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NUREG-1801, Rev. 2

Generic Aging Lessons Learned (GALL) Report

Final Report

Office of Nuclear Reactor Regulation

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Generic Aging Lessons Learned (GALL) Report

Final Report

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ABSTRACT

NUREG-1801, “The Generic Aging Lessons Learned (GALL) Report” (GALL Report), contains the staff's generic evaluation of the existing plant programs and documents the technical basis for determining where existing programs are adequate without modification and where existing programs should be augmented for the period of extended operation. The evaluation results documented in the GALL Report indicate that many of the existing programs are adequate to manage the aging effects for structures or components for license renewal without change. The GALL Report also contains recommendations on specific areas for which existing programs should be augmented for license renewal. An applicant may reference the GALL Report in a license renewal application to demonstrate that the programs at the applicant’s facility correspond to those reviewed and approved in the GALL Report. The GALL Report should be treated as an approved topical report. However, if an applicant takes credit for a program in the GALL Report, it is incumbent on the applicant to ensure that the conditions and operating experience at the plant are bounded by the conditions and operating experience for which the GALL Report program was evaluated. If these bounding conditions are not met, it is incumbent on the applicant to address the additional effects of aging and augment the GALL Report aging management program(s) as appropriate. The staff will verify that the applicant’s programs are consistent with those described in the GALL Report and/or with plant conditions and operating experience during the performance of an aging management program audit and review. The focus of the balance of the staff’s review of a license renewal application is on those programs that an applicant has enhanced to be consistent with the GALL Report, those programs for which the applicant has taken an exception to the program described in the GALL Report, and plant-specific programs not described in the GALL Report. The information in the GALL Report has been incorporated into the NUREG-1800, “Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants,” as directed by the Commission, to improve the efficiency of the license renewal process.

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ABBREVIATIONS

ACAR	aluminum conductor aluminum alloy reinforced
ACRS	aluminum conductor steel reinforced
ACI	American Concrete Institute
ADS	automatic depressurization system
AFW	auxiliary feedwater
ALARA	as low as reasonably achievable
AMP	aging management program
AMR	aging management review
ANSI	American National Standards Institute
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
B&PV	boiler and pressure vessel
B&W	Babcock & Wilcox
BWR	boiling water reactor
BWRVIP	Boiling Water Reactor Vessel and Internals Project
CASS	cast austenitic stainless steel
CB	core barrel
CCCW	closed-cycle cooling water
CE	Combustion Engineering
CEA	control element assembly
CFR	Code of Federal Regulations
CFS	core flood system
CLB	current licensing basis
CRD	control rod drive
CRDM	control rod drive mechanism
CRDRL	control rod drive return line
CRGT	control rod guide tube
CVCS	chemical and volume control system
DC	direct current
DHR	decay heat removal
DSCSS	drywell and suppression chamber spray system
EDG	emergency diesel generator
EPDM	ethylene propylene diene monosomer
EPR	ethylene-propylene rubber
EPRI	Electric Power Research Institute

EQ	environmental qualification
FAC	flow-accelerated corrosion
FERC	Federal Energy Regulatory Commission
FRN	Federal Register Notice
FSAR	Final Safety Analysis Report
FW	feedwater
GALL	Generic Aging Lessons Learned
GE	General Electric
GL	generic letter
HDPE	high density polyethylene
HELBs	high-energy line breaks
HP	high pressure
HPCI	high-pressure coolant injection
HPCS	high-pressure core spray
HPSI	high-pressure safety injection
HVAC	heating, ventilation, and air conditioning
I&C	instrumentation and control
IASCC	irradiation assisted stress corrosion cracking
IC	isolation condenser
ID	inside diameter
IEB	inspection and enforcement bulletin
IEEE	Institute of Electrical and Electronics Engineers
IGA	intergranular attack
IGSCC	intergranular stress corrosion cracking
IN	information notice
INPO	Institute of Nuclear Power Operations
IPA	integrated plant assessment
IR	insulation resistance
IRM	intermediate range monitor
ISI	inservice inspection
LER	licensee event report
LG	lower grid
LOCA	loss of coolant accident
LP	low pressure
LPCI	low-pressure coolant injection
LPCS	low-pressure core spray

