable for continued service, the areas containing such flaws, degradation, or repairs shall be reexamined during the next inspection period, in accordance with Examination Category E-C. When these reexaminations reveal that the flaws, areas of degradation, or repairs remain essentially unchanged for three consecutive inspection periods, these areas no longer require augmented examination in accordance with Examination Category E-C.

   IWE-2430 specifies that (a) examinations performed during any one inspection that reveal flaws or areas of degradation exceeding the acceptance standards are to be extended to include an additional number of examinations within the same category approximately equal to the initial number of examinations, and (b) when additional flaws or areas of degradation that exceed the acceptance standards are revealed, all of the remaining examinations within the same category are to be performed to the extent specified in Table IWE-2500-1 for the inspection interval. Alternatives to these examinations are provided in 10 CFR 50.55a(b)(2)(ix)(D).

6. ***Acceptance Criteria:*** IWE-3000 provides acceptance standards for components of steel containments and liners of concrete containments. Table IWE-3410-1 presents criteria to evaluate the acceptability of the containment components for service following the preservice examination and each inservice examination. This table specifies the acceptance standard for each examination category. Most of the acceptance standards rely on visual examinations. Areas that are suspect require an engineering evaluation or require correction by repair or replacement. For some examinations, such as augmented examinations, numerical values are specified for the acceptance standards. For the containment steel shell or liner, material loss exceeding 10% of the nominal containment wall thickness, or material loss that is projected to exceed 10% of the nominal containment wall thickness before the next examination, are documented. Such areas are to be accepted by engineering evaluation or corrected by repair or replacement in accordance with IWE-3122.

7. ***Corrective Actions:*** Subsection IWE states that components whose examination results indicate flaws or areas of degradation that do not meet the acceptance standards listed in Table-3410-1 are acceptable if an engineering evaluation indicates that the flaw or area of degradation is nonstructural in nature or has no effect on the structural integrity of the containment. Except as permitted by 10 CFR 50.55a(b)(ix)(D), components that do not meet the acceptance standards are subject to additional examination requirements, and the components are repaired or replaced to the extent necessary to meet the acceptance standards of IWE-3000. For repair of components within the scope of Subsection IWE, IWE-3124 states that repairs and reexaminations are to comply with IWA-4000. IWA-4000 provides repair specifications for pressure retaining components including metal containments and metallic liners of concrete containments. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address corrective actions.

8. ***Confirmation Process:*** When areas of degradation are identified, an evaluation is performed to determine whether repair or replacement is necessary. If the evaluation determines that repair or replacement is necessary, Subsection IWE specifies confirmation that appropriate corrective actions have been completed and are effective. Subsection IWE states that repairs and reexaminations are to comply with the requirements of IWA-4000. Reexaminations are conducted in accordance with the requirements of IWA-2200, and the recorded results are to demonstrate that the repair meets the acceptance standards set forth in Table IWE-3410-1. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process.

9. ***Administrative Controls:*** IWA-6000 provides specifications for the preparation, submittal, and retention of records and reports. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address administrative controls.

10. ***Operating Experience:*** ASME Section XI, Subsection IWE was incorporated into 10 CFR 50.55a in 1996. Prior to this time, operating experience pertaining to degradation of steel components of containment was gained through the inspections required by 10 CFR Part 50, Appendix J and ad hoc inspections conducted by licensees and the Nuclear Regulatory Commission (NRC). NRC Information Notice (INs) 86-99, 88-82 and 89-79 described occurrences of corrosion in steel containment shells. NRC Generic Letter (GL) 87-05 addressed the potential for corrosion of boiling water reactor (BWR) Mark I steel

drywells in the "sand pocket region." More recently, NRC IN 97-10 identified specific locations where concrete containments are susceptible to liner plate corrosion. The program is to consider the liner plate and containment shell corrosion concerns described in these generic communications. Implementation of the ISI requirements of Subsection IWE, in accordance with 10 CFR 50.55a, is a necessary element of aging management for steel components of steel and concrete containments through the period of extended operation.

References

10 CFR Part 50, Appendix J, *Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors,* Office of the Federal Register, National Archives and Records Administration, 2000.

10 CFR 50.55a, *Codes and Standards,* Office of the Federal Register, National Archives and Records Administration, 2000.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components,* Subsection IWA, *General Requirements,* 2001 edition including the 2002 and 2003 Addenda, The ASME Boiler and Pressure Vessel Code, The American Society of Mechanical Engineers, New York, NY.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components,* Subsection IWB, *Requirements for Class 1 Components of Light-Water Cooled Power Plants,* 2001 edition including the 2002 and 2003 Addenda , The ASME Boiler and Pressure Vessel Code, The American Society of Mechanical Engineers, New York, NY.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components,* Subsection IWC, *Requirements for Class 2 Components of Light-Water Cooled Power Plants,* 2001 edition including the 2002 and 2003 Addenda, The ASME Boiler and Pressure Vessel Code, The American Society of Mechanical Engineers, New York, NY.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components,* Subsection IWE, *Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants,* 2001 edition including the 2002 and 2003 Addenda, The ASME Boiler and Pressure Vessel Code, The American Society of Mechanical Engineers, New York, NY.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components,* Subsection IWL, *Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants,* 2001 edition including the 2002 and 2003 Addenda, The ASME Boiler and Pressure Vessel Code, The American Society of Mechanical Engineers, New York, NY.

NRC Generic Letter 87-05, *Request for Additional Information Assessment of Licensee Measures to Mitigate and/or Identify Potential Degradation of Mark I Drywells,* U.S. Nuclear Regulatory Commission, March 12, 1987.

NRC Information Notice 86-99, *Degradation of Steel Containments,* U.S. Nuclear Regulatory Commission, December 8, 1986 and Supplement 1, February 14, 1991.

NRC Information Notice 88-82, *Torus Shells with Corrosion and Degraded Coatings in BWR Containments,* U.S. Nuclear Regulatory Commission, October 14,1988 and Supplement 1, May 2, 1989.

NRC Information Notice 89-79, *Degraded Coatings and Corrosion of Steel Containment Vessels,* U.S. Nuclear Regulatory Commission, December 1, 1989 and Supplement 1, June 29, 1989.

NRC Information Notice 97-10, *Liner Plate Corrosion in Concrete Containment,* U.S. Nuclear Regulatory Commission, March 13, 1997.

XI.S7 RG 1.127, INSPECTION OF WATER-CONTROL STRUCTURES ASSOCIATED WITH NUCLEAR POWER PLANTS

Item	Locator	Comment	Justification
1	Program Description Footnote	Add conditional statement acknowledging use of structures monitoring program for certain structures for plants committed to RG 1.127.	Some plants committed to RG 1.127 already use structures monitoring program for certain structures (e.g., Intake Structure Concrete). The change will reduce the likelihood of exceptions to the program.

XI.S7 RG 1.127, INSPECTION OF WATER-CONTROL STRUCTURES ASSOCIATED WITH NUCLEAR POWER PLANTS

Program Description

Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.127, Revision 1, "Inspection of Water-Control Structures Associated with Nuclear Power Plants," describes an acceptable basis for developing an inservice inspection and surveillance program for dams, slopes, canals, and other water-control structures associated with emergency cooling water systems or flood protection of nuclear power plants. The RG 1.127 program addresses age-related deterioration, degradation due to extreme environmental conditions, and the effects of natural phenomena that may affect water-control structures. The RG 1.127 program recognizes the importance of periodic monitoring and maintenance of water-control structures so that the consequences of age-related deterioration and degradation can be prevented or mitigated in a timely manner.

RG 1.127 provides detailed guidance for the licensee's inspection program for water-control structures, including guidance on engineering data compilation, inspection activities, technical evaluation, inspection frequency, and the content of inspection reports. Water-control structures covered by the RG 1.127 program include concrete structures; embankment structures; spillway structures and outlet works; reservoirs; cooling water channels and canals, and intake and discharge structures; and safety and performance instrumentation. RG 1.127 delineates current NRC practice in evaluating inservice inspection programs for water-control structures. The attributes of an acceptable aging management program (AMP) for license renewal are described below.

For plants not committed to RG 1.127, Revision 1, aging management of water-control structures may be included in the Structures Monitoring Program (XI.S6). ***Even if plant is committed to RG 1.127, Revision 1, aging management of certain structures and components may be included in the Structures Monitoring Program (XI.S6).*** However, details pertaining to water-control structures are to incorporate the attributes described herein.

Evaluation and Technical Basis

1. ***Scope of Program:*** RG 1.127 applies to water-control structures associated with emergency cooling water systems or flood protection of nuclear power plants. The water-control structures included in the RG 1.127 program are concrete structures; embankment structures; spillway structures and outlet works; reservoirs; cooling water channels and canals, and intake and discharge structures; and safety and performance instrumentation.

2. ***Preventive Action:*** No preventive actions are specified; RG 1.127 is a monitoring program.

3. ***Parameters Monitored or Inspected:*** RG 1.127 identifies the parameters to be monitored and inspected for water-control structures. The parameters vary depending on the particular structure. Parameters to be monitored and inspected for concrete structures include cracking, movements (e.g., settlement, heaving, deflection), conditions at junctions with abutments and embankments, erosion, cavitation, seepage, and leakage. Parameters to be monitored and inspected for earthen embankment structures include settlement, depressions, sink holes, slope stability (e.g., irregularities in alignment and variances from originally constructed slopes), seepage, proper functioning of drainage systems, and degradation of slope protection features. Further details of parameters to be monitored and

inspected for these and other water-control structures are specified in Section C.2 of RG 1.127.

4. ***Detection of Aging Effects:*** Visual inspections are primarily used to detect degradation of water-control structures. In some cases, instruments have been installed to measure the behavior of water-control structures. RG 1.127 indicates that the available records and readings of installed instruments are to be reviewed to detect any unusual performance or distress that may be indicative of degradation. RG 1.127 describes periodic inspections, to be performed at least once every five years. Similar intervals of five years are specified in ACI 349.3R for inspection of structures continually exposed to fluids or retaining fluids. Such intervals have been shown to be adequate to detect degradation of water-control structures before they have a significant effect on plant safety. RG 1.127 also describes special inspections immediately following the occurrence of significant natural phenomena, such as large floods, earthquakes, hurricanes, tornadoes, and intense local rainfalls.

5. ***Monitoring and Trending:*** Water-control structures are monitored by periodic inspection as described in RG 1.127. In addition to monitoring the aging effects identified in Attribute (3) above, inspections also monitor the adequacy and quality of maintenance and operating procedures. RG 1.127 does not discuss trending.

6. ***Acceptance Criteria:*** Acceptance criteria to evaluate the need for corrective actions are not specified in RG 1.127. However, the "Evaluation Criteria" provided in Chapter 5 of ACI 349.3R-96 provides acceptance criteria (including quantitative criteria) for determining the adequacy of observed aging effects and specifies criteria for further evaluation . Although not required, plant-specific acceptance criteria based on Chapter 5 of ACI 349.3R-96 are acceptable. Acceptance criteria for earthen structures such as dams, canals, and embankments are to be consistent with programs falling within the regulatory jurisdiction of the Federal Energy Regulatory Commission (FERC) or the U.S. Army Corps of Engineers.

7. ***Corrective Actions:*** RG 1.127 recommends that the licensee's inservice inspection and surveillance program include periodic inspections of water-control structures to identify deviations in structural conditions due to age-related deterioration and degradation from the original design basis. When findings indicate that significant changes have occurred, the conditions are to be evaluated. This includes a technical assessment of the causes of distress or abnormal conditions, an evaluation of the behavior or movement of the structure, and recommendations for remedial or mitigating measures. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address corrective actions.

8. ***Confirmation Process:*** As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process.

9. ***Administrative Controls:*** As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address administrative controls.

10. ***Operating Experience:*** Degradation of water-control structures has been detected, through RG 1.127 programs, at a number of nuclear power plants, and in some cases, it has required remedial action. No loss of intended functions has resulted from these occurrences. Therefore, it can be concluded that the inspections implemented in accordance with the

guidance in RG 1.127 have been successful in detecting significant degradation before loss of intended function occurs.

NOTE: For dam inspection and maintenance, programs under the regulatory jurisdiction of FERC or the U.S. Army Corps of Engineers, continued through the period of extended operation, will be adequate for the purpose of aging management. For programs not falling under the regulatory jurisdiction of FERC or the U.S. Army Corps of Engineers, the staff will evaluate the effectiveness of the aging management program based on compatibility to the common practices of the FERC and Corps programs.

References

ACI Standard 349.3R-96, *Evaluation of Existing Nuclear Safety-Related Concrete Structures,* American Concrete Institute.

NRC Regulatory Guide 1.127, *Inspection of Water-Control Structures Associated with Nuclear Power Plants,* Revision 1, U.S. Nuclear Regulatory Commission, March 1978.

CHAPTER 2

SCOPING AND SCREENING METHODOLOGY FOR IDENTIFYING STRUCTURES AND COMPONENTS SUBJECT TO AGING MANAGEMENT REVIEW AND IMPLEMENTATION RESULTS

2.1 SCOPING AND SCREENING METHODOLOGY

Item	Locator	Comment	Justification
1	2.1.2.1	In this and other paragraphs, Revision 3 of NEI 95-10 is referenced. Draft Regulatory Guide (RG) DG-1140 (RG 1.188) references NEI 95-10, Rev. 5. This is a generic comment as Revision 3 is referenced in several sections of the SRP-LR.	Editorial
2	2.1.2.2	Change Regulatory Guide 1.1888 to Regulatory Guide 1.188.	Editorial; an 8 too far.
3	Table 2.1-4(a) and (b)	Table needs to be changed to match NEI 95-10 Rev.5 Table 4.1-1	Consistency
4	Table 2.1-4(b),	In the Intended Function column, Electrical Continuity should not be bold type.	Editorial

2.1 SCOPING AND SCREENING METHODOLOGY

Review Responsibilities

Primary - Branch responsible for quality assurance
Secondary - Branches responsible for systems, as appropriate

2.1.1 Areas of Review

This section addresses the scoping and screening methodology for license renewal. As required by 10 CFR 54.21(a)(2), the applicant, in its integrated plant assessment (IPA), is to describe and justify methods used to identify systems, structures, and components (SSCs) subject to an aging management review (AMR). The SSCs subject to AMR are those that perform an intended function, as described on 10 CFR 54.4 and meet two criteria:

1. They perform such functions without moving parts or without a change in configuration or properties, as set forth in 10 CFR 54.21(a)(1)(i), (denoted as "passive" components and structures in this SRP), and

2. They are not subject to replacement based on a qualified life or specified time period, as set forth in 10 CFR 54.21(a)(1)(ii), (denoted as "long-lived" structures and components).

The identification of the SSCs within the scope of license renewal is called "scoping." For those SSCs within the scope of license renewal, the identification of "passive," "long-lived" structures and components that are subject to an AMR is called "screening."

To verify that the applicant has properly implemented its methodology, the staff reviews the implementation results separately, following the guidance in Sections 2.2 through 2.5.

The following areas relating to the applicant's scoping and screening methodology are reviewed.

2.1.1.1 Scoping

The methodology used by the applicant to implement the scoping requirements of 10 CFR 54.4, "Scope," is reviewed.

2.1.1.2 Screening

The methodology used by the applicant to implement the "screening" requirements of 10 CFR 54.21(a)(1) is reviewed.

2.1.2 Acceptance Criteria

The acceptance criteria for the areas of review are based on the following regulations:

- 10 CFR 54.4(a) as it relates to the identification of plant SSCs within the scope of the rule;

- 10 CFR 54.4(b) as it relates to the identification of the intended functions of plant SSCs determined to be within the scope of the rule; and

- 10 CFR 54.21(a)(1) and (a)(2) as they relate to the methods utilized by the applicant to identify plant structures and components subject to an AMR.

Specific criteria necessary to determine whether the applicant has met the relevant requirements of 10 CFR 54.4(a), 54.4(b), 54.21(a)(1), and 54.21(a)(2) are as follows.

2.1.2.1 Scoping

The scoping methodology used by the applicant should be consistent with the process described in Section 3.0, "Identify the SSCs Within the Scope of License Renewal and Their Intended Functions," of NEI 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule," Rev. *5* ~~3~~ (Ref. 1), or the justification provided by the applicant for any exceptions should provide a reasonable basis for the exception.

2.1.2.2 Screening

The screening methodology used by the applicant should be consistent with the process described in Section 4.1, "Identification of Structures and Components Subject to an Aging Management Review and Intended Functions," of NEI 95-10, Rev. 3 (Ref. 1) as referenced by Regulatory Guide 1.188~~8~~.

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2.1.3 Review Procedures

Preparation for the review of the scoping and screening methodology employed by the applicant should include the following:

- Review of the NRC's safety evaluation report (SER) that was issued along with the operating license for the facility. This review is conducted for the purpose of familiarization with the principal design criteria for the facility and its CLB, as defined in 10 CFR 54.3(a).

- Review of Chapters 1 through 12 of the Updated Final Safety Analysis Report (UFSAR) and the facility's technical specifications for the purposes of familiarization with the facility design and the nomenclature that is applied to SSCs within the facility (including the bases for such nomenclature). During this review, the SSCs should be identified that are relied upon to remain functional during and after design basis events (DBEs), as defined in 10 CFR 50.49(b)(1)(ii), for which the facility was designed, to ensure that the functions described in 10 CFR 54.4(a)(1) are successfully accomplished. This review should also yield information regarding seismic Category I SSCs as defined in Regulatory Guide 1.29, "Seismic Design Classification" (Ref. 2). For a newer plant, this information is typically contained in Section 3.2.1, "Seismic Classification," of the UFSAR consistent with the Standard Review Plan (NUREG-0800) (Ref. 3).

- Review of Chapter 15 (or equivalent) of the UFSAR to identify the anticipated operational occurrences and postulated accidents that are explicitly evaluated in the accident analyses for the facility. During this review, the SSCs that are relied upon to remain functional during and following design basis events (as defined in 10 CFR 50.49(b)(1)) to ensure the functions described in 10 CFR 54.4(a)(1) should be identified.

- The set of design basis events as defined in the rule is not limited to Chapter 15 (or equivalent) of the UFSAR. Examples of design basis events that may not be described in this chapter include external events, such as floods, storms, earthquakes, tornadoes, or hurricanes, and internal events, such as a high-energy-line break. Information regarding design basis events as defined in 10 CFR 50.49(b)(1) may be found in any chapter of the facility UFSAR, the Commission's regulations, NRC orders, exemptions, or license conditions within the CLB. These sources should also be reviewed to identify systems, structures, and components that are relied upon to remain functional during and following design basis events (as defined in 10 CFR 50.49(b)(1)) to ensure the functions described in 10 CFR 54.4(a)(1).

- Review of the facility's Probabilistic Risk Analysis (PRA) Summary Report that was prepared by the licensee in response to Generic Letter (GL) 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR 50.54(f)," dated November 23, 1988 (Ref. 4). This review should yield additional information regarding the impact of the Individual Plant Examination (IPE) on the CLB for the facility. While the LR Rule is "deterministic," the NRC in the statement of considerations (SOC) accompanying the Rule also states: "In license renewal, probabilistic methods may be most useful, on a plant-specific basis, in helping to assess the relative importance of structures and components that are subject to an aging management review by helping to draw attention to specific vulnerabilities (e.g., results of an IPE or IPEEE)" (60 FR 22468). For example, the reviewer should focus on IPE information pertaining to plant changes or modifications that are initiated by the licensee in accordance with the requirements of 10 CFR 50.59 or 10 CFR 50.90.

- Review of the results of the facility's Individual Plant Examination of External Events (IPEEE) study conducted as a follow-up to the IPE performed as a result of GL 88-20 to identify any changes or modifications made to the facility in accordance with the requirements of 10 CFR 50.59 or 10 CFR 50.90.

- Review of the applicant's docketed correspondence related to the following regulations:

  (a) 10 CFR 50.48, "Fire Protection,"

  (b) 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants,"

  (c) 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," [applicable to pressurized water reactor (PWR) plants].

  (d) 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients without Scram Events for Light-Water-Cooled Nuclear Power Plants," and

(e) 10 CFR 50.63, "Loss of All Alternating Current Power." [applicable to pressurized water reactor (PWR) plants].

Other staff members are reviewing the applicant's scoping and screening results separately following the guidance in Sections 2.2 through 2.5. The reviewer should keep these other staff members informed of findings that may affect their review of the applicant's scoping and screening results. The reviewer should coordinate this sharing of information through the license renewal project manager.

2.1.3.1 Scoping

Once the information delineated above has been gathered, the reviewer reviews the applicant's methodology to determine whether its depth and breadth are sufficiently comprehensive to identify the SSCs within the scope of license renewal, and the structures and components requiring an AMR. Because "[t]he CLB represents the evolving set of requirements and commitments for a specific plant that are modified as necessary over the life of a plant to ensure continuation of an adequate level of safety" (60 FR 22465, May 8, 1995), the regulations, orders, license conditions, exemptions, and technical specifications defining functional requirements for facility SSCs that make up an applicant's CLB should be considered as the initial input into the scoping process. 10 CFR 50.49 defines DBEs as conditions of normal operation, including anticipated operational occurrences, DBAs, external events, and natural phenomena for which the plant must be designed to ensure (1) the integrity of the reactor pressure boundary, (2) the capability to shut down the reactor and maintain it in safe shutdown condition, or (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to those referred to in 10 CFR 50.34(a)(1), 50.67(b)(2), or 100.11, as applicable. Therefore, to determine the safety-related SSCs that are within the scope of the rule under 10 CFR 54.4 (a)(1), the applicant must identify those SSCs that are relied upon to remain functional during and following these DBEs, consistent with the CLB of the facility. Most licensees have developed lists or database that identify systems, structures and components relied on for compliance with other regulations in a manner consistent with the CLB of their facilities. Consistent with the licensing process and regulatory criteria used to develop such lists or databases, licensees should build upon these information sources to satisfy 10 CFR Part 54 requirements.

With respect to technical specifications, the NRC states (60 FR 22467):

> *The Commission believes that there is sufficient experience with its policy on technical specifications to apply that policy generically in revising the license renewal rule consistent with the Commission's desire to credit existing regulatory programs. Therefore, the Commission concludes that the technical specification limiting conditions for operation scoping category is unwarranted and has deleted the requirement that identifies systems, structures, and components with operability requirements in technical specifications as being within the scope of the license renewal review.*

Therefore, the applicant need not consider its technical specifications and applicable limiting conditions of operation when scoping for license renewal. This is not to say that the events and functions addressed within the applicant's technical specifications can be excluded in determining the SSCs within the scope of license renewal solely on the basis of such an event's inclusion in the technical specifications. Rather, those SSCs governed by an applicant's technical specifications that are relied upon to remain functional during a DBE, as identified within the applicant's UFSAR, applicable NRC regulations, license conditions, NRC orders, and exemptions, need to be included within the scope of license renewal.

For licensee commitments, such as licensee responses to NRC bulletins, generic letters, or enforcement actions, and those documented in staff safety evaluations or licensee event reports, and which make up the remainder of an applicant's CLB, many of the associated SSCs need not be considered under license renewal. Generic communications, safety evaluations, and other similar documents found on the docket are not regulatory requirements, and commitments made by a licensee to address any associated safety concerns are not typically considered to be design requirements. However, any generic communication, safety evaluation, or licensee commitment that specifically identifies or describes a function associated with a system, structure, or component necessary to fulfill the requirement of a particular regulation, order, license condition, and/or exemption may need to be considered when scoping for license renewal. For example, NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification," states:

> *The licensing basis according to 10 CFR 50.55a for all PWRs requires that the licensee meet the American Society of Mechanical Engineers Boiler and Pressure Vessel Code Sections III and XI and to reconcile the pipe stresses and fatigue evaluation when any significant differences are observed between measured data and the analytical results for the hypothesized conditions. Staff evaluation indicates that the thermal stratification phenomenon could occur in all PWR surge lines and may invalidate the analyses supporting the integrity of the surge line. The staff's concerns include unexpected bending and thermal striping (rapid oscillation of the thermal boundary interface along the piping inside surface) as they affect the overall integrity of the surge line for its design life (e.g., the increase of fatigue).*

Therefore, this bulletin specifically describes conditions that may affect compliance with the requirements associated with 10 CFR 50.55a and functions specifically related to this regulation that must be considered in the scoping process for license renewal.

An applicant may take an approach in scoping and screening that combines similar components from various systems. For example, containment isolation valves from various systems may be identified as a single system for purposes of license renewal.

Staff from branches responsible for systems may be requested to assist in reviewing the plant design basis and intended function(s), as necessary.

The reviewer should verify that the applicant's scoping methods document the actual information sources used (for example, those identified in Table 2.1-1).

Table 2.1-2 contains specific staff guidance on certain subjects of scoping.

2.1.3.1.1 Safety-Related

The applicant's methodology is reviewed to ensure that the safety-related SSCs are identified to satisfactorily accomplish any of the intended functions identified in 10 CFR 54.4(a)(1). The reviewer must ascertain how, and to what extent, the applicant incorporated the information in the CLB for the facility in its methodology. Specifically, the reviewer should review the application, as well as all other relevant sources of information outlined above, to identify the set of plant-specific conditions of normal operation, DBAs, external events, and natural phenomena for which the plant must be designed to ensure the following functions:

- The integrity of the reactor coolant pressure boundary;
- The capability to shut down the reactor and maintain it in a safe shutdown condition; or
- The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the guidelines in 10 CFR 50.34(a)(1), 50.67(b)(2), or 100.11, as applicable.

2.1.3.1.2 Nonsafety-Related

The applicant's methodology is reviewed to ensure that nonsafety-related SSCs whose failure could prevent satisfactory accomplishment of any of the functions identified in 10 CFR 54.4(a)(1) are identified as being within the scope of license renewal.

The scoping criterion under 10 CFR 54.4(a)(2), in general, is intended to identify those nonsafety-related SSCs that support safety-related functions. More specifically, this scoping criterion requires an applicant to identify all nonsafety-related SSCs whose failure could prevent satisfactory accomplishment of the applicable functions of the SSCs identified under 10 CFR 54.4(a)(1). Section III.c(iii) of the SOC (60 FR 22467) clarifies the NRC's intent for this requirement in the following statement:

> *The inclusion of nonsafety-related systems, structures, and components whose failure could prevent other systems, structures, and components from accomplishing a safety function is intended to provide protection against safety function failure in cases where the safety-related structure or component is not itself impaired by age-related degradation but is vulnerable to failure from the failure of another structure or component that may be so impaired.*

In addition, Section III.c(iii) of the SOC provides the following guidance to assist an applicant in determining the extent to which failures must be considered when applying this scoping criterion:

> *Consideration of hypothetical failures that could result from system interdependencies that are not part of the current licensing bases and that have not been previously experienced is not required. [...] However, for some license renewal applicants, the Commission cannot exclude the possibility that hypothetical failures that <u>are part of the CLB</u> may require consideration of second-, third-, or fourth-level support systems.*

Therefore, to satisfy the scoping criterion under 10 CFR 54.4(a)(2), the applicant must identify those nonsafety-related SSCs (including certain second-, third-, or fourth-level support systems) whose failures are considered in the CLB and could prevent the satisfactory accomplishment of a safety-related function identified under 10 CFR 54.4(a)(1). In order to identify such systems, the applicant should consider those failures identified in (1) the documentation that makes up its CLB, (2) plant-specific operating experience, and (3) industry-wide operating experience that is specifically applicable to its facility. The applicant need not consider hypothetical failures that are not part of the CLB, have not been previously experienced, or are not applicable to its facility.

In part, 10 CFR 54(a)(2) requires that the applicant consider all nonsafety-related SSCs whose failure could prevent satisfactory accomplishment of any of the functions identified in 10 CFR 54.4(a)(1)(i), 10 CFR 54.4(a)(1)(ii), or 10 CFR 54.4(a)(1)(iii) to be within the scope of license renewal. By letters dated December 3, 2001, and March 15, 2002, the NRC issued a staff position to NEI which provided staff guidance for determining what SSCs meet the 10 CFR 54.4(a)(2) criterion. The December 3, 2001 letter, "License Renewal Issue: Scoping of Seismic II/I Piping Systems," provided specific examples of operating experience which identified pipe failure events (summarized in Information Notice (IN) 2001-09, "Main Feedwater System Degradation in Safety-Related ASME Code Class 2 Piping Inside the Containment of a Pressurized Water Reactor") and the approaches the NRC considers acceptable to determine which piping systems should be included in scope based on the 10 CFR 54.4(a)(2) criterion. The March 15, 2002 letter, "License Renewal Issue: Guidance on the Identification and Treatment of Structures, Systems, and Components Which Meet 10 CFR 54.4(a)(2)," further described the staff's recommendations for the evaluation of non-piping SSCs to determine which additional nonsafety-related SSCs are within scope. The position states that the applicants should not consider hypothetical failures, but rather should base their evaluation on the plant's CLB, engineering judgment and analyses, and relevant operating experience. The paper further describes operating experience as all documented plant-specific and industry-wide experience that can be used to determine the plausibility of a failure. Documentation would include NRC generic communications and event reports, plant-specific condition reports, industry reports such as significant operating experience reports (SOERs), and engineering evaluations.

For example, the safety classification of a pipe at certain locations, such as valves, may change throughout its length in the plant. In these instances, the applicant should identify the safety-related portion of the pipe as being within the scope of license renewal under 10 CFR 54.4(a)(1). However, the entire pipe run, including associated piping anchors, may have been analyzed as part of the CLB to establish that it could withstand DBE loads. If this is the case, a failure in the pipe run or in the associated piping anchors could render the safety-related portion of the piping unable to perform its intended function under CLB design conditions. Therefore, the reviewer must verify that the applicant's methodology would include (1) the remaining nonsafety-related piping up to its anchors and (2) the associated piping anchors as being within the scope of license renewal under 10 CFR 54.4(a)(2).

In order to comply, in part, with the requirements of 10 CFR 54.4(a)(2), all applicants must include in scope all NSR piping attached directly to SR piping (within scope) up to a defined anchor point consistent with the plant CLB. This anchor point may be served by a true anchor (a device or structure which ensures forces and moments are restrained in three (3) orthogonal directions) or an equivalent anchor, such as a large piece of plant equipment (e.g., a heat exchanger,) determined by an evaluation of the plant-specific piping design (i.e., design documentation, such as piping stress analysis for the facility).

Applicants should be able to define an equivalent anchor consistent with their CLB (e.g., described in the UFSAR or other CLB documentation), which is being credited for the 10 CFR 54.4(a)(2) evaluation, and be able to describe the structures and components that are part of the NSR piping segment boundary up to and including the anchor point or equivalent anchor point within scope of the rule.

There may be isolated cases where an equivalent anchor point for a particular piping segment is not clearly described within the existing CLB information. In those instances the applicant may use a combination of restraints or supports such that the NSR piping and associated structures and components attached to SR piping is included in scope up to a boundary point which encompasses at least two (2) supports in each of three (3) orthogonal directions.

It is important to note that the scoping criterion under 10 CFR 54.4(a)(2) specifically applies to those functions "identified in paragraphs (a)(1)(i), (ii), and (iii)" of 10 CFR 54.4 and does not apply to functions identified in 10 CFR 54.4(a)(3), as discussed below.

2.1.3.1.3 "Regulated Events"

The applicant's methodology is reviewed to ensure that structures, systems and components (SSCs) relied on in safety analyses or plant evaluations to perform functions that demonstrate compliance with the requirements of the fire protection, environmental qualification, pressurized thermal shock (PTS), anticipated transients without scram (ATWS), and station blackout (SBO) regulations are identified. The reviewer should review the applicant's docketed correspondence associated with compliance of the facility with these regulations.

The scoping criteria in 10 CFR 54.4(a)(3) require an applicant to consider *"[a]ll SSC relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the [specified] Commission regulations[.]"* In addition, Section III.c(iii) (60 FR 22467) of the SOC states that the NRC intended to limit the potential for unnecessary expansion of the review for SSCs that meet the scoping criteria under 10 CFR 54.4(a)(3) and provides additional guidance that qualifies what is meant by "*those SSCs relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the Commission regulations*" in the following statement:

> *[T]he Commission intends that this [referring to 10 CFR 54.4(a)(3)] scoping category include all SSC whose function is relied upon to demonstrate compliance with these Commission[ ] regulations. An applicant for license renewal should rely on the plant's current licensing bases, actual plant-specific experience, industry-wide operating experience, as appropriate, and existing engineering evaluations to determine those SSC that are the initial focus of license renewal review.*

Therefore, all SSCs that are relied upon in the plant's CLB (as defined in 10 CFR 54.3), plant-specific experience, industry-wide experience (as appropriate), and safety analyses or plant evaluations to perform a function that demonstrates compliance with NRC regulations identified under 10 CFR 54.4(a)(3), are required to be included within the scope of the rule. For example, if a nonsafety-related diesel generator is required for safe shutdown under the fire protection plan, the diesel generator and all SSCs specifically relied upon for that generator to comply with NRC regulations shall be included within the scope of license renewal under 10 CFR 54.4(a)(3). Such SSCs may include, but should not be limited to, the cooling water system or systems relied upon for operability, the diesel support pedestal, and any applicable power supply cable specifically required for safe shutdown in the event of a fire.

In addition, the last sentence of the second paragraph in Section III.c(iii) of the SOC provides the following guidance for limiting the application of the scoping criterion under 10 CFR 54.4(a)(3) as it applies to the use of hypothetical failures:

Consideration of hypothetical failures that could result from system interdependencies, that are not part of the current licensing bases and that have not been previously experienced is not required. (60 FR 22467)

The SOC does not provide any additional guidance relating to the use of hypothetical failures or the need to consider second-, third-, or fourth-level support systems for scoping under 10 CFR 54.4(a)(3). Therefore, in the absence of any guidance, an applicant need not consider hypothetical failures or second-, third-, or fourth-level support systems in determining the SSCs within the scope of the rule under 10 CFR 54.4(a)(3). For example, if a nonsafety-related diesel generator is relied upon only to remain functional to demonstrate compliance with the NRC SBO regulation, the applicant need not consider the following SSCs: (1) an alternate/backup cooling water system, (2) non-seismically-qualified building walls, or (3) an overhead segment of non-seismically-qualified piping (in a Seismic II/I configuration). This guidance is not intended to exclude any support system (whether identified by an applicant's CLB, or as indicated from actual plant-specific experience, industrywide experience [as applicable], safety analyses, or plant evaluations) that is specifically relied upon for compliance with, the applicable NRC regulation. For example, if analysis of a nonsafety-related diesel generator (relied upon to demonstrate compliance with an applicable NRC regulation) specifically relies upon a second cooling system to cool the diesel generator jacket water cooling system for the generator to be operable, then both cooling systems must be included within the scope of the rule under 10 CFR 54.4(a)(3).

The applicant is required to identify the SSCs whose functions are relied upon to demonstrate compliance with the regulations identified in 10 CFR 54.4(a)(3) (that is, whose functions were credited in the analysis or evaluation). Mere mention of an SSC in the analysis or evaluation does not necessarily constitute support of an intended function as required by the regulation.

For environmental qualification, the reviewer verifies that the applicant has indicated that the environmental qualification equipment is that equipment already identified by the licensee under 10 CFR 50.49(b), that is, equipment relied upon in safety analyses or plant evaluations to demonstrate compliance with NRC regulations for environmental qualification (10 CFR 50.49).

The PTS regulation is applicable only to PWRs. If the renewal application is for a PWR and the applicant relies on a Regulatory Guide 1.154 (Ref. 5) analysis to satisfy 10 CFR 50.61, as described in the plant's CLB, the reviewer verifies that the applicant's methodology would include SSCs relied on in that analysis.

For SBO, the reviewer verifies that the applicant's methodology would include those SSCs relied upon during the "coping duration" and "recovery" phase of an SBO event (Ref. 6). In addition, because 10 CFR 50.63(c)(1)(ii) and its associated guidance in Regulatory Guide 1.155 include procedures to recover from an SBO that include offsite and onsite power, the plant system portion of the offsite power system that is used to connect the plant to the offsite power source should also be included within the scope of the rule.

2.1.3.2 Screening

Once the SSCs within the scope of license renewal have been identified, the next step is determining which structures and components are subject to an AMR (i.e., "screening") (Ref. 1).

2.1.3.2.1 "Passive"

The reviewer reviews the applicant's methodology to ensure that "passive" structures and components are identified as those that perform their intended functions without moving parts or a change in configuration or properties in accordance with 10 CFR 54.21(a)(1)(i). The description of "passive" may also be interpreted to include structures and components that do not display "a change in state." 10 CFR 54.21(a)(1)(i) provides specific examples of structures and components that do or do not meet the criterion. The reviewer verifies that the applicant's screening methodology includes consideration of the intended functions of structures and components consistent with the plant's CLB, as typified in Tables 2.1-4(a) and (b), respectively. (Ref. 1).

The license renewal rule focuses on "passive" structures and components because structures and components that have passive functions generally do not have performance and condition characteristics that are as readily observable as those that perform active functions. "Passive" structures and components, for the purpose of the license renewal rule, are those that perform an intended function, as described in 10 CFR 54.4, without moving parts or without a change in configuration or properties (Ref. 2). The description of "passive" may also be interpreted to include structures and components that do not display "a change of state."

Table 2.1-5 provides a list of typical structures and components identifying whether they meet 10 CFR 54.21(a)(1)(i).

10 CFR 54.21(a)(1)(i) explicitly excludes instrumentation, such as pressure transmitters, pressure indicators, and water level indicators, from an AMR. The applicant does not have to identify pressure-retaining boundaries of this instrumentation because 10 CFR 54.21(a)(1)(i) excludes this instrumentation without exception, unlike pumps and valves. Further, instrumentation is sensitive equipment and degradation of its pressure retaining boundary would be readily determinable by surveillance and testing (Ref.6). If an applicant determines that certain structures and components listed in Table 2.1-5 as meeting 10 CFR 54.21(a)(1)(i) do not meet that requirement for its plant, the reviewer reviews the applicant's basis for that determination.

2.1.3.2.2 "Long-Lived"

The applicant's methodology is reviewed to ensure that "long-lived" structures and components are identified as those that are not subject to periodic replacement based on a qualified life or specified time period. Passive structures and components that are not replaced on the basis of a qualified life or specified time period require an AMR.

Replacement programs may be based on vendor recommendations, plant experience, or any means that establishes a specific replacement frequency under a controlled program. Section f(i)(b) of the SOC provides the following guidance for identifying "long-lived" structures and components:

> *In sum, a structure or component that is not replaced either (i) on a specified interval based upon the qualified life of the structure or component or (ii) periodically in accordance with a specified time period is deemed by § 54.21(a)(1)(ii) of this rule to be "long-lived," and therefore subject to the § 54.21(a)(3)aging management review [60 FR 22478].*

A qualified life does not necessarily have to be based on calendar time. A qualified life based on run time or cycles are examples of qualified life references that are not based on calendar time (Ref. 3).

Structures and components that are replaced on the basis of performance or condition are not generically excluded from an AMR. Rather, performance or condition monitoring may be evaluated later in the IPA as programs to ensure functionality during the period of extended operation. On this topic, Section f(i)(b) of the SOC provides the following guidance:

> *It is important to note, however, that the Commission has decided not to generically exclude passive structures and components that are replaced based on performance or condition from an aging management review. Absent the specific nature of the performance or condition replacement criteria and the fact that the Commission has determined that the components with "passive" functions are not as readily monitorable as components with active functions, such generic exclusion is not appropriate. However, the Commission does not intend to preclude a license renewal applicant from providing site-specific justification in a license renewal application that a replacement program on the basis of performance or condition for a passive structure or component provides reasonable assurance that the intended function of the passive structure or component will be maintained in the period of extended operation. [60 FR 22478]*

2.1.4 Evaluation Findings

If the review of the information in the license renewal application is complete, and the reviewer has determined that it is satisfactory and in accordance with the acceptance criteria in Subsection 2.1.2, a statement of the following type should be included in the staff's safety evaluation report:

> The staff concludes that there is reasonable assurance that the applicant's methodology for identifying the systems, structures, and components within the scope of license renewal and the structures and components requiring an aging management review is consistent with the requirements of 10 CFR 54.4 and 10 CFR 54.21(a)(1).

2.1.5 Implementation

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of NRC regulations, the method described herein will be used by the staff in its evaluation of conformance with NRC regulations.

2.1.6 References

1. NEI 95-10, Rev. 5 "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule," Nuclear Energy Institute, January 2005.

2. Regulatory Guide 1.29, Rev. 2, "Seismic Design Classification," September 1978.

3. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," July 1981.

4. Generic Letter (GL) 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities-10 CFR 50.54(f)," dated November 23, 1988.

5. Regulatory Guide 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors," January 1987.

6. Deleted.

7. NUREG-1723, "Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Stations, Units 1, 2, and 3," March 2000.

8. Letter to Douglas J. Walters, Nuclear Energy Institute, from Christopher I. Grimes, NRC, dated August 5, 1999.

9. Summary of December 8, 1999, Meeting with the Nuclear Energy Institute (NEI) on License Renewal Issue (LR) 98-12, "Consumables," Project No. 690, January 21, 2000.

10. Letter to William R. McCollum, Jr., Duke Energy Corporation, from Christopher I. Grimes, NRC, dated October 8,1999.

11. NEI 95-10, Rev. 0, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule," Nuclear Energy Institute, March 1, 1996.

12. Letter to Alan Nelson, Nuclear Energy Institute, and David Lochbaum, Union of Concerned Scientists, from Christopher I. Grimes, NRC, "License Renewal Issue: Scoping of Seismic II/I Piping Systems," dated December 3, 2001.

13. Letter to Alan Nelson, Nuclear Energy Institute, and David Lochbaum, Union of Concerned Scientists, from Christopher I. Grimes, NRC, "License Renewal Issue: Guidance on the Identification and Treatment of Structures, Systems, and Components Which Meet 10 CFR 54.4(a)(2)," dated March 15, 2002.

14. Letter to Alan Nelson, Nuclear Energy Institute, and David Lochbaum, Union of Concerned Scientists, from Christopher I. Grimes, NRC, "Staff Guidance on Scoping of Equipment Relied on to Meet the Requirements of the Station Blackout (SBO) Rule (10 CFR 50.63) for License Renewal (10 CFR 54.4(a)(3))," dated April 1, 2002.

Table 2.1-1. Sample Listing of Potential Information Sources

Verified databases (databases that are subject to administrative controls to assure and maintain the integrity of the stored data or information)
Master equipment lists (including NSSS vendor listings)
Q-lists
Updated Final Safety Analysis Reports
Piping and instrument diagrams
NRC Orders, Exemptions, or License Conditions for the facility
Design-basis documents
General arrangement or structural outline drawings
Probabilistic risk assessment summary report
Maintenance rule compliance documentation
Design-basis event evaluations (including plant-specific 10 CFR 50.59 evaluation procedures)
Emergency operating procedures
Docketed correspondence
System interaction commitments
Technical specifications
Environmental qualification program documents
Regulatory compliance reports (including Safety Evaluation Reports)
Severe Accident Management Guidelines

Table 2.1-2. Specific Staff Guidance on Scoping

Issue	Guidance
Commodity groups	The applicant may also group like structures and components into commodity groups. Examples of commodity groups are pipe supports and cable trays. The basis for grouping structures and components can be determined by such characteristics as similar function, similar design, similar materials of construction, similar aging management practices, or similar environments. If the applicant uses commodity groups, the reviewer verifies that the applicant has described the basis for the groups.
Complex assemblies	Some structures and components, when combined, are considered a complex assembly (for example, diesel generator starting air skids or heating, ventilating, and air conditioning refrigerant units). For purposes of performing an AMR, it is important to clearly establish the boundaries of review. An applicant should establish the boundaries for such assemblies by identifying each structure and component that makes up the complex assembly and determining whether or not each structure and component is subject to an AMR (Ref. 1). NEI 95-10, Revision 0, Appendix C, Example 5 (Ref. 11), illustrates how the evaluation boundary for a control room chiller complex assembly might be determined. The control room chillers were purchased as skid mounted equipment. These chillers are part of the control room chilled water system. There are two (2) control room chillers. Each is a 100% capacity refrigeration unit. The functions of the control room chillers are: to provide a reliable source of chilled water at a maximum temperature of 44°F, to provide a pressure boundary for the control room chilled water system, to provide a pressure boundary for the service water system, and to provide a pressure boundary for the refrigerant. All of these functions are considered intended functions. Typically, control room chillers are considered as one functional unit; however, for purposes of evaluating the effects of aging, it is necessary to consider the individual components. Therefore, the boundary of each control room chiller is established as follows: 1. At the inlet and outlet flanges of the service water system connections on the control room chiller condenser. Connected piping is part of the service water system. 2. At the inlet and outlet flanges of the control room chilled water system piping connections on the control room chiller evaporator. Connected piping is part of the control room chilled water system. 3. For electrical power supplies, the boundary is the output terminals on the circuit breakers supplying power to the skid. This includes the cables from the circuit breaker to the skid and applies for 480 VAC and 120 VAC. 4. The interface for instrument air supplies is at the instrument air tubing connection to the pressure control regulators, temperature controllers and transmitters, and solenoid valves located on the skid. The tubing from the instrument air header to the device on the skid is part of the instrument air system. 5. The interface with the annunciator system is at the external connection of the contacts of the device on the skid (limit switch, pressure switch, level switch, etc.) that indicates the alarm condition. The cables are part of the annunciator system. Based on the boundary established, the following components would be subject to an aging management review: condenser, evaporator, economizer, chiller refrigerant piping, refrigerant expansion orifice, foundations and bolting, electrical cabinets, cables, conduit, trays and supports, valves

Table 2.1-2. Specific Staff Guidance on Scoping (continued)

Issue	Guidance
Hypothetical failures	For 10 CFR 54.4(a)(2), an applicant should consider those failures identified in (1) the documentation that makes up its CLB, (2) plant-specific operating experience, and (3) industry-wide operating experience that is specifically applicable to its facility. The applicant need not consider hypothetical failures that are not part of CLB and that have not been previously experienced. For example, an applicant should consider including (1) the portion of a fire protection system identified in the UFSAR that supplies water to the refueling floor that is relied upon in a DBA analysis as an alternate source of cooling water that can be used to mitigate the consequences from the loss of spent fuel pool cooling, (2) a nonsafety-related, non-seismically-qualified building whose intended function as described in the plant's CLB is to protect a tank that is relied upon as an alternate source of cooling water needed to mitigate the consequences of a DBE, and (3) a segment of nonsafety-related piping identified as a Seismic II/I component in the applicant's CLB (Ref. 8).
Cascading	For 10 CFR 54.4(a)(3), an applicant need not consider hypothetical failures or second-, third, or fourth-level support systems. For example, if a nonsafety-related diesel generator is only relied upon to remain functional to demonstrate compliance with the NRC's SBO regulations, an applicant may not need to consider (1) an alternate/backup cooling water system, (2) the diesel generator non-seismically-qualified building walls, or (3) an overhead segment of non-seismically-qualified piping (in a Seismic II/I configuration). An applicant may not exclude any support system (identified by its CLB, actual plant-specific experience, industry-wide experience, as applicable, or existing engineering evaluations) that is specifically relied upon for compliance with, or operation within, applicable NRC regulation. For example, if the analysis of a nonsafety-related diesel generator (relied upon to demonstrate compliance with an applicable NRC regulation) specifically relies upon a second cooling system to cool the diesel generator jacket water cooling system for the diesel to be operable, then both cooling systems must be included within the scope of the rule (Ref. 8).

Table 2.1-3. Specific Staff Guidance on Screening

Issue	Guidance
Consumables	Consumables may be divided into the following four categories for the purpose of license renewal: (a) packing, gaskets, component seals, and O-rings; (b) structural sealants; (c) oil, grease, and component filters; and (d) system filters, fire extinguishers, fire hoses, and air packs. The consumables in both categories (a) and (b) are considered as subcomponents and are not explicitly called out in the scoping and screening procedures. Rather, they are implicitly included at the component level (e.g., if a valve is identified as being in scope, a seal in that valve would also be in scope as a subcomponent of that valve). For category (a), the applicant would generally be able to exclude these subcomponents using a clear basis, such as the example of ASME Section III not being relied on for pressure boundary. For category (b), these subcomponents may perform functions without moving parts or a change in configuration, and they are not typically replaced. The applicant's structural AMP should address these items with respect to an AMR program on a plant-specific basis. The consumables in category (c) are usually short-lived and periodically replaced, and can normally be excluded from an AMR on that basis. Likewise, the consumables that fall within category (d) are typically replaced based on performance or condition monitoring that identifies whether these components are at the end of their qualified lives and may be excluded, on a plant-specific basis, from AMR under 10 CFR 54.21(a)(1)(ii). The applicant should identify the standards that are relied on for the replacement as part of the methodology description (for example, NFPA standards for fire protection equipment) (Ref. 9).
Heat exchanger intended functions	Both the pressure boundary and heat transfer functions for heat exchangers should be considered because heat transfer may be a primary safety function of these components. There may be a unique aging effect associated with different materials in the heat exchanger parts that are associated with the heat transfer function and not the pressure boundary function. Normally the programs that effectively manage aging effects of the pressure boundary function can, in conjunction with the procedures for monitoring heat exchanger performance, effectively manage aging effects applicable to the heat transfer function (Ref. 10).
Multiple functions	Structures and components may have multiple functions. The intended functions as delineated in 10 CFR 54.4(b) are to be reviewed for license renewal. For example, a flow orifice that is credited in a plant's accident analysis to limit flow would have two intended functions. One intended function is pressure boundary. The other intended function is to limit flow. The reviewer verifies that the applicant has considered multiple functions in identifying structure and component intended functions.

Table 2.1-4(a) Typical "Passive" Structure Intended Functions

Structures	
Intended Function	**Description**
Direct Flow	Provide spray shield or curbs for directing flow (e.g., safety injection flow to containment sump)
Expansion/Separation	Provide for thermal expansion and/or seismic separation
Fire Barrier	Provide rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant
Flood Barrier	Provide flood protection barrier (internal and external flooding event)
Gaseous Release Path	Provide path for release of filtered and unfiltered gaseous discharge
Heat Sink	Provide heat sink during station blackout or design-basis accidents
HELB Shielding	Provide shielding against high-energy line breaks
Missile Barrier	Provide missile barrier (internally or externally generated)
~~Non-S/R Structural Support~~	~~Provide structural support to nonsafety-related components whose failure could prevent satisfactory accomplishment of any of the required safety-related functions~~
Pipe Whip Restraint	Provide pipe whip restraint
Pressure Relief	Provide over-pressure protection
Shelter, Protection	Provide shelter/protection to safety-related components
Shielding	Provide shielding against radiation
Shutdown Cooling Water	Provide source of cooling water for plant shutdown
Structural Pressure Barrier	Provide pressure boundary or essentially leaktight barrier to protect public health and safety in the event of any postulated design-basis events.
~~Structural Support for Criterion (a)(1) equipment~~	~~Provide structural support and/or functional support to safety-related equipment~~

Table 2.1-4(b) Typical "Passive" Component Intended Functions

Components	
Intended Function	**Description**
Absorb Neutrons	Absorb neutrons
Electrical Continuity	Provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals
Insulate (electrical)	Insulate and support an electrical conductor
Filter	Provide filtration
Heat Transfer	Provide heat transfer
Leakage Boundary (Spatial)	Nonsafety-related component that maintains mechanical and structural integrity to prevent spatial interactions that could cause failure of safety-related SSCs
Pressure Boundary	Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered, or provide fission product barrier for containment pressure boundary, or provide containment isolation for fission product retention
Spray	Convert fluid into spray
Structural Integrity (Attached)	Nonsafety-related component that maintains mechanical and structural integrity to provide structural support to attached safety-related piping and components
Structural Support	Provide structural ~~and / or functional~~ support to safety-related components ***and/or nonsafety-related components***
Throttle	Provide flow restriction

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Table 2.1-5. Typical Structures, Components, and Commodity Groups, and 10 CFR 54.21(a)(1)(i) Determinations for Integrated Plant Assessment

Item	Category	Structure, Component, or Commodity Grouping	Structure, Component, or Commodity Group Meets 10 CFR 54.21(a)(1)(i) (Yes/No)
1	Structures	Category I Structures	Yes
2	Structures	Primary Containment Structure	Yes
3	Structures	Intake Structures	Yes
4	Structures	Intake Canal	Yes
5	Structures	Other Non-Category I Structures Within the Scope of License Renewal	Yes
6	Structures	Equipment Supports and Foundations	Yes
7	Structures	Structural Bellows	Yes
8	Structures	Controlled Leakage Doors	Yes
9	Structures	Penetration Seals	Yes
10	Structures	Compressible Joints and Seals	Yes
11	Structures	Fuel Pool and Sump Liners	Yes
12	Structures	Concrete Curbs	Yes
13	Structures	Offgas Stack and Flue	Yes
14	Structures	Fire Barriers	Yes
15	Structures	Pipe Whip Restraints and Jet Impingement Shields	Yes
16	Structures	Electrical and Instrumentation and Control Penetration Assemblies	Yes
17	Structures	Instrumentation Racks, Frames, Panels, and Enclosures	Yes
18	Structures	Electrical Panels, Racks, Cabinets, and Other Enclosures	Yes
19	Structures	Cable Trays and Supports	Yes
20	Structures	Conduit	Yes
21	Structures	Tube Track	Yes
22	Structures	Reactor Vessel Internals	Yes
23	Structures	ASME Class 1 Hangers and Supports	Yes
24	Structures	Non-ASME Class 1 Hangers and Supports	Yes
25	Structures	Snubbers	No
26	Reactor Coolant Pressure Boundary Components (Note: the components of the RCPB are defined by each plant's CLB and site specific documentation	ASME Class 1 Piping	Yes
27	Reactor Coolant Pressure Boundary Components	Reactor Vessel	Yes
28	Reactor Coolant Pressure Boundary Components	Reactor Coolant Pumps	Yes (Casing)

Table 2.1-5. Typical Structures, Components, and Commodity Groups, and 10 CFR 54.21(a)(1)(i) Determinations for Integrated Plant Assessment (continued)

Item	Category	Structure, Component, or Commodity Grouping	Structure, Component, or Commodity Group Meets 10 CFR 54.21(a)(1)(i) (Yes/No)
29	Reactor Coolant Pressure Boundary Components	Control Rod Drives	No
30	Reactor Coolant Pressure Boundary Components	Control Rod Drive Housing	Yes
31	Reactor Coolant Pressure Boundary Components	Steam Generators	Yes
32	Reactor Coolant Pressure Boundary Components	Pressurizers	Yes
33	Non-Class I Piping Components	Underground Piping	Yes
34	Non-Class I Piping Components	Piping in Low Temperature Demineralized Water Service	Yes
35	Non-Class I Piping Components	Piping in High Temperature Single Phase Service	Yes
36	Non-Class I Piping Components	Piping in Multiple Phase Service	Yes
37	Non-Class I Piping Components	Service Water Piping	Yes
38	Non-Class I Piping Components	Low Temperature Gas Transport Piping	Yes
39	Non-Class I Piping Components	Stainless Steel Tubing	Yes
40	Non-Class I Piping Components	Instrument Tubing	Yes
41	Non-Class I Piping Components	Expansion Joints	Yes
42	Non-Class I Piping Components	Ductwork	Yes
43	Non-Class I Piping Components	Sprinklers Heads	Yes
44	Non-Class I Piping Components	Miscellaneous Appurtenances (Includes fittings, couplings, reducers, elbows, thermowells, flanges, fasteners, welded attachments, etc.)	Yes
45	Pumps	ECCS Pumps	Yes (Casing)
46	Pumps	Service Water and Fire Pumps	Yes (Casing)
47	Pumps	Lube Oil and Closed Cooling Water Pumps	Yes (Casing)
48	Pumps	Condensate Pumps	Yes (Casing)
49	Pumps	Borated Water Pumps	Yes (Casing)
50	Pumps	Emergency Service Water Pumps	Yes (Casing)
51	Pumps	Submersible Pumps	Yes (Casing)
52	Turbines	Turbine Pump Drives (excluding pumps)	Yes (Casing)
53	Turbines	Gas Turbines	Yes (Casing)

Table 2.1-5. Typical Structures, Components, and Commodity Groups, and 10 CFR 54.21(a)(1)(i) Determinations for Integrated Plant Assessment (continued)

Item	Category	Structure, Component, or Commodity Grouping	Structure, Component, or Commodity Group Meets 10 CFR 54.21(a)(1)(i) (Yes/No)
54	Turbines	Controls (Actuator and Overspeed Trip)	No
55	Engines	Fire Pump Diesel Engines	No
56	Emergency Diesel Generators	Emergency Diesel Generators	No
57	Heat Exchangers	Condensers	Yes
58	Heat Exchangers	HVAC Coolers	Yes
59	Heat Exchangers	Primary Water System Heat Exchangers	Yes
60	Heat Exchangers	Treated Water System Heat Exchangers	Yes
61	Heat Exchangers	Closed Cooling Water System Heat Exchangers	Yes
62	Heat Exchangers	Lubricating Oil System Heat Exchangers	Yes
63	Heat Exchangers	Raw Water System Heat Exchangers	Yes
64	Heat Exchangers	Containment Atmospheric System Heat Exchangers	Yes
65	Miscellaneous Process Components	Gland Seal Blower	No
66	Miscellaneous Process Components	Recombiners	The applicant shall identify the intended function and apply the IPA process to determine if the grouping is active or passive.
67	Miscellaneous Process Components	Flexible Connectors	Yes
68	Miscellaneous Process Components	Strainers	Yes
69	Miscellaneous Process Components	Rupture Disks	Yes
70	Miscellaneous Process Components	Steam Traps	Yes
71	Miscellaneous Process Components	Restricting Orifices	Yes
72	Miscellaneous Process Components	Air Compressor	No
73	Electrical and I&C	Alarm Unit (e.g., fire detection devices)	No
74	Electrical and I&C	Analyzers (e.g., gas analyzers, conductivity analyzers)	No
75	Electrical and I&C	Annunciators (e.g., lights, buzzers, alarms)	No
76	Electrical and I&C	Batteries	No

Table 2.1-5. Typical Structures, Components, and Commodity Groups, and 10 CFR 54.21(a)(1)(i) Determinations for Integrated Plant Assessment (continued)

Item	Category	Structure, Component, or Commodity Grouping	Structure, Component, or Commodity Group Meets 10 CFR 54.21(a)(1)(i) (Yes/No)
77	Electrical and I&C	Cables and Connections, Bus, electrical portions of Electrical and I&C Penetration Assemblies (e.g., electrical penetration assembly cables and connections, connectors, electrical splices, fuse holders, terminal blocks, power cables, control cables, instrument cables, insulated cables, communication cables, uninsulated ground conductors, transmission conductors, isolated-phase bus, nonsegregated-phase bus, segregated-phase bus, switchyard bus)	Yes
78	Electrical and I&C	Chargers, Converters, Inverters (e.g., converters-voltage/current, converters-voltage/pneumatic, battery chargers/inverters, motor-generator sets)	No
79	Electrical and I&C	Circuit Breakers (e.g., air circuit breakers, molded case circuit breakers, oil-filled circuit breakers)	No
80	Electrical and I&C	Communication Equipment (e.g., telephones, video or audio recording or playback equipment, intercoms, computer terminals, electronic messaging, radios, transmission line traps and other power-line carrier equipment)	No
81	Electrical and I&C	Electric Heaters	No Yes for a Pressure Boundary if applicable
82	Electrical and I&C	Heat Tracing	No
83	Electrical and I&C	Electrical Controls and Panel Internal Component Assemblies (may include internal devices such as, but not limited to, switches, breakers, indicating lights, etc.) (e.g., main control board, HVAC control board)	No
84	Electrical and I&C	Elements, RTDs, Sensors, Thermocouples, Transducers (e.g., conductivity elements, flow elements, temperature sensors, radiation sensors,watt transducers, thermocouples, RTDs, vibration probes, amp transducers, frequency transducers, power factor transducers, speed transducers, var. transducers, vibration transducers, voltage transducers)	No Yes for a Pressure Boundary if applicable
85	Electrical and I&C	Fuses	No

Table 2.1-5. **Typical Structures, Components, and Commodity Groups, and 10 CFR 54.21(a)(1)(i) Determinations for Integrated Plant Assessment (continued)**

Item	Category	Structure, Component, or Commodity Grouping	Structure, Component, or Commodity Group Meets 10 CFR 54.21(a)(1)(i) (Yes/No)
86	Electrical and I&C	Generators, Motors (e.g., emergency diesel generators, ECCS and emergency service water pump motors, small motors, motor-generator sets, steam turbine generators, combustion turbine generators, fan motors, pump motors, valve motors, air compressor motors)	No
87	Electrical and I&C	High-voltage Insulators (e.g., porcelain switchyard insulators, transmission line insulators)	Yes
88	Electrical and I&C	Surge Arresters (e.g., switchyard surge arresters, lightning arresters, surge suppressers, surge capacitors, protective capacitors)	No
89	Electrical and I&C	Indicators (e.g., differential pressure indicators, pressure indicators, flow indicators, level indicators, speed indicators, temperature indicators, analog indicators, digital indicators, LED bar graph indicators, LCD indicators)	No
90	Electrical and I&C	Isolators (e.g., transformer isolators, optical isolators, isolation relays, isolating transfer diodes)	No
91	Electrical and I&C	Light Bulbs (e.g., indicating lights, emergency lighting, incandescent light bulbs, fluorescent light bulbs)	No
92	Electrical and I&C	Loop Controllers (e.g., differential pressure indicating controllers, flow indicating controllers, temperature controllers, controllers, speed controllers, programmable logic controller, single loop digital controller, process controllers, manual loader, selector station, hand/auto station, auto/manual station)	No
93	Electrical and I&C	Meters (e.g., ammeters, volt meters, frequency meters, var meters, watt meters, power factor meters, watt-hour meters)	No
94	Electrical and I&C	Power Supplies	No
95	Electrical and I&C	Radiation Monitors (e.g., area radiation monitors, process radiation monitors)	No
96	Electrical and I&C	Recorders (e.g., chart recorders, digital recorders, events recorders)	No

Table 2.1-5. Typical Structures, Components, and Commodity Groups, and 10 CFR 54.21(a)(1)(i) Determinations for Integrated Plant Assessment (continued)

Item	Category	Structure, Component, or Commodity Grouping	Structure, Component, or Commodity Group Meets 10 CFR 54.21(a)(1)(i) (Yes/No)
97	Electrical and I&C	Regulators (e.g., voltage regulators)	No
98	Electrical and I&C	Relays (e.g., protective relays, control/logic relays, auxiliary relays)	No
99	Electrical and I&C	Signal Conditioners	No
100	Electrical and I&C	Solenoid Operators	No
101	Electrical and I&C	Solid-State Devices (e.g., transistors, circuit boards, computers)	No
102	Electrical and I&C	Switches (e.g., differential pressure indicating switches, differential pressure switches, pressure indicator switches, pressure switches, flow switches, conductivity switches, level indicating switches, temperature indicating switches, temperature switches, moisture switches, position switches, vibration switches, level switches, control switches, automatic transfer switches, manual transfer switches, manual disconnect switches, current switches, limit switches, knife switches)	No
103	Electrical and I&C	Switchgear, Load Centers, Motor Control Centers, Distribution Panel Internal Component Assemblies (may include internal devices such as, but not limited to, switches, breakers, indicating lights, etc.) (e.g., 4.16 kV switchgear, 480V load centers, 480V motor control centers, 250 VDC motor control centers, 6.9 kV switchgear units, 240/125V power distribution panels)	No
104	Electrical and I&C	Transformers (e.g., instrument transformers, load center transformers, small distribution transformers, large power transformers, isolation transformers, coupling capacitor voltage transformers)	No
105	Electrical and I&C	Transmitters (e.g., differential pressure transmitters, pressure transmitters, flow transmitters, level transmitters, radiation transmitters, static pressure transmitters)	No
106	Valves	Hydraulic Operated Valves	Yes (Bodies)
107	Valves	Explosive Valves	Yes (Bodies)
108	Valves	Manual Valves	Yes (Bodies)
109	Valves	Small Valves	Yes (Bodies)
110	Valves	Motor-Operated Valves	Yes (Bodies)
111	Valves	Air-Operated Valves	Yes (Bodies)

Table 2.1-5. Typical Structures, Components, and Commodity Groups, and 10 CFR 54.21(a)(1)(i) Determinations for Integrated Plant Assessment (continued)

Item	Category	Structure, Component, or Commodity Grouping	Structure, Component, or Commodity Group Meets 10 CFR 54.21(a)(1)(i) (Yes/No)
112	Valves	Main Steam Isolation Valves	Yes (Bodies)
113	Valves	Small Relief Valves	Yes (Bodies)
114	Valves	Check Valves	Yes (Bodies)
115	Valves	Safety Relief Valves	Yes (Bodies)
116	Valves	Dampers	No
117	Tanks	Air Accumulators	Yes
118	Tanks	Discharge Accumulators (Dampers)	Yes
119	Tanks	Boron Acid Storage Tanks	Yes
120	Tanks	Above Ground Oil Tanks	Yes
121	Tanks	Underground Oil Tanks	Yes
122	Tanks	Demineralized Water Tanks	Yes
123	Tanks	Neutron Shield Tank	Yes
124	Fans	Ventilation Fans	No
125	Fans	Other Fans	No
126	Miscellaneous	Emergency Lighting	No
127	Miscellaneous	Hose Stations	Yes

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3.1 AGING MANAGEMENT OF REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

Item	Locator	Comment	Justification
1	3.1.2.2.2.1	Clarify intent of NRC Information Notice (IN) 90-04	Refer to comments for NUREG-1801 Vol. 2 Items IV.D1-16 and IV.D2-10. Discussion does not clearly convey the nature of the degradation mechanism, which is related to a very specific set of conditions, not to most SGs. IN 90-04 indicates that pitting corrosion on the surface served as corrosion fatigue crack initiation sites, not that pitting corrosion resulted in sufficient degradation to cause loss of component function. Further, this degradation mode has been limited to isolated cases of weld-zone cracking in Westinghouse Model 44 and 51 SGs, where a high stress region exists in the area of the shell to transition cone weld.
2	3.1.2.2.4.2	Delete redundant text	Editorial
3	3.1.2.2.4.3	Delete mention of cyclic loading and loss of material	The subject of this item is cracking due to SCC and IGSCC. As such, requirements to augment AMP for cracking due to cyclic loading or for loss of material due to pitting and crevice corrosion do not belong or are addressed elsewhere. Firstly, cyclic loading of isolation condenser components is not an aging effect identified in NUREG-1801 Volume 2. Secondly, NUREG-1801 Vol. 2 Item IV.C1-6 (R-16) addresses LOM. Table 3.1.1, Item 5 (R-16) lists 3.1.2.2.2.2 as the applicable SRP-LR section and not 3.1.2.2.4.3.
4	3.1.2.2.8	Delete "due to line" to correct sentence.	Editorial correction. Either delete words or determine what is missing from the 2nd sentence.
5	3.1.2.2.18	Eliminate specific fluence level	Editorial – fluence level is not required part of the description, neutron flux would be adequate terminology.
6	Table 3.1-1	Odd page footers are for NUREG-1801.	Editorial

7	Table 3.1-1, Table 3.2-1, Table 3.3-1, Table 3.4-1 Note: This comment actually applies to the rollup Tables of GALL Volume 1., which then become the tables in the SRP	When addressing Further Evaluation Recommended issues, avoid mixing internal and external environments for the same component. For example, Table 3.2-1, ID 2 addresses "Steel components (including piping and ducting) exposed to external condensation or outside air, internally or externally to indoor uncontrolled air." Table 3.2-1, ID 11 addresses "Steel piping, piping components, and piping elements (internal surfaces) and ducting closure bolting exposed to condensation (internal, treated water, or air –indoor uncontrolled (external)." In this case, the words "internally or" should be deleted from ID 1 and the word "external" should be replaced with "internal." Of course the related items from NUREG-1801, Volume 2, that deal with external surfaces should point to ID 1; while those dealing with internal surfaces should point to ID 11. If the NUREG-1801, Volume 2, related items are unclear about which surface is being addressed, those items should be clarified.	Typically the plant-specific program applied to internal surfaces is not the same program as applied to external surfaces. Addressing the plant-specific programs to be used would be simplified if separate LRA discussions were permitted for external and internal surfaces. The suggested changes would also eliminate a source of confusion regarding which surfaces are being addressed in GALL roll-up line items.
7	Table 3.1-1	Delete Loose Parts Monitoring and Neutron Noise rows from table.	These programs are not credited in GALL.

3.1 AGING MANAGEMENT OF REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

Review Responsibilities

Primary - Branch assigned responsibility by PM as described in SRP-LR section 3.0

3.1.1 Areas of Review

This review plan section addresses the aging management review (AMR) of the reactor vessel, internals, and reactor coolant system. For a recent vintage plant, the information related to the reactor vessel, internals, and reactor coolant system is contained in Chapter 5, "Reactor Coolant System and Connected Systems," of the plant's final safety analysis report (FSAR), consistent with the Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (NUREG-0800) (Ref. 1). For older plants, the location of applicable information is plant-specific because an older plant's FSAR may have predated NUREG-0800.

The reactor vessel, internals, and reactor coolant system includes the reactor vessel and internals. For BWRs, this system also includes the reactor coolant recirculation system and portions of other systems connected to the pressure vessel extending to the first isolation valve outside of containment or to the first anchor point. These connected systems include residual heat removal, low-pressure core spray, high-pressure core spray, low-pressure coolant injection, high-pressure coolant injection, reactor core isolation cooling, isolation condenser, reactor coolant cleanup, feedwater, and main steam. For PWRs, the reactor coolant system includes the primary coolant loop, the pressurizer and pressurizer relief tank, and the steam generators. The connected systems for PWRs include the residual heat removal or low pressure injection system, core flood spray or safety injection tank, chemical and volume control system or high pressure injection system, and sampling system.

The responsible review organization is to review the following LRA AMR and AMP items assigned to it, per SRP-LR section 3.0, for review:

<u>AMRs</u>

- AMRs consistent with the GALL report, for which further evaluation is not recommended
- AMRs consistent with the GALL report, for which further evaluation is recommended
- AMRs not consistent with the GALL report

<u>AMPs</u>

- AMPs consistent with GALL AMPs (with or without exceptions)
- Plant-specific AMPs

<u>FSAR Supplement</u>

- In addition, the responsible review organization is to review the FSAR supplement associated with each assigned AMP.

3.1.2 Acceptance Criteria

The acceptance criteria for the areas of review describe methods for determining whether the applicant has met the requirements of the NRC's regulations in 10 CFR 54.21.

3.1.2.1 AMR Results Consistent with the GALL Report for Which No Further Evaluation is Recommended

The aging management review and acceptable aging management programs applicable to the reactor vessel, internals, and reactor coolant system are described and evaluated in Chapter IV of the GALL report (Ref. 2).

The applicant's LRA should provide sufficient information so that the NRC reviewer is able to confirm that the specific AMR line-item and the associated AMP are consistent with the cited GALL AMR line-item. The staff reviewer should then confirm that the LRA AMR line-item is consistent with the GALL line-item to which it is compared.

If the applicant identifies an exception to the cited GALL AMP, the LRA should include a basis demonstrating how the criteria of 10 CFR 54.21(a)(3) would still be met. The NRC reviewer should then confirm that the AMP with all exceptions would satisfy the criteria of 10 CFR 54.21(a)(3). If, while reviewing the AMP, the reviewer identifies a difference from the GALL AMP, this difference should be reviewed and dispositioned as if it were an exception identified by the applicant in its LRA. The disposition of all LRA-defined exceptions and staff-identified differences should be documented.

The LRA should identify any enhancements that are needed to permit an existing aging management program to be declared consistent with the GALL AMP to which the LRA AMP is compared. The reviewer is to confirm both that the enhancement, if implemented, would allow the existing plant aging management program to be consistent with the GALL AMP and also that the applicant has a commitment in the FSAR supplement to implement the enhancement prior to the period of extended operation. The reviewer should document the disposition of all enhancements.

3.1.2.2 AMR Results Consistent with the GALL Report for Which Further Evaluation is Recommended

The basic acceptance criteria defined in 3.1.2.1 apply to all of the AMRs and AMPs reviewed as part of this section. In addition, if the GALL AMR line-item to which the LRA AMR line-item is compared identifies that "further evaluation is recommended," then additional criteria apply as identified by the GALL report for each of the following aging effect/aging mechanism combinations.

3.1.2.2.1 Cumulative Fatigue Damage (BWR/PWR)

Fatigue is a time-limited aging analysis (TLAA) as defined in 10 CFR 54.3. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c)(1). The evaluation of this TLAA is addressed separately in Section 4.3.

3.1.2.2.2 Loss of Material due to General, Pitting, and Crevice Corrosion (BWR/PWR)

1. Loss of material due to general, pitting, and crevice corrosion could occur in the steel PWR steam generator shell assembly. The existing program relies on control of chemistry to mitigate corrosion and ISI to detect loss of material. The extent and schedule of the existing steam generator inspections are designed to ensure that flaws cannot attain a depth sufficient to threaten the integrity of the welds. However, according to NRC Information Notice (IN) 90-04 (Ref. 4), the program may not be sufficient to detect pitting and crevice

corrosion, ***if general and pitting corrosion of the shell is known to exist.*** The GALL report recommends augmented inspection to manage this aging effect. ***Furthermore, the GALL Report clarifies that this issue is limited to Westinghouse Model 44 and 51 Steam Generators where a high stress region exists at the shell to transition cone weld.*** Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

2. Loss of material due to pitting and crevice corrosion could occur in stainless steel BWR isolation condenser components. General, pitting, and crevice corrosion could occur in steel BWR isolation condenser components. The existing program relies on control of reactor water chemistry to mitigate corrosion and on ASME Section XI inservice inspection (ISI). However, the existing program should be augmented to detect loss of material due to general, pitting or crevice corrosion. The GALL report recommends an augmented program to include temperature and radioactivity monitoring of the shell-side water, and eddy current testing of tubes to ensure that the component's intended function will be maintained during the period of extended operation. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3.1.2.2.3 Loss of Fracture Toughness due to Neutron Irradiation Embrittlement (BWR/PWR)

1. Neutron irradiation embrittlement is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation for all ferritic materials that have a neutron fluence greater than 10^{17} n/cm2 (E >1 MeV) at the end of the license renewal term. Certain aspects of neutron irradiation embrittlement are TLAAs as defined in 10 CFR 54.3. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c)(1). The evaluation of this TLAA is addressed separately in Section 4.2.

2. Loss of fracture toughness due to neutron irradiation embrittlement could occur in BWR and PWR reactor vessels. A reactor vessel materials surveillance program monitors neutron irradiation embrittlement of the reactor vessel. Reactor vessel surveillance programs are plant specific, depending on matters such as the composition of limiting materials, availability of surveillance capsules, and projected fluence levels. In accordance with 10 CFR Part 50, Appendix H, an applicant is required to submit its proposed withdrawal schedule for approval prior to implementation. Thus, further staff evaluation is required for license renewal. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3. Loss of fracture toughness due to neutron irradiation embrittlement could occur in Westinghouse and B&W baffle/former bolts and screws. The GALL report recommends no further aging management review if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation~~,~~ submit an inspection plan for reactor internals to the NRC for review and approval.

3.1.2.2.4 Cracking due to Stress Corrosion Cracking and Intergranular Stress Corrosion Cracking (BWR)

1. Cracking due to stress corrosion cracking (SCC) and intergranular stress corrosion cracking [IGSCC]) could occur in small-bore steel and stainless steel reactor coolant system and connected system piping less than Nominal Pipe Size (NPS) 4. The existing program relies on ASME Section XI ISI and on control of water chemistry to mitigate SCC. The GALL report recommends that a plant-specific destructive examination or a nondestructive examination (NDE) that permits inspection of the inside surfaces of the piping is to be conducted to ensure that cracking has not occurred and the component intended function will be maintained during the extended period of operation. The AMPs should be augmented by verifying that service-induced weld cracking is not occurring in the small-bore piping less than NPS 4, including pipe, fittings, and branch connections..