LPM	loose part monitoring
LPRM	low-power range monitor
LPSI	low-pressure safety injection
LRAAI	license renewal applicant action items
LRT	leak rate test
LWR	light water reactor
MFW	main feedwater
MIC	microbiologically influenced corrosion
MS	main steam
MSR	moisture separator/reheater
MT	magnetic particle testing
NDE	nondestructive examination
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NPAR	nuclear plant aging research
NPS	nominal pipe size
NRC	Nuclear Regulatory Commission
NRMS	normalized root mean square
NSAC	Nuclear Safety Analysis Center
NSSS	nuclear steam supply system
NUMARC	Nuclear Management and Resources Council
OCCW	open-cycle cooling water
OD	outside diameter
ODSCC	outside diameter stress corrosion cracking
OM	operation and maintenance
PT	penetrant testing
PVC	polyvinyl chloride
PWR	pressurized water reactor
PWSCC	primary water stress corrosion cracking
QA	quality assurance
RCCA	rod control cluster assemblies
RCIC	reactor core isolation cooling
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system

RG	Regulatory Guide
RHR	residual heat removal
RMS	root mean square
RWC	reactor water cleanup
RWST	refueling water storage tank
RWT	refueling water tank
SAW	submerged arc weld
SCC	stress corrosion cracking
SDC	shutdown cooling
SFP	spent fuel pool
SG	steam generator
S/G	standards and guides
SIL	services information letter
SIT	safety injection tank
SLC	standby liquid control
SOER	significant operating experience report
SR	silicon rubber
SRM	source range monitor
SRM	staff requirements memorandum
SRP-LR	standard review plan for license renewal
SS	stainless steel
SSC	systems, structures, and components
TGSCC	transgranular stress corrosion cracking
TLAA	time-limited aging analysis
UCS	Union of Concerned Scientists
UHS	ultimate heat sink
USI	unresolved safety issue
UT	ultrasonic testing
UV	ultraviolet
XPLE	cross-linked polyethylene

INTRODUCTION

NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," is referenced as a technical basis document in NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants" (SRP-LR). The GALL Report lists generic aging management reviews (AMRs) of systems, structures, and components (SSCs) that may be in the scope of license renewal applications (LRAs) and identifies aging management programs (AMPs) that are determined to be acceptable to manage aging effects of SSCs in the scope of license renewal, as required by 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants." If an applicant takes credit for a program in the GALL Report, it is incumbent on the applicant to ensure that the conditions and operating experience at the plant are bounded by the conditions and operating experience for which the GALL Report was evaluated. If these bounding conditions are not met, it is incumbent on the applicant to address the additional effects of aging and augment the GALL report AMPs as appropriate.

If an LRA references the GALL Report as the approach used to manage aging effect(s), the NRC staff will use the GALL Report as a basis for the LRA assessment consistent with guidance specified in the SRP-LR.

BACKGROUND

Revision 0 of the GALL Report

By letter dated March 3, 1999, the Nuclear Energy Institute (NEI) documented the industry's views on how existing plant programs and activities should be credited for license renewal. The issue can be summarized as follows:

To what extent should the staff review existing programs relied on for license renewal to determine whether an applicant has demonstrated reasonable assurance that such programs will be effective in managing the effects of aging on the functionality of structures and components during the period of extended operation?

In a staff paper (SECY-99-148, "Credit for Existing Programs for License Renewal") dated June 3, 1999, the staff described options for crediting existing programs and recommended one option that the staff believed would improve the efficiency of the license renewal process.