2. Cracking due to SCC and IGSCC could occur in the stainless steel and nickel alloy BWR reactor vessel flange leak detection lines. ~~The GALL report recommends~~ The GALL report recommends that a plant-specific aging management program be evaluated because existing programs may not be capable of mitigating or detecting cracking due to SCC. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan)

3. Cracking due to SCC and IGSCC could occur in steel and stainless steel BWR isolation condenser components. The existing program relies on control of reactor water chemistry to mitigate SCC and on ASME Section XI inservice inspection (ISI). However, the existing program should be augmented to detect cracking due to SCC ~~and cyclic loading or loss of material due to pitting and crevice corrosion~~. The GALL report recommends an augmented program to include temperature and radioactivity monitoring of the shell-side water, and eddy current testing of tubes to ensure that the component's intended function will be maintained during the period of extended operation. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3.1.2.2.5 Crack Growth due to Cyclic Loading (PWR)

Crack growth due to cyclic loading could occur in reactor vessel shell forgings clad with stainless steel using a high-heat-input welding process. Growth of intergranular separations (underclad cracks) in the heat affected zone under austenitic stainless steel cladding is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation for all the SA 508-Cl 2 forgings where the cladding was deposited with a high heat input welding process. The methodology for evaluating the underclad flaw should be consistent with the current well-established flaw evaluation procedure and criterion in the ASME Section XI Code. See the Standard Review Plan, Section 4.7, "Other Plant-Specific Time-Limited Aging Analysis," for generic guidance for meeting the requirements of 10 CFR 54.21(c).

3.1.2.2.6 Loss of Fracture Toughness due to Neutron Irradiation Embrittlement and Void Swelling (PWR)

Loss of fracture toughness due to neutron irradiation embrittlement and void swelling could occur in stainless steel, cast austenitic stainless steel, nickel alloy, and PH stainless steel forging reactor vessel internals. The GALL report recommends no further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.).

3.1.2.2.7 Cracking due to Stress Corrosion Cracking (PWR)

1. Cracking due to SCC could occur in the PWR stainless steel reactor vessel flange leak detection lines and bottom-mounted instrument guide tubes. The GALL report recommends further evaluation to ensure that these aging effects are adequately managed. The GALL report recommends that a plant-specific aging management program be evaluated because existing programs may not be capable of mitigating or detecting cracking due to SCC. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

2. Cracking due to SCC could occur in Class 1 PWR cast austenitic stainless steel (CASS) reactor coolant system piping, piping components, and piping elements. The GALL report recommends maintenance of reactor water chemistry in accordance with the guidelines of TR-105714 and further evaluation for piping, piping components, and piping elements that do not meet the material guidelines of NUREG-0313 (Ref. 5). For piping, piping components, and piping elements that do not meet the NUREG-0313 guidelines, the GALL report recommends that the program 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. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3. Cracking due to stress corrosion cracking (SCC) could occur in PWR small-bore stainless steel reactor coolant system and connected system piping less than NPS 4. The existing program relies on ASME Section XI ISI and on control of water chemistry to mitigate SCC. The GALL report recommends that a plant-specific destructive examination or a nondestructive examination (NDE) that permits inspection of the inside surfaces of the piping be conducted to ensure that cracking has not occurred and the component intended function will be maintained during the period of extended operation. The AMPs should be augmented by verifying that cracking is not occurring in the small-bore piping less than NPS 4, including pipe, fittings, and branch connections.

3.1.2.2.8 Cracking due to Cyclic Loading (BWR)

Cracking due to cyclic loading could occur in the stainless steel BWR jet pump sensing lines. The GALL report recommends that a plant specific aging management program be evaluated to mitigate or detect cracking ~~due to line~~. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3.1.2.2.9 Loss of Preload due to Stress Relaxation (PWR)

Loss of preload due to stress relaxation could occur in stainless steel and nickel alloy reactor vessel internals screws, bolts, tie rods, and hold-down springs. The GALL report recommends no further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.

3.1.2.2.10 Loss of Material due to Erosion (PWR)

Loss of material due to erosion could occur in steel steam generator feedwater impingement plates and supports. The GALL report recommends further evaluation of a plant-specific aging management program to ensure that this aging effect is adequately managed. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3.1.2.2.11 Cracking due to Flow-Induced Vibration (BWR)

Cracking due to flow-induced vibration could occur for the BWR stainless steel steam dryers. The GALL report recommends further evaluation of a plant-specific aging management program to ensure that this aging effect is adequately managed. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3.1.2.2.12 Cracking due to Thermal and Mechanical Loading (BWR/PWR)

Cracking due to thermal and mechanical loading could occur in Class 1 small-bore steel (BWR), steel with stainless steel cladding, and stainless steel reactor coolant system and connected system piping less than NPS 4. The existing program relies on ASME Section XI ISI to manage cracking due to thermal and mechanical loading. However, Inservice Inspection for Class 1 components Inspection in accordance with ASME Section XI does not require volumetric examination of pipes less than NPS 4. Therefore, a plant-specific destructive examination or a nondestructive examination (NDE) that permits inspection of the inside surfaces of the piping is to be conducted to ensure that cracking has not occurred and the component intended function will be maintained during the extended period of operation. A one-time inspection of a sample of locations is an acceptable method to ensure that the aging effect is not occurring and the component's intended function will be maintained during the period of extended operation. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3.1.2.2.13 Cracking due to Primary Water Stress Corrosion Cracking (PWR)

1. Cracking due to PWSCC could occur in PWR components made of nickel alloy or having nickel alloy cladding, including reactor coolant pressure boundary components and penetrations inside the RCS such as pressurizer heater sheathes and sleeves, nozzles, and other internal components. The GALL report recommends no further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to implement applicable (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff-accepted industry guidelines

2. Cracking due to PWSCC could occur in nickel alloy of PWR core support pads (or core guide lugs) and pressurizer spray heads. The GALL report recommends no further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to submit a plant-specific AMP delineating commitments to Orders, Bulletins, or Generic Letters that inspect stipulated components for cracking of wetted surfaces.

3.1.2.2.14 Wall Thinning due to Flow-accelerated Corrosion (PWR)

1. Wall thinning due to flow-accelerated corrosion could occur in steel feedwater inlet rings and supports. As noted in Combustion Engineering (CE) Information Notice (IN) 90-04 and NRC IN 91-19 and LER 50-362/90-05-01, this form of degradation has been detected only in certain CE System 80 steam generators. The GALL report recommends further evaluation to ensure that this aging effect is adequately managed. The GALL report recommends that a plant-specific aging management program be evaluated because existing programs may not be capable of mitigating or detecting wall thinning due to flow-accelerated corrosion. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

2. Wall thinning due to flow-accelerated corrosion could occur in the steel tube support lattice bars of PWR steam generators. The GALL report recommends no further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to submit, for NRC review and approval, an inspection plan for tube support lattice bars as based upon staff approved NEI 97-06 guidelines, or other alternative regulatory basis for steam generator degradation management, at least 24 months prior to the extended period.

3.1.2.2.15 Changes in Dimensions due to Void Swelling (PWR)

Changes in dimensions due to void swelling could occur in PWR reactor internal components. The GALL report recommends no further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.

3.1.2.2.16 Cracking due to Stress Corrosion Cracking and Primary Water Stress Corrosion Cracking (PWR)

Cracking due to SCC and PWSCC could occur in PWR stainless steel, cast austenitic stainless steel, nickel alloy, and steel with stainless steel or nickel alloy cladding pressurizer components; steam generator upper and lower heads, tubesheets, and CRD pressure housing components. The GALL report recommends no further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to implement applicable (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff-accepted industry guidelines

3.1.2.2.17 Cracking due to Stress Corrosion Cracking, Primary Water Stress Corrosion Cracking, and Irradiation-Assisted Stress Corrosion Cracking (PWR)

Cracking due to Stress Corrosion Cracking, Primary Water Stress Corrosion Cracking, and Irradiation-Assisted Stress Corrosion Cracking could occur in PWR stainless steel and nickel alloy reactor vessel internals. The existing program relies on control of water chemistry to mitigate these effects. However, the existing program should be augmented to manage these aging effects on the intended function of reactor vessel internals components. The GALL report recommends that no further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.

3.1.2.2.18 Cracking due to Stress Corrosion Cracking and Irradiation-Assisted Stress Corrosion Cracking (PWR)

1. Cracking due to stress corrosion cracking and irradiation-assisted stress corrosion cracking could occur in PWR stainless steel baffle former bolts and screws exposed to ***neutron flux*** ~~high fluence (>1 x 10e21 n/cm2, E >0.1 MeV)~~. The GALL report recommends that no further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.

2. Cracking due to stress corrosion cracking and irradiation-assisted stress corrosion cracking could occur in PWR stainless steel, cast austenitic stainless steel, and PH stainless steel forging reactor vessel internals components. The GALL report recommends no further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval

3.1.2.2.19 Quality Assurance for Aging Management of Nonsafety-Related Components

Acceptance criteria are described in Branch Technical Position IQMB-1 (Appendix A.2 of this standard review plan).

3.1.2.3 AMR Results Not Consistent with or Not Addressed in GALL Report

Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3.1.2.4 FSAR Supplement

The summary description of the programs and activities for managing the effects of aging for the period of extended operation in the FSAR supplement should be appropriate such that later changes can be controlled by 10 CFR 50.59. The description should contain information associated with the bases for determining that aging effects will be managed during the period of extended operation. The description should also contain any future aging management activities, including enhancements and commitments, to be completed before the period of extended operation. Examples of the type of information to be included are provided in Table 3.1-2 of this standard review plan.

3.1.3 Review Procedures

For each area of review, the following review procedures are to be followed.

3.1.3.1 AMR Results Consistent with the GALL Report for which no Further Evaluation is Recommended

The applicant may reference the GALL report in its license renewal application, as appropriate, and demonstrate that the aging management reviews and programs at its facility are consistent with those reviewed and approved in the GALL report. The reviewer should not conduct a re-review of the substance of the matters described in the GALL report. If the applicant has provided the information necessary to adopt the finding of program acceptability as described and evaluated in the GALL report, the staff should find acceptable the applicant's reference to GALL in its license renewal application. In making this determination, the reviewer verifies that the applicant has provided a brief description of the system, components, materials, and environment. The reviewer also confirms that the applicant has stated that the applicable aging effects and industry and plant-specific operating experience have been reviewed by the applicant and are evaluated in the GALL Report.

Furthermore, the reviewer should confirm that the applicant has addressed operating experience identified after the issuance of the GALL report. Performance of this review requires the reviewer to confirm that the applicant has identified those aging effects for the reactor vessel, internals, and reactor coolant system components that are contained in GALL as applicable to its plant.

The reviewer confirms that the applicant has identified the appropriate AMPs as described and evaluated in the GALL report. If the applicant commits to an enhancement to make its aging management program consistent with a GALL AMP, then the reviewer is to confirm that this enhancement when implemented will indeed make the LRA AMP consistent with the GALL AMP. If an aging management program in the LRA identifies an exception to the GALL AMP to which it is claiming to be consistent, the reviewer is to confirm that the LRA AMP with the exception will satisfy the criteria of 10 CFR 54.21(a)(3). If the reviewer identifies a difference, not identified by the LRA, between the LRA AMP and the GALL AMP, with which the LRA claims to be consistent, the reviewer should confirm that the LRA AMP with this difference satisfies 10 CFR 54.21(a)(3). The reviewer should document the basis for accepting enhancements, exceptions or differences. The AMPs evaluated in GALL pertinent to the reactor vessel, internals, and reactor coolant system components are summarized in Table 3.1-1 of this standard review plan. In this table, the ID column provides a row identifier useful in matching the information presented in the corresponding table in the GALL report Vol. 1. The Related Item column identifies the item number in the GALL report Vol. 2, Chapters II through VIII, presenting detailed information summarized by this row.

3.1.3.2 AMR Results Consistent with the GALL Report for Which Further Evaluation is Recommended

The basic review procedures defined in 3.1.3.1 apply to all of the AMRs and AMPs provided in this section. In addition, if the GALL AMR line-item to which the LRA AMR line-item is compared identifies that "further evaluation is recommended," then additional criteria apply as identified by the GALL report for each of the following aging effect/aging mechanism combinations.

3.1.3.2.1 Cumulative Fatigue Damage (BWR/PWR)

Fatigue is a TLAA as defined in 10 CFR 54.3. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c)(1). The staff reviews the evaluation of this TLAA separately following the guidance in Section 4.3 of this standard review plan.

3.1.3.2.2 Loss of Material due to General, Pitting, and Crevice Corrosion (BWR/PWR)

1. The GALL report recommends further evaluation for the management of loss of material due to general, pitting, and crevice corrosion of the steel PWR steam generator shell assembly. The existing program relies on control of reactor water chemistry to mitigate corrosion and on ISI for detection. Based on NRC IN 90-04 (Ref. 4), if general, pitting, and crevice corrosion of the shell exists, the existing program requirements may not be sufficient to detect loss of material due to these effects, and additional inspection procedures may be necessary. The reviewer verifies on a case-by-case basis that the applicant has proposed a program that will manage loss of material due to pitting and crevice corrosion by providing enhanced inspection and supplemental methods to detect loss of material and ensure that the component intended function will be maintained during the period of extended operation.

2. The GALL report recommends an augmented program to include temperature and radioactivity monitoring of the shell-side water and eddy current testing of tubes for the management of loss of material due to general, pitting, and crevice corrosion in BWR isolation condenser components. The existing program relies on control of reactor water chemistry to mitigate corrosion and on ASME Section XI ISI for detection. However, the inspection requirements should be augmented to detect loss of material due to general, pitting, and crevice corrosion, and an augmented program to include temperature and radioactivity monitoring of the shell-side water and eddy current testing of tubes is recommended to ensure that the component's intended function will be maintained during the period of extended operation. The reviewer verifies on a case-by-case basis that the applicant has proposed an augmented program that will manage loss of material due to general, pitting, and crevice corrosion and ensure that the component intended function will be maintained during the period of extended operation.

3.1.3.2.3 Loss of Fracture Toughness due to Neutron Irradiation Embrittlement (BWR/PWR)

1. Neutron irradiation embrittlement is a TLAA as defined in 10 CFR 54.3. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c)(1). The staff reviews the evaluation of this TLAA following the guidance in Section 4.2 of this standard review plan.

2. The GALL report recommends further evaluation of the reactor vessel materials surveillance program for the period of extended operation. Neutron embrittlement of the reactor vessel is monitored by a reactor vessel materials surveillance program. Reactor vessel surveillance programs are plant specific, depending on matters such as the composition of limiting materials, availability of surveillance capsules, and projected fluence levels. In accordance with 10 CFR Part 50, Appendix H, an applicant must submit its proposed withdrawal schedule for approval prior to implementation. Thus, further staff evaluation is required for license renewal. The reviewer verifies on a case-by-case basis that the applicant has proposed an adequate reactor vessel materials surveillance program for the period of extended operation. Specific criteria for an acceptable AMP are provided in chapter XI, Section M31 of the GALL report.

3. The GALL report recommends no further evaluation for the management of loss of fracture toughness due to neutron irradiation embrittlement of Westinghouse and B&W baffle/former bolts and screws if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval. The staff reviews the application to confirm the FSAR supplement contains the appropriate commitment to ensure that an adequate program will be in place for the management of these aging effects. If the FSAR supplement does not contain the appropriate commitment, the staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of this aging effect.

3.1.3.2.4 Cracking due to Stress Corrosion Cracking and Intergranular Stress Corrosion Cracking (BWR)

1. For steel and stainless steel BWR small-bore piping less than NPS 4 in., including pipe, fittings, and branch connections, the GALL report recommends Inservice Inspection for Class 1 components, Water Chemistry for BWR, and a plant-specific destructive examination or a nondestructive examination (NDE) that permits inspection of component inside surfaces for the management of cracking due to SCC and IGSCC. Inspection in accordance with ASME Section XI does not require volumetric examination of pipes less than NPS 4 and the existing program should be augmented by verifying that cracking is not occurring. The staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

2. The GALL report recommends that a plant-specific aging management program is to be evaluated to manage cracking due to SCC and IGSCC in stainless steel and nickel alloy BWR reactor vessel flange leak detection lines. The staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

3. The GALL report recommends an augmented program to include temperature and radioactivity monitoring of the shell-side water, and eddy current testing of tubes for the management of cracking due to SCC, IGSCC and cyclic loading or loss of material due to pitting and crevice corrosion of the steel and stainless steel BWR isolation condenser components. The existing program relies on control of reactor water chemistry to mitigate SCC and on ASME Section XI inservice inspection (ISI) to detect leakage. However, the existing program should be augmented to detect cracking due to SCC, IGSCC and cyclic loading or loss of material due to pitting and crevice corrosion. The staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

3.1.3.2.5 Crack Growth due to Cyclic Loading (PWR)

The GALL report recommends further evaluation of programs to manage crack growth due to cyclic loading in reactor vessel shell. Growth of intergranular separations (underclad cracks) in the heat affected zone under austenitic stainless steel cladding is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation for all the SA 508-Cl 2 forgings where the cladding was deposited with a high heat input welding process. The methodology for evaluating the underclad flaw should be consistent with the current well-established flaw evaluation procedure and criterion in the ASME Section XI Code. The Standard Review Plan, Section 4.7, "Other Plant-Specific Time-Limited Aging Analysis," provides generic guidance for meeting the requirements of 10 CFR 54.21(c). The staff reviews the evaluation of this TLAA separately following the guidance in Section 4.7 of this standard review plan.

3.1.3.2.6 Loss of Fracture Toughness due to Neutron Irradiation Embrittlement and Void Swelling (PWR)

The GALL report recommends no further evaluation of programs to manage loss of fracture toughness due to neutron irradiation embrittlement and void swelling in stainless steel, cast austenitic stainless steel, nickel alloy, and PH stainless steel forging reactor vessel internals if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval. The staff reviews the application to confirm the FSAR supplement contains the appropriate commitment to ensure that an adequate program will be in place for the management of these aging effects. If the FSAR supplement does not contain the appropriate commitment, the staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

3.1.3.2.7 Cracking due to Stress Corrosion Cracking (PWR)

1. The GALL report recommends that a plant-specific aging management program is to be evaluated to manage cracking due to stress corrosion cracking in stainless steel PWR reactor vessel flange leak detection lines and bottom-mounted instrument guide tubes. The staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

2. The GALL report recommends maintenance of reactor water chemistry in accordance with the guidelines of TR-105714 and further evaluation for CASS piping, piping components, and piping elements that do not meet the material guidelines of NUREG-0313 (Ref. 5). For components that do not meet the NUREG-0313 guidelines, the GALL report recommends that the program 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. The staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

3. For stainless steel PWR small-bore piping less than NPS 4 in., including pipe, fittings, and branch connections, the GALL report recommends Inservice Inspection for Class 1 components, maintenance of reactor water chemistry in accordance with the guidelines of TR-102134, Water Chemistry for PWR, and a plant-specific destructive examination or a nondestructive examination (NDE) that permits inspection of component inside surfaces for the management of cracking due to SCC and IGSCC. Inspection in accordance with ASME Section XI does not require volumetric examination of pipes less than NPS 4 and the existing program should be augmented by verifying that cracking is not occurring. The staff

reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

3.1.3.2.8 Cracking due to Cyclic Loading (BWR)

The GALL report recommends that a plant specific aging management program be evaluated for the management of cracking due to cyclic loading in stainless steel BWR jet pump sensing lines. The staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

3.1.3.2.9 Loss of Preload due to Stress Relaxation (PWR)

The GALL report recommends further evaluation of loss of preload due to stress relaxation that could occur in stainless steel and nickel alloy reactor vessel internals screws, bolts, tie rods, and hold-down springs. The GALL report recommends no further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval. The staff reviews the application to confirm the FSAR supplement contains the appropriate commitment to ensure that an adequate program will be in place for the management of these aging effects. If the FSAR supplement does not contain the appropriate commitment, the staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of this aging effect.

3.1.3.2.10 Loss of Material due to Erosion (PWR)

The GALL report recommends further evaluation of a plant-specific aging management program for the management of loss of material due to erosion of steam generator feedwater impingement plates and supports. The staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

3.1.3.2.11 Cracking due to Flow-Induced Vibration (BWR)

The GALL report recommends further evaluation of a plant-specific aging management program for the management of cracking due to flow-induced vibration of BWR stainless steel steam dryers. The staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

3.1.3.2.12 Cracking due to Thermal and Mechanical Loading (BWR/PWR)

The GALL report recommends ASME Section XI ISI and a plant-specific destructive examination or a nondestructive examination (NDE) that permits inspection of the inside surfaces of small-bore (less than NPS 4) Class 1 steel (BWR), steel with stainless steel clad, and stainless steel piping for the management of cracking due to thermal and mechanical loading. Inspection in accordance with ASME Section XI does not require volumetric examination of small-bore reactor coolant system and connected system piping less than NPS 4 and the existing program should be augmented by verifying that service-induced weld cracking is not occurring in the piping, fittings, and branch connections. The GALL report recommends that the inspection include a representative sample of the system population, and, where practical and prudent, focus on the bounding or lead components most susceptible to aging due to time in service, severity of operating conditions, and lowest design margin. Actual inspection locations should be based on physical accessibility, exposure levels, NDE examination techniques, and locations identified in Nuclear Regulatory Commission (NRC) Information Notice (IN) 97-46. Combinations of NDE, including visual, ultrasonic, and surface techniques, should be performed by qualified personnel following procedures consistent with the ASME Code and 10 CFR Part 50 Appendix B. Follow-up of unacceptable inspection findings should include expansion of the inspection sample size and locations. The staff reviews the program to confirm that it includes measures to determine if unacceptable degradation is occurring. If an applicant proposes a one-time inspection of select components and susceptible locations to ensure that cracking is not occurring, the reviewer verifies that the proposed inspection will be performed using techniques similar to ASME Code and ASTM standards including visual, ultrasonic, and surface techniques (Refs. 6 and 7) to ensure that the component's intended function will be maintained during the period of extended operation.

3.1.3.2.13 Cracking due to Primary Water Stress Corrosion Cracking (PWR)

1. The GALL report recommends no further aging management review is necessary to manage cracking due to PWSCC of PWR components made of nickel alloy or having nickel alloy cladding, including reactor coolant pressure boundary components and penetrations inside the RCS such as pressurizer heater sheathes and sleeves, nozzles, and other internal components, if the applicant provides a commitment in the FSAR supplement to implement applicable (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff-accepted industry guidelines

   The staff reviews the application to confirm the FSAR supplement contains the appropriate commitment to ensure that an adequate program will be in place for the management of PWSCC. If the FSAR supplement does not contain the appropriate commitment, the staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of PWSCC.

2. The GALL report recommends no further aging management review is necessary to manage cracking due to PWSCC of PWR core support pads (or core guide lugs) and pressurizer spray heads if the applicant provides a commitment in the FSAR supplement to implement applicable (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff-accepted industry guidelines

   The staff reviews the application to confirm the FSAR supplement contains the appropriate commitment to ensure that an adequate program will be in place for the management of PWSCC. If the FSAR supplement does not contain the appropriate commitment, the staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of PWSCC.

3.1.3.2.14 Wall Thinning due to Flow-accelerated Corrosion (PWR)

1. The GALL report recommends that a plant-specific aging management program be evaluated to manage loss of material due to wall thinning in the feedwater inlet ring and supports. As noted in Combustion Engineering (CE) Information Notice (IN) 90-04 and NRC IN 91-19 and LER 50-362/90-05-01, this form of degradation has been detected only in

certain CE System 80 steam generators. The staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

2. The GALL report recommends no further aging management review is necessary to manage wall thinning due to flow-accelerated corrosion in the steel tube support lattice bars of PWR steam generators if the applicant provides a commitment in the FSAR supplement to submit, for NRC review and approval, an inspection plan for tube support lattice bars as based upon staff approved NEI 97-06 guidelines, or other alternative regulatory basis for steam generator degradation management, at least 24 months prior to the extended period.

   The staff reviews the application to confirm the FSAR supplement contains the appropriate commitment to ensure that an adequate program will be in place for the management of this aging effect. If the FSAR supplement does not contain the appropriate commitment, the staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of this aging effect.

3.1.3.2.15 Changes in Dimensions due to Void Swelling (PWR)

The GALL report recommends no further aging management review is necessary to manage changes in dimensions due to void swelling in reactor internal components if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.

The staff reviews the application to confirm the FSAR supplement contains the appropriate commitment to ensure that an adequate program will be in place for the management of this aging effect. If the FSAR supplement does not contain the appropriate commitment, the staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of this aging effect.

3.1.3.2.16 Cracking due to Stress Corrosion Cracking and Primary Water Stress Corrosion Cracking (PWR)

The GALL report recommends no further aging management review is necessary to manage cracking due to SCC and PWSCC of PWR stainless steel, cast austenitic stainless steel, nickel alloy, and steel with stainless steel or nickel alloy cladding pressurizer components; steam generator upper and lower heads, tubesheets, and CRD pressure housing components if the applicant provides a commitment in the FSAR supplement to implement applicable (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff-accepted industry guidelines

The staff reviews the application to confirm the FSAR supplement contains the appropriate commitment to ensure that an adequate program will be in place for the management of PWSCC. If the FSAR supplement does not contain the appropriate commitment, the staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of SCC and PWSCC.

3.1.3.2.17 Cracking due to Stress Corrosion Cracking, Primary Water Stress Corrosion Cracking, and Irradiation-Assisted Stress Corrosion Cracking (PWR)

The GALL report recommends further evaluation of programs to manage cracking due to Stress Corrosion Cracking, Primary Water Stress Corrosion Cracking, and Irradiation-Assisted Stress Corrosion Cracking in stainless steel and nickel alloy reactor vessel internals. The existing program relies on control of water chemistry to mitigate these effects. However, the existing program should be augmented to manage these aging effects on the intended function of reactor vessel internals components. The GALL report recommends that no further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.

The staff reviews the application to confirm the FSAR supplement contains the appropriate commitment to ensure that an adequate program will be in place for the management of these aging effects. If the FSAR supplement does not contain the appropriate commitment, the staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

3.1.3.2.18 Cracking due to Stress Corrosion Cracking and Irradiation-Assisted Stress Corrosion Cracking (PWR)

1. The GALL report recommends further evaluation of programs to manage cracking due to stress corrosion cracking and irradiation-assisted stress corrosion cracking in stainless steel baffle former bolts and screws exposed to high fluence (>1 x 10e21 n/cm2, E >0.1 MeV). The GALL report recommends that no further aging management review is necessary if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.

   The staff reviews the application to confirm the FSAR supplement contains the appropriate commitment to ensure that an adequate program will be in place for the management of these aging effects. If the FSAR supplement does not contain the appropriate commitment, the staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

2. The GALL report recommends no further aging management review is necessary to manage cracking due to stress corrosion cracking and irradiation-assisted stress corrosion cracking in stainless steel, cast austenitic stainless steel, and PH stainless steel forging reactor vessel internals components if the applicant provides a commitment in the FSAR supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.

The staff reviews the application to confirm the FSAR supplement contains the appropriate commitment to ensure that an adequate program will be in place for the management of these aging effects. If the FSAR supplement does not contain the appropriate commitment, the staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

3.1.3.2.19 Quality Assurance for Aging Management of Nonsafety-Related Components

The applicant's aging management programs for license renewal should contain the elements of corrective actions, the confirmation process, and administrative controls. Safety-related components are covered by 10 CFR Part 50 Appendix B, which is adequate to address these program elements. However, Appendix B does not apply to nonsafety-related components that are subject to an AMR for license renewal. Nevertheless, the applicant has the option to expand the scope of its 10 CFR Part 50 Appendix B program to include these components and address the associated program elements. If the applicant chooses this option, the reviewer verifies that the applicant has documented such a commitment in the FSAR supplement. If the applicant chooses alternative means, the branch responsible for quality assurance should be requested to review the applicant's proposal on a case-by-case basis.

3.1.3.3 AMR Results Not Consistent with or Not Addressed in GALL Report

The reviewer should confirm that the applicant, in the license renewal application, has identified applicable aging effects, listed the appropriate combination of materials and environments, and aging management programs that will adequately manage the aging effects. The aging management program credited could be an AMP that is described and evaluated in the GALL report or a plant-specific program. Review procedures are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3.1.3.4 FSAR Supplement

The reviewer confirms that the applicant has provided information equivalent to that in Table 3.1-2 in the FSAR supplement for aging management of the reactor vessel, internals, and reactor coolant system for license renewal. The reviewer also confirms that the applicant has provided information equivalent to that in Table 3.1-2 in the FSAR supplement for Subsection 3.1.3.3, "Aging Management Evaluations that Are Different from or Not Addressed in the GALL Report."

The staff expects to impose a license condition on any renewed license to require the applicant to update its FSAR to include this FSAR supplement at the next update required pursuant to 10 CFR 50.71(e)(4). As part of the license conditions until the FSAR update is complete, the applicant may make changes to the programs described in its FSAR supplement without prior NRC approval, provided that the applicant evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59.

As noted in Table 3.1-2, an applicant need not incorporate the implementation schedule into its FSAR. However, the reviewer should confirm that the applicant has identified and committed in the license renewal application to any future aging management activities, including enhancements and commitments to be completed before the period of extended operation. The staff expects to impose a license condition on any renewed license to ensure that the applicant will complete these activities no later than the committed date.

3.1.4 Evaluation Findings

If the reviewer has confirmed that the applicant has provided information sufficient to satisfy the provisions of this review plan section, an evaluation finding similar to the following text should be included in the staff's safety evaluation report:

> On the basis of its review, the staff concludes that the applicant has adequately identified the aging effects and the AMPs credited with managing these aging effects for the reactor vessel, internals and reactor coolant system, such that there is reasonable assurance that the component intended functions will be maintained consistent with the CLB during the period of extended operation. The staff also reviewed the applicable FSAR supplement program descriptions and concludes that the FSAR supplement provides an adequate program description of the AMPs credited for managing aging effects, as required by 10 CFR 54.21(d).

3.1.5 Implementation

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the NRC's regulations, the method described herein will be used by the staff in its evaluation of conformance with NRC regulations.

3.1.6 References

1. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, July 1981.

2. Draft NUREG-1801, "Generic Aging Lessons Learned (GALL)," U.S. Nuclear Regulatory Commission, Revision 1, September 2005.

3. NEI 97-06, "Steam Generator Program Guidelines," Nuclear Energy Institute, December 1997.

4. NRC Information Notice 90-04, "Cracking of the Upper Shell-to-Transition Cone Girth Welds in Steam Generators," U.S. Nuclear Regulatory Commission, January 26, 1990.

5. NUREG-0313, Rev. 2, "Technical Report on Material Selection and Processing Guidelines for BRW Coolant Pressure Boundary Piping," U.S. Nuclear Regulatory Commission, January 1988.

6. EPRI TR-107569-V1R5, "PWR Steam Generator Examination Guidelines, Rev. 5," Electric Power Research Institute, September 1997.

7. NRC Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes," U.S. Nuclear Regulatory Commission, June 1974.

8. NRC Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes (for Comment)," U.S. Nuclear Regulatory Commission, May 1976.

9. NRC Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," U.S. Nuclear Regulatory Commission, August 3, 1995.

10. NRC Information Notice 90-10, "Primary Water Stress Corrosion Cracking (PWSCC) of Inconel 600," U.S. Nuclear Regulatory Commission, February 23, 1990.

11. NRC Information Notice 90-30, "Ultrasonic Inspection Techniques for Dissimilar Metal Welds," U.S. Nuclear Regulatory Commission, May 1, 1990.

12. NRC Generic Letter 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning," May 2, 1989.

13. NSAC-202L-R2, "Recommendations for an Effective Flow-accelerated Corrosion Program," Electric Power Research Institute, April 1999.

14. NRC Information Notice 96-11, "Ingress of Demineralizer Resins Increase Potential for Stress Corrosion Cracking of Control Rod Drive Mechanism Penetrations," February 14, 1996.

15. NRC Generic Letter 97-06, "Degradation of Steam Generator Internals," U.S. Nuclear Regulatory Commission, December 30, 1997.

16. BWRVIP-29 (EPRI TR-103515), *BWR Water Chemistry Guidelines-Revision 3, Normal and Hydrogen Water Chemistry*, Electric Power Research Institute, Palo Alto, CA, February 1994.

17. 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.

18. EPRI TR-105714, *PWR primary Water Chemistry Guidelines-Revision 3*, Electric Power Research Institute, Palo Alto, CA, Nov. 1995.

19. NRC Generic Letter 88-01, *NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping*, January 25, 1988.

20. NRC Generic Letter 97-01, *Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations*, April 1, 1997.

21. NRC Information Notice 97-46, *Unisolable Crack in High-Pressure Injection Piping*, July 9, 1997.

22. NRC Regulatory Guide 1.99, Rev. 2, *Radiation Embrittlement of Reactor Vessel Materials,* May 1988.

23. NUREG-0619, *BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking*, U.S. Nuclear Regulatory Commission, November 1980.

24. 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.

25. EPRI TR-104213, *Bolted Joint Maintenance & Application Guide,* Electric Power Research Institute, Palo Alto, CA, December 1995.

26. NEI letter dated Dec. 11, 1998, Dave Modeen to Gus Lainas, "Responses to NRC Requests for Additional Information (RAIs) on GL 97-01."

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

Table 3.1-1. Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report

ID	Type	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Related Item
1	BWR/ PWR	Reactor coolant pressure boundary closure bolting, head closure studs, support skirts and attachment welds, pressurizer relief tank components, steam generator components, and reactor vessel internals	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA (See subsection 3.1.2.2.1)	R-13 R-18 R-33 R-46 R-53 R-54 R-70 R-73 R-91
2	BWR/ PWR	Reactor coolant pressure boundary components, steam generator tubes and sleeves, reactor vessel internals, pressurizer components	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c) and environmental effects are to be addressed for Class 1 components	Yes, TLAA (See subsection 3.1.2.2.1)	R-04 R-189 R-45
3	BWR/ PWR	Pump and valve closure bolting	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c) check Code limits for allowable cycles (less than 7000 cycles) of thermal stress range	Yes, TLAA (See subsection 3.1.2.2.1)	R-28
4	PWR	Steel steam generator shell assembly	Loss of material due to general, pitting and crevice corrosion	Inservice inspection and water chemistry	Yes, detection of aging effects is to be evaluated (See subsection 3.1.2.2.2.1)	R-34
5	BWR	Stainless steel; steel isolation condenser tube side components exposed to reactor coolant	Loss of material due to general (steel only), pitting and crevice corrosion	Inservice inspection, water chemistry, and plant-specific verification program	Yes, detection of aging effects is to be evaluated (See subsection 3.1.2.2.2.2)	R-16

Table 3.1-1. Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report						
ID	**Type**	**Component**	**Aging Effect/Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Related Item**
6	BWR/ PWR	Reactor vessel beltline shell, nozzles, and welds	Loss of fracture toughness due to neutron irradiation embrittlement	TLAA, evaluated in accordance with Appendix G of 10 CFR 50 and RG 1.99. The applicant may choose to demonstrate that the materials of the nozzles are not controlling for the TLAA evaluations.	Yes, TLAA (See subsection 3.1.2.2.3.1)	R-62 R-67 R-81 R-84
7	BWR/ PWR	Reactor vessel beltline shell, nozzles, and welds; safety injection nozzles	Loss of fracture toughness due to neutron irradiation embrittlement	Reactor vessel surveillance	Yes, plant specific (See subsection 3.1.2.2.3.2)	R-63 R-82 R-86
8	PWR	Westinghouse stainless steel baffle former bolts	Loss of fracture toughness due to neutron irradiation embrittlement	FSAR supplement commitment to (1) participate in industry RVI aging programs (2) implement applicable results (3) submit for NRC approval > 24 months before the extended period an RVI inspection plan based on industry recommendation.	No, but licensee commitment to be confirmed (See subsection 3.1.2.2.3.3)	R-128
9	BWR	Steel and stainless steel Class 1 piping, fittings and branch connections < NPS 4 exposed to reactor coolant	Cracking due to Stress corrosion cracking and intergranular stress corrosion cracking	Inservice Inspection, Water chemistry, and a plant specific examination	Yes, parameters monitored/inspected and detection of aging effects are to be evaluated (See subsection 3.1.2.2.4.1)	R-03

Table 3.1-1. Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report						
ID	**Type**	**Component**	**Aging Effect/Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Related Item**
10	BWR	Stainless steel and nickel alloy reactor vessel flange leak detection line	Cracking due to Stress corrosion cracking and intergranular stress corrosion cracking	A plant-specific aging management program is to be evaluated because existing programs may not be capable of mitigating or detecting crack initiation and growth due to SCC in the vessel flange leak detection line.	Yes, plant specific (See subsection 3.1.2.2.4.2)	R-61
11	BWR	Stainless steel; steel isolation condenser tube side components exposed to reactor coolant	Cracking due to Stress corrosion cracking and intergranular stress corrosion cracking	Inservice inspection, water chemistry, and plant-specific verification program	Yes, detection of aging effects is to be evaluated (See subsection 3.1.2.2.4.3)	R-15
12	PWR	Reactor vessel shell fabricated of SA508-Cl 2 forgings clad with stainless steel using a high-heat-input welding process	Crack growth due to cyclic loading	TLAA	Yes, TLAA (See subsection 3.1.2.2.5)	R-85

Table 3.1-1. Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report						
ID	**Type**	**Component**	**Aging Effect/Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Related Item**
13	PWR	Stainless steel, cast austenitic stainless steel, nickel alloy, and PH stainless steel forging reactor vessel internals components (except Westinghouse baffle former bolts) exposed to reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement, void swelling	FSAR supplement commitment to (1) participate in industry RVI aging programs (2) implement applicable results (3) submit for NRC approval > 24 months before the extended period an RVI inspection plan based on industry recommendation.	No, but licensee commitment to be confirmed (See subsection 3.1.2.2.6)	R-122 R-127 R-132 R-135 R-141 R-157 R-161 R-164 R-169 R-178 R-188 R-196 R-200 R-205 R-212 R-216
14	PWR	Stainless steel reactor vessel flange leak detection line and bottom-mounted instrument guide tubes	Cracking due to stress corrosion cracking	Plant specific	Yes, plant specific (See subsection 3.1.2.2.7.1)	R-74 RP-13
15	PWR	Class 1 cast austenitic stainless steel (CASS) piping, piping components, and piping elements	Cracking due to stress corrosion cracking	Water chemistry (and plant specific for components that do not meet the material guidelines of NUREG-0313).	Yes, plant specific for components that do not meet the material guidelines of NUREG-0313 (See subsection 3.1.2.2.7.2)	R-05

Table 3.1-1. Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report						
ID	**Type**	**Component**	**Aging Effect/Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Related Item**
16	PWR	Stainless steel Class 1 piping, fittings and branch connections < NPS 4 exposed to reactor coolant	Cracking due to stress corrosion cracking	Inservice Inspection, Water chemistry, and a plant specific examination	Yes, parameters monitored/inspected and detection of aging effects are to be evaluated (See subsection 3.1.2.2.7.3)	R-02
17	BWR	Stainless steel jet pump sensing line	Cracking due to cyclic loading	Plant specific	Yes, plant specific (See subsection 3.1.2.2.8)	R-102
18	PWR	Stainless steel, nickel alloy reactor vessel internals screws, bolts, tie rods, and hold-down springs	Loss of preload due to stress relaxation	FSAR supplement commitment to (1) participate in industry RVI aging programs (2) implement applicable results (3) submit for NRC approval > 24 months before the extended period an RVI inspection plan based on industry recommendation.	No, but licensee commitment to be confirmed (See subsection 3.1.2.2.9)	R-108 R-114 R-129 R-136 R-137 R-154 R-165 R-184 R-192 R-197 R-201 R-207 R-213
19	PWR	Steel steam generator feedwater impingement plate and support	Loss of material due to erosion	Plant specific	Yes, plant specific (See subsection 3.1.2.2.10)	R-39
20	BWR	Stainless steel steam dryers exposed to reactor coolant	Cracking due to flow-induced vibration	Plant specific	Yes, plant specific (See subsection 3.1.2.2.11)	RP-18

Table 3.1-1. Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report

ID	Type	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Related Item
21	BWR/ PWR	BWR steel and stainless steel Class 1 piping, fittings and branch connections < NPS 4 exposed to reactor coolant; PWR stainless steel and steel with stainless steel cladding Class 1 piping, fittings and branch connections < NPS 4	Cracking due to thermal and mechanical loading	Inservice Inspection and a plant specific examination (one-time inspection)	Yes, parameters monitored/inspected and detection of aging effects are to be evaluated (See subsection 3.1.2.2.12)	R-55 R-57
22	PWR	Nickel alloy reactor coolant pressure boundary penetrations; primary side nozzles and safe ends, pressurizer steam space nozzles, heater sheaths and sleeves, heater bundle diaphragm plate, and manways and flanges; steam generator divider plate	Cracking due to primary water stress corrosion cracking	Inservice Inspection and Water Chemistry, and for Alloy 600, FSAR supplement commitment to implement applicable plant commitments to (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff-accepted industry guidelines.	No, but licensee commitments to be confirmed (See subsection 3.1.2.2.13.1)	R-01 R-06 R-218 R-75 R-89 R-90 RP-21 RP-22
23	PWR	Nickel alloy, cast austenitic stainless steel, stainless steel pressurizer spray head, core support pads/core guide lugs	Cracking due to primary water stress corrosion cracking	Water Chemistry and One-Time Inspection or Inservice Inspection and provide commitment in FSAR supplement to submit AMP delineating commitments to Orders, Bulletins, or Generic Letters that inspect stipulated components for cracking of wetted surfaces.	No, unless licensee commitments need to be confirmed (See subsection 3.1.2.2.13.2)	R-24 R-88

Table 3.1-1. Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report						
ID	**Type**	**Component**	**Aging Effect/Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Related Item**
24	PWR (CE)	Steel steam generator feedwater inlet ring and supports	Wall thinning due to flow-accelerated corrosion	Combustion Engineering (CE) System 80 steam generator feedwater ring inspection	Yes, plant specific (See subsection 3.1.2.2.14.1)	R-51
25	PWR	Steel tube support lattice bars exposed to secondary feedwater/ steam	Wall thinning due to flow-accelerated corrosion	FSAR supplement commitment to submit an inspection plan for tube support lattice bars for NRC review and approval, at least 24 months prior to the extended period.	No, but licensee commitment to be confirmed (See subsection 3.1.2.2.14.2)	R-41

Table 3.1-1. Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report

ID	Type	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Related Item
26	PWR	Stainless steel, cast austenitic stainless steel, nickel alloy, PH stainless steel forging reactor vessel internals components (except Westinghouse baffle former bolts)	Changes in dimensions due to void swelling	FSAR supplement commitment to (1) participate in industry RVI aging programs (2) implement applicable results (3) submit for NRC approval > 24 months before the extended period an RVI inspection plan based on industry recommendation.	No, but licensee commitment to be confirmed (See subsection 3.1.2.2.15)	R-107 R-110 R-113 R-117 R-119 R-121 R-124 R-126 R-131 R-134 R-139 R-144 R-147 R-151 R-158 R-160 R-163 R-168 R-174 R-177 R-182 R-187 R-195 R-199 R-204 R-211 R-215

Table 3.1-1. Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report

ID	Type	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Related Item
27	PWR	Stainless steel, cast austenitic stainless steel, nickel alloy, steel with stainless steel or nickel alloy cladding pressurizer components; steam generator upper and lower heads, tubesheets; CRD pressure housing	Cracking due to stress corrosion cracking, primary water stress corrosion cracking	Inservice Inspection and Water Chemistry, and for Alloy 600, FSAR supplement commitment to implement applicable plant commitments to (1) NRC Orders, Bulletins and Generic Letters associated with nickel alloys and (2) staff-accepted industry guidelines.	No, but licensee commitments to be confirmed (See subsection 3.1.2.2.16)	R-25 R-35 R-76
28	PWR	Stainless steel and nickel alloy reactor vessel internals (except Westinghouse baffle former bolts)	Cracking due to stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking	Water Chemistry and FSAR supplement commitment to (1) participate in industry RVI aging programs (2) implement applicable results (3) submit for NRC approval > 24 months before the extended period an RVI inspection plan based on industry recommendation.	No, but licensee commitment to be confirmed (See subsection 3.1.2.2.17)	R-112 R-118 R-133 R-150 R-162 R-167 R-186 R-194 R-203 R-210
29	PWR	Westinghouse stainless steel baffle former bolts and screws exposed to high fluence (>1 x 10e21 n/cm2, E >0.1 MeV)	Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking	FSAR supplement commitment to (1) participate in industry RVI aging programs (2) implement applicable results (3) submit for NRC approval > 24 months before the extended period an RVI inspection plan based on industry recommendation.	No, but licensee commitment to be confirmed (See subsection 3.1.2.2.18.1)	R-125

Table 3.1-1. Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report

ID	Type	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Related Item
30	PWR	Stainless steel, cast austenitic stainless steel PH stainless steel forging reactor vessel internals components (except Westinghouse baffle former bolts)	Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking	Water Chemistry and FSAR supplement commitment to (1) participate in industry RVI aging programs (2) implement applicable results (3) submit for NRC approval > 24 months before the extended period an RVI inspection plan based on industry recommendation.	No, but licensee commitment to be confirmed (See subsection 3.1.2.2.18.2)	R-106 R-109 R-116 R-120 R-123 R-130 R-138 R-143 R-146 R-149 R-155 R-159 R-166 R-172 R-173 R-175 R-176 R-180 R-181 R-185 R-193 R-202 R-209 R-214
31	BWR	Stainless steel and nickel alloy penetrations for control rod drive stub tubes instrumentation, jet pump instrument, standby liquid control, flux monitor, and drain line exposed to reactor coolant	Cracking due to stress corrosion cracking, Intergranular stress corrosion cracking, cyclic loading	BWR penetrations and water chemistry	No	R-69

Table 3.1-1. Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report

ID	Type	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Related Item
32	BWR	Stainless steel, cast austenitic stainless steel, and nickel alloy piping, piping components, and piping elements greater than or equal to 4 NPS; nozzle safe ends and associated welds	Cracking due to Stress corrosion cracking and intergranular stress corrosion cracking	BWR Stress Corrosion Cracking and Water Chemistry	No	R-20 R-21 R-22 R-68
33	BWR	Stainless steel, nickel alloy vessel shell attachment welds exposed to reactor coolant	Cracking due to Stress corrosion cracking and intergranular stress corrosion cracking	BWR vessel ID attachment welds and water chemistry	No	R-64
34	BWR	Stainless steel fuel supports and control rod drive assemblies control rod drive housing exposed to reactor coolant	Cracking due to Stress corrosion cracking and intergranular stress corrosion cracking	BWR vessel internals and water chemistry	No	R-104
35	BWR	Stainless steel, cast austenitic stainless steel, nickel alloy core shroud, core plate, core plate bolts, support structure, top guide, core spray lines, spargers, jet pump assemblies, control rod drive housing, nuclear instrumentation guide tubes	Cracking due to stress corrosion cracking, intergranular stress corrosion cracking, irradiation-assisted stress corrosion cracking	BWR vessel internals and water chemistry	No	R-100 R-105 R-92 R-93 R-96 R-97 R-98 R-99
36	BWR	Steel (with or without stainless steel cladding) control rod drive return line nozzles exposed to reactor coolant	Cracking due to cyclic loading	CRD return line nozzle	No	R-66
37	BWR	Steel (with or without stainless steel cladding) feedwater nozzles exposed to reactor coolant	Cracking due to cyclic loading	Feedwater nozzle	No	R-65
38	BWR	Steel piping, piping components, and piping elements exposed to reactor coolant	Wall thinning due to flow-accelerated corrosion	Flow-accelerated corrosion	No	R-23

Table 3.1-1. Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report						
ID	**Type**	**Component**	**Aging Effect/Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Related Item**
39	BWR	Nickel alloy core shroud and core plate access hole cover (welded and mechanical covers)	Cracking due to stress corrosion cracking, intergranular stress corrosion cracking, irradiation-assisted stress corrosion cracking	Inservice inspection and water chemistry	No	R-95
40	BWR	Steel top head enclosure (without cladding) top head nozzles (vent, top head spray or RCIC, and spare) exposed to reactor coolant	Loss of material due to general, pitting and crevice corrosion	Inservice inspection and water chemistry	No	R-59
41	BWR	Nickel alloy core shroud and core plate access hole cover (welded and mechanical covers)	Cracking due to stress corrosion cracking, intergranular stress corrosion cracking, irradiation-assisted stress corrosion cracking	Inservice inspection, water chemistry, and augmented inspection of the access hole cover welds	No	R-94
42	BWR	High-strength low alloy steel top head closure studs and nuts exposed to air with reactor coolant leakage	Cracking due to Stress corrosion cracking and intergranular stress corrosion cracking	Reactor head closure studs	No	R-60
43	BWR	Jet pump assembly castings; orificed fuel support	Loss of fracture toughness due to thermal aging and neutron irradiation embrittlement	Thermal aging and neutron irradiation embrittlement	No	R-101 R-103

Table 3.1-1. Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report

ID	Type	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Related Item
44	BWR/ PWR	Reactor coolant pressure boundary (RCPB) valve closure bolting, manway and holding bolting, and closure bolting in high-pressure and high-temperature systems	Cracking due to stress corrosion cracking, loss of material due to wear, loss of preload due to stress relaxation	Bolting Integrity	No	R-10 R-11 R-12 R-26 R-27 R-29 R-32 R-78 R-79 R-80
45	BWR/ PWR	Copper alloy piping, piping components, and piping elements exposed to closed cycle cooling water	Loss of material due to pitting, crevice, and galvanic corrosion	Closed-cycle cooling water system and One-Time Inspection	No	RP-11
46	BWR/ PWR	Steel piping, piping components, and piping elements exposed to closed cycle cooling water	Loss of material due to general, pitting and crevice corrosion	Closed-cycle cooling water system and One-Time Inspection	No	RP-10
47	BWR/ PWR	Cast austenitic stainless steel Class 1 pump casings, and valve bodies and bonnets exposed to reactor coolant >250°C (>482°F)	Loss of fracture toughness due to thermal aging embrittlement	Inservice inspection. Thermal aging susceptibility screening is not necessary, inservice inspection requirements are sufficient for managing these aging effects. ASME Code Case N-481 also provides an alternative for pump casings.	No	R-08
48	BWR/ PWR	Copper alloy >15% Zn piping, piping components, and piping elements exposed to closed cycle cooling water	Loss of material due to selective leaching	Selective Leaching of Materials	No	RP-12
49	BWR/ PWR	Cast austenitic stainless steel piping and CRD pressure housings	Loss of fracture toughness due to thermal aging embrittlement	Thermal aging embrittlement of CASS	No	R-52 R-77

Table 3.1-1. Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report

ID	Type	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Related Item
50	PWR	Steel reactor coolant pressure boundary external surfaces	Loss of material due to boric acid corrosion	Boric acid corrosion	No	R-17
51	PWR	Steel steam generator steam nozzle and safe end, feedwater nozzle and safe end, AFW nozzles and safe ends exposed to secondary feedwater/steam	Wall thinning due to flow-accelerated corrosion	Flow-accelerated corrosion	No	R-37 R-38
52	PWR	Stainless steel and nickel alloy reactor vessel internals exposed to reactor coolant	Loss of material due to wear	Inservice inspection	No	R-115 R-142 R-148 R-152 R-156 R-170 R-179 R-190 R-208 R-87
53	PWR	Stainless steel, steel pressurizer integral support exposed to air with metal temperature up to 288°C (550°F)	Cracking due to cyclic loading	Inservice inspection	No	R-19
54	PWR	Stainless steel, steel with stainless steel cladding reactor coolant system cold leg, hot leg, surge line, and spray line piping and fittings exposed to reactor coolant	Cracking due to cyclic loading	Inservice inspection	No	R-56
55	PWR	Steel steam generator secondary manways and handholds	Loss of material due to erosion	Inservice inspection	No	R-31
56	PWR	Stainless steel flux thimble tubes	Loss of material due to wear	Inservice inspection and recommendations of NRC IEB 88-09	No	R-145

Table 3.1-1. Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report

ID	Type	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Related Item
57	PWR	Stainless steel, cast austenitic stainless steel, nickel alloy and associated welds and buttering nozzle safe ends inlet outlet safety injection exposed to reactor coolant	Cracking due to stress corrosion cracking, primary water stress corrosion cracking	Inservice inspection and water chemistry	No	R-83
58	PWR	Stainless steel Class 1 piping, fittings, primary nozzles, safe ends, manways, flanges, CRD housing; pressurizer heater sheaths, sleeves, heater bundle diaphragm plate; pressurizer relief tank components; steam generator divider plate	Cracking due to stress corrosion cracking	Inservice inspection and water chemistry	No	R-07 R-14 R-217 R-30 RP-17
59	PWR	Steel with stainless steel or nickel alloy cladding; or stainless steel pressurizer components exposed to reactor coolant	Cracking due to cyclic loading	Inservice inspection and water chemistry	No	R-58
60	PWR	High-strength low alloy steel closure head stud assembly exposed to air with reactor coolant leakage	Cracking due to stress corrosion cracking; loss of material due to wear	Reactor head closure studs	No	R-71 R-72
61	PWR	Chrome plated nickel alloy, and stainless steel, steam generator anti-vibration bars exposed to secondary feedwater/ steam	Cracking due to stress corrosion cracking, loss of material due to crevice corrosion and fretting	Steam generator tubing integrity and water chemistry	No	RP-14 RP-15
62	PWR	Nickel alloy steam generator tubes, repair sleeves, and tube plugs exposed to reactor coolant	Cracking due to primary water stress corrosion cracking	Steam generator tubing integrity and water chemistry	No	R-40 R-44

Table 3.1-1. Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report

ID	Type	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Related Item
63	PWR	Nickel alloy steam generator tubes and sleeves exposed to secondary feedwater/ steam	Cracking due to OD stress corrosion cracking and intergranular attack, loss of material due to fretting and wear	Steam generator tubing integrity and water chemistry	No	R-47 R-48 R-49
64	PWR	Nickel alloy steam generator tubes and sleeves exposed to phosphate chemistry in secondary feedwater/ steam	Loss of material due to wastage and pitting corrosion	Steam generator tubing integrity and water chemistry	No	R-50
65	PWR	Steel steam generator tube support plate, tube bundle wrapper exposed to secondary feedwater/steam	Loss of material due to erosion, general, pitting, and crevice corrosion, ligament cracking due to corrosion	Steam generator tubing integrity and water chemistry	No	R-42 RP-16
66	PWR	Nickel alloy steam generator tubes exposed to secondary feedwater/ steam	Denting due to corrosion of steel tube support plate	Steam generator tubing integrity and water chemistry and, for plants that could experience denting at the upper support plates, evaluate potential for rapidly propagating cracks and then develop and take corrective actions consistent with Bulletin 88-02.	No	R-43

Table 3.1-1. Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report

ID	Type	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Related Item
67	PWR	Reactor vessel internals (CASS)	Loss of fracture toughness due to thermal aging and neutron irradiation embrittlement	Thermal aging and neutron irradiation embrittlement	No	R-111 R-140 R-153 R-171 R-183 R-191 R-206
68	PWR	Cast austenitic stainless steel, steel with stainless steel cladding Class 1 pump casings and valve bodies exposed to reactor coolant	Cracking due to stress corrosion cracking	Water chemistry (and Inservice Inspection for components that do not meet the material guidelines of NUREG-0313).	No	R-09
69	PWR	Nickel alloy steam generator components such as, secondary side nozzles (vent, drain, and instrumentation) exposed to secondary feedwater/ steam	Cracking due to stress corrosion cracking	Water Chemistry and One-Time Inspection or Inservice Inspection.	No	R-36
70	BWR/ PWR	Stainless steel piping, piping components, and piping elements exposed to air with borated water leakage or gas	None	None	NA - No AEM or AMP	RP-07
71	BWR/ PWR	Stainless steel, cast austenitic stainless steel, and nickel alloy piping, piping components, and piping elements exposed to air – indoor uncontrolled (external)	None	None	NA - No AEM or AMP	RP-02 RP-03 RP-04
72	BWR/ PWR	Steel and stainless steel piping, piping components, and piping elements in concrete	None	None	NA - No AEM or AMP	RP-01 RP-06

Table 3.1-1. Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report						
ID	**Type**	**Component**	**Aging Effect/Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Related Item**
73	PWR	Stainless steel piping, piping components, and piping elements exposed to air with borated water leakage or gas	None	None	NA - No AEM or AMP	RP-05

Table 3.1-2. FSAR Supplement for Aging Management of Reactor Vessel, Internals, and Reactor Coolant System		
Program	**Description of Program**	**Implementation Schedule***
ISI	The program consists of periodic volumetric, surface, and/or visual examination of components and their supports for assessment, signs of degradation, and corrective actions. This program is in accordance with ASME Section XI, 2001 edition through the 2002 and 2003 addenda.	Existing program
Water chemistry	To mitigate aging effects on component surfaces that are exposed to water as process fluid, chemistry programs are used to control water chemistry for impurities (e.g., chloride, fluoride, and sulfate) that accelerate corrosion. This program relies on monitoring and control of water chemistry to keep peak levels of various contaminants below the system-specific limits based on EPRI guidelines of TR-103515 for water chemistry in BWRs, TR-105714 for primary water chemistry in PWRs, and TR-102134 for secondary water chemistry in PWRs.	Existing program
One-time inspection	To verify the effectiveness of the water chemistry control program by determining if the aging effect is not occurring or the aging effect is progressing slowly so that the that the intended function will be maintained during the period of extended operation, a one-time inspection of small-bore piping less than NPS 4, including pipe, fittings, and branch connections, using suitable techniques at the most susceptible locations is performed. Actual inspection locations should be based on physical accessibility, exposure levels, and NDE techniques, and locations identified in NRC IN 97-46.	Inspection should be completed before the period of extended operation.
Bolting integrity	This program consists of guidelines on materials selection, strength and hardness properties, installation procedures, lubricants and sealants, corrosion considerations in the selection and installation of pressure-retaining bolting for nuclear applications, and enhanced inspection techniques. This program relies on the bolting integrity program delineated in NUREG-1339 and industry's recommendations delineated in EPRI NP-5769, with the exceptions noted in NUREG-1339 for safety-related bolting and in EPRI TR-104213 for pressure retaining bolting and structural bolting.	Existing program

Table 3.1-2. FSAR Supplement for Aging Management of Reactor Vessel, Internals, and Reactor Coolant System (continued)		
Program	**Description of Program**	**Implementation Schedule***
Reactor vessel surveillance	Periodic testing of metallurgical surveillance samples is used to monitor the progress of neutron embrittlement of the reactor pressure vessel as a function of neutron fluence, in accordance with Regulatory Guide (RG) 1.99, Rev. 2.	The surveillance capsule withdrawal schedule should be revised before the period of extended operation.
Boric acid corrosion	The program consists of (1) visual inspection of external surfaces that are potentially exposed to borated water leakage, (2) timely discovery of leak path and removal of the boric acid residues, (3) assessment of the damage, and (4) follow-up inspection for adequacy. This program is implemented in response to GL 88-05.	Existing program
Thermal aging and neutron irradiation embrittlement of cast austenitic stainless steel	The program consists of (1) determination of the susceptibility of cast austenitic stainless steel components to thermal aging embrittlement, (2) accounting for the synergistic effects of thermal aging and neutron irradiation, and (3) implementing a supplemental examination program, as necessary.	Program should be implemented before the period of extended operation.
Reactor Head Closure Studs	This program includes inservice inspection ISI. For boiling water reactors (BWRs), this program also includes additional preventive actions and inspection techniques.	Existing program
Flow-accelerated corrosion	The program consists of the following: (1) conduct appropriate analysis and baseline inspection, (2) determine extent of thinning and replace/repair components, and (3) perform follow up inspections to confirm or quantify and take longer-term corrective actions. This program is in response to NRC GL 89-08.	Existing Program
Quality assurance	The 10 CFR Part 50, Appendix B program provides for corrective actions, confirmation process, and administrative controls for aging management programs for license renewal. The scope of this existing program will be expanded to include nonsafety-related structures and components that are subject to an AMR for license renewal.	Program should be implemented before the period of extended operation.
Vessel closure head penetration	The program assesses degradation of CRD mechanism nozzle and other vessel closure head penetrations, and consists of a review of the scope and schedule of inspection, including the leakage detection system, to assure detection of cracks before the loss of intended function of the penetrations. This is in response to NRC GL 97-01.	Existing program

Table 3.1-2. FSAR Supplement for Aging Management of Reactor Vessel, Internals, and Reactor Coolant System (continued)		
Program	**Description of Program**	**Implementation Schedule***
BWR Control Rod Drive Return Line Nozzle	The AMP monitors the effects of cracking on the intended function of the component by detection and sizing of cracks by ISI in accordance with the NUREG-0619 and alternative recommendation of GE NE-523-A71-0594. NUREG-0619 specifies UT of the entire nozzle and penetration testing (PT) of varying portions of the blend radius and bore. GE NE-523-A71-0594 specifies UT of specific regions of the blend radius and bore. UT techniques and personnel qualification are according to the guidelines of GE NE-523-A71-0594.	Program should be implemented before the period of extended operation.
Steam generator tube integrity	This program consists of SG inspection scope, frequency, acceptance criteria for the plugging and repair of flawed tubes in accordance with the plant technical specifications and includes commitments to NEI 97-06.	Existing program
~~Loose part monitoring~~	~~The program consists of loose part monitoring of reactor vessel and primary coolant systems in accordance with ASME OM-S/G-1997 standards. The program addresses methods, intervals, parameters to be measured and evaluated, and records requirements.~~	~~Existing program~~
~~Neutron noise monitoring~~	~~The program consists of neutron noise monitoring for the detection of loss of axial preload at the core support barrel's upper support flange, and can detect physical displacement and motion of reactor internals in accordance with ASME OM-S/G-1997 standards. The program addresses methods, intervals, parameters to be measured and evaluated, acceptance criteria, and records requirements.~~	~~Existing program~~
BWR Vessel Internals	The program includes (a) inspection and flaw evaluation in conformance with the guidelines of applicable and staff-approved boiling water reactor vessel and internals project (BWRVIP) documents and (b) monitoring and control of reactor coolant water chemistry in accordance with the guidelines of BWRVIP-29 (EPRI TR-103515) to ensure the long-term integrity and safe operation of boiling water reactor (BWR) vessel internal components.	Existing program
Plant-specific AMP	The description should contain information associated with the basis for determining that aging effects will be managed during the period of extended operation.	Program should be implemented before the period of extended operation.

Table 3.1-2. FSAR Supplement for Aging Management of Reactor Vessel, Internals, and Reactor Coolant System (continued)		
Program	**Description of Program**	**Implementation Schedule***
BWR Vessel ID Attachment Welds	The program includes (a) inspection and flaw evaluation in conformance with the guidelines of staff-approved boiling water reactor vessel and internals project (BWRVIP)-48 and (b) monitoring and control of reactor coolant water chemistry in accordance with the guidelines of BWRVIP-29 (EPRI TR-103515).	
BWR Stress Corrosion Cracking	The program to manage intergranular stress corrosion cracking (IGSCC) in boiling water reactor (BWR) coolant pressure boundary piping made of stainless steel (SS) is delineated in NUREG-0313, Rev. 2, and Nuclear Regulatory Commission (NRC) Generic Letter (GL) 88-01 and its Supplement 1. The program includes (a) preventive measures to mitigate IGSCC and (b) inspections to monitor IGSCC and its effects.	Existing program
BWR Penetrations	The program includes (a) inspection and flaw evaluation in conformance with the guidelines of staff-approved boiling water reactor vessel and internals project (BWRVIP)-49 and BWRVIP-27 documents and (b) monitoring and control of reactor coolant water chemistry in accordance with the guidelines of BWRVIP-29 (EPRI TR-103515) to ensure the long-term integrity and safe operation of boiling water reactor (BWR) vessel internal components.	Existing program
Nickel-Alloy Nozzles and Penetrations	The program includes (a) primary water stress corrosion cracking (PWSCC) susceptibility assessment to identify susceptible components, (b) monitoring and control of reactor coolant water chemistry to mitigate PWSCC, and (c) inservice inspection ISI of reactor vessel head penetrations to monitor PWSCC and its effect on the intended function of the component. For susceptible penetrations and locations, the program includes an industry wide, integrated, long-term inspection program based on the industry responses to NRC Generic Letter (GL) 97-01.	Existing program
Thermal Aging of Cast Austenitic Stainless Steel	This program includes (a) determination of the susceptibility of cast austenitic stainless steel components to thermal aging embrittlement and (b) for potentially susceptible components aging management is accomplished through either enhanced volumetric examination or plant- or component-specific flaw tolerance evaluation.	Existing program

Table 3.1-2. FSAR Supplement for Aging Management of Reactor Vessel, Internals, and Reactor Coolant System (continued)		
Program	**Description of Program**	**Implementation Schedule***
PWR Vessel Internals	The program includes (a) augmentation of the inservice inspection (ISI) to include enhanced VT-1 examinations of non-bolted components, and other demonstrated acceptable methods for bolted components for certain susceptible or limiting components or locations, and (b) monitoring and control of reactor coolant water chemistry in accordance with the EPRI guidelines in TR-105714 to ensure the long-term integrity and safe operation of pressurized water reactor (PWR) vessel internal components.	Program should be implemented before the period of extended operation.
BWR Feedwater Nozzle	This program includes (a) enhancing inservice inspection (ISI) specified in the American Society of Mechanical Engineers (ASME) Code, Section XI, with the recommendation of General Electric (GE) NE-523-A71-0594 to perform periodic ultrasonic testing inspection of critical regions of the BWR feedwater nozzle.	Existing program

* An applicant need not incorporate the implementation schedule into its FSAR. However, the reviewer should verify that the applicant has identified and committed in the license renewal application to any future aging management activities to be completed before the period of extended operation. The staff expects to impose a license condition on any renewed license to ensure that the applicant will complete these activities no later than the committed date.

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3.5 AGING MANAGEMENT OF CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORTS

The changes marked in this section support the consolidation of tables in GALL Chapter III. No further bases are provided. The changes are marked using WORD revision tracking features.

3.5 AGING MANAGEMENT OF CONTAINMENTS, STRUCTURES, AND COMPONENT SUPPORTS

Review Responsibilities

Primary - Branch assigned responsibility by PM as described in SRP-LR section 3.0

3.5.1 Areas of Review

This review plan section addresses the aging management review (AMR) for structures and component supports. For a recent vintage plant, the information related to structures and component supports is contained in Chapter 3, "Design of Structures, Components, Equipment, and Systems," of the plant's FSAR, consistent with the "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (NUREG-0800) (Ref. 1). For older vintage plants, the location of applicable information is plant-specific because an older plant's FSAR may have predated NUREG-0800. The scope of this section is PWR and BWR containment structures; Class I structures; and component supports. The PWR containment structures consist of concrete (reinforced or prestressed) and steel containments. The BWR containment structures consist of Mark I steel containments, Mark II concrete (reinforced or prestressed) and steel containments, and Mark III concrete and steel containments (Ref. 2).

The Class I and Class 2 structures are organized into three groups: Group 1: BWR reactor building, PWR shield building, control room/building, auxiliary building, diesel generator building, radwaste building, turbine building, switchgear room, yard structures (auxiliary feedwater pump house, utility/piping tunnels, security lighting poles, manholes, duct banks), SBO structures (transmission towers, startup transformer circuit breaker foundation, electrical enclosure), containment internal structures, excluding refueling canal, fuel storage facility, refueling canal; BWR unit vent stack; Group 2: water-control structures (e.g., intake structure, cooling tower, and spray pond); Group 3: Tanks (concrete and steel) and missile barriers, and (Ref. 2).

The component supports are organized into two groups: Group B1: supports for ASME piping and components and Class MC (BWR Containment Supports) components; Group B2: Other supports *(All other supports except as stated in Group B1)* (Ref. 2).

The responsible review organization is to review the following LRA AMR and AMP items assigned to it, per SRP-LR section 3.0, for review:

<u>**AMRs**</u>
- AMRs consistent with the GALL report, for which further evaluation is not recommended
- AMRs consistent with the GALL report, for which further evaluation is recommended
- AMRs not consistent with the GALL report

<u>**AMPs**</u>
- AMPs consistent with GALL AMPs (with or without exceptions)
- Plant-specific AMPs

<u>**FSAR Supplement**</u>
- In addition, the responsible review organization is to review the FSAR supplement associated with each assigned AMP.

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3.5.2 Acceptance Criteria

The acceptance criteria for the areas of review describe methods for determining whether the applicant has met the requirements of the NRC's regulations in 10 CFR 54.21.

3.5.2.1 AMR Results Consistent with the GALL Report for Which No Further Evaluation is Recommended

The aging management review and acceptable aging management programs applicable to structures and component supports are described and evaluated in Chapters II and III of the GALL report (Ref. 2).

The applicant's LRA should provide sufficient information so that the NRC reviewer is able to confirm that the specific AMR line-item and the associated AMP are consistent with the cited GALL AMR line-item. The staff reviewer should then confirm that the LRA AMR line-item is consistent with the GALL line-item to which it is compared.

If the applicant identifies an exception to the cited GALL AMP, the LRA should include a basis demonstrating how the criteria of 10 CFR 54.21(a)(3) would still be met. The NRC reviewer should then confirm that the AMP with all exceptions would satisfy the criteria of 10 CFR 54.21(a)(3). If, while reviewing the AMP, the reviewer identifies a difference from the GALL AMP, this difference should be reviewed and dispositioned as if it were an exception identified by the applicant in its LRA. The disposition of all LRA-defined exceptions and staff-identified differences should be documented.

The LRA should identify any enhancements that are needed to permit an existing aging management program to be declared consistent with the GALL AMP to which the LRA AMP is compared. The reviewer is to confirm both that the enhancement, if implemented, would allow the existing plant aging management program to be consistent with the GALL AMP and also that the applicant has a commitment in the FSAR supplement to implement the enhancement prior to the period of extended operation. The reviewer should document the disposition of all enhancements.

3.5.2.2 AMR Results Consistent with the GALL Report for Which Further Evaluation is Recommended

The basic acceptance criteria defined in 3.5.2.1 apply to all of the AMRs and AMPs reviewed as part of this section. In addition, if the GALL AMR line-item to which the LRA AMR line-item is compared identifies that "further evaluation is recommended," then additional criteria apply as identified by the GALL report for each of the following aging effect/aging mechanism combinations.

3.5.2.2.1 PWR and BWR Containments

3.5.2.2.1.1 Aging of Inaccessible Concrete Areas

Cracking, spalling, and increases in porosity and permeability due to aggressive chemical attack; and cracking, spalling, loss of bond, and loss of material due to corrosion of embedded steel could occur in inaccessible areas of PWR concrete and steel containments; BWR Mark II concrete containments; and Mark III concrete and steel containments. The GALL report recommends further evaluation to manage the aging effects for inaccessible areas if the environment is aggressive.

3.5.2.2.1.2 Cracks and Distortion due to Increased Stress Levels from Settlement; Reduction of Foundation Strength due to Erosion of Porous Concrete Subfoundations, if Not Covered by Structures Monitoring Program

Cracking, distortion, and increase in component stress level due to settlement could occur in PWR concrete and steel containments and BWR Mark II concrete containments and Mark III concrete and steel containments. Also, reduction of foundation strength due to erosion of porous concrete subfoundations could occur in all types of PWR and BWR containments. Some plants may rely on a de-watering system to lower the site ground water level. If the plant's CLB credits a de-watering system, the GALL report recommends verification of the continued functionality of the de-watering system during the period of extended operation. The GALL report recommends no further evaluation if this activity is included in the scope of the applicant's structures monitoring program.

3.5.2.2.1.3 Reduction of Strength and Modulus of Concrete Structures due to Elevated Temperature

Reduction of strength and modulus of elasticity due to elevated temperatures could occur in PWR concrete and steel containments and BWR Mark II concrete containments and Mark III concrete and steel containments. The GALL report recommends further evaluation if any portion of the concrete containment components exceeds specified temperature limits, i.e., general area temperature 66°C (150°F) and local area temperature 93°C (200°F).

3.5.2.2.1.4 Loss of Material due to General, Pitting and Crevice Corrosion in Inaccessible Areas of Steel Containment Shell or Liner Plate

Loss of material due to general, pitting and crevice corrosion could occur in inaccessible areas of the steel containment shell or the steel liner plate for all types of PWR and BWR containments. The GALL report recommends further evaluation of plant-specific programs to manage this aging effect for inaccessible areas if specific criteria defined in the GALL report cannot be satisfied.

3.5.2.2.1.5 Loss of Prestress due to Relaxation, Shrinkage, Creep, and Elevated Temperature

Loss of prestress forces due to relaxation, shrinkage, creep, and elevated temperature for PWR prestressed concrete containments and BWR Mark II prestressed concrete containments is a TLAA as defined in 10 CFR 54.3. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c). The evaluation of this TLAA is addressed separately in Section 4.5 of this standard review plan.

3.5.2.2.1.6 Cumulative Fatigue Damage

If included in the current licensing basis, fatigue analyses of containment steel liner plates and steel containment shells (including welded joints) and penetrations (including penetration sleeves, dissimilar metal welds, and penetration bellows) for all types of PWR and BWR containments and BWR vent header and downcomers are TLAAs as defined in 10 CFR 54.3. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c). The evaluation of this TLAA is addressed separately in Section 4.6 of this standard review plan.

3.5.2.2.1.7 Cracking due to Cyclic Loading and Stress Corrosion Cracking
Cracking of containment penetrations (including penetration sleeves, penetration bellows, and dissimilar metal welds) due to cyclic loading or SCC could occur in all types of PWR and BWR containments. Cracking could also occur in vent line bellows, vent headers and downcomers due to SCC for BWR containments. A visual VT-3 examination would not detect such cracks. Moreover, stress corrosion cracking is a concern for dissimilar metal welds. The GALL report recommends further evaluation of the inspection methods implemented to detect these aging effects.

3.5.2.2.1.8 Scaling, Cracking, and Spalling due to Freeze-Thaw; and Expansion and Cracking due to Reaction with Aggregate

Scaling, cracking, and spalling due to freeze-thaw could occur in PWR and BWR concrete containments; and expansion and cracking due to reaction with aggregate could occur in concrete elements of PWR and BWR concrete and steel containments. Further evaluation is not necessary if stated conditions in NUREG-1801 are satisfied for inaccessible areas.

3.5.2.2.2 Class I Structures

3.5.2.2.2.1 Aging of Structures Not Covered by Structures Monitoring Program

The GALL report recommends further evaluation of certain structure/aging effect combinations if they are not covered by the structures monitoring program. This includes (1) scaling, cracking, and spalling due to repeated freeze-thaw for Groups 1 and 3 structures; (2) scaling, cracking, spalling and increase in porosity and permeability due to leaching of calcium hydroxide and aggressive chemical attack for Groups 1 and 3 structures; (3) expansion and cracking due to reaction with aggregates for Groups 1 and 3 structures; (4) cracking, spalling, loss of bond, and loss of material due to general, pitting and crevice corrosion of embedded steel for Groups 1 and 3 structures; (5) cracks and distortion due to increase in component stress level from settlement for Groups 1 and 3 structures; (6) reduction of foundation strength due to erosion of porous concrete subfoundation for Groups 1 and 3 structures; (7) loss of material due to general, pitting and crevice corrosion of structural steel components for Groups 1 and 3 structures; (8) loss of strength and modulus of concrete structures due to elevated temperatures for Groups 1; and (9) cracking due to SCC and loss of material due to crevice corrosion of stainless steel liner for tanks in Group 3 structure. Further evaluation is necessary only for structure/aging effect combinations not covered by the structures monitoring program.

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Technical details of the aging management issue are presented in Subsection 3.5.2.2.1.2 for items (5) and (6) and Subsection 3.5.2.2.1.3 for item (8).

Loss of material (spalling, scaling) and cracking due to freeze-thaw could occur in below-grade inaccessible concrete areas for Groups 1 and 3 structures; and expansion and cracking due to reaction with aggregates could occur in below-grade inaccessible concrete areas for Groups 1 and 3 structures. The GALL report recommends further evaluation of plant-specific programs to manage the aging effects for inaccessible areas if specific criteria defined in the GALL report cannot be satisfied.

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3.5.2.2.2.2 Aging Management of Inaccessible Areas

Cracking, spalling, and increases in porosity and permeability due to aggressive chemical attack; and cracking, spalling, loss of bond, and loss of material due to corrosion of embedded steel could occur in below-grade inaccessible concrete areas. The GALL report recommends further evaluation to manage these aging effects in inaccessible areas of Groups 1 and 3 structures.

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3.5.2.2.3 Component Supports

3.5.2.2.3.1 Aging of Supports Not Covered by Structures Monitoring Program

The GALL report recommends further evaluation of certain component support/aging effect combinations if they are not covered by the structures monitoring program. This includes (1) reduction in concrete anchor capacity due to degradation of the surrounding concrete, for Groups B1 and B2 supports; (2) loss of material due to environmental corrosion, for Groups B2 supports; and (3) reduction/loss of isolation function due to degradation of vibration isolation elements for mechanical equipment, in Group B2 supports. Further evaluation is necessary only for structure/aging effect combinations not covered by the structures monitoring program.

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3.5.2.2.3.2 Cumulative Fatigue Damage due to Cyclic Loading

Fatigue of component support members, anchor bolts, and welds for Groups B1.1, B1.2, and B1.3 component supports is a TLAA as defined in 10 CFR 54.3 only if a CLB fatigue analysis exists. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c). The evaluation of this TLAA is addressed separately in Section 4.3 of this standard review plan.

3.5.2.2.4 Quality Assurance for Aging Management of Nonsafety-Related Components

Acceptance criteria are described in Branch Technical Position IQMB-1 (Appendix A.2 of this standard review plan.)

3.5.2.3 AMR Results Not Consistent with or Not Addressed in GALL Report

Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan.)

3.5.2.4 FSAR Supplement

The summary description of the programs and activities for managing the effects of aging for the period of extended operation in the FSAR supplement should be appropriate such that later changes can be controlled by 10 CFR 50.59. The description should contain information associated with the bases for determining that aging effects are managed during the period of extended operation. The description should also contain any future aging management activities, including enhancements and commitments, to be completed before the period of extended operation. Examples of the type of information required are provided in Table 3.5-2 of this standard review plan.

3.5.3 Review Procedures

For each area of review, the following review procedures are to be followed:

3.5.3.1 AMR Results Consistent with the GALL Report for Which No Further Evaluation is Recommended

The applicant may reference the GALL report in its license renewal application, as appropriate, and demonstrate that the aging management reviews and programs at its facility are consistent with those reviewed and approved in the GALL report. The reviewer should not conduct a re-review of the substance of the matters described in the GALL report. If the applicant has provided the information necessary to adopt the finding of program acceptability as described and evaluated in the GALL report, the staff should find acceptable the applicant's reference to GALL in its license renewal application. In making this determination, the reviewer confirms that the applicant has provided a brief description of the system, components, materials, and environment. The reviewer also confirms that the applicant has stated that the applicable aging effects and industry and plant-specific operating experience have been reviewed by the applicant and are evaluated in the GALL Report.

Furthermore, the reviewer should confirm that the applicant has addressed operating experience identified after the issuance of the GALL report. Performance of this review requires the reviewer to confirm that the applicant has identified those aging effects for the structures and component supports that are contained in GALL as applicable to its plant.

The reviewer confirms that the applicant has identified the appropriate AMPs as described and evaluated in the GALL report. If the applicant commits to an enhancement to make its aging management program consistent with a GALL AMP, then the reviewer is to confirm that this enhancement when implemented will indeed make the LRA AMP consistent with the GALL AMP. If an aging management program in the LRA identifies an exception to the GALL AMP to which it is claiming to be consistent, the reviewer is to confirm that the LRA AMP with the exception will satisfy the criteria of 10 CFR 54.21(a)(3). If the reviewer identifies a difference, not identified by the LRA, between the LRA AMP and the GALL AMP, with which the LRA claims to be consistent, the reviewer should confirm that the LRA AMP with this difference satisfies 10 CFR 54.21(a)(3). The reviewer should document the basis for accepting enhancements, exceptions or differences. The AMPs evaluated in GALL pertinent to the structures and component supports are summarized in Table 3.5-1 of this standard review plan. In this table, the ID column provides a row identifier useful in matching the information presented in the corresponding table in the GALL report Vol. 1. The Related Item column identifies the item number in the GALL report Vol. 2, Chapters II through VIII, presenting detailed information summarized by this row.

3.5.3.2 AMR Results Consistent with the GALL Report for Which Further Evaluation is Recommended

The basic review procedures defined in 3.5.3.1 apply to all of the AMRs and AMPs provided in this section. In addition, if the GALL AMR line-item to which the LRA AMR line-item is compared identifies that "further evaluation is recommended," then additional criteria apply as identified by the GALL report for each of the following aging effect/aging mechanism combinations.

3.5.3.2.1 PWR and BWR Containments

3.5.3.2.1.1 Aging of Inaccessible Concrete Areas

The GALL report recommends further evaluation of programs to manage aging effects in inaccessible areas. Possible effects due to leaching of calcium hydroxide and aggressive chemical attack are cracking, spalling, and increases in porosity and permeability. Possible effects due to corrosion of embedded steel in PWR concrete and steel containments and BWR Mark II concrete containments and Mark III concrete and steel containments are cracking, spalling, loss of bond, and loss of material. The current aging management programs that involve detecting aging effects in inaccessible areas consist of Section XI, Subsection IWL examinations of 1992 or later edition of ASME code (Ref. 3), which is in accordance with the requirements of, and is approved in, 10 CFR 50.55a. However, Subsection IWL exempts from examination portions of the concrete containments that are inaccessible (e.g., foundation, exterior walls below grades, concrete covered by liner)

To cover the inaccessible areas, 10 CFR 50.55a(b)(2)(ix) requires that the licensee evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas. In addition, the GALL report recommends further evaluation to manage the aging effects for inaccessible areas if the below-grade environment is aggressive. Periodic monitoring of below-grade water chemistry (including consideration of potential seasonal variations) is an acceptable approach to demonstrate that the below-grade environment is aggressive or non-aggressive. The GALL recommends that examination of representative samples of below-grade concrete, when excavated for any reason, be performed, if the below-grade environment is aggressive. The reviewer reviews the applicant's proposed aging management program to verify that, where appropriate, an effective inspection program will be implemented to ensure that the aging effects in inaccessible areas are adequately managed during the period of extended operation.

3.5.3.2.1.2 Cracks and distortion due to increased stress levels from settlement; Reduction of Foundation Strength due to Erosion of Porous Concrete Subfoundations, if Not Covered by Structures Monitoring Program

If applicable to the applicant's plant, the GALL report recommends aging management of (1) cracks and distortion due to increases in component stress level from settlement for PWR concrete and steel containments and BWR Mark II concrete containments and Mark III concrete and steel containments and (2) reduction of foundation strength due to erosion of porous concrete subfoundations for all types of PWR and BWR containments If a de-watering system is relied upon for control of settlement and erosion, then proper functioning of the de-watering system should be monitored for the period of extended operation. The reviewer confirms that, if the applicant's plant credits a de-watering system in its CLB, the applicant has committed to monitor the functionality of the de-watering system under the applicant's structures monitoring program. If not, the reviewer evaluates the plant-specific program for monitoring the de-watering system during the period of extended operation.

3.5.3.2.1.3 Reduction of Strength and Modulus of Concrete Structures due to Elevated Temperature

The GALL report recommends further evaluation of programs to manage reduction of strength and modulus of concrete structures due to elevated temperature for PWR concrete and steel containments and BWR Mark II concrete containments and Mark III concrete and steel containments. The GALL report notes that the implementation of Subsection IWL examinations and 10 CFR 50.55a would not be able to detect the reduction of concrete strength and modulus due to elevated temperature and also notes that no mandated aging management exists for managing this aging effect.

The GALL report recommends that a plant-specific evaluation be performed if any portion of the concrete containment components exceeds specified temperature limits, viz., general temperature 66°C (150°F) and local area temperature 93°C (200°F). The reviewer verifies that the applicant's discussion in the renewal application indicates that the affected PWR and BWR containment components are not exposed to temperature that exceeds the temperature limits [operating temperature <66°C (150°F), local area temperature <93°C (200°F)]. For concrete containment components that operate above these temperature limits, the reviewer reviews the applicant's proposed programs on a case-by-case basis to ensure that the effects of elevated temperature will be managed during the period of extended operation.

3.5.3.2.1.4 Loss of Material due to General, Pitting and Crevice Corrosion in Inaccessible Areas of Steel Containment Shell or Liner Plate

The GALL report identifies programs to manage loss of material due to general, pitting and crevice corrosion of the steel containment shell or the steel liner plate for all types of PWR and BWR containments. The aging management program consists of ASME Section XI, Subsection IWE (Ref. 4) and the requirements of 10 CFR 50.55a for inaccessible areas. Subsection IWE exempts from examination portions of the containments that are inaccessible, such as embedded or inaccessible portions of steel liners and steel containment shells, piping, and valves penetrating or attaching to the containment.

To cover the inaccessible areas, 10 CFR 50.55a(b)(2)(ix) requires that the licensee shall evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas. In addition, the GALL report recommends further evaluation of plant-specific programs to manage the aging effects for inaccessible areas if specific criteria defined in the GALL report cannot be satisfied. The reviewer reviews the applicant's proposed aging management program to confirm that, where appropriate, an effective inspection program has been developed and implemented to ensure that the aging effects in inaccessible areas are adequately managed.

3.5.3.2.1.5 Loss of Prestress due to Relaxation, Shrinkage, Creep, and Elevated Temperature

The GALL report identifies loss of prestress as a TLAA to be performed for the period of license renewal. The reviewer reviews the evaluation of this TLAA separately, following the guidance in Section 4.5 of this standard review plan.

3.5.3.2.1.6 Cumulative Fatigue Damage

Fatigue analyses included in current licensing basis for the containment liner plate and penetrations are TLAAs as defined in 10 CFR 54.3. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c). The evaluation of this TLAA is addressed separately in Section 4.6 of this standard review plan.

3.5.3.2.1.7 Cracking due to Cyclic Loading and Stress Corrosion Cracking

The GALL report recommends further evaluation of programs to manage cracking of containment penetrations (including penetration sleeves, penetration bellows, and dissimilar metal welds) due to cyclic loading or SCC for all types of PWR and BWR containments. A similar recommendation for further evaluation of programs to manage cracking of vent line bellows, vent headers and downcomers due to SCC is also provided for BWR containments. Containment ISI and leak rate testing may not be sufficient to detect cracks, especially for dissimilar metal welds. The reviewer should evaluate the applicant's proposed programs to confirm that adequate inspection methods will be implemented to ensure that cracks are detected.

3.5.3.2.1.8 Scaling, Cracking, and Spalling due to Freeze-Thaw; and Expansion and Cracking due to Reaction with Aggregate

The GALL report recommends further evaluation only if the stated conditions are not satisfied for inaccessible concrete. This includes scaling, cracking, and spalling due to freeze-thaw; and expansion and cracking due to reaction with aggregate for concrete elements of PWR and BWR containments. The reviewer should confirm that the applicant has satisfied the conditions for inaccessible concrete as identified in the GALL report. Otherwise, the reviewer reviews the applicant's proposed aging management program to verify that, where appropriate, an effective inspection program has been developed and implemented to ensure that these aging effects in inaccessible areas are adequately managed.

3.5.3.2.2 Class I Structures

3.5.3.2.2.1 Aging of Structures Not Covered by Structures Monitoring Program

The GALL report recommends further evaluation of certain structure/aging effect combinations if they are not covered by the structures monitoring program. This includes (1) scaling, cracking, and spalling due to repeated freeze-thaw for Groups 1 and 3 structures; (2) scaling, cracking, spalling and increase in porosity and permeability due to leaching of calcium hydroxide and aggressive chemical attack for Groups 1 and 3 structures; (3) expansion and cracking due to reaction with aggregates for Groups 1 and 3 structures; (4) cracking, spalling, loss of bond, and loss of material due to general, pitting and crevice corrosion of embedded steel for Groups 1 and 3 structures; (5) cracks and distortion due to increase in component stress level from settlement for Groups 1 and 3 structures; (6) reduction of foundation strength due to erosion of porous concrete subfoundation for Groups 1 and 3 structures; (7) loss of material due to general, pitting and crevice corrosion of structural steel components for Groups 1 and 3 structures; (8) loss of strength and modulus of concrete structures due to elevated temperatures for Group 1; and (9) cracking due to SCC and loss of material due to crevice corrosion of stainless steel liner for tanks in Groups 3 structures. Further evaluation is necessary only for structure/aging effect combinations not covered by the structures monitoring program.

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The aging management program consists of a structures monitoring program to confirm that the CLB is maintained through periodic testing and inspection of critical plant structures, systems, and components. The reviewer confirms that the applicant has identified the structure/aging effect combinations not within the scope of the applicant's structures monitoring program developed in accordance with the guidance provided in NUMARC 93-01, Rev. 2 (Ref. 5) and RG 1.160, Rev. 2 (Ref. 6). The applicant may choose to expand the scope of its structures monitoring program to include these structure/aging effect combinations. Otherwise, the reviewer evaluates the applicant's proposed program in accordance with the guidance in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan.)

The GALL report recommends further evaluation of plant-specific programs to manage the aging effects for inaccessible areas if specific criteria defined in the GALL report cannot be satisfied. The following degradations are managed: loss of material (spalling, scaling) and cracking due to freeze-thaw for Groups 1 and 3 structures; and expansion and cracking due to reaction with aggregates for Groups 1 and 3 structures. The reviewer reviews the aging management program on a case-by-case basis to ensure that the intended functions will be maintained during the period of the extended operation.

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3.5.3.2.2.2 Aging Management of Inaccessible Areas

The GALL report recommends further evaluation of aging management for inaccessible concrete areas, such as foundation and exterior walls below grade exposed to an aggressive environment. The following degradations are managed: cracking, spalling, and increases in porosity and permeability due to aggressive chemical attack; and cracking, spalling, loss of bond, and loss of material due to corrosion of embedded steel for Groups 1 and 3 structures. Periodic monitoring of below-grade water chemistry (including consideration of potential seasonal variations) is an acceptable approach to demonstrate that the below-grade environment is aggressive or non-aggressive. The GALL recommends that examination of representative samples of below-grade concrete, when excavated for any reason, be performed, if the below-grade environment is aggressive. The reviewer reviews the aging management program on a case-by-case basis to ensure that the intended functions will be maintained during the period of the extended operation.

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3.5.3.2.3 Component Supports

3.5.3.2.3.1 Aging of Supports Not Covered by Structures Monitoring Program

The GALL report recommends further evaluation of certain component support/aging effect combinations if they are not covered by the structures monitoring program. This includes (1) reduction in concrete anchor capacity due to degradation of the surrounding concrete, for Groups B1 and B2 supports; (2) loss of material due to environmental corrosion, for Groups B2-B5 supports; and (3) reduction/loss of isolation function due to degradation of vibration isolation elements in Mechanical equipment, for Group B2 supports. Further evaluation is necessary only for structure/aging effect combinations not covered by the structures monitoring program.

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The aging management program consists of a structures monitoring program to verify that the CLB is maintained through periodic testing and inspection of critical plant structures, systems, and components. The reviewer confirms that the applicant has identified the component support/aging effect combinations not within the scope of the applicant's structures monitoring program developed in accordance with the guidance provided in NUMARC 93-01, Rev. 2 (Ref. 5) and RG 1.160, Rev. 2 (Ref. 6). The applicant may choose to expand the scope of its structures monitoring program to include these component support/aging effect combinations. Otherwise, the reviewer evaluates the applicant's proposed program in accordance with the guidance in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3.5.3.2.3.2 Cumulative Fatigue Damage

Fatigue of support members, anchor bolts, and welds for Groups B1.1, B1.2, and B1.3 component supports is a TLAA as defined in 10 CFR 54.3 only if a CLB fatigue analysis exists. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c). The evaluation of this TLAA is addressed separately in Section 4.3 of this standard review plan.

3.5.3.2.4 Quality Assurance for Aging Management of Nonsafety-Related Components

The applicant's aging management programs for license renewal should contain the elements of corrective actions, the confirmation process, and administrative controls. Safety-related components are covered by 10 CFR Part 50 Appendix B, which is adequate to address these program elements. However, Appendix B does not apply to nonsafety-related components that are subject to an AMR for license renewal. Nevertheless, an applicant has the option to expand the scope of its 10 CFR Part 50 Appendix B program to include these components and address these program elements. If the applicant chooses this option, the reviewer verifies that the applicant has documented such a commitment in the FSAR supplement. If the applicant chooses alternative means, the branch responsible for quality assurance should be requested to review the applicant's proposal on a case-by-case basis.

3.5.3.3 AMR Results Not Consistent with or Not Addressed in GALL Report

The reviewer should confirm that the applicant, in the license renewal application, has identified applicable aging effects, listed the appropriate combination of materials and environments, and aging management programs that will adequately manage the aging effects. The aging management program credited could be an AMP that is described and evaluated in the GALL report or a plant-specific program. Review procedures are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3.5.3.4 FSAR Supplement

The reviewer confirms that the applicant has provided information equivalent to that in Table 3.5-2 in the FSAR supplement for aging management of the Structures and Component Supports for license renewal. The reviewer also confirms that the applicant has provided information equivalent to that in Table 3.5-2 in the FSAR supplement for SRP-LR Subsection 3.5.3.3, "AMR Results Not Consistent with or Not Addressed in the GALL Report,"

The staff expects to impose a license condition on any renewed license to require the applicant to update its FSAR to include this FSAR supplement at the next update required pursuant to 10 CFR 50.71(e)(4). As part of the license condition until the FSAR update is complete, the applicant may make changes to the programs described in its FSAR supplement without prior NRC approval, provided that the applicant evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59.

As noted in Table 3.5-2, an applicant need not incorporate the implementation schedule into its FSAR. However, the reviewer should confirm that the applicant has identified and committed in the license renewal application to any future aging management activities, including enhancements and commitments, to be completed before the period of extended operation. The staff expects to impose a license condition on any renewed license to ensure that the applicant will complete these activities no later than the committed date.

3.5.4 Evaluation Findings

If the reviewer has confirmed that the applicant has provided information sufficient to satisfy the provisions of this review plan section, an evaluation finding similar to the following text should be included in the staff's safety evaluation report:

> On the basis of its review, the staff concludes that the applicant has adequately identified the aging effects and the AMPs credited with managing these aging effects for the structures and component supports systems, such that there is reasonable assurance that the component intended functions will be maintained consistent with the CLB during the period of extended operation. The staff also reviewed the applicable FSAR supplement program descriptions and concludes that the FSAR supplement provides an adequate program description of the AMPs credited for managing aging effects, as required by 10 CFR 54.21(d).

3.5.5 Implementation

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the NRC's regulations, the method described herein will be used by the staff in its evaluation of conformance with NRC regulations.

3.5.6 References

1. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, July 1981.

2. Draft NUREG-1801, "Generic Aging Lessons Learned (GALL)," U.S. Nuclear Regulatory Commission, Revision 1, September 2005.

3. American Society of Mechanical Engineers, ASME Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, Subsection IWL, Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants, 1992 edition with 1992 addenda, or 1995 edition with 1996 addenda. The ASME Boiler and Pressure Vessel Code, The American Society of Mechanical Engineers, New York, NY.

4. American Society of Mechanical Engineers, ASME Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, Subsection IWE, Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants, 1992 edition with 1992 addenda, or 1995 edition with 1996 addenda. The ASME Boiler and Pressure Vessel Code, The American Society of Mechanical Engineers, New York, NY.

5. NUMARC 93-01, Rev. 2, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" [Line-In/Line-Out Version], Nuclear Energy Institute, April 1996.

6. NRC Regulatory Guide 1.160, Revision 2, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," March 1997.

Table 3.5-1. **Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of the GALL Report**

ID	Type	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Related Item
Common Components of All Types of PWR and BWR Containment						
1	BWR/PWR	Penetration sleeves, penetration bellows, dissimilar metal welds, and downcomers	Cumulative fatigue damage (CLB fatigue analysis exists)	TLAA evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA (see Subsection 3.5.2.2.1.6)	C-13
2	BWR/PWR	Penetration sleeves, bellows, dissimilar metal welds, and downcomers.	Cracking due to cyclic loading, or cracking due to SCC	Containment ISI and Containment leak rate test	Yes, detection of aging effects is to be evaluated (see Subsection 3.5.2.2.1.7)	C-14, C-15
3	BWR/PWR	Penetration sleeves, penetration bellows, and dissimilar metal welds	Loss of material due to general, pitting and crevice corrosion	Containment ISI and Containment leak rate test	No	C-12
4	BWR/PWR	Personnel airlock, equipment hatch and CRD hatch	Loss of material due to general, pitting and crevice corrosion	Containment ISI and Containment leak rate test	No	C-16
5	BWR/PWR	Personnel airlock, equipment hatch and CRD hatch	Loss of leak tightness in closed position due to mechanical wear of locks, hinges and closure mechanism	Containment leak rate test and Plant Technical Specifications	No	C-17
6	BWR/PWR	Seals, gaskets, and moisture barriers	Loss of sealant and leakage through containment due to deterioration of joint seals, gaskets, and moisture barriers	Containment ISI and Containment leak rate test	No	C-18

Table 3.5-1. Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of the GALL Report

ID	Type	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Related Item
PWR Concrete (Reinforced and Prestressed) and Steel Containment BWR Concrete (Mark II and III) and Steel (Mark I, II, and III) Containment						
7	BWR/PWR	Concrete elements: foundation, walls, dome.	Aging of accessible and inaccessible concrete areas due to aggressive chemical attack, and corrosion of embedded steel	Containment ISI and for inaccessible concrete, an examination of representative samples of below-grade concrete, when excavated for any reason, be performed, if the below-grade environment is aggressive	Yes, if the environment is aggressive (see Subsection 3.5.2.2.1.1)	C-03, C-05
8	BWR/PWR	Concrete elements: foundation, walls, dome.	Aging of accessible and inaccessible concrete areas due to leaching of calcium hydroxide.	Containment ISI	No, if concrete was constructed as stated for inaccessible areas.	C-02
9	BWR/PWR	Concrete elements: All	Cracks and distortion due to increased stress levels from settlement	Structures Monitoring	No, if within the scope of the applicant's structures monitoring program (see Subsection 3.5.2.2.1.2)	C-06

Table 3.5-1. Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of the GALL Report

ID	Type	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Related Item
10	BWR/PWR	Concrete elements: foundation	Reduction in foundation strength due to erosion of porous concrete subfoundation	Structures Monitoring	No, if within the scope of the applicant's structures monitoring program (see Subsection 3.5.2.2.1.2)	C-07
11	BWR/PWR	Concrete elements: foundation, dome, and wall	Reduction of strength and modulus due to elevated temperature	Plant specific	Yes, for any portions of concrete containment that exceed specified temperature limits (see Subsection 3.5.2.2.1.3)	C-08
12	BWR/PWR	Prestressed containment: tendons and anchorage components	Loss of prestress due to relaxation, shrinkage, creep, and elevated temperature	TLAA evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA (see Subsection 3.5.2.2.1.5)	C-11
13	BWR/PWR	Steel elements: liner plate, containment shell downcomers, drywell support skirt, ECCS suction header	Loss of material due to general, pitting and crevice corrosion in accessible and inaccessible areas	Containment ISI and Containment leak rate test	Yes, if corrosion is significant for inaccessible areas (see Subsection 3.5.2.2.1.4)	C-09, C-19
14	BWR	Steel elements: vent header, drywell head, torus, downcomers, pool shell	Cumulative fatigue damage (CLB fatigue analysis exists)	TLAA evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA (see Subsection 3.5.2.2.1.6)	C-13, C-21

Table 3.5-1. Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of the GALL Report

ID	Type	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Related Item
15	BWR/PWR	Steel elements: protected by coating	Loss of material due to general, pitting and crevice corrosion in accessible areas only	Protective coating monitoring and maintenance	No	C-12, C-19
16	BWR/PWR	Prestressed containment: tendons and anchorage components	Loss of material due to corrosion of prestressing tendons and anchorage components	Containment ISI	No	C-10
17	BWR/PWR	Concrete elements: foundation, dome, and wall	Scaling, cracking, and spalling due to freeze-thaw; expansion and cracking due to reaction with aggregate	Containment ISI	No, if stated conditions are satisfied for inaccessible areas (see Subsection 3.5.2.2.1.8)	C-01, C-04
18	BWR	Steel elements: vent line bellows, vent headers, downcomers	Cracking due to cyclic loads or Cracking due to SCC	Containment ISI and Containment leak rate test	Yes, detection of aging effects is to be evaluated (see Subsection 3.5.2.2.1.7)	C-20, C-22
19	BWR	Steel elements: Suppression chamber liner	Cracking due to SCC	Containment ISI and Containment leak rate test	No	C-24
20	BWR	Steel elements: drywell head and downcomer pipes	Fretting and lock up due to wear	Containment ISI	No	C-23

Table 3.5-1. Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of the GALL Report

ID	Type	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Related Item
Class I Structures						
21	BWR/PWR	All Groups except Group 2; accessible and inaccessible interior/exterior concrete, steel & lubrite components	All types of aging effects	Structures Monitoring	No, if within the scope of the applicant's structures monitoring program and a plant-specific aging management program is required for inaccessible areas as stated (see Subsection 3.5.2.2.2.1)	T-01, T-03, T-04, T-06, T-11,
22	BWR/PWR	Groups 1 and 3; inaccessible concrete components, such as exterior walls below grade and foundation	Aging of inaccessible concrete areas due to aggressive chemical attack, and corrosion of embedded steel	Examination of representative samples of below-grade concrete, when excavated for any reason, be performed, if the below-grade environment is aggressive	Yes, if environment is aggressive (see Subsection 3.5.2.2.2.2)	T-05, T-07
23	BWR/PWR	Groups 1 and 3; inaccessible concrete components, such as exterior walls below grade and foundation	Aging of inaccessible concrete areas due to leaching of calcium hydroxide	None, if concrete was constructed as stated.	No, if concrete was constructed as stated for inaccessible areas.	T-02

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Table 3.5-1. Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of the GALL Report

ID	Type	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Related Item
24	BWR/PWR	Group 2: all accessible/ inaccessible concrete, metal, and earthen components	All types of aging effects, including loss of material due to abrasion, cavitation, and corrosion	Inspection of Water-Control Structures or FERC/US Army Corps of Engineers dam inspections and maintenance. *and/or Structures Monitoring*	No	T-15, T-16, T-17, T-18, T-19, T-20, T-21, T-22
25	BWR/PWR	Group 1: Fuel pool liners	Cracking due to SCC and loss of material due to pitting and crevice corrosion	Water Chemistry Program and Monitoring of spent fuel pool water level	No	T-14
26	BWR/PWR	Groups 1 and 2: all masonry block walls	Cracking due to restraint shrinkage, creep, and aggressive environment	Masonry Wall */Structures Monitoring*	No	T-12
27	BWR/PWR	Groups 1 and 3: foundation	Cracks and distortion due to increased stress levels from settlement	Structures Monitoring	No, if within the scope of the applicant's structures monitoring program (see Subsection 3.5.2.2.1.2)	T-08

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Table 3.5-1. Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of the GALL Report

ID	Type	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Related Item
28	BWR/PWR	Groups 1-3, foundation	Reduction in foundation strength due to erosion of porous concrete subfoundation	Structures Monitoring	No, if within the scope of the applicant's structures monitoring program (see Subsection 3.5.2.2.1.2)	T-09
29	BWR/PWR	Groups 1: concrete	Reduction of strength and modulus due to elevated temperature	Plant-specific	Yes, for any portions of concrete that exceed specified temperature limits (see Subsection 3.5.2.2.1.3)	T-10
30	BWR/PWR	Groups 3: Tank liners	Cracking due to SCC; Loss of material due to pitting and crevice corrosion	Plant-specific	Yes, plant-specific (see subsection 3.5.2.2.1.7)	T-23
319	BWR/PWR	Group 2: Seals, gaskets, and moisture barriers	Loss of sealing due to deterioration of seals, gaskets, and moisture barriers (caulking, flashing, and other sealants)	Structures Monitoring	No	TP-7
32	BWR/PWR	All groups except 2: Radial beam seats in BWR drywell; RPV support shoes for PWR with nozzle supports; Steam generator supports	Lock-up due to wear	ISI or Structures monitoring	No, if within the scope of the applicant's structures monitoring program or ISI	T-13

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Table 3.5-1. Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of the GALL Report

ID	Type	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Related Item
Component Supports						
33	BWR/PWR	All Groups: support members: anchor bolts, concrete surrounding anchor bolts, welds, grout pad, bolted connections, etc.	Aging of component supports	Structures Monitoring	No, if within the scope of the applicant's structures monitoring program (see Subsection 3.5.2.2.3.1)	T-29, T-30, T-31, TP-6, TP-8
34	BWR/ PWR	All Groups: stainless steel support members: anchor bolts, concrete surrounding anchor bolts, welds, grout pad, bolted connections, etc.	None	None	NA-no AE/M or AMP	TP-4, TP-5
35	BWR/PWR	Groups B1, support members: anchor bolts, welds	Cumulative fatigue damage (CLB fatigue analysis exists)	TLAA evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA (see Subsection 3.5.2.2.3.2)	T-26
36	PWR	All Groups: support members: anchor bolts, welds	Loss of material due to boric acid corrosion	Boric acid corrosion	No	T-25, TP-3
37	BWR/PWR	Groups B1; support members: anchor bolts, welds, spring hangers, guides, stops, and vibration isolators; radial beam seats in BWR drywell, RPV support shoes for PWR with nozzle supports, other supports	Loss of material due to general and pitting corrosion; loss of mechanical function due to corrosion, distortion, dirt, overload, etc.	ISI Structures Monitoring	No No, if within the applicant's structures monitoring program.	T-24, T-28, TP-1, TP-2

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Table 3.5-1. **Summary of Aging Management Programs for Structures and Component Supports Evaluated in Chapters II and III of the GALL Report**

ID	Type	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Related Item
38	BWR/PWR	Group B1: high strength low-alloy bolts (for ASME Class 1 piping suoort)	Cracking due to SCC	Bolting integrity	No	T-27

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Table 3.5-2. FSAR Supplement for Aging Management of Structures and Component Supports

Program	Description of Program	Implementation Schedule*
PWR and BWR Containment		
Containment inservice inspection (Containment ISI)	The ASME Section XI, Subsection IWL program consists of periodic visual inspection of concrete surfaces for reinforced and prestressed concrete containments, and periodic visual inspection and sample tendon testing of unbonded post-tensioning systems for prestressed concrete containments, for signs of degradation, assessment of damage and corrective actions. Measured tendon lift-off forces are compared to predicted tendon forces calculated in accordance with RG 1.35.1. The ASME Section XI, Subsection IWE program consists of periodic visual, surface, and volumetric inspection of pressure retaining components of steel and concrete containments for signs of degradation, assessment of damage and corrective actions. This program is in accordance with ASME Section XI, Subsections IWE and IWL, 1992 edition including 1992 addenda or 1995 edition, including 1996 addenda.	Existing program
Containment leak rate test (LRT)	This program consists of monitoring of leakage rates through containment liner/welds, penetrations, fittings, and other access openings for detecting degradation of containment pressure boundary. Corrective actions are taken if leakage rates exceed acceptance criteria. This program is implemented in accordance with 10 CFR Part 50 Appendix J, RG 1.163 and NEI 94-01, Rev. 0.	Existing program
Protective coating monitoring and maintenance	This program consists of guidance for selection, application, inspection, and maintenance of protective coatings. This program is implemented in accordance with RG 1.54, Rev. 0 or Rev. 1.	Existing program
Class I Structures		
Inspection of water-control structures	The program consists of inspection and surveillance program for dams, slopes, canals, intake structure and other water-control structures associated with emergency cooling water systems or flood protection based on RG 1.127, Rev. 1.	Existing program

Table 3.5-2. FSAR Supplement for Aging Management of Structures and Component Supports (continued)

Program	Description of Program	Implementation Schedule*
Monitoring of leakage in fuel storage facility	This activity consists of periodic monitoring of leak chase system drain lines and leak detection sump of fuel storage facility and refueling channel to detect SCC and crevice corrosion of stainless steel liners. Alternately, the pool water level may be monitored for evidence of leakage. This activity augments the Water Chemistry Program for aging management of the spent fuel pool liner.	Existing program
Water chemistry (BWR/PWR)	To mitigate aging effects on component surfaces that are exposed to water as process fluid, chemistry programs are used to control water impurities (e.g., chloride, fluoride, and sulfate) that accelerate corrosion. The water chemistry program relies on monitoring and control of water chemistry based on EPRI guidelines of TR-103515 for water chemistry in BWRs and TR-102134 for secondary water chemistry in PWRs.	Existing program
Masonry wall	This program consists of inspections, based on IE Bulletin 80-11 and plant-specific monitoring proposed by IN 87-67, for managing cracking of masonry walls.	Existing program
Component Supports		
Inservice inspection (ISI)	This program consists of periodic visual examination of component supports for signs of degradation, evaluation, and corrective actions. This program is in accordance with ASME Section XI, Subsection IWF, 1989 edition through 1995 edition, including 1996 addenda.	Existing program
Boric acid corrosion (PWR)	The program consists of (1) visual inspection of external surfaces that are potentially exposed to borated water leakage, (2) timely discovery of leak path and removal of the boric acid residues, (3) assessment of the damage, and (4) follow-up inspection for adequacy. This program is implemented in response to GL 88-05.	Existing program

Table 3.5-2. FSAR Supplement for Aging Management of Structures and Component Supports (continued)

Program	Description of Program	Implementation Schedule*
Bolting integrity (BWR/PWR)	This program consists of guidelines on materials selection, strength and hardness properties, installation procedures, lubricants and sealants, corrosion considerations in the selection and installation of pressure-retaining bolting for nuclear applications, and enhanced inspection techniques. This program relies on the bolting integrity program delineated in NUREG-1339 and industry's recommendations delineated in EPRI NP-5769, with the exceptions noted in NUREG-1339 for safety-related bolting and in EPRI TR-104213 for pressure retaining bolting and structural bolting.	Existing program
Class I Structures and Component Supports		
Structures monitoring	The program consists of periodic inspection and monitoring the condition of structures and structure component supports to ensure that aging degradation leading to loss of intended functions will be detected and that the extent of degradation can be determined. This program is implemented in accordance with NUMARC 93-01, Rev. 2 and RG 1.160, Rev. 2.	Existing program
PWR and BWR Containment, Class I Structures, and Component Supports		
Quality assurance	The 10 CFR Part 50 App. B program provides for corrective actions, confirmation process, and administrative controls for aging management programs for license renewal. The scope of this existing program will be expanded to include nonsafety-related structures and components that are subject to an AMR for license renewal.	Program should be implemented before the period of extended operation

* An applicant need not incorporate the implementation schedule into its FSAR. However, the reviewer should verify that the applicant has identified and committed in the license renewal application to any future aging management activities to be completed before the period of extended operation. The staff expects to impose a license condition on any renewed license to ensure that the applicant will complete these activities no later than the committed date.

3.6 AGING MANAGEMENT OF ELECTRICAL AND INSTRUMENTATION AND CONTROLS

Item	Locator	Comment	Justification
1	Sections 3.6.2.2.2, 3.6.2.2.3, and 3.6.2.2.4, and Sections 3.6.3.2.2, 3.6.3.2.3, and 3.6.3.2.4 on	These sections are noted as "Deleted." Rather than include these sections as they are presently annotated, delete them entirely.	Some applicants prefer to align the LRA section numbers of Further Evaluation Recommended items so that they correspond to the paragraphs in NUREG-1800. Including these deleted sections would make it more difficult and awkward to achieve one-to-one correspondence.

3.6 AGING MANAGEMENT OF ELECTRICAL AND INSTRUMENTATION AND CONTROLS

Review Responsibilities

Primary - Branches assigned responsibility by PM as described in SRP-LR section 3.0

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3.6.1 Areas of Review

This review plan section addresses the aging management review (AMR) of the electrical and instrumentation and controls (I&C). For a recent vintage plant, the information related to the Electrical and I&C is contained in Chapter 7, "Instrumentation and Controls," and Chapter 8, "Electric Power," of the plant's FSAR, consistent with the "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (NUREG-0800) (Ref. 1). For older plants, the location of applicable information is plant-specific because an older plant's FSAR may have predated NUREG-0800. Typical electrical and I&C components that are subject to an AMR for license renewal are electrical cables and connections.

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The responsible review organization is to review the following LRA AMR and AMP items assigned to it, per SRP-LR section 3.0, for review:

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<u>AMRs</u>

- AMRs consistent with the GALL report, for which further evaluation is not recommended
- AMRs consistent with the GALL report, for which further evaluation is recommended
- AMRs not consistent with the GALL report

<u>AMPs</u>

- AMPs consistent with GALL AMPs (with or without exceptions)
- Plant-specific AMPs

<u>FSAR Supplement</u>

- In addition, the responsible review organization is to review the FSAR supplement associated with each assigned AMP.

3.6.2 Acceptance Criteria

The acceptance criteria for the areas of review describe methods for determining whether the applicant has met the requirements of the NRC's regulations in 10 CFR 54.21.

3.6.2.1 AMR Results Consistent with the GALL Report for Which No Further Evaluation is Recommended

The aging management review and acceptable aging management programs applicable to the electrical and I&C components are described and evaluated in Chapter VI of the GALL report (Ref. 2).

The applicant's LRA should provide sufficient information so that the NRC reviewer is able to confirm that the specific AMR line-item and the associated AMP are consistent with the cited GALL AMR line-item. The staff reviewer should then confirm that the LRA AMR line-item is consistent with the GALL line-item to which it is compared.

If the applicant identifies an exception to the cited GALL AMP, the LRA should include a basis demonstrating how the criteria of 10 CFR 54.21(a)(3) would still be met. The NRC reviewer should then confirm that the AMP with all exceptions would satisfy the criteria of 10 CFR 54.21(a)(3). If, while reviewing the AMP, the reviewer identifies a difference from the GALL AMP, this difference should be reviewed and dispositioned as if it were an exception identified by the applicant in its LRA. The disposition of all LRA-defined exceptions and staff-identified differences should be documented.

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The LRA should identify any enhancements that are needed to permit an existing aging management program to be declared consistent with the GALL AMP to which the LRA AMP is compared. The reviewer is to confirm both that the enhancement, if implemented, would allow the existing plant aging management program to be consistent with the GALL AMP and also that the applicant has a commitment in the FSAR supplement to implement the enhancement prior to the period of extended operation. The reviewer should document the disposition of all enhancements.

3.6.2.2 AMR Results Consistent with the GALL Report for Which Further Evaluation is Recommended

The basic acceptance criteria defined in 3.6.2.1 apply to all of the AMRs and AMPs reviewed as part of this section. In addition, if the GALL AMR line-item to which the LRA AMR line-item is compared identifies that "further evaluation is recommended," then additional criteria apply as identified by the GALL report for each of the following aging effect/aging mechanism combinations.

3.6.2.2.1 Electrical Equipment Subject to Environmental Qualification

Environmental qualification is a TLAA as defined in 10 CFR 54.3. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c)(1). The evaluation of this TLAA is addressed separately in Section 4.4 of this standard review plan.

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3.6.2.2.2 Degradation of Insulator Quality due to Presence of Any Salt Deposits and Surface Contamination; and Loss of Material due to Mechanical Wear

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Degradation of insulator quality due to presence of any salt deposits and surface contamination; and loss of material due to mechanical wear caused by wind blowing on transmission conductors could occur in high voltage insulators. The GALL report recommends further evaluation of a plant-specific aging management program to ensure that this aging effect is adequately managed. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3.6.2.2.3 Loss of Material due to Wind Induced Abrasion and Fatigue; Loss of Conductor Strength due to Corrosion; and Increased Resistance of Connection due to Oxidation or Loss of Pre-load

Loss of material due to wind induced abrasion and fatigue; loss of conductor strength due to corrosion; and increased resistance of connection due to oxidation or loss of pre-load could occur in transmission conductors and connections; and in switchyard bus and connections. The GALL report recommends further evaluation of a plant-specific aging management program to ensure that this aging effect is adequately managed. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3.6.2.2.4 Quality Assurance for Aging Management of Nonsafety-Related Components

Acceptance criteria are described in Branch Technical Position IQMB-1 (Appendix A.2 of this standard review plan).

3.6.2.3 AMR Results Not Consistent with or Not Addressed in GALL Report

Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3.6.2.4 FSAR Supplement

The summary description of the programs and activities for managing the effects of aging for the period of extended operation in the FSAR supplement should be appropriate such that later changes can be controlled by 10 CFR 50.59. The description should contain information associated with the bases for determining that aging effects are managed during the period of extended operation. The description should also contain any future aging management activities, including enhancements and commitments, to be completed before the period of extended operation. Examples of the type of information required are provided in Table 3.6-2 of this standard review plan.

3.6.3 Review Procedures

For each area of review, the following review procedures are to be followed:

3.6.3.1 AMR Results Consistent with the GALL Report for Which No Further Evaluation is Recommended

The applicant may reference the GALL report in its license renewal application, as appropriate, and demonstrate that the aging management reviews and programs at its facility are consistent with those reviewed and approved in the GALL report. The reviewer should not conduct a re-review of the substance of the matters described in the GALL report. If the applicant has provided the information necessary to adopt the finding of program acceptability as described and evaluated in the GALL report, the staff should find acceptable the applicant's reference to GALL in its license renewal application. In making this determination, the reviewer confirms that the applicant has provided a brief description of the system, components, materials, and environment. The reviewer also confirms that the applicant has stated that the applicable aging effects and industry and plant-specific operating experience have been reviewed by the applicant and are evaluated in the GALL Report.

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Furthermore, the reviewer should confirm that the applicant has addressed operating experience identified after the issuance of the GALL report. Performance of this review requires the reviewer to confirm that the applicant has identified those aging effects for the electrical and I&C components that are contained in GALL as applicable to its plant.

The reviewer confirms that the applicant has identified the appropriate AMPs as described and evaluated in the GALL report. If the applicant commits to an enhancement to make its aging management program consistent with a GALL AMP, then the reviewer is to confirm that this enhancement when implemented will indeed make the LRA AMP consistent with the GALL AMP. If an aging management program in the LRA identifies an exception to the GALL AMP to which it is claiming to be consistent, the reviewer is to confirm that the LRA AMP with the exception will satisfy the criteria of 10 CFR 54.21(a)(3). If the reviewer identifies a difference, not identified by the LRA, between the LRA AMP and the GALL AMP, with which the LRA claims to be consistent, the reviewer should confirm that the LRA AMP with this difference satisfies 10 CFR 54.21(a)(3). The reviewer should document the basis for accepting enhancements, exceptions or differences. The AMPs evaluated in GALL pertinent to the electrical and I&C components are summarized in Table 3.6-1 of this standard review plan. In this table, the ID column provides a row identifier useful in matching the information presented in the corresponding table in the GALL report Vol. 1. The Related Item column identifies the item number in the GALL report Vol. 2, Chapters II through VIII, presenting detailed information summarized by this row.

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3.6.3.2 AMR Results Consistent with the GALL Report for Which Further Evaluation is Recommended

The basic review procedures defined in 3.6.3.1 apply to all of the AMRs and AMPs provided in this section. In addition, if the GALL AMR line-item to which the LRA AMR line-item is compared identifies that "further evaluation is recommended," then additional criteria apply as identified by the GALL report for each of the following aging effect/aging mechanism combinations.

3.6.3.2.1 Electrical Equipment Subject to Environmental Qualification

Environmental qualification is a TLAA as defined in 10 CFR 54.3. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c)(1). The staff reviews the evaluation of this TLAA separately following the guidance in Section 4.4 of this standard review plan.

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3.6.3.2.2 Degradation of Insulator Quality due to Presence of Any Salt Deposits and Surface Contamination; and Loss of Material due to Mechanical Wear

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The GALL report recommends a plant-specific aging management program for the management of degradation of insulator quality due to presence of any salt deposits and surface contamination, and loss of material due to mechanical wear caused by wind blowing on transmission conductors in high voltage insulators. The staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

3.6.3.2.3 Loss of Material due to Wind Induced Abrasion and Fatigue; Loss of Conductor Strength due to Corrosion; and Increased Resistance of Connection due to Oxidation or Loss of Pre-load

The GALL report recommends a plant-specific aging management program for the management of loss of material due to wind induced abrasion and fatigue; loss of conductor strength due to corrosion; and increased resistance of connection due to oxidation or loss of pre-load in transmission conductors and connections; and in switchyard bus and connections. The staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

3.6.3.2.4 Quality Assurance for Aging Management of Nonsafety-Related Components

The applicant's aging management programs for license renewal should contain the elements of corrective actions, the confirmation process, and administrative controls. Safety-related components are covered by 10 CFR Part 50, Appendix B, which is adequate to address these program elements. However, Appendix B does not apply to non safety-related components that are subject to an AMR for license renewal. Nevertheless, the applicant has the option to expand the scope of its 10 CFR Part 50, Appendix B program to include these components and address these program elements. If the applicant chooses this option, the reviewer verifies that the applicant has documented such a commitment in the FSAR supplement. If the applicant chooses alternative means, the branch responsible for quality assurance should be requested to review the applicant's proposal on a case-by-case basis.

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3.6.3.3 AMR Results Not Consistent with or Not Addressed in GALL Report

The reviewer should confirm that the applicant, in the license renewal application, has identified applicable aging effects, listed the appropriate combination of materials and environments, and aging management programs that will adequately manage the aging effects. The aging management program credited could be an AMP that is described and evaluated in the GALL report or a plant-specific program. Review procedures are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3.6.3.4 FSAR Supplement

The reviewer confirms that the applicant has provided information equivalent to that in Table 3.6-2 in the FSAR supplement for aging management of the Electrical and I&C System for license renewal. The reviewer also confirms that the applicant has provided information equivalent to that in Table 3.6-2 in the FSAR supplement for SRP-LR Subsection 3.6.3.3, "Aging Management Evaluations that Are Different from or Not Addressed in the GALL Report."

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The staff expects to impose a license condition on any renewed license to require the applicant to update its FSAR to include this FSAR supplement at the next update required pursuant to 10 CFR 50.71(e)(4). As part of the license condition until the FSAR update is complete, the applicant may make changes to the programs described in its FSAR supplement without prior NRC approval, provided that the applicant evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59.

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As noted in Table 3.6-2, an applicant need not incorporate the implementation schedule into its FSAR. However, the reviewer should confirm that the applicant has identified and committed in the license renewal application to any future aging management activities, including enhancements and commitments to be completed before the period of extended operation. The staff expects to impose a license condition on any renewed license to ensure that the applicant will complete these activities no later than the committed date.

3.6.4 Evaluation Findings

If the reviewer has confirmed that the applicant has provided information sufficient to satisfy the provisions of this review plan section, an evaluation finding similar to the following text should be included in the staff's safety evaluation report:

> On the basis of its review, the staff concludes that the applicant has adequately identified the aging effects and the AMPs credited with managing these aging effects for the engineered safety features systems, such that there is reasonable assurance that the component intended functions will be maintained consistent with the CLB during the period of extended operation. The staff also reviewed the applicable FSAR supplement program descriptions and concludes that the FSAR supplement provides an adequate program description of the AMPs credited for managing aging effects, as required by 10 CFR 54.21(d).

3.6.5 Implementation

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the NRC's regulations, the method described herein will be used by the staff in its evaluation of conformance with NRC regulations.

3.6.6 References

1. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, July 1981.

2. Draft NUREG-1801, "Generic Aging Lessons Learned (GALL)," U.S. Nuclear Regulatory Commission, Revision 1, September 2005.

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Table 3.6-1. Summary of Aging Management Programs for the Electrical Components Evaluated in Chapter VI of the GALL Report

ID	Type	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Related Item
1	BWR/PWR	Electrical equipment subject to 10 CFR 50.49 environmental qualification (EQ) requirements	Degradation due to various aging mechanisms	Environmental qualification of electric components	Yes, TLAA (see Subsection 3.6.2.2.1)	L-05
2	BWR/PWR	Electrical cables, connections and fuse holders (insulation) not subject to 10 CFR 50.49 EQ requirements	Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced insulation resistance (IR); electrical failure due to thermal/ thermoxidative degradation of organics; radiolysis and photolysis (ultraviolet [UV] sensitive materials only) of organics; radiation-induced oxidation; and moisture intrusion	Aging management program for electrical cables and connections not subject to 10 CFR 50.49 EQ requirements	No	L-01, LP-03

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Table 3.6-1. Summary of Aging Management Programs for the Electrical Components Evaluated in Chapter VI of the GALL Report

ID	Type	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Related Item
3	BWR/PWR	Electrical cables and connections used in instrumentation circuits not subject to 10 CFR 50.49 EQ requirements that are sensitive to reduction in conductor insulation resistance (IR)	Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced IR; electrical failure due to thermal/thermoxidative degradation of organics; radiation-induced oxidation; and moisture intrusion	Aging management program for electrical cables and connections used in instrumentation circuits not subject to 10 CFR 50.49 EQ requirements	No	L-02
4	BWR/PWR	Inaccessible medium-voltage (2 kV to 15 kV) cables (e.g., installed in conduit or direct buried) not subject to 10 CFR 50.49 EQ requirements	Formation of water trees, localized damage leading to electrical failure (breakdown of insulation); water trees due to moisture intrusion	Aging management program for inaccessible medium-voltage cables not subject to 10 CFR 50.49 EQ requirements	No	L-03
5	PWR	Electrical connectors not subject to 10 CFR 50.49 EQ requirements that are exposed to borated water leakage	Corrosion of connector contact surfaces due to intrusion of borated water	Boric acid corrosion	No	L-04
6	BWR/PWR	Fuse holders – metallic clamp	Fatigue due to ohmic heating, thermal cycling, electrical transients, frequent manipulation, vibration, chemical contamination, corrosion, and oxidation	Aging management program for fuse holders	No	LP-01

Table 3.6-1. Summary of Aging Management Programs for the Electrical Components Evaluated in Chapter VI of the GALL Report

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ID	Type	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Related Item
7	BWR/PWR	Phase bus - Bus/connections	Loosening of bolted connections due to thermal cycling and ohmic heating	Aging management program for bus duct	No	LP-04
8	BWR/PWR	Phase bus – Insulation/insulators	Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced insulation resistance (IR); electrical failure due to thermal/ thermoxidative degradation of organics/thermoplastics, radiation-induced oxidation; moisture/debris intrusion, and ohmic heating	Aging management program for bus duct	No	LP-05
9	BWR/PWR	Phase bus – Enclosure assemblies	Loss of material due to general corrosion	Structures Monitoring Program	No	LP-06
10	BWR/PWR	Phase bus – Enclosure assemblies	Hardening and loss of strength/ elastomers degradation	Structures Monitoring Program	No	LP-10

Table 3.6-1. Summary of Aging Management Programs for the Electrical Components Evaluated in Chapter VI of the GALL Report

ID	Type	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Related Item
11	BWR/PWR	High voltage insulators	Degradation of insulation quality due to presence of any salt deposits and surface contamination; Loss of material caused by mechanical wear due to wind blowing on transmission conductors	Plant specific	Yes, plant specific (see subsection 3.6.2.2.5)	LP-07, LP-11
12	BWR/PWR	Transmission conductors and connections, Switchyard bus and connections	Loss of material due to wind induced abrasion and fatigue; Loss of conductor strength due to corrosion; Increased resistance of connection due to oxidation or loss of pre-load	Plant specific	Yes, plant specific (see subsection 3.6.2.2.6)	LP-08, LP-09
13	BWR/PWR	Cable Connections – Metallic parts	Loosening of bolted connections due to thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination, corrosion, and oxidation	Aging management program for electrical cable connections not subject to 10 CFR 50.49 environmental qualification requirements	No	LP-12
14	BWR/PWR	Fuse Holders (Not Part of a Larger Assembly) Insulation material	None	None	N/A	LP-02

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Table 3.6-2. FSAR Supplement for Aging Management of Electrical and Instrumentation and Control System

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Program	Description of Program	Implementation Schedule*
Aging management program for non-environmentally qualified electrical cables and connections exposed to an adverse localized environment caused by heat, radiation, or moisture.	Accessible electrical cables and connections installed in adverse localized environments are visually inspected at least once every 10 years for cable and connection jacket surface anomalies, such as embrittlement, discoloration, cracking, swelling, or surface contamination, which are precursor indications of conductor insulation aging degradation from heat, radiation or moisture. An adverse localized environment is a condition in a limited plant area that is significantly more severe than the specified service condition for the electrical cable or connection.	First inspection for license renewal should be completed before the period of extended operation
Aging management program for non-environmentally qualified electrical cables and connections used in instrumentation circuits that are sensitive to reduction in conductor insulation resistance, and are exposed to an adverse localized environment caused by heat, radiation, or moisture.	Electrical cables and connections used in circuits with sensitive, low-level signals, such as radiation monitoring and nuclear instrumentation, are calibrated as part of the instrumentation loop calibration at the normal calibration frequency, which provides sufficient indication of the need for corrective actions based on acceptance criteria related to instrumentation loop performance. The review of calibration results is performed once every 10 years. In cases where cables are not part of calibration or surveillance program, a proven cable test (such as insulation resistance tests, time domain reflectometry tests, or other tests judged to be effective) for detecting deterioration of the insulation system are performed. The test frequency is based on engineering evaluation not to exceed 10 years.	First review of calibration results or cable tests for license renewal should be completed before the period of extended operation

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Table 3.6-2. FSAR Supplement for Aging Management of Electrical and Instrumentation and Control System

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Program	Description of Program	Implementation Schedule*
Aging management program for non-environmentally qualified inaccessible medium-voltage cables exposed to an adverse localized environment caused by moisture and voltage exposure	In-scope, medium-voltage cables exposed to significant moisture and significant voltage are tested at least once every 10 years to provide an indication of the condition of the conductor insulation. The specific type of test performed will be determined prior to the initial test, and is to be a proven test for detecting deterioration of the insulation system due to wetting, such as power factor, partial discharge, or polarization index, as described in EPRI TR-103834-P1-2, or other testing that is state-of-the-art at the time the test is performed. Significant moisture is defined as periodic exposures that last more than a few days (e.g., cable in standing water). Periodic exposures that last less than a few days (e.g., normal rain and drain) are not significant. Significant voltage exposure is defined as being subjected to system voltage for more than 25% of the time. The moisture and voltage exposures described as significant in these definitions are not significant for medium-voltage cables that are designed for these conditions (e.g., continuous wetting and continuous energization are not significant for submarine cables). In addition, inspection for water collection is performed following significant moisture exposure due to weather related events. However, the inspection frequency should not exceed two years.	First tests or first inspections for license renewal should be completed before the period of extended operation

Table 3.6-2. FSAR Supplement for Aging Management of Electrical and Instrumentation and Control System (continued)

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Program	Description of Program	Implementation Schedule*
Boric acid corrosion.	The program consists of (1) visual inspection of external surfaces that are potentially exposed to borated water leakage, (2) timely discovery of leak path and removal of the boric acid residues, (3) assessment of the damage, and (4) follow-up inspection for adequacy. This program is implemented in response to GL 88-05.	Existing program
Plant-specific AMP	The description should contain information associated with the basis for determining that aging effects will be managed during the period of extended operation.	Program should be implemented before the period of extended operation.
Aging management program for fuse holders	Fuse holders within the scope of license renewal will be tested at least once every 10 years to provide an indication of degradation of the metallic clamp portion of the fuse holders. Testing may include thermography, contact resistance testing, or other appropriate testing methods.	First tests for license renewal should be completed before the period of extended operation
Aging management program for bus ducts	Internal portions of bus ducts are visually inspected at least once every 10 years for cracks, corrosion, foreign debris, excessive dust buildup, and evidence of water intrusion. The bus insulating system is inspected for signs of embrittlement, cracking, melting, swelling, or discoloration, which may indicates overheating or aging degradation. The (internal) bus supports are inspected for structural integrity and signs of cracks. A sample of accessible bolted connections on the internal bus work is checked for proper torque, or check for connection resistance.	First inspection for license renewal should be completed before the period of extended operation.
Aging management program for non-environmentally qualified electrical cable connections	A representative sample of electrical cable connections within the scope of license renewal will be tested at least once every 10 years. Testing may include thermography, contact resistance testing, or other appropriate testing methods.	First tests for license renewal should be completed before the period of extended operation
Structures monitoring	The program consists of periodic inspection and monitoring the condition of structures and structure component supports to ensure that aging degradation leading to loss of intended functions will be detected and that the extent of degradation can be determined. This program is implemented in accordance with NUMARC 93-01, Rev. 2 and RG 1.160, Rev. 2.	Existing program

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Table 3.6-2. FSAR Supplement for Aging Management of Electrical and Instrumentation and Control System (continued)

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Program	Description of Program	Implementation Schedule*
Quality assurance	The 10 CFR Part 50, Appendix B program provides for corrective actions, the confirmation process, and administrative controls for aging management programs for license renewal. The scope of this existing program will be expanded to include non safety-related structures and components that are subject to an AMR for license renewal.	Program should be implemented before the period of extended operation

* An applicant need not incorporate the implementation schedule into its FSAR. However, the reviewer should verify that the applicant has identified and committed in the license renewal application to any future aging management activities to be completed before the period of extended operation. The staff expects to impose a license condition on any renewed license to ensure that the applicant will complete these activities no later than the committed date.

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CHAPTER 4

TIME-LIMITED AGING ANALYSES

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4.1 IDENTIFICATION OF TIME-LIMITED AGING ANALYSES

Item	Locator	Comment	Justification
1	Table 4.1-3,	Split up the multiple TLAAs listed in rows one and five of the table, i.e., Make "Low-temperature overpressure protection (LTOP) analyses," "Flow induced vibration endurance limit," transient cycle count assumptions," and "ductility reduction of fracture toughness for the reactor vessel internals" separate lines on the table.	These TLAAs will be easier to address on a one-by-one basis, for example, by reference to a specific section of the LRA. Lumping them together has necessitated referencing multiple LRA sections for one line item.

4.1 IDENTIFICATION OF TIME-LIMITED AGING ANALYSES

Review Responsibilities

Primary - Branch responsible for the TLAA issues
Secondary - Other branches responsible for engineering, as appropriate

4.1.1 Areas of Review

This review plan section addresses the identification of time-limited aging analyses (TLAAs). The technical review of TLAAs is addressed in section 4.2 through 4.7. As explained in more detail below, the list of TLAAs are certain plant-specific safety analyses that are based on an explicitly assumed 40-year plant life (for example, aspects of the reactor vessel design). Pursuant to 10 CFR 54.21(c)(1), a license renewal applicant is required to provide a list of TLAAs, as defined in 10 CFR 54.3. The area relating to the identification of TLAAs is reviewed.

TLAAs may have developed since issuance of a plant's operating license. As indicated in 10 CFR 54.30, the adequacy of the plant's CLB, which includes TLAAs, is not an area within the scope of the license renewal review. Any question regarding the adequacy of the CLB must be addressed under the backfit rule (10 CFR 50.109) and is separate from the license renewal process.

In addition, pursuant to 10 CFR 54.21(c)(2), an applicant must provide a list of plant-specific exemptions granted under 10 CFR 50.12 that are based on TLAAs. However, the initial license renewal applicants have found no such exemptions for their plants.

It is an applicant's option to include more analyses than those required by 10 CFR 54.21(c)(1). The staff should focus its review to confirm that the applicant did not omit any TLAAs, as defined in 10 CFR 54.3.

4.1.2 Acceptance Criteria

The acceptance criteria for the areas of review described in Subsection 4.1.1 of this review plan section delineate acceptable methods for meeting the requirements of the NRC's regulations in 10 CFR 54.21(c)(1). For the applicant's list of exemptions to be acceptable, the staff should have reasonable assurance that there has been no omission of TLAAs from that list.

Pursuant to 10 CFR 54.3, TLAAs are those licensee calculations and analyses that:

1. Involve systems, structures, and components within the scope of license renewal, as delineated in 10 CFR 54.4(a);

2. Consider the effects of aging;

3. Involve time-limited assumptions defined by the current operating term, for example, 40 years;

4. Were determined to be relevant by the licensee in making a safety determination;

5. Involve conclusions or provide the basis for conclusions related to the capability of the system, structure, or component to perform its intended function(s), as delineated in 10 CFR 54.4(b); and

6. Are contained or incorporated by reference in the CLB.

4.1.3 Review Procedures

For each area of review described in Subsection 4.1.1, the reviewer should adhere to the following review procedures:

The reviewer should use the plant UFSAR and other CLB documents, such as staff SERs, in performing the review. The reviewer should select analyses that the applicant did not identify as TLAAs that are likely to meet the six criteria identified in Subsection 4.1.2. The reviewer verifies that the selected analyses, not identified by the applicant as TLAAs, do not meet at least one of the following criteria (Ref. 1).

Sections 4.2 through 4.6 identify typical types of TLAAs for most plants. Information on the licensee's methodology for identifying TLAAs may also be useful in identifying calculations that did not meet six criteria below.

1. Involve systems, structures, and components within the scope of license renewal, as delineated in 10 CFR 54.4(a). Chapter 2 of this standard review plan provides the reviewer guidance on the scoping and screening methodology, and on plant level and various system level scoping results.

2. Consider the effects of aging. The effects of aging include, but are not limited to: loss of material, loss of toughness, loss of prestress, settlement, cracking, and loss of dielectric properties.

3. Involve time-limited assumptions defined by the current operating term (for example, 40 years). The defined operating term should be explicit in the analysis. Simply asserting that a component is designed for a service life or plant life is not sufficient. The assertion should be supported by calculations or other analyses that explicitly include a time limit.

4. Were determined to be relevant by the licensee in making a safety determination. Relevancy is a determination that the applicant should make based on a review of the information available. A calculation or analysis is relevant if it can be shown to have a direct bearing on the action taken as a result of the analysis performed. Analyses are also relevant if they provide the basis for a licensee's safety determination and, in the absence of the analyses, the licensee might have reached a different safety conclusion.

5. Show capability of the system, structure, or component to perform its intended function(s), as delineated. Involve conclusions or provide the basis for conclusions related to 10 CFR 54.4(b). Analyses that do not affect the intended functions of systems, structures, or components are not TLAAs.

6. Are contained or incorporated by reference in the CLB. The CLB includes the technical specifications as well as design basis information (as defined in 10 CFR 50.2) or licensee commitments documented in the plant-specific documents contained or incorporated by reference in the CLB including, but not limited to: the FSAR, NRC SERs, the fire protection

plan/hazards analyses, correspondence to and from the NRC, the quality assurance plan, and topical reports included as references to the FSAR. Calculations and analyses that are not in the CLB or not incorporated by reference are not TLAAs. If a code of record is in the FSAR for particular groups of structures or components, reference material includes all calculations called for by that code of record for those structures and components.

TLAAs that need to be addressed are not necessarily those analyses that have been previously reviewed or approved by the NRC. The following examples illustrate TLAAs that need to be addressed and were not previously reviewed and approved by the NRC:

- The FSAR states that the design complies with a certain national code and standard. A review of the code and standard reveals that it calls for an analysis or calculation. Some of these calculations or analysis will be TLAAs. The actual calculation was performed by the licensee to meet the code and standard. The specific calculation was not referenced in the FSAR. The NRC had not reviewed the calculation.

- In response to a generic letter, a licensee submitted a letter to the NRC committing to perform a TLAA that would address the concern in the generic letter. The NRC had not documented a review of the licensee's response and had not reviewed the actual analysis.

The following examples illustrate analyses that are *not* TLAAs and need not be addressed under 10 CFR 54.21(c):

- Population projections (Section 2.1.3 of NUREG-0800) (Ref. 2).

- Cost-benefit analyses for plant modifications.

- Analysis with time-limited assumptions defined short of the current operating term of the plant, for example, an analysis for a component based on a service life that would not reach the end of the current operating term.

The number and type of TLAAs vary depending on the plant-specific CLB. All six criteria set forth in 10 CFR 54.3 (and repeated in Subsection 4.1.2 of this review plan section) must be satisfied to conclude that a calculation or analysis is a TLAA. Table 4.1-1 provides examples of how these six criteria may be applied (Ref. 1). Table 4.1-2 provides a list of potential TLAAs (60 FR 22480). Table 4.1-3 provides a list of other plant-specific TLAAs that have been identified by the initial license renewal applicants. Tables 4.1-2 and 4.1-3 provide examples of analyses that potentially could be TLAAs for a particular plant. However, TLAAs are plant-specific and depend on an applicant's CLB. It is not expected that all applicants would identify all the analyses in these tables as TLAAs for their plants. Also, an applicant may have performed specific TLAAs for its plant that are not shown in these tables.

Staff members from other branches of the Division of Engineering will be reviewing the application in their assigned areas without examining the identification of TLAAs. However, they may come across situations in which they may question why the applicant did not identify certain analyses as TLAAs. The reviewer should coordinate the resolution of any such questions with these other staff members and determine whether these analyses should be evaluated as TLAAs.

In order to determine whether there is reasonable assurance that the applicant has identified the TLAAs for its plant, the reviewer should find that the analyses omitted from the applicant's list are not TLAAs.

Should an applicant identify a TLAA that is also a basis for a plant-specific exemption granted pursuant to 10 CFR 50.12 and the exemption is in effect, the reviewer verifies that the applicant has also identified that exemption pursuant to 10 CFR 54.21(c)(2). However, the initial license renewal applicants have found no such exemptions for their plants.

4.1.4 Evaluation Findings

The reviewer determines whether the applicant has provided sufficient information to satisfy the provisions of this review plan section, and whether the staff's evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report, as appropriate:

> The staff concludes that the applicant has provided an acceptable list of TLAAs as defined in 10 CFR 54.3, and that no 10 CFR 50.12 exemptions have been granted on the basis of a TLAA, as defined in 10 CFR 54.3.

4.1.5 Implementation

Except in those cases in which the applicant proposes an acceptable alternative method, the method described herein will be used by the staff in its evaluation of conformance with NRC regulations.

4.1.6 References

1. NEI 95-10, Revision 3, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 – The License Renewal Rule," Nuclear Energy Institute, March 2001.

2. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports Nuclear Power Plants," July 1981.

Table 4.1-1. Identification of Potential Time-Limited Aging Analyses and Basis for Disposition

Example	Disposition
NRC correspondence requests a utility to justify that unacceptable cumulative wear did not occur during the design life of control rods.	Does not qualify as a TLAA because the design life of control rods is less than 40 years. Therefore, does not meet criterion (3) of the TLAA definition in 10 CFR 54.3.
Maximum wind speed of 100 mph is expected to occur once per 50 years.	Not a TLAA because it does not involve an aging effect.
Correspondence from the utility to the NRC states that the membrane on the containment basemat is certified by the vendor to last for 40 years.	The membrane was not credited in any safety evaluation, and therefore the analysis is not considered a TLAA. This example does not meet criterion (4) of the TLAA definition in 10 CFR 54.3.
Fatigue usage factor for the pressurizer surge line was determined not to be an issue for the current license period in response to NRC Bulletin 88-11.	This example is a TLAA because it meets all 6 criteria in the definition of TLAA in 10 CFR 54.3. The utility's fatigue design basis relies on assumptions defined by the 40-year operating life for this component, which is the current operating term.
Containment tendon lift-off forces are calculated for the 40-year life of the plant. These data are used during Technical Specification surveillance for comparing measured to predicted lift-off forces.	This example is a TLAA because it meets all 6 criteria of the TLAA definition in 10 CFR 54.3. The lift-off force curves are currently limited to 40-year values, and are needed to perform a required Technical Specification surveillance.

Table 4.1-2. Potential Time-Limited Aging Analyses

Reactor vessel neutron embrittlement
Concrete containment tendon prestress
Metal fatigue
Environmental qualification of electrical equipment
Metal corrosion allowance
Inservice flaw growth analyses that demonstrate structure stability for 40 years
Inservice local metal containment corrosion analyses
High-energy line-break postulation based on fatigue cumulative usage factor

Table 4.1-3. Additional Examples of Plant-Specific TLAAs as Identified by the Initial License Renewal Applicants

Intergranular separation in the heat-affected zone (HAZ) of reactor vessel low-alloy steel under austenitic SS cladding. ~~Low-temperature overpressure protection (LTOP) analyses~~
Low-temperature overpressure protection (LTOP) analyses
Fatigue analysis for the main steam supply lines to the turbine-driven auxiliary feedwater pumps
Fatigue analysis of the reactor coolant pump flywheel
Fatigue analysis of polar crane
Flow-induced vibration endurance limit~~, transient cycle count assumptions, and ductility reduction of fracture toughness~~ for the reactor vessel internals
Transient cycle count assumptions for the reactor vessel internals
Ductility reduction of fracture toughness for the reactor vessel internals
Leak before break
Fatigue analysis for the containment liner plate
Containment penetration pressurization cycles
Reactor vessel circumferential weld inspection relief (BWR)

4.2 REACTOR VESSEL NEUTRON EMBRITTLEMENT ANALYSIS

Item	Locator	Comment	Justification
1	4.2.2.1.2	Correct typographical error in last sentence.	Editorial
2	4.2.3.1.1.2.2	Provide flexibility in information provided for equivalent margins analysis	Many Applicants will not have fluence predictions at 1-inch, but rather at ¾-T. The applicant should provide the ¼-T and ¾-T data at the time of submittal. This is based upon the NRC's request in an RAI for Farley.

4.2 REACTOR VESSEL NEUTRON EMBRITTLEMENT ANALYSIS

Review Responsibilities

Primary - Branch responsible for the TLAA issues
Secondary - Branch responsible for reactor systems

4.2.1 Areas of Review

During plant service, neutron irradiation reduces the fracture toughness of ferritic steel in the reactor vessel beltline region of light-water nuclear power reactors. Areas of review to ensure that the reactor vessel has adequate fracture toughness to prevent brittle failure during normal and off-normal operating conditions are (1) upper-shelf energy, (2) PTS for PWRs, (3) heat-up and cool-down (pressure-temperature limits) curves, and (4) BWR Vessel and Internals Project (VIP) VIP-05 analysis for elimination of circumferential weld inspection and analysis of the axial welds.

The adequacy of the analyses for these four areas is reviewed for the period of extended operation.

The branch responsible for reactor systems should review neutron fluence and dosimetry information in the application.

4.2.2 Acceptance Criteria

The acceptance criteria for the areas of review described in Subsection 4.2.1 of this review plan section delineate acceptable methods for meeting the requirements of the NRC's regulation in 10 CFR 54.21(c)(1).

4.2.2.1 Time-Limited Aging Analysis

Pursuant to 10 CFR 54.21(c)(1)(i) - (iii), an applicant must demonstrate one of the following:

(i) The analyses remain valid for the period of extended operation;

(ii) The analyses have been projected to the end of the extended period of operation; or

(iii) The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

For the first three areas of review for the analysis of reactor vessel neutron embrittlement, the specific acceptance criteria depend on the applicant's choice of 10 CFR 54.21(c)(1)(i), (ii), or (iii).

4.2.2.1.1 Upper-Shelf Energy

10 CFR Part 50 Appendix G (Ref. 1) paragraph IV.A.1 requires that the reactor vessel beltline materials must have a Charpy upper-shelf energy of no less than 68 J (50 ft-lb) throughout the life of the reactor vessel, unless otherwise approved by the NRC. An applicant may take any one of the following three approaches:

4.2.2.1.1.1 10 CFR 54.21(c)(1)(i)

The existing upper-shelf energy analysis remains valid during the period of extended operation because the neutron fluence projected to the end of the period of extended operation is bound by the fluence assumed in the existing analysis.

4.2.2.1.1.2 10 CFR 54.21(c)(1)(ii)

The upper-shelf energy is reevaluated to consider the period of extended operation in accordance with 10 CFR Part 50, Appendix G.

4.2.2.1.1.3 10 CFR 54.21(c)(1)(iii)

Acceptance criteria under 10 CFR 54.21(c)(1)(iii) have yet to be developed. They will be evaluated on a case-by-case basis to ensure that the aging effects will be managed such that the intended function(s) will be maintained during the period of extended operation.

4.2.2.1.2 Pressurized Thermal Shock (for PWRs)

For PWRs, 10 CFR 50.61 (Ref. 2) requires that the "reference temperature" for reactor vessel beltline materials evaluated at end of life (EOL) fluence, RT_{PTS}, be less than the "PTS screening criteria" at the expiration date of the operating license, unless otherwise approved by the NRC. The "PTS screening criteria" are 132°C (270°F) for plates, forgings, and axial weld materials, and 149°C (300°F) for circumferential weld materials. The regulations require updating of the PTS assessment upon a request for a change in the expiration date of a facility's operating license. Therefore, the RT_{PTS} value must be calculated for the entire life of the facility, including the period of extended operation. The PTS TL~~L~~AA may be handled as follows.

4.2.2.1.2.1 10 CFR 54.21(c)(1)(i)

The existing PTS analysis remains valid during the period of extended operation because the neutron fluence projected to the end of the period of extended operation is bound by the fluence assumed in the existing analysis.

4.2.2.1.2.2 10 CFR 54.21(c)(1)(ii)

The PTS analysis is reevaluated to consider the period of extended operation in accordance with 10 CFR 50.61. An analysis is performed in accordance with RG 1.154 (Ref. 3) if the "PTS screening criteria" in 10 CFR 50.61 are exceeded during the period of extended operation.

4.2.2.1.2.3 10 CFR 54.21(c)(1)(iii)

The staff position for license renewal on this option is described in a May 27, 2004 letter from L.A.Reyes (EDO) to the Commission (Ref. 13) which states that if the applicant does not extend the TLAA, the applicant should provide an assessment of the current licensing basis TLAA for PTS, a discussion of the flux reduction program implemented in accordance with 10 CFR 50.61(b)(3), if necessary, and an identification of the viable options that exist for managing the aging effect in the future.

4.2.2.1.3 Pressure-Temperature (P-T) Limits

10 CFR Part 50, Appendix G (Ref. 1) requires that the reactor pressure vessel (RPV) be maintained within established pressure-temperature (P-T) limits including during any condition of normal operation. This includes heatup and cooldown. These limits specify the maximum allowable pressure as a function of reactor coolant temperature. As the reactor pressure vessel becomes embrittled and its fracture toughness is reduced, the allowable pressure (given the required minimum temperature) is reduced. P-T limits may be handled as follows.

4.2.2.1.3.1 10 CFR 54.21(c)(1)(i)

The existing P-T limits are valid during the period of extended operation because the neutron fluence projected to the end of the period of extended operation is bound by the fluence assumed in the existing analysis.

4.2.2.1.3.2 10 CFR 54.21(c)(1)(ii)

The P-T limits are reevaluated to consider the period of extended operation in accordance with 10 CFR Part 50, Appendix G (Ref. 1).

4.2.2.1.3.3 10 CFR 54.21(c)(1)(iii)

Not applicable: Updated P-T limits for the period of extended operation must be available prior to entering the period of extended operation. (It is not necessary to implement P-T limits to carry the reactor vessel through 60 years at the time of application. The updated limits must be contained in a pressure-temperature limit report (PTLR) or in the technical specification (TS) prior to the period of extended operation.)

4.2.2.1.4 Elimination of Circumferential Weld Inspection (for BWRs)

Some BWRs have an approved technical alternative which eliminates the reactor vessel circumferential shell weld inspections for the current license term because they satisfy the limiting conditional failure probability for the circumferential welds at the expiration of the current license, based on BWRVIP-05 and the extent of neutron embrittlement (Refs. 4-6). An applicant for renewal of a license to operate such a BWR may provide justification to extend this relief into the period of extended operation in accordance with BWRVIP-74 (Ref 7). The staff's review of BWRVIP-74 (Ref. 7) is contained in an October 18, 2001 letter to C.Terry, BWRVIP Chairman (Ref. 11). Section A.4.5 of Report BWRVIP-74 indicates that the staff's SER conservatively evaluated BWR RPV's to have 64 effective full power years (EFPY), which is 10 EFPY greater than what is realistically expected for the end of the license renewal period. Since this was a generic analysis, a licensee relying on BWRVIP-74 should provide plant-specific information to demonstrate that the circumferential beltline weld materials meet the criteria specified in the report and that operator training and procedures will be utilized during the license renewal term to limit the frequency for cold over-pressure events.

4.2.2.1.5 Axial Welds (for BWRs)

The staff's SER contained in a letter to Carl Terry dated March 7, 2000, "Supplement to Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report" (Ref. 8) discussed the staff's concern related to RPV failure frequency for axial welds and the BWRVIP's analysis of the RPV failure frequency of axial welds. The SER indicates that the RPV failure frequency due to failure of the limiting axial welds in the BWR fleet at the end of 40 years of operation is less than 5 x 10^{-6} per reactor year, given the assumptions on flaw density, distribution, and location described in the SER. Since the BWRVIP analysis was generic, a licensee relying on BWRVIP-74 should monitor axial beltline weld embrittlement. The applicant may provide plant-specific information to demonstrate that the axial beltline weld materials at the extended period of operation meet the criteria specified in the report or have a program to monitor axial weld embrittlement relative to the values specified by the staff in its May 7, 2000, (Ref. 8) letter.

4.2.2.2 FSAR Supplement

The specific criterion for meeting 10 CFR 54.21(d) is:

The summary description of the evaluation of TLAAs for the period of extended operation in the FSAR supplement is appropriate such that later changes can be controlled by 10 CFR 50.59. The description should contain information associated with the TLAAs regarding the basis for determining that the applicant has made the demonstration required by 10 CFR 54.21(c)(1).

4.2.3 Review Procedures

For each area of review described in Subsection 4.2.1, the following review procedures should be followed.

4.2.3.1 Time-Limited Aging Analysis

For the first three areas of review for the analysis of reactor vessel neutron embrittlement, the review procedures depend on the applicant's choice of 10 CFR 54.21(c)(1)(i), (ii), or (iii). For each area, the applicant's three options under section 54.21(c)(1) are discussed in turn, as follows.

4.2.3.1.1 Upper-Shelf Energy

4.2.3.1.1.1 10 CFR 54.21(c)(1)(i)

The projected neutron fluence at the end of the period of extended operation is reviewed to verify that it is bound by the fluence assumed in the existing upper-shelf energy analysis.

4.2.3.1.1.2 10 CFR 54.21(c)(1)(ii)

The documented results of the revised upper-shelf energy analysis based on the projected neutron fluence at the end of the period of extended operation is reviewed for compliance with 10 CFR Part 50, Appendix G. The applicant may use RG 1.99 Rev. 2 (Ref. 9) to project upper-shelf energy to the end of the period of extended operation. The applicant may also use ASME Code Section XI Appendix K (Ref. 10) for the purpose of performing an equivalent margins analysis to demonstrate that adequate protection for ductile failure will be maintained to the end of the period of extended operation. The staff should review the applicant's methodology for this evaluation.

The staff should confirm that the applicant has provided sufficient information for all Upper Shelf Energy (USE) and/or equivalent margins analysis calculations for the period of extended operation as follows:

Neutron Fluence: The applicant should have identified: (1) the neutron fluence at the 1/4T location for each beltline material at the expiration of the license renewal period; (2) the methodology used in determining the neutron fluence, and identified (3) whether the methodology followed the guidance in Regulatory Guide (RG) 1.190 (Ref. 12).

To confirm that the USE analysis meets the requirements of Appendix G of 10 CFR Part 50 at the end of the license renewal period, the staff should determine whether:

1. For each beltline material, the applicant has provided the unirradiated Charpy USE,and the projected Charpy USE at the end of the license renewal period, and whether the drop in Charpy USE was determined using the limit lines in Figure 2 of RG 1.99, Revision 2 or from surveillance data and the percentage copper.

2. If an equivalent margins analysis was used to demonstrate compliance with the USE requirements in Appendix G of 10 CFR Part 50, the applicant has provided the analysis or identified an approved topical report that contains the analysis. Information the staff will consider to assess the equivalent margins analysis includes: the unirradiated USE (if available) for the limiting material, its copper content, the fluence (1/4T, and at ***3/4T or*** 1 inch depth), the EOLE USE (if available), the operating temperature in the downcomer at full power, the vessel radius, the vessel wall thickness, the J-applied analysis for Service Level C and D, the vessel accumulation pressure, and the vessel bounding heatup/cooldown rate during normal operation.

For Boiling Water Reactors, the staff should confirm that the beltline materials are evaluated in accordance with Renewal Applicant Action Items 10, 11 and 12 in the staff's SER, for BWRVIP-74 (Letter to C. Terry dated October 18, 2001) (Ref.11). The applicant should also identify whether there are two or more surveillance material samples available that are relevant to the RPV beltline materials. If there are two or more data points for a surveillance material, the applicant should provide analyses of the data to determine whether the data is consistent with the RG 1.99, Revision 2 methodology that was utilized in the BWRVIP-74 analyses.

4.2.3.1.1.3 10 CFR 54.21(c)(1)(iii)

The applicant's proposal to demonstrate that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation will be reviewed on a case-by-case basis.

4.2.3.1.2 Pressurized Thermal Shock (for PWRs)

4.2.3.1.2.1 10 CFR 54.21(c)(1)(i)

The projected neutron fluence at the end of the period of extended operation is reviewed to verify that it is bound by the fluence assumed in the existing PTS analysis.

4.2.3.1.2.2 10 CFR 54.21(c)(1)(ii)

The documented results of the revised PTS analysis based on the projected neutron fluence at the end of the period of extended operation is reviewed for compliance with 10 CFR 50.61.

The staff should confirm that the applicant has provided sufficient information for Pressurized Thermal Shock for the period of extended operation as follows:

Neutron Fluence: Identified the neutron fluence at the inside surface and the 1/4T location for each beltline material at the expiration of the license renewal period. Identified the methodology used in determining the neutron fluence and identified whether the methodology followed the guidance in Regulatory Guide (RG) 1.190 (Ref. 12).

There are two methodologies from 10 CFR 50.61 that can be used in the PTS analysis based on the projected neutron fluence at the end of the period of extended operation. RT_{NDT} is the reference temperature (NDT means nil-ductility temperature) used as an indexing parameter to determine the fracture toughness and the amount of embrittlement of a material. RT_{PTS} is the reference temperature used in the PTS analysis and is related to RT_{NDT} at the end of the facility's operating license.

The first methodology does not rely on plant-specific surveillance data to calculate delta RT_{NDT} (i.e., the mean value of the adjustment or shift in reference temperature caused by irradiation). The delta RT_{NDT} is determined by multiplying a chemistry factor from the tables in 10 CFR 50.61 by a fluence factor calculated from the neutron flux using an equation.

The second methodology relies on plant-specific surveillance data to determine the delta RT_{NDT}. In this methodology, two or more sets of surveillance data are needed. A surveillance datum consists of a measured delta RT_{NDT} for a corresponding neutron fluence. 10 CFR 50.61 specifies a procedure and a criterion for determining whether the surveillance data are credible. For the surveillance data to be defined as credible, the difference in the predicted value and the measured value for delta RT_{NDT} must be less than 28°F for weld metal. When a credible surveillance data set exists, the chemistry factor can be determined from these data in lieu of a value from the table in 10 CFR 50.61. Then the standard deviation of the increase in the RT_{NDT} can be reduced from 28°F to 14°F for welds.

To confirm that the Pressurized Thermal Shock analysis results in RT_{PTS} values below the screening criteria in 10 CFR 50.61 at the end of the license renewal period, the applicant should provide the following:

1. For each beltline material provide the unirradiated RT_{NDT}, the method of calculating the unirradiated RT_{NDT} (either generic or plant-specific), the margin, the chemistry factor, the method of calculating the chemistry factor, the mean value for the shift in transition temperature and the RT_{PTS} value.

2. If there are two or more data for a surveillance material that is from the same heat of material as the beltline material, provide analyses to determine whether the data are credible in accordance with RG 1.99, Revision 2 and whether the margin value used in the analysis is appropriate.

3. If there are two or more data for a surveillance material that is not from the same heat of material as the beltline material, provide analyses of the data to determine whether the data are consistent with the RG 1.99, Revision 2 methodology.

If the "PTS screening criteria" in 10 CFR 50.61 are exceeded during the period of extended operation, an analysis based on RG 1.154 (Ref. 3) is reviewed.

4.2.3.1.2.3 10 CFR 54.21(c)(1)(iii)

The applicant's proposal to demonstrate that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation will be reviewed on a case-by-case basis.

The license renewal application should provide an assessment of the current licensing basis TLAA for PTS, a discussion of the flux reduction program implemented in accordance with §50.61(b)(3), if necessary, and an identification of the viable options that exist for managing the aging effect in the future.

A. The applicant should explain its core management plans (e.g., operation with a low leakage core design and/or integral burnable neutron absorbers) from now through the end of the period of extended operation. Based on this core management strategy, the applicant should:

   (1) Identify the material in the RPV which has limiting RT_{PTS} value,
   (2) Provide the projected fluence value for the limiting material at end of license extended (EOLE),
   (3) Provide the projected RT_{PTS} value for the limiting material at EOLE, and
   (4) Provide the projected date and fluence values at which the limiting material will exceed the screening criteria in §50.61.

B. The applicant should discuss aging management programs that it intends to implement which will actively "manage" the condition of the facility's RPV, and hence, the risk associated with PTS. This discussion would be expected to address, at least, the facility's reactor pressure vessel material surveillance program.

C. The applicant should briefly discuss the options that it is considering with respect to "resolving" the PTS issue through EOLE. It is anticipated that this discussion would include some or all of the following:

   (1) Plant modifications (e.g., heating of ECCS injection water) which could limit the risk associated with postulated PTS events [see §50.61(b)(4) and/or (b)(6)],
   (2) More detailed safety analyses (e.g., using Regulatory Guide 1.154) which may be performed to show that the PTS risk for the facility is acceptably low through EOLE [see §50.61(b)(4)],
   (3) More advanced material property evaluation (e.g., use of Master Curve technology) to demonstrate greater fracture resistance for the limiting material [applies to §50.61(b)(4)] and/or,
   (4) The potential for RPV thermal annealing in accordance with §50.66 [see §50.61(b)(7)].

4.2.3.1.3 Pressure-Temperature (P-T) Limits

4.2.3.1.3.1 10 CFR 54.21(c)(1)(i)

The documented results of the projected neutron fluence at the end of the period of extended operation is reviewed to verify that it is bound by the embrittlement assumed in the existing P-T limit analysis.

4.2.3.1.3.2 10 CFR 54.21(c)(1)(ii)

The documented results of the revised P-T limit analysis based on the projected reduction in fracture toughness at the end of the period of extended operation is reviewed for compliance with 10 CFR Part 50 Appendix G.

4.2.3.1.3.3 10 CFR 54.21(c)(1)(iii)

Not applicable.

4.2.3.1.4 Elimination of Circumferential Weld Inspection (for BWRs)

To demonstrate that the vessel has not been embrittled beyond the basis for the technical alternative and that cold over-pressure events are not likely to occur during the license renewal term, the applicant should provide: (1) a comparison of the neutron fluence, initial RT_{NDT}, chemistry factor amounts of copper and nickel, delta RT_{NDT}, and mean RT_{NDT} of the limiting circumferential weld at the end of the license renewal period to the 64 EFPY reference case in Appendix E of the staff's SER (Ref. 11), (2) an estimate of conditional failure probability of the RPV at the end of the license renewal term based on the comparison of the mean RT_{NDT} for the limiting circumferential welds and the reference case, and (3) a description of procedures and training that will be utilized during the license renewal term to limit the frequency of cold over-pressure events to the amount specified in the staff's SER (Ref. 11). The staff should ensure that the applicant's plant is bound by the BWRVIP-74 analysis and that the applicant has committed to actions that are the basis for the staff approval.

4.2.3.1.5 Axial Welds (for BWRs)

To demonstrate that the vessel has not been embrittled beyond the basis for the staff and BWRVIP analyses, the applicant should provide: (1) a comparison of the neutron fluence, initial RT_{NDT}, chemistry factor amounts of copper and nickel, delta RT_{NDT}, and mean RT_{NDT} of the limiting axial weld at the end of the license renewal period to the reference case in the BWRVIP and staff analyses and (2) an estimate of conditional failure probability of the RPV at the end of the license renewal term based on the comparison of the mean RT_{NDT} for the limiting axial welds and the reference case. If this comparison does not indicate that the RPV failure frequency for axial welds is less than 5×10^{-6} per reactor year, the applicant should provide a probabilistic analysis to determine the RPV failure frequency for axial welds. The staff should ensure that the applicant's plant is bounded by the BWRVIP analysis or that the applicant has committed to a program to monitor axial weld embrittlement relative to the values specified by the staff in its May 7, 2000, letter.

4.2.3.2 FSAR Supplement

The reviewer verifies that the applicant has provided information to be included in the FSAR supplement that includes a summary description of the evaluation of the reactor vessel neutron embrittlement TLAA. Table 4.2-1 of this review plan section contains examples of acceptable FSAR supplement information for this TLAA. The reviewer verifies that the applicant has provided a FSAR supplement with information equivalent to that in Table 4.2-1.

The staff expects to impose a license condition on any renewed license to require the applicant to update its FSAR to include this FSAR supplement at the next update required pursuant to

10 CFR 50.71(e)(4). As part of the license condition, until the FSAR update is complete, the applicant may make changes to the programs described in its FSAR supplement without prior NRC approval, provided that the applicant evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59. If the applicant updates the FSAR to include the final FSAR supplement before the license is renewed, no condition will be necessary.

As noted in Table 4.2-1, an applicant need not incorporate the implementation schedule into its FSAR. However, the review should verify that the applicant has identified and committed in the license renewal application to any future aging management activities, including enhancements and commitments to be completed before the period of extended operation. The staff expects to impose a license condition on any renewed license to ensure that the applicant will complete these activities no later than the committed date.

4.2.4 Evaluation Findings

The reviewer determines whether the applicant has provided sufficient information to satisfy the provisions of this review plan section and whether the staff's evaluation supports conclusions of the following type, depending on the applicant's choice of 10 CFR 54.21(c)(1)(i), (ii), or (iii), to be included in the staff's safety evaluation report:

> The staff concludes that the applicant has provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1), that, for the reactor vessel neutron embrittlement TLAA, [choose which is appropriate] (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the period of extended operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation. The staff also concludes that the FSAR supplement contains an appropriate summary description of the reactor vessel neutron embrittlement TLAA evaluation for the period of extended operation as reflected in the license condition.

4.2.5 IMPLEMENTATION

Except in those cases in which the applicant proposes an acceptable alternative method, the method described herein will be used by the staff in its evaluation of conformance with NRC regulations.

4.2.6 References

1. 10 CFR Part 50 Appendix G, "Fracture Toughness Requirements."

2. 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events."

3. Regulatory Guide 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors," January 1987.

4. BWRVIP-05, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," Boiling Water Reactor Owners Group, September 28, 1995.

5. Letter to Carl Terry of Niagara Mohawk Power Company, from Gus C. Lainas of NRC, dated July 28, 1998.

6. Generic Letter 98-05, "Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief from Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds," Nuclear Regulatory Commission, November 10, 1998.

7. BWRVIP-74, "BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines," Boiling Water Reactor Owners Group, September 1999.

8. Letter to Carl Terry of Niagara Mohawk Power Company, from Jack R. Strosnider, Jr., of NRC, dated March 7, 2000.

9. Regulatory Guide 1.99 Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials," May, 1988.

10. Appendix K of ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."

11. Letter to Carl Terry of Niagara Mohawk Power Company, BWRVIP Chairman, from Christopher Grimes, of NRC, dated October 18, 2001.

12. Regulatory Guide 1.190 Rev. 0, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001

13. Letter to the Commission from L.A.Reyes (EDO), dated May 27, 2004 (ADAMS accession number ML041190564)

Table 4.2-1. Examples of FSAR Supplement for Reactor Vessel Neutron Embrittlement TLAA Evaluation

TLAA	Description of Evaluation	Implementation Schedule*
Upper-shelf energy	10 CFR Part 50 Appendix G paragraph IV.A.1 requires that the reactor vessel beltline materials must have Charpy upper-shelf energy of no less than 50 ft-lb (68 J) throughout the life of the reactor vessel unless otherwise approved by the NRC. The upper-shelf energy has been determined to exceed 50 ft-lb (68 J) to the end of the period of extended operation.	Completed
Pressurized thermal shock (for PWRs)	For PWRs, 10 CFR 50.61 requires the "reference temperature RT_{PTS}" for reactor vessel beltline materials be less than the "PTS screening criteria" at the expiration date of the operating license unless otherwise approved by the NRC. The "PTS screening criteria" are 270°F (132°C) for plates, forgings, and axial weld materials, or 300°F (149°C) for circumferential weld materials. The "reference temperature" has been determined to be less than the "PTS screening criteria" at the end of the period of extended operation.	Completed
Pressure-temperature (P-T) limits	10 CFR Part 50 Appendix G requires that heatup and cooldown of the RPV be accomplished within established P-T limits. These limits specify the maximum allowable pressure as a function of reactor coolant temperature. As the RPV becomes embrittled and its fracture toughness is reduced, the allowable pressure is reduced. 10 CFR Part 50 Appendix G requires periodic update of P-T limits based on projected embrittlement and data from a material surveillance program. The P-T limits will be updated to consider the period of extended operation.	Update should be completed before the period of extended operation.
Elimination of circumferential weld inspection and analysis of axial welds (for BWRs)	NRC has granted relief from the reactor vessel circumferential shell weld inspections because the applicant has demonstrated through plant-specific analysis that the plant meets BWRVIP-74 as approved by the NRC and has provided sufficient information that the probability of vessel failure due to embrittlement of axial welds is low.	Completed

* An applicant need not incorporate the implementation schedule into its FSAR. However, the reviewer should verify that the applicant has identified and committed in the license renewal application to any future aging management activities to be completed before the period of extended operation. The staff expects to impose a license condition on any renewed license to ensure that the applicant will complete these activities no later than the committed date.

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4.3 METAL FATIGUE ANALYSIS

Item	Locator	Comment	Justification
1	4.3.1.1.1, 4.3.2.1.1.2, 4.3.3.1.1.2	Correct stated Code limit for CUF	The Code, and many applicants CLB, allow for a CFUF equal to unity over the service life.
2	4.3.2.1.2.2, 4.3.3.1.2.2	Allow for stress reduction factors that may differ from those in Table 4.3-1	The applicant's code of record is bounding in their CLB. Table 4.3-1 appears to be appropriate for most cases, but it may not be for all cases.
4	4.3.3.2.3	Provide flexibility for future references for environmental life correction factors	Leaves room for improvement in the research into the phenomenon and improvements in calculational methods.

4.3 METAL FATIGUE ANALYSIS

Review Responsibilities

Primary - Branch responsible for the TLAA issues
Secondary - None

4.3.1 Areas of Review

A metal component subjected to cyclic loading at loads less than the static design load may fail because of fatigue. Metal fatigue of components may have been evaluated based on an assumed number of transients or cycles for the current operating term. The validity of such metal fatigue analysis is reviewed for the period of extended operation.

The metal fatigue analysis review includes, as appropriate, a review of in service flaw growth analyses, reactor vessel underclad cracking analysis, reactor vessel internals fatigue analysis, postulated high energy line break, leak-before-break, RCP flywheel, and metal bellows.

4.3.1.1 Time-Limited Aging Analysis

Metal components may be designed or analyzed based on requirements in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code or the American National Standards Institute (ANSI) guidance. These codes contain explicit metal fatigue or cyclic considerations based on TLAAs.

4.3.1.1.1 ASME Section III, Class 1

ASME Class 1 components, which include core support structures, are analyzed for metal fatigue. ASME Section III (Ref. 1) requires a fatigue analysis for Class 1 components that considers all transient loads based on the anticipated number of transients. A Section III Class 1 fatigue analysis requires the calculation of the "cumulative usage factor" (CUF) based on the fatigue properties of the materials and the expected fatigue service of the component. The ASME Code limits the CUF to a value of less ***or equal to*** than one for acceptable fatigue design. The fatigue resistance of these components during the period of extended operation is an area of review.

4.3.1.1.2 ANSI B31.1

ANSI B31.1 (Ref. 2) applies only to piping. It does not call for an explicit fatigue analysis. It specifies allowable stress levels based on the number of anticipated thermal cycles. The specific allowable stress reductions due to thermal cycles are listed in Table 4.3-1. For example, the allowable stress would be reduced by a factor of 1.0, i.e., no reduction, for piping that is not expected to experience more than 7,000 thermal cycles during plant service, but would be reduced to half of the maximum allowable static stress for 100,000 or more thermal cycles. The fatigue resistance of these components during the period of extended operation is an area of review.

4.3.1.1.3 Other Evaluations Based on CUF

The codes also contain metal fatigue analysis rules based on a CUF calculation [the 1969 edition of ANSI B31.7 (Ref. 3) for Class 1 piping, ASME NC-3200 vessels, ASME NE-3200

Class MC components, and metal bellows designed to ASME NC-3649.4(e)(3), ND-3649.4(e)(3), or NE-3366.2(e)(3)]. For these components, the discussion relating to ASME Section III, Class 1 in Subsection 4.3.1.1.1 of this review plan section applies.

4.3.1.1.4 ASME Section III, Class 2 and 3

ASME Section III, Class 2 and 3 piping cyclic design requirements are similar to the guidance in ANSI B31.1. The discussion relating to B31.1 in Subsection 4.3.1.1.2 of this review plan section applies.

4.3.1.2 Generic Safety Issue

The fatigue design criteria for nuclear power plant components have changed as the industry consensus codes and standards have developed. The fatigue design criteria for a specific component depend on the version of the design code that applied to that component, i.e., the code of record. There is a concern that the effects of the reactor coolant environment on the fatigue life of components were not adequately addressed by the code of record.

The NRC has decided that the adequacy of the code of record relating to metal fatigue is a potential safety issue to be addressed by the current regulatory process for operating reactors (Refs. 4 and 5). The effects of fatigue for the initial 40-year reactor license period were studied and resolved under Generic Safety Issue (GSI)-78, "Monitoring of Fatigue Transient Limits for reactor coolant system," and GSI-166, "Adequacy of Fatigue Life of Metal Components" (Ref. 6). GSI-78 addressed whether fatigue monitoring was necessary at operating plants. As part of the resolution of GSI-166, an assessment was made of the significance of the more recent fatigue test data on the fatigue life of a sample of components in plants where Code fatigue design analysis had been performed. The efforts on fatigue life estimation and ongoing issues under GSI-78 and GSI-166 for 40-year plant life were addressed separately under a staff generic task action plan (Refs. 7 and 8). The staff documented its completion of the fatigue action plan in SECY-95-245 (Ref. 9).

SECY-95-245 was based on a study described in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components" (Ref. 10). In NUREG/CR-6260, sample locations with high fatigue usage were evaluated. Conservatisms in the original fatigue calculations, such as actual cycles versus assumed cycles, were removed, and the fatigue usage was recalculated using a fatigue curve considering the effects of the environment. The staff found that most of the locations would have a CUF of less than the ASME Code limit of 1.0 for 40 years. On the basis of the component assessments, supplemented by a 40-year risk study, the staff concluded that a backfit of the environmental fatigue data to operating plants could not be justified. However, because the staff was less certain that sufficient excessive conservatisms in the original fatigue calculations could be removed to account for an additional 20 years of operation for renewal, the staff recommended in SECY-95-245 that the samples in NUREG/CR-6260 should be evaluated considering environmental effects for license renewal. GSI-190, "Fatigue Evaluation of Metal Components for 60-year Plant Life," was established to address the residual concerns of GSI-78 and GSI-166 regarding the environmental effects on fatigue of pressure boundary components for 60 years of plant operation.

The scope of GSI-190 included design basis fatigue transients. It studied the probability of fatigue failure and its effect on core damage frequency (CDF) of selected metal components for 60-year plant life. The results showed that some components have cumulative probabilities of crack initiation and through-wall growth that approach one within the 40- to 60-year period. The maximum failure rate (through-wall cracks per year) was in the range of 10^{-2} per year, and those failures were generally associated with high cumulative usage factor locations and components with thinner walls, i.e., pipes more vulnerable to through-wall cracks. In most cases, the leakage from these through-wall cracks is small and not likely to lead to core damage. It was concluded that no generic regulatory action is necessary and that GSI-190 is resolved based on results of probabilistic analyses and sensitivity studies, interactions with the industry (NEI and EPRI), and different approaches available to licensees to manage the effects of aging (Refs. 11 and 12).

However, the calculations supporting resolution of this issue, which included consideration of environmental effects, indicate the potential for an increase in the frequency of pipe leaks as plants continue to operate. Thus, the staff concluded that licensees are to address the effects of coolant environment on component fatigue life as aging management programs are formulated in support of license renewal.

The applicant's consideration of the effects of coolant environment on component fatigue life for license renewal is an area of review.

4.3.1.3 FSAR Supplement

Detailed information on the evaluation of TLAAs is contained in the renewal application. A summary description of the evaluation of TLAAs for the period of extended operation is contained in the applicant's FSAR supplement. The FSAR supplement is an area of review.

4.3.2 Acceptance Criteria

The acceptance criteria for the areas of review described in Subsection 4.3.1 of this review plan section delineate acceptable methods for meeting the requirements of the NRC's regulations in 10 CFR 54.21(c)(1).

4.3.2.1 Time-Limited Aging Analysis

Pursuant to 10 CFR 54.21(c)(1)(i) - (iii), an applicant must demonstrate one of the following:

(i) the analyses remain valid for the period of extended operation,

(ii) the analyses have been projected to the end of the extended period of operation, or

(iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Specific acceptance criteria for metal fatigue are:

4.3.2.1.1 ASME Section III, Class 1

For components designed or analyzed to ASME Class 1 requirements, the acceptance criteria, depending on the applicant's choice of 10 CFR 54.21(c)(1)(i), (ii), or (iii), are:

4.3.2.1.1.1 10 CFR 54.21(c)(1)(i)

The existing CUF calculations remain valid because the number of assumed transients would not be exceeded during the period of extended operation.

4.3.2.1.1.2 10 CFR 54.21(c)(1)(ii)

The CUF calculations have been reevaluated based on an increased number of assumed transients to bound the period of extended operation. The resulting CUF remains less than ***or equal to*** unity for the period of extended operation.

4.3.2.1.1.3 10 CFR 54.21(c)(1)(iii)

In Chapter X of the GALL report (Ref. 13), the staff has evaluated a program for monitoring and tracking the number of critical thermal and pressure transients for the selected reactor coolant system components. The staff has determined that this program is an acceptable aging management program to address metal fatigue of the reactor coolant system components according to 10 CFR 54.21(c)(1)(iii). The GALL report may be referenced in a license renewal application and should be treated in the same manner as an approved topical report. In referencing the GALL report, the applicant should indicate that the material referenced is applicable to the specific plant involved and should provide the information necessary to adopt the finding of program acceptability as described and evaluated in the report. The applicant should also verify that the approvals set forth in the GALL report for the generic program apply to the applicant's program.

4.3.2.1.2 ANSI B31.1

For piping designed or analyzed to B31.1, the acceptance criteria, depending on the applicant's choice of 10 CFR 54.21(c)(1)(i), (ii), or (iii), are:

4.3.2.1.2.1 10 CFR 54.21(c)(1)(i)

The existing fatigue strength reduction factors remain valid because the number of cycles would not be exceeded during the period of extended operation.

4.3.2.1.2.2 10 CFR 54.21(c)(1)(ii)

The fatigue strength reduction factors have been reevaluated based on an increased number of assumed thermal cycles and ***the stress reduction factors (e.g.,*** Table 4.3-1***) given in the applicant's code of record*** to bound the period of extended operation. The adjusted fatigue strength reduction factors are such that the component design basis remains valid during the period of extended operation.

4.3.2.1.2.3 10 CFR 54.21(c)(1)(iii)

The effects of aging on the intended function(s) will be adequately managed for the period of extended operation. The component could be replaced and the allowable stresses for the replacement will be sufficient as specified by the code during the period of extended operation.

Alternative acceptance criteria under 10 CFR 54.21(c)(1)(iii) have yet to be developed. They will be evaluated on a case-by-case basis to ensure that the aging effects will be managed such that the intended functions(s) will be maintained during the period of extended operation.

4.3.2.1.3 Other Evaluations Based on CUF

The acceptance criteria in Subsection 4.3.2.1.1 of this review plan section apply.

4.3.2.1.4 ASME Section III, Class 2 and 3

The acceptance criteria in Subsection 4.3.2.1.2 of this review plan section apply.

4.3.2.2 Generic Safety Issue

The staff recommendation for the closure of GSI-190 is contained in a December 26, 1999 memorandum from Ashok Thadani to William Travers (Ref. 11). The staff recommended that licensees address the effects of the coolant environment on component fatigue life as aging management programs are formulated in support of license renewal. One method acceptable to the staff for satisfying this recommendation is to assess the impact of the reactor coolant environment on a sample of critical components. These critical components should include, as a minimum, those selected in NUREG/CR-6260 (Ref. 10). The sample of critical components can be evaluated by applying environmental correction factors to the existing ASME Code fatigue analyses. Formulas for calculating the environmental life correction factors for carbon and low-alloy steels are contained in NUREG/CR-6583 (Ref. 14) and those for austenitic SSs are contained in NUREG/CR-5704 (Ref. 15).

4.3.2.3 FSAR Supplement

The specific criterion for meeting 10 CFR 54.21(d) is:

> The summary description of the evaluation of TLAAs for the period of extended operation in the FSAR supplement is appropriate such that later changes can be controlled by 10 CFR 50.59. The description should contain information associated with the TLAAs regarding the basis for determining that the applicant has made the demonstration required by 10 CFR 54.21(c)(1).

4.3.3 Review Procedures

For each area of review described in Subsection 4.3.1, the following review procedures should be followed:

4.3.3.1 Time-Limited Aging Analysis

4.3.3.1.1 ASME Section III, Class 1

For components designed or analyzed to ASME Class 1 requirements, the review procedures, depending on the applicant's choice of 10 CFR 54.21(c)(1)(i), (ii), or (iii), are:

4.3.3.1.1.1 10 CFR 54.21(c)(1)(i)

The operating transient experience and a list of the assumed transients used in the existing CUF calculations for the current operating term are reviewed to ensure that the number of assumed transients would not be exceeded during the period of extended operation.

4.3.3.1.1.2 10 CFR 54.21(c)(1)(ii)

The operating transient experience and a list of the increased number of assumed transients projected to the end of the period of extended operation are reviewed to ensure that the transient projection is adequate. The revised CUF calculations based on the projected number of assumed transients are reviewed to ensure that the CUF remains less than ***or equal to*** one at the end of the period of extended operation.

The code of record should be used for the reevaluation, or the applicant may update to a later code edition pursuant to 10 CFR 50.55a. In the latter case, the reviewer verifies that the requirements in 10 CFR 50.55a are met.

4.3.3.1.1.3 10 CFR 54.21(c)(1)(iii)

The applicant may reference the GALL report in its license renewal application, as appropriate. The review should verify that the applicant has stated that the report is applicable to its plant with respect to its program that monitors and tracks the number of critical thermal and pressure transients for the selected reactor coolant system components. The reviewer verifies that the applicant has identified the appropriate program as described and evaluated in the GALL report. The reviewer also ensures that the applicant has stated that its program contains the same program elements that the staff evaluated and relied upon in approving the corresponding generic program in the GALL report. No further staff evaluation is necessary.

4.3.3.1.2 ANSI B31.1

For piping designed or analyzed to ANSI B31.1 guidance, the review procedures, depending on the applicant's choice of 10 CFR 54.21(c)(1)(i), (ii), or (iii), are:

4.3.3.1.2.1 10 CFR 54.21(c)(1)(i)

The operating cyclic experience and a list of the assumed thermal cycles used in the existing allowable stress determination are reviewed to ensure that the number of assumed thermal cycles would not be exceeded during the period of extended operation.

4.3.3.1.2.2 10 CFR 54.21(c)(1)(ii)

The operating cyclic experience and a list of the increased number of assumed thermal cycles projected to the end of the period of extended operation are reviewed to ensure that the thermal cycle projection is adequate. The revised allowable stresses based on the projected number of assumed thermal cycles and ***the stress reduction factors given in the applicant's code of record*** ~~Table 4.3-1~~ are reviewed to ensure that they remain sufficient as specified by the code during the period of extended operation. ***Typical stress reduction factors based on thermal cycles are given in Table 4.3-1.***

The ***applicant's*** code of record should be used for the reevaluation, or the applicant may use the criteria of 10 CFR 50.55a. In the latter case, the reviewer verifies that the requirements in 10 CFR 50.55a are met.

4.3.3.1.2.3 10 CFR 54.21(c)(1)(iii)

The applicant's proposed program to ensure that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation is reviewed. If the applicant proposed a component replacement before it exceeds the assumed thermal cycles, the reviewer verifies that the allowable stresses for the replacement will remain sufficient as specified by the code during the period of extended operation. Other applicant-proposed programs will be reviewed on a case-by-case basis.

4.3.3.1.3 Other Evaluations Based on CUF

The review procedures in Subsection 4.3.3.1.1 of this review plan section apply.

4.3.3.1.4 ASME Section III, Class 2 and 3

The review procedures in Subsection 4.3.3.1.2 of this review plan section apply.

4.3.3.2 Generic Safety Issue

The reviewer verifies that the applicant has addressed the staff recommendation for the closure of GSI-190 contained in a December 26, 1999 memorandum from Ashok Thadani to William Travers (Ref. 11). The reviewer verifies that the applicant has addressed the effects of the coolant environment on component fatigue life as aging management programs are formulated in support of license renewal. If an applicant has chosen to assess the impact of the reactor coolant environment on a sample of critical components, the reviewer verifies the following:

1. The critical components include, as a minimum, those selected in NUREG/CR-6260 (Ref. 10).

2. The sample of critical components have been evaluated by applying environmental correction factors to the existing ASME Code fatigue analyses.

3. Formulas for calculating the environmental life correction factors are those contained in NUREG/CR–6583 (Ref. 14) for carbon and low-alloy steels, and in NUREG/CR-5704 (Ref. 15) for austenitic SSs***, or an approved technical equivalent***.

4.3.3.3 FSAR Supplement

The reviewer verifies that the applicant has provided information, to be included in the FSAR supplement, that includes a summary description of the evaluation of the metal fatigue TLAA. Table 4.3-2 contains examples of acceptable FSAR supplement information for this TLAA. The reviewer verifies that the applicant has provided a FSAR supplement with information equivalent to that in Table 4.3-2. The staff expects to impose a license condition on any renewed license to require the applicant to update its FSAR to include this FSAR supplement at the next update required pursuant to 10 CFR 50.71(e)(4). As part of the license condition, until the FSAR update is complete, the applicant may make changes to the programs described in its FSAR supplement without prior NRC approval, provided that the applicant evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59. If the applicant updates the FSAR to include the final FSAR supplement before the license is renewed, no condition will be necessary.

As noted in Table 4.3-2, an applicant need not incorporate the implementation schedule into its FSAR. However, the review should verify that the applicant has identified and committed in the license renewal application to any future aging management activities, including enhancements and commitments to be completed before the period of extended operation.

The staff expects to impose a license condition on any renewed license to ensure that the applicant will complete these activities no later than the committed date.

4.3.4 Evaluation Findings

The reviewer determines whether the applicant has provided sufficient information to satisfy the provisions of this review plan section and whether the staff's evaluation supports conclusions of the following type, depending on the applicant's choice of 10 CFR 54.21(c)(1)(i), (ii), or (iii), to be included in the staff's safety evaluation report:

> The staff concludes that the applicant has provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1), that, for the metal fatigue TLAA, [choose which is appropriate] (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the period of extended operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation. The staff also concludes that the FSAR supplement contains an appropriate summary description of the metal fatigue TLAA evaluation for the period of extended operation as reflected in the license condition.

4.3.5 Implementation

Except in those cases in which the applicant proposes an acceptable alternative method, the method described herein will be used by the staff in its evaluation of conformance with NRC regulations.

4.3.6 References

1. ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Power Plant Components," American Society of Mechanical Engineers.

2. ANSI/ASME B31.1, "Power Piping," American National Standards Institute.

3. ANSI/ASME B31.7-1969, "Nuclear Power Piping," American National Standards Institute.

4. SECY-93-049, "Implementation of 10 CFR Part 54, 'Requirements for Renewal of Operating Licenses for Nuclear Power Plants,'" March 1, 1993.

5. Staff Requirements Memorandum from Samuel J. Chilk, dated June 28, 1993.

6. NUREG-0933, "A Prioritization of Generic Safety Issues," Supplement 20, July 1996.

7. Letter from William T. Russell of NRC to William Rasin of the Nuclear Management and Resources Council, dated July 30, 1993.

8. SECY-94-191, "Fatigue Design of Metal Components," July 26, 1994.

9. SECY-95-245, "Completion of The Fatigue Action Plan," September 25, 1995.

10. NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," March 1995.

11. Letter from Ashok C. Thadani of the Office of Nuclear Regulatory Research to William D. Travers, Executive Director of Operations, dated December 26, 1999.

12. NUREG/CR-6674, "Fatigue Analysis of Components for 60-Year Plant Life," June 2000.

13. NUREG-1801, "Generic Aging Lessons Learned (GALL)," U.S. Nuclear Regulatory Commission, March 2001.

14. NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," March 1998.

15. NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," April 1999.

Table 4.3-1. Stress Range Reduction Factors

Number of Equivalent Full Temperature Cycles	Stress Range Reduction Factor
7,000 and less	1.0
7,000 to 14,000	0.9
14,000 to 22,000	0.8
22,000 to 45,000	0.7
45,000 to 100,000	0.6
100,000 and over	0.5

Table 4.3-2. Example of FSAR Supplement for Metal Fatigue TLAA Evaluation

10 CFR 54.21(c)(1)(iii) Example

TLAA	Description of Evaluation	Implementation Schedule*
Metal fatigue	The aging management program monitors and tracks the number of critical thermal and pressure test transients, and monitors the cycles for the selected reactor coolant system components. The aging management program will address the effects of the coolant environment on component fatigue life by assessing the impact of the reactor coolant environment on a sample of critical components that include, as a minimum, those components selected in NUREG/CR-6260. The sample of critical components can be evaluated by applying environmental correction factors to the existing ASME Code fatigue analyses. Formulas for calculating the environmental life correction factors are contained in NUREG/CR-6583 for carbon and low-alloy steels and in NUREG/CR-5704 for austenitic SSs.	Evaluation should be completed before the period of extended operation

* An applicant need not incorporate the implementation schedule into its FSAR. However, the reviewer should verify that the applicant has identified and committed in the license renewal application to any future aging management activities to be completed before the period of extended operation. The staff expects to impose a license condition on any renewed license to ensure that the applicant will complete these activities no later than the committed date.

4.4 ENVIRONMENTAL QUALIFICATION (EQ) OF ELECTRIC EQUIPMENT

Item	Locator	Comment	Justification
1	4.4.1	Revise 2nd paragraph	Revised wording provides proper characterization of the licensing bases of plants with respect to mechanical equipment qualification programs and how those programs would be treated in license renewal.

4.4 ENVIRONMENTAL QUALIFICATION (EQ) OF ELECTRIC EQUIPMENT

Review Responsibilities

Primary - Branch responsible for electrical engineering
Secondary – Plant Systems Branch (Mechanical Equipment only)

4.4.1 Areas of Review

The NRC has established environmental qualification requirements in 10 CFR Part 50, Appendix A, Criterion 4, and 10 CFR 50.49. Section 50.49 specifically requires each nuclear power plant licensee to establish a program to qualify certain electric equipment (not including equipment located in mild environments) so that such equipment, in its end-of-life condition, will meet its performance specifications during and following design basis accidents under the most severe environmental conditions postulated at the equipment's location after such an accident. Such conditions include, among others, conditions resulting from loss of coolant accidents (LOCAs), high energy line breaks (HELBs), and post-LOCA radiation. Equipment qualified by test must be preconditioned by aging to its end-of-life condition (i.e., the condition at the end of the current operating term). Those components with a qualified life equal to or greater than the duration of the current operating term are covered by TLAAs.

In a related subject, ***the CLB of*** some nuclear power plants ***requires an environmental qualification program for*** ~~have~~ mechanical equipment ***as a means to demonstrate compliance*** ~~that was qualified in accordance with~~ ***with*** the provisions of Criterion 4 of Appendix A to 10 CFR Part 50. If a plant has ***such an environmental qualification program for*** mechanical equipment, ***the program may establish a qualified life for the mechanical equipment.*** ~~it is typically documented in the plant's master EQ list.~~ ***Those analyses that establish a component's qualified life equal to or greater than the duration of the current operating term (but less than the cumulative period of plant operation) meet the definition of TLAAs pursuant to 10 CFR 54.3.*** If ***such TLAAs exist for*** these qualified mechanical ***components*** ~~equipment require a performance of a TLAA, it~~ ***the TLAAs*** should be ***included in Section 4.7 of the application, and the review should be*** performed in accordance with the provisions of SRP-LR Section 4.7, "Other Plant-Specific Time-Limited Aging Analyses." ~~If a TLAA of qualified mechanical equipment is necessary, usually it will involve assessments of the environmental effects on components such as seals, gaskets, lubricants, fluids for hydraulic systems, or diaphragms.~~

[There are no other recommended changes to Section 4.4]

4.5 CONCRETE CONTAINMENT TENDON PRESTRESS ANALYSIS

Item	Locator	Comment	Justification
1	4.5.1	Clarify requirements for evaluating existing TLAA	10CFR54 does not require TLAAs to be created to address this aging effect; however analyses/calculations to address the aging effect of loss of prestress in containment tendons generally exist and generally meet the definition of a TLAA as given by 10CFR54.3.
2	4.5.2.1.1, 4.5.2.1.2, 4.5.3.1.2, Table 4.5-1	Update acceptance criteria	Updated to meet the requirement per ASME Section XI, Sub-section IWL Acceptance criteria
3	4.5.2.1.3	Acknowledge GALL as guidance rather than requirement.	ISG 1
4	4.5.3.1.3	Address evaluations outside GALL guidance	ISG 1
5	4.5.6	Update Reference 4	Editorial
6	Table 4.5-1	10 CFR 54.21(c)(1)(i) and 10 CFR 54.21(c)(1)(ii) Examples are repeat information , so it should be consolidated in one item.	Updated to meet the requirement per ASME Section XI, Sub-section IWL Acceptance criteria

4.5 CONCRETE CONTAINMENT TENDON PRESTRESS ANALYSIS

Review Responsibilities

Primary - Branch responsible for structural engineering
Secondary - None

4.5.1 Areas of Review

The prestressing tendons in prestressed concrete containments lose their prestressing forces with time due to creep and shrinkage of concrete, and relaxation of the prestressing steel. During the design phase, engineers estimate these losses to arrive at the end of operating life (Refs. 1 and 2), normally forty years. The operating experiences with the trend of prestressing forces indicate that the prestressing tendons lose their prestressing forces at a rate higher than predicted due to sustained high temperature (Ref. 3). Thus, it is necessary to ***ensure that the applicant addresses existing*** ~~perform~~ TLAAs for the extended period of operation.

The adequacy of the prestressing forces in prestressed concrete containments is reviewed for the period of extended operation.

4.5.2 Acceptance Criteria

The acceptance criteria for the area of review described in Subsection 4.5.1 of this review plan section delineate acceptable methods for meeting the requirements of the NRC's regulations in 10 CFR 54.21(c)(1).

4.5.2.1 Time-Limited Aging Analysis

Pursuant to 10 CFR 54.21(c)(1)(i) - (iii), an applicant must demonstrate one of the following:

(i) The analyses remain valid for the period of extended operation;

(ii) The analyses have been projected to the end of the extended period of operation; or

(iii) The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Accordingly, the specific options for satisfying the acceptance criterion are:

4.5.2.1.1 10 CFR 54.21(c)(1)(i)

The existing prestressing force evaluation remains valid because (1) losses of the prestressing force are less than the predicted losses as evidenced from the trend lines constructed from the recent inspection, (2) the period of evaluation covers the period of extended operation, and (3) the trend lines of the measured prestressing forces remain above the ***minimum required prestress force specified at anchorage*** ~~predicted lower limit (PLL)~~ for each group of tendons for the period of extended operation.

4.5.2.1.2 10 CFR 54.21(c)(1)(ii)

The ***trend line*** ~~predicted lower limits (PLLs)~~ of prestressing forces for each group of tendons developed for 40 years of operation should be extended to 60 years. The applicant should demonstrate that the trend lines of the measured prestressing forces will stay above the ~~PLLs and the~~ design Minimum Required Value (MRV) in the CLB for each group of tendons during the period of extended operation (Ref. 4). If this cannot be done, the applicant should develop a systematic plan for retensioning selected tendons so that the trend lines will remain above the ***minimum required prestress force specified at anchorage*** ~~PLLs~~ for each group of tendons during the period of extended operation, or perform a reanalysis of containment to demonstrate design adequacy.

4.5.2.1.3 10 CFR 54.21(c)(1)(iii)

In Chapter X of the GALL report (Ref. 4), the staff has evaluated a program that assesses the concrete containment tendon prestressing forces, and has determined that it is an acceptable aging management program to address concrete containment tendon prestress according to 10 CFR 54.21(c)(1)(iii), except for operating experience. The GALL report recommends further evaluation of the applicant's operating experience related to the containment prestress force. ***However, the GALL report contains one acceptable way and not the only way to manage aging for license renewal.***

The GALL report may be referenced in a license renewal application, and should be treated in the same manner as an approved topical report. In referencing the GALL report, an applicant should indicate that the material referenced is applicable to the specific plant involved and should provide the information necessary to adopt the finding of program acceptability as described and evaluated in the report. An applicant should also verify that the approvals set forth in the GALL report for the generic program apply to the applicant's program.

4.5.2.2 FSAR Supplement

The specific criterion for meeting 10 CFR 54.21(d) is:

The summary description of the evaluation of TLAAs for the period of extended operation in the FSAR supplement is appropriate such that later changes can be controlled by 10 CFR 50.59. The description should contain information associated with the TLAAs regarding the basis for determining that the applicant has made the demonstration required by 10 CFR 54.21(c)(1).

4.5.3 Review Procedures

For each area of review described in Subsection 4.5.1 of this review plan section, the following review procedures should be followed:

4.5.3.1 Time-Limited Aging Analysis

For a concrete containment prestressing tendon system, the review procedures, depending on the applicant's choice of 10 CFR 54.21(c)(1)(i), (ii), or (iii), are:

4.5.3.1.1 10 CFR 54.21(c)(1)(i)

The results of a recent inspection to measure the amount of prestress loss are reviewed to ensure that the reduction of prestressing force is less than the predicted loss in the existing analysis. The reviewer verifies that the trend line of the measured prestressing force when plotted on the predicted prestressing force curve shows that the existing analysis will cover the period of extended operation.

4.5.3.1.2 10 CFR 54.21(c)(1)(ii)

The reviewer reviews the trend lines of the measured prestressing forces to ensure that individual tendon lift-off forces (rather than average lift-off forces of the tendon group) are considered in the regression analysis, as discussed in IN 99-10 (Ref. 3). Either the reviewer verifies that the trend lines will stay above the ***minimum required*** ~~PLL~~ prestressing forces for each group of tendons during the period of extended operation or, if the trend lines fall below the ***minimum required prestressing forces*** ~~PLL~~ during this period, the reviewer verifies that the applicant has a systematic plan for retensioning the tendons to ensure that the trend lines will return to being, and remain, above the ***minimum required prestressing forces*** ~~PLL~~ for each group of tendons during the period of extended operation. If the applicant chooses to reanalyze the containment, the reviewer verifies that the design adequacy is maintained in the period of extended operation.

4.5.3.1.3 10 CFR 54.21(c)(1)(iii)

An applicant may reference the GALL report in its license renewal application, as appropriate. The review should verify that the applicant has stated that the report is applicable to its plant with respect to its program that assesses the concrete containment tendon prestressing forces. The reviewer verifies that the applicant has identified the appropriate program as described and evaluated in the GALL report. The reviewer also ensures that the applicant has stated that its program contains the same program elements that the staff evaluated and relied upon in approving the corresponding generic program in the GALL report.

The GALL report recommends further evaluation of the applicant's operating experience related to the containment prestress force. The applicant's program should incorporate the relevant operating experience that occurred at the applicant's plant as well as at other plants. The applicant should consider applicable portions of the experience with prestressing systems described in Information Notice 99-10 (Ref. 3). Tendon operating experience could vary among plants with prestressed concrete containments. The difference could be due to the prestressing system design (for example, button-heads, wedge or swaged anchorages), environment, or type of reactor (PWR or BWR). The reviewer reviews the applicant's program to verify that the applicant has adequately considered plant-specific operating experience.

If the applicant does not reference the GALL report in its renewal application, additional staff evaluation is necessary to determine whether the applicant's program is acceptable for this area of review.

4.5.3.2 FSAR Supplement

The reviewer verifies that the applicant has provided information, to be included in the FSAR supplement, that includes a summary description of the evaluation of tendon prestress TLAA. Table 4.5-1 of this review plan section contains examples of acceptable FSAR supplement information for this TLAA. The reviewer verifies that the applicant has provided a FSAR supplement with information equivalent to that in Table 4.5-1.

The staff expects to impose a license condition on any renewed license to require the applicant to update its FSAR to include this FSAR supplement at the next update required pursuant to 10 CFR 50.71(e)(4). As part of the license condition, until the FSAR update is complete, the applicant may make changes to the programs described in its FSAR supplement without prior NRC approval, provided that the applicant evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59.

As noted in Table 4.5-1, an applicant need not incorporate the implementation schedule into its FSAR. However, the review should verify that the applicant has identified and committed in the license renewal application to any future aging management activities, including enhancements and commitments to be completed before the period of extended operation. The staff expects to impose a license condition on any renewed license to ensure that the applicant will complete these activities no later than the committed date.

4.5.4 Evaluation Findings

The reviewer verifies that the applicant has provided sufficient information to satisfy the provisions of this review plan section and that the staff's evaluation supports conclusions of the following type, depending on the applicant's choice of 10 CFR 54.21(c)(1)(i), (ii), or (iii), to be included in the staff's safety evaluation report:

> The staff concludes that the applicant has provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1), that, for the concrete containment tendon prestress TLAA, [choose which is appropriate] (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the period of extended operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation. The staff also concludes that the FSAR supplement contains an appropriate description of the concrete containment tendon prestress TLAA evaluation for the period of extended operation as reflected in the license condition.

4.5.5 Implementation

Except in those cases in which the applicant proposes an acceptable alternative method, the method described herein will be used by the staff in its evaluation of conformance with NRC regulations.

4.5.6 References

1. Regulatory Guide 1.35, Rev. 3, "Inspection of Ungrouted Tendons in Prestressed Concrete Containments," July 1990.

2. Regulatory Guide 1.35.1, "Determining Prestressing Forces for Inspection of Prestressed Concrete Containments," July 1990.

3. NRC Information Notice 99-10, "Degradation of Prestressing Tendon Systems in Prestressed Concrete Containments," April 1999.

4. NUREG-1801, "Generic Aging Lessons Learned (GALL)," U.S. Nuclear Regulatory Commission, ~~July 2001.~~ ***Revision 1, September 2005.***

Table 4.5-1. Examples of FSAR Supplement for Concrete Containment Tendon Prestress TLAA Evaluation

10 CFR 54.21(c)(1)(i) ***and 10 CFR 54.21(c)(1)(ii)*** **Example**

TLAA	Description of Evaluation	Implementation Schedule*
Concrete containment tendon prestress	The prestressing tendons are used to impart compressive forces in the prestressed concrete containments to resist the internal pressure inside the containment that would be generated in the event of a LOCA. The prestressing forces generated by the tendons diminish over time due to losses in prestressing forces in the tendons and in the surrounding concrete. The prestressing force evaluation has been determined to remain valid to the end of the period of extended operation, and the trend lines of the measured prestressing forces will stay above the ***minimum required prestressing force*** ~~PLLs~~ for each group of tendons to the end of this period.	Completed

~~**10 CFR 54.21(c)(1)(ii) Example**~~

~~TLAA~~	~~Description of Evaluation~~	~~Implementation Schedule*~~
~~Concrete containment tendon prestress~~	~~The prestressing tendons are used to impart compressive forces in the prestressed concrete containments to resist the internal pressure inside the containment that would be generated in the event of a LOCA. The prestressing forces generated by the tendons diminish over time due to losses in prestressing forces in the tendons and in the surrounding concrete. The prestressing forces have been reevaluated, showing that the trend lines of the measured prestressing forces will stay above the PLLs for each group of tendons to the end of the period of extended operation.~~	~~Completed~~

10 CFR 54.21(c)(1)(iii) Example

TLAA	Description of Evaluation	Implementation Schedule*
Concrete containment tendon prestress	The prestressing tendons are used to impart compressive forces in the prestressed concrete containments to resist the internal pressure inside the containment that would be generated in the event of a LOCA. The prestressing forces generated by the tendons diminish over time due to losses of prestressing forces in the tendons and in the surrounding concrete. The aging management program developed to monitor the prestressing forces should ensure that, during each inspection, the trend lines of the measured prestressing forces show that they meet the requirements of 10 CFR 50.55a(b)(2)(ix)(B). If the trend lines cross the ***minimum required prestressing force*** ~~PLLs~~, corrective actions will be taken. The program will also incorporate any plant-specific and industry operating experience.	Program should be implemented before the period of extended operation.

* An applicant need not incorporate the implementation schedule into its FSAR. However, the reviewer should verify that the applicant has identified and committed in the license renewal application to any future aging management activities to be completed before the period of extended operation. The staff expects to impose a license condition on any renewed license to ensure that the applicant will complete these activities no later than the committed date.

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4.6 CONTAINMENT LINER PLATE, METAL CONTAINMENTS, AND PENETRATIONS FATIGUE ANALYSIS

Item	Locator	Comment	Justification
1	4.6.1.1.1, 4.6.2.1.1.3, 4.6.3.1.1.3, Table 4.6-1	Correct stated Code limit for CUF	Code allows CFUF to be equal to 1.0 for the entire service life of the bellows. Many calculations on bellows are done to show that an extremely large number of the design cycles may be experienced without CUF = 1.0. IN this case, there is no TLAA as the assumption is not based upon the original license term.

4.6 CONTAINMENT LINER PLATE, METAL CONTAINMENTS, AND PENETRATIONS FATIGUE ANALYSIS

Review Responsibilities

Primary - Branch responsible for structural engineering
Secondary - Branch responsible for mechanical engineering

4.6.1 Areas of Review

The interior surface of a concrete containment structure is lined with thin metallic plates to provide a leak-tight barrier against the uncontrolled release of radioactivity to the environment, as required by 10 CFR Part 50. The thickness of the liner plates is generally between 1/4 in. (6.2 mm) and 3/8 in. (9.5 mm). The liner plates are attached to the concrete containment wall by stud anchors or structural rolled shapes or both. The design process assumes that the liner plates do not carry loads. However, normal loads, such as from concrete shrinkage, creep, and thermal changes, imposed on the concrete containment structure, are transferred to the liner plates through the anchorage system. Internal pressure and temperature loads are directly applied to the liner plates. Thus, under design-base conditions, the liner plates could experience significant strains. Some plants may have metal containments instead of concrete containments with liner plates.

Fatigue of the liner plates or metal containments may be considered in the design based on an assumed number of loading cycles for the current operating term. The cyclic loads include reactor building interior temperature variation during the heatup and cooldown of the reactor coolant system, a LOCA, annual outdoor temperature variations, thermal loads due to the high energy containment penetration piping lines (such as steam and feedwater lines), seismic loads, and pressurization due to periodic Type A integrated leak rate tests.

High energy piping penetrations and the fuel transfer canal in some plants are equipped with bellow assemblies. These are designed to accommodate relative movements between the containment wall (including the liner) and the adjoining structures. The penetrations have sleeves (up to 10 feet in length, with a 2 to 3-inch annulus around the piping) to penetrate the concrete containment wall and allow movement of the piping system. Dissimilar metal welds connect the piping penetrations to the bellows to provide leak-tight penetrations.

The containment liner plates, metal containments, penetration sleeves (including dissimilar metal welds), and penetration bellows may be designed in accordance with requirements of Section III of the ASME Boiler and Pressure Vessel Code. If a plant's code of record requires a fatigue analysis, then this analysis may be a TLAA and must be evaluated in accordance with 10 CFR 54.21(c)(1) to ensure that the effects of aging on the intended functions will be adequately managed for the period of extended operation.

The adequacy of the fatigue analyses of the containment liner plates (including welded joints), metal containments, penetration sleeves, dissimilar metal welds, and penetration bellows is reviewed in this review plan section for the period of extended operation. The fatigue analyses of the pressure boundary of process piping are reviewed separately following the guidance in Section 4.3, "Metal Fatigue," of this review plan.

4.6.1.1 Time-Limited Aging Analysis

The containment liner plates (including welded joints), metal containments, penetration sleeves, dissimilar metal welds, and penetration bellows may be designed and/or analyzed in accordance with ASME code requirements. The ASME code contains explicit metal fatigue or cyclic considerations based on TLAAs. Specific requirements are contained in the design code of reference for each plant.

4.6.1.1.1 ASME Section III, MC or Class 1

ASME Section III Division 2, "Code for Concrete Reactor Vessel and Containments," Subsection CC, "Concrete Containment," and Division 1, Subsection NE, "Class MC Components," (Ref. 1) require a fatigue analysis for liner plates, metal containments, and penetrations that considers all cyclic loads based on the anticipated number of cycles. Containment components may also be designed to ASME Section III Class 1 requirements. A Section III, MC or Class 1 fatigue analysis requires the calculation of the CUF based on the fatigue properties of the materials and the expected fatigue service of the component. The ASME code limits the CUF to a value less than ***or equal to*** one for acceptable fatigue design. The fatigue resistance of the liner plates or metal containments, and penetrations during the period of extended operation is an area of review.

4.6.1.1.2 Other Evaluations Based on CUF

Other evaluations also contain metal fatigue analysis requirements based on a CUF calculation, such as metal bellows designed to ASME NC-3649.4(e)(3) or NE-3366.2(e)(3). For these cases, the discussion relating to ASME Section III, MC or Class 1, in Subsection 4.6.1.1.1 of this review plan section, applies.

4.6.1.2 FSAR Supplement

Detailed information on the evaluation of TLAAs is contained in the renewal application. A summary description of the evaluation of TLAAs for the period of extended operation is contained in the applicant's FSAR supplement. The FSAR supplement is an area of review.

4.6.2 Acceptance Criteria

The acceptance criteria for the areas of review described in Subsection 4.6.1 of this review plan section delineate acceptable methods for meeting the requirements of the NRC's regulations in 10 CFR 54.21(c)(1).

4.6.2.1 Time-Limited Aging Analysis

Pursuant to 10 CFR 54.21(c)(1), an applicant must demonstrate one of the following:

(i) The analyses remain valid for the period of extended operation;

(ii) The analyses have been projected to the end of the extended period of operation; or

(iii) The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Specific acceptance criteria for fatigue of containment liner plates, metal containments, liner plate weld joints, dissimilar metal welds, penetration sleeves, and penetration bellows are:

4.6.2.1.1 ASME Section III, MC or Class 1

For containment liner plates, metal containments, and penetrations designed or analyzed to ASME MC or Class 1 requirements, the acceptance criteria, depending on the applicant's choice of 10 CFR 54.21(c)(1)(i), (ii), or (iii), are:

4.6.2.1.1.1 10 CFR 54.21(c)(1)(i)

The existing CUF calculations remain valid because the number of assumed cyclic loads will not be exceeded during the period of extended operation.

4.6.2.1.1.2 10 CFR 54.21(c)(1)(ii)

CLB fatigue analysis, per ASME Code Section III, was conducted for a 40-year life. The CUF calculations should be reevaluated based on an increased number of assumed cyclic loads to cover the period of extended operation. All cyclic loads considered in the original fatigue analyses (including Type A and Type B leak rate tests) should be reevaluated and revised as necessary. The revised analysis should show that the CUF will not exceed one, as required by the ASME code, during the period of extended operation.

4.6.2.1.1.3 10 CFR 54.21(c)(1)(iii)

The effects of aging on the intended function(s) will be adequately managed for the period of extended operation. The component could be replaced; the CUF for the replacement must be less than ***or equal to*** one during the period of extended operation.

An alternative aging management program provided by the applicant will be evaluated on a case-by-case basis to ensure that the aging effects will be managed such that the intended functions(s) will be maintained during the period of extended operation. In cases where a mitigation or inspection program is proposed, the aging management program may be evaluated against the 10 elements described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

4.6.2.1.2 Other Evaluations Based on CUF

The acceptance criteria in Subsection 4.6.2.1.2 of this review plan section apply.

4.6.2.2 FSAR Supplement

The specific criterion for meeting 10 CFR 54.21(d) is:

The summary description of the evaluation of TLAAs for the period of extended operation in the FSAR supplement is appropriate such that later changes can be controlled by 10 CFR 50.59. The description should contain information associated with the TLAAs regarding the basis for determining that the applicant has made the demonstration required by 10 CFR 54.21(c)(1).

4.6.3 Review Procedures

For each area of review described in Subsection 4.6.1 of this review plan section, the following review procedures should be followed:

4.6.3.1 Time-Limited Aging Analysis

4.6.3.1.1 ASME Section III, MC or Class 1

For containment liner plates, metal containments, and penetrations designed or analyzed to ASME MC or Class 1 requirements, the review procedures, depending on the applicant's choice of 10 CFR 54.21(c)(1)(i), (ii), or (iii), are:

4.6.3.1.1.1 10 CFR 54.2l(c)(1)(i)

The number of assumed transients used in the existing CUF calculations for the current operating term is compared to the extrapolation to 60 years of operation of the number of operating transients experienced to date. The comparison confirms that the number of transients in the existing analyses will not be exceeded during the period of extended operation.

4.6.3.1.1.2 10 CFR 54.21(c)(1)(ii)

Operating transient experience and a list of the increased number of assumed cyclic loads projected to the end of the period of extended operation are reviewed to ensure that the cyclic load projection is adequate. The revised CUF calculations based on the projected number of assumed cyclic loads are reviewed to ensure that the CUF remains less than one at the end of the period of extended operation.

The code of record should be used for the reevaluation, or the applicant may update to a later code edition pursuant to 10 CFR 50.55a. In the latter case, the reviewer verifies that the requirements in 10 CFR 50.55a are met.

4.6.3.1.1.3 10 CFR 54.21(c)(1)(iii)

The applicant's proposed aging management program to ensure that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation is reviewed. If the applicant proposed a component replacement before its CUF exceeds one, the reviewer verifies that the CUF for the replacement will remain less than ***or equal to*** one during the period of extended operation.

Other applicant proposed programs will be reviewed on a case-by-case basis.

4.6.3.1.2 Other Evaluations Based on CUF

The review procedures in Subsection 4.6.3.1 of this review plan section apply.

4.6.3.2 FSAR Supplement

The reviewer verifies that the applicant has provided information, to be included in the FSAR supplement, that includes a summary description of the evaluation of containment liner plate, metal containments, and penetrations fatigue TLAA. Table 4.6-1 of this review plan section contains examples of acceptable FSAR supplement information for this TLAA. The reviewer verifies that the applicant has provided a FSAR supplement with information equivalent to that in Table 4.6-1.

The staff expects to impose a license condition on any renewed license to require the applicant to update its FSAR to include this FSAR supplement at the next update required pursuant to 10 CFR 50.71(e)(4). As part of the license condition, until the FSAR update is complete, the applicant may make changes to the programs described in its FSAR supplement without prior NRC approval, provided that the applicant evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59.

As noted in Table 4.6-1, the applicant need not incorporate the implementation schedule into its FSAR. However, the review should verify that the applicant has identified and committed in the license renewal application to any future aging management activities, including enhancements and commitments to be completed before the period of extended operation. The staff expects to impose a license condition on any renewed license to ensure that the applicant will complete these activities no later than the committed date.

4.6.4 Evaluation Findings

The reviewer verifies that the applicant has provided sufficient information to satisfy the provisions of this review plan section and that the staff's evaluation supports conclusions of the following type, depending on the applicant's choice of 10 CFR 54.21(c)(1)(i), (ii), or (iii), to be included in the staff's safety evaluation report:

> The staff evaluation concludes that the applicant has provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1), that, for the containment liner plate or metal containment, and penetrations fatigue TLAA, [choose which is appropriate] (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the period of extended operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation. The staff also concludes that the FSAR supplement contains an appropriate summary description of the containment liner plate or metal containment, and penetrations fatigue TLAA evaluation for the period of extended operation as reflected in the license condition.

4.6.5 Implementation

Except in those cases in which the applicant proposes an acceptable alternative method, the method described herein will be used by the staff in its evaluation of conformance with NRC regulations.

4.6.6 References

1. ASME Boiler and Pressure Vessel Code, Section III, Division 2, "Code for Concrete Reactor Vessels and Containments," Subsection CC, "Concrete Containment," and Division 1, Subsection NE, "MC Components," American Society of Mechanical Engineers, New York, New York, 1989 or other editions as approved in 10 CFR 50.55a.

Table 4.6-1. Examples of FSAR Supplement for Containment Liner Plates, Metal Containments, and Penetrations Fatigue TLAA Evaluation

10 CFR 54.21(c)(1)(i) Example

TLAA	Description of Evaluation	Implementation Schedule*
Containment liner plates (or metal containment) and penetrations fatigue	The containment liner plates (or metal containment), liner weld joints, penetration sleeves, dissimilar metal welds, and penetration bellows provide a leak-tight barrier. A Section III, MC or Class 1 fatigue analysis limits the CUF to a value less than ***or equal to*** one for acceptable fatigue design. The existing CUF evaluation has been determined to remain valid because the number of assumed cyclic loads would not be exceeded during the period of extended operation.	Completed

10 CFR 54.21(c)(1)(ii) Example

TLAA	Description of Evaluation	Implementation Schedule*
Containment liner plates (or metal containment) and penetrations fatigue	The containment liner plates (or metal containment), liner weld joints, penetration sleeves, dissimilar metal welds, and penetration bellows provide a leak-tight barrier. A Section III, MC or Class 1 fatigue analysis limits the CUF to a value less than ***or equal to*** one for acceptable fatigue design. The CUF calculations have been reevaluated based on an increased number of assumed cyclic loads to cover the period of extended operation. The revised CUF will not exceed one during the period of extended operation.	Completed

10 CFR 54.21(c)(1)(iii) Example

TLAA	Description of Evaluation	Implementation Schedule*
Containment liner plates (or metal containment) and penetrations fatigue	The containment liner plates (or metal containment), liner weld joints, penetration sleeves, dissimilar metal welds, and penetration bellows provide a leak-tight barrier. A Section III, MC or Class 1 fatigue analysis limits the CUF to a value less than ***or equal to*** one for acceptable fatigue design. If the component is replaced, the CUF for the replacement will be shown to be less than one during the period of extended operation.	Program should be implemented before the period of extended operation.

Note: All containment components need not meet the same requirement. It is likely that the liner plate and the bellows may be evaluated per 10CFR54.21(c)(1)(i), while high energy penetrations may be evaluated per 10CFR54.21(c)(1)(ii).

* An applicant need not incorporate the implementation schedule into its FSAR. However, the reviewer should verify that the applicant has identified and committed in the license renewal application to any future aging management activities to be completed before the period of extended operation. The staff expects to impose a license condition on any renewed license to ensure that the applicant will complete these activities no later than the committed date.

4.7 OTHER PLANT-SPECIFIC TIME-LIMITED AGING ANALYSES

Item	Locator	Comment	Justification
1	4.7.3	Reorganize 1st paragraph	This paragraph appears to be written backwards. Alternative paragraph given with the intent to focus on the Rule and its requirements.

4.7 OTHER PLANT-SPECIFIC TIME-LIMITED AGING ANALYSES

Review Responsibilities

Primary - Branch responsible for the TLAA issues
Secondary - Other branches responsible for systems, as appropriate

4.7.1 Areas of Review

There are certain plant-specific safety analyses that may have been based on an explicitly assumed 40-year plant life (for example, aspects of the reactor vessel design) and may, therefore, be time-limited aging analyses (TLAAs.) Pursuant to 10 CFR 54.21(c), a license renewal applicant is required to evaluate TLAAs. The definition of TLAAs is provided in 10 CFR 54.3 and in Section 4.1 of this standard review plan.

TLAAs may have evolved since issuance of a plant's operating license, and are plant-specific. As indicated in 10 CFR 54.30, the adequacy of the plant's CLB, which includes TLAAs, is not an area within the scope of the license renewal review. Any question regarding the adequacy of the CLB must be addressed under the backfit rule (10 CFR 50.109) and is separate from the license renewal process.

License renewal reviews focus on the period of extended operation. Pursuant to 10 CFR 54.30, if the reviews required by 10 CFR 54.21(a) or (c) show that there is not reasonable assurance during the current license term that licensed activities will be conducted in accordance with the CLB, the licensee is required to take measures under its current license to ensure that the intended function of those systems, structures, or components will be maintained in accordance with the CLB throughout the term of the current license. The adequacy of the measures for the term of the current license is not within the scope of the license renewal review.

Pursuant to 10 CFR 54.21(c), an applicant must provide a listing of TLAAs and plant-specific exemptions that are based on TLAAs. The staff reviews the applicant's identification of TLAAs and exemptions separately, following the guidance in Section 4.1 of this standard review plan.

Based on lessons learned in the review of the initial license renewal applications, the staff has developed review procedures for the evaluation of certain TLAAs. If an applicant identifies these TLAAs as applicable to its plant, the staff reviews them separately, following the guidance in Sections 4.2 through 4.6. The staff reviews other TLAAs that are identified by the applicant, following the generic guidance in this review plan section. For particular systems, the staff from branches responsible for those systems may be requested to assist in the review, as appropriate.

The following areas relating to a TLAA are reviewed:

4.7.1.1 Time-Limited Aging Analysis

The evaluation of the TLAA for the period of extended operation is reviewed.

4.7.1.2 FSAR Supplement

The FSAR supplement summarizing the evaluation of the TLAA for the period of extended operation in accordance with 10 CFR 54.21(d) is reviewed.

4.7.2 Acceptance Criteria

The acceptance criteria for the areas of review described in Subsection 4.7.1 of this review plan section delineate acceptable methods for meeting the requirements of the NRC's regulations in 10 CFR 54.21(c)(1).

4.7.2.1 Time-Limited Aging Analysis

Pursuant to 10 CFR 54.21(c)(1)(i) - (iii), an applicant must demonstrate one of the following for the TLAAs:

(i) The analyses remain valid for the period of extended operation;

(ii) The analyses have been projected to the end of the extended period of operation; or

(iii) The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

4.7.2.2 FSAR Supplement

The specific criterion for meeting 10 CFR 54.21(d) is:

The summary description of the evaluation of TLAAs for the period of extended operation in the FSAR supplement is appropriate such that later changes can be controlled by 10 CFR 50.59. The description should contain information associated with the TLAAs regarding the basis for determining that the applicant has made the demonstration required by 10 CFR 54.21 (c)(1).

4.7.3 Review Procedures

For certain applicants, plant-specific analyses may meet the definition of a TLAA as given in 10CFR54.3. The concern for License Renewal is that these analyses may not have properly considered the length of the extended period of operation, the consideration of which may change conclusions with regard to safety and the capability of SSCs within the scope of the Rule to perform or one or more safety functions. The review of these TLAAs will provide the assurance that the aging effect is properly addressed through the period of extended operation.~~The requirement for evaluation of TLAAs captures, for review of applications for license renewal, certain plant-specific aging analyses that are explicitly based on the duration of the current operating license of the plant. The concern is that these aging analyses do not cover the period of extended operation. Unless these analyses are evaluated, there is no assurance that the systems, structures, and components addressed by these analyses can perform their intended function(s) during the period of extended operation.~~

For each area of review described in Subsection 4.7.1 of this review plan section, the following review procedures are followed:

4.7.3.1 Time-Limited Aging Analysis

For each TLAA identified, the review procedures depend on the applicant's choice of methods of compliance from those identified in 10 CFR 54.21(c)(1)(i), (ii), or (iii), as follows:

4.7.3.1.1 10 CFR 54.21(c)(1)(i)

Justification provided by the applicant is reviewed to verify that the existing analyses are valid for the period of extended operation. The existing analyses should be shown to be bounding even during the period of extended operation.

The applicant should describe the TLAA with respect to the objectives of the analysis, assumptions used in the analysis, conditions, acceptance criteria, relevant aging effects, and intended function(s). The applicant should show that (1) conditions and assumptions used in the analysis already address the relevant aging effects for the period of extended operation, and (2) acceptance criteria are maintained to provide reasonable assurance that the intended function(s) is maintained for renewal. Thus, no reanalysis is necessary for renewal.

In some instances, the applicant may identify activities to be performed to verify the assumption basis of the calculation, such as cycle counting. An evaluation of that activity should be provided by the applicant. The reviewer should assure that the applicant's activity is sufficient to confirm the calculation assumptions for the 60-year period.

If the TLAA must be modified or recalculated to extend the period of evaluation to consider the period of extended operation, the reevaluation should be addressed under 10 CFR 54.21(c)(1)(ii).

4.7.3.1.2 10 CFR 54.21(c)(1)(ii)

The documented results of the revised analyses are reviewed to verify that their period of evaluation is extended such that they are valid for the period of extended operation, for example, 60 years. The applicable analysis technique can be the one that is in effect in the plant's CLB at the time of filing of the renewal application.

The applicant may recalculate the TLAA using a 60-year period to show that the TLAA acceptance criteria continue to be satisfied for the period of extended operation. The applicant may also revise the TLAA by recognizing and reevaluating any overly conservative conditions and assumptions. Examples include relaxing overly conservative assumptions in the original analysis, using new or refined analytical techniques, and performing the analysis using a 60-year period. The applicant shall provide a sufficient description of the analysis and document the results of the reanalysis to show that it is satisfactory for the 60-year period.

As applicable, the plant's code of record should be used for the reevaluation, or the applicant may update to a later code edition pursuant to 10 CFR 50.55a. In the latter case, the reviewer verifies that the requirements in 10 CFR 50.55a are met.

In some cases, the applicant may identify activities to be performed to verify the assumption basis of the calculation, such as cycle counting. An evaluation of that activity should be provided by the applicant. The reviewer should assure that the applicant's activity is sufficient to confirm the calculation assumptions for the 60-year period.

4.7.3.1.3 10 CFR 54.21(c)(1)(iii)

Under this option, the applicant would propose to manage the aging effects associated with the TLAA by an aging management program in the same manner as would be described in the IPA in 10 CFR 54.21(a)(3). The reviewer reviews the applicant's aging management program to verify that the effects of aging on the intended function(s) will be adequately managed consistent with the CLB for the period of extended operation.

The applicant should identify the structures and components associated with the TLAA. The TLAA should be described with respect to the objectives of the analysis, conditions, assumptions used, acceptance criteria, relevant aging effects, and intended function(s). In cases where a mitigation or inspection program is proposed, the reviewer may use the guidance provided in Branch Technical Position RLSB-1 of this standard review plan to ensure that the effects of aging on the structure and component intended function(s) are adequately managed for the period of extended operation.

4.7.3.2 FSAR Supplement

The reviewer verifies that the applicant has provided information, to be included in the FSAR supplement, that includes a summary description of the evaluation of each TLAA. Each such summary description is reviewed to verify that it is appropriate such that later changes can be controlled by 10 CFR 50.59. The description should contain information that the TLAAs have been dispositioned for the period of extended operation. Sections 4.2 through 4.6 of this standard review plan contain examples of acceptable FSAR supplement information for TLAA evaluation.

The staff expects to impose a license condition on any renewed license to require the applicant to update its FSAR to include this FSAR supplement at the next update required pursuant to 10 CFR 50.71(e)(4). As part of the license condition, until the FSAR update is complete, the applicant may make changes to the programs described in its FSAR supplement without prior NRC approval, provided that the applicant evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59.

As noted in Sections 4.2 through 4.6, an applicant need not incorporate the implementation schedule into its FSAR. However, the review should verify that the applicant has identified and committed in the license renewal application to any future aging management activities, including enhancements and commitments to be completed before the period of extended operation.

The staff expects to impose a license condition on any renewed license to ensure that the applicant will complete these activities no later than the committed date.

4.7.4 Evaluation Findings

The reviewer verifies that the applicant has provided sufficient information to satisfy the provisions of this review plan section and that the staff's evaluation supports conclusions of the following type, depending on the applicant's choice of 10 CFR 54.21(c)(1)(i), (ii), or (iii), to be included in the staff's safety evaluation report:

> The staff concludes that the applicant has provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1), that, for the (name of specific) TLAA, [choose which is appropriate] (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the period of

extended operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation. The staff also concludes that the FSAR supplement contains an appropriate summary description of this TLAA evaluation for the period of extended operation as reflected in the license condition.

4.7.5 Implementation

Except in those cases in which the applicant proposes an acceptable alternative method, the method described herein will be used by the staff in its evaluation of conformance with NRC regulations.

4.7.6 References

None

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