By a staff requirements memorandum (SRM), dated August 27, 1999, the Commission approved the staff's recommendation and directed the staff to focus the staff review guidance in the SRP-LR on areas where existing programs should be augmented for license renewal. The staff would develop a GALL Report to document the staff's evaluation of generic existing programs. The GALL Report would document the staff's basis for determining which existing programs are adequate without modification and which existing programs should be augmented for license renewal. The GALL Report would be referenced in the SRP-LR as a basis for determining the adequacy of existing programs.

The GALL Report (Revision 0) is built on a previous report, NUREG/CR-6490, "Nuclear Power Plant Generic Aging Lessons Learned (GALL)," which is a systematic compilation of plant aging information. The GALL Report (Revision 0) extended the information in NUREG/CR-6490 to provide an evaluation of the adequacy of AMPs for license renewal. The NUREG/CR-6490 report was based on information in over 500 documents: Nuclear Plant Aging Research (NPAR) program reports sponsored by the Office of Nuclear Regulatory Research, Nuclear Management and Resources Council (NUMARC, now NEI) industry reports addressing license renewal for major structures and components, licensee event reports (LERs), information notices, generic letters, and bulletins. The staff also considered information contained in the reports provided by the Union of Concerned Scientists (UCS) in a letter dated May 5, 2000.

Following the general format of NUREG-0800 for major plant sections, except for refueling water, chilled water, residual heat removal, condenser circulating water, and condensate storage system in pressurized water reactor (PWR) and boiling water reactor (BWR) power plants, the staff reviewed the aging effects on components and structures, identified the relevant existing programs, and evaluated program attributes to manage aging effects for license renewal. The GALL Report (Revision 0) was prepared with the technical assistance of Argonne National Laboratory and Brookhaven National Laboratory. As directed in the SRM, the GALL Report (Revision 0) had the benefit of the experience of the staff members who conducted the review of the initial LRAs. Also, as directed in the SRM, the staff sought stakeholders' participation in the development of this report. The staff held many public meetings and workshops to solicit input from the public. The staff also requested comments from the public on the draft improved license renewal guidance documents, including the GALL Report, in the Federal Register Notice, Vol. 65, No. 170, August 31, 2000. The staff's analysis of stakeholder comments is documented in NUREG-1739. These documents can be found online at http://www.nrc.gov/reading-rm/doc-collections/.

Revision 1 of the GALL Report

Based on lessons learned from the reviews of LRAs and other public input including industry comments, the NRC staff proposed changes to the GALL Report (Revision 0) to make the GALL Report (Revision 1) more efficient. A preliminary version of Revision 1 of the GALL Report was posted on the NRC public web page on September 30, 2004. The draft revisions of the GALL Report (Vol. 1 and Vol. 2) were further refined and issued for public comment on January 31, 2005. The staff also held public meetings with stakeholders to facilitate dialogue and to discuss comments. The staff subsequently took into consideration comments received (see NUREG-1832) and incorporated its dispositions into the September 2005 version of the GALL Report (Revision 1).

Revision 2 of the GALL Report

Based on further lessons learned from the reviews of LRAs, operating experience obtained after Revision 1 was issued, and other public input including industry comments, the NRC staff proposed changes to the GALL Report (Revision 1). A preliminary version of Revision 2 of the GALL Report was posted on the NRC public web page on December 23, 2009. The draft revision of the GALL Report was further refined and issued for public comment on May 18, 2010. The staff held public meetings with stakeholders to facilitate dialogue and to discuss comments. The staff subsequently took into consideration comments received (see NUREG-1950) and incorporated their dispositions into the December 2010, Revision 2 of the GALL Report.

Revision 2 – Operating Experience Evaluation

The extended operation of nuclear reactors necessitates a thorough analysis of existing experience. An operating experience review was performed by NRC staff to identify necessary additions or modifications to the GALL Report based on this experience. Both domestic and foreign operating experience was reviewed.

The staff from the Division of License Renewal (DLR) analyzed operating experience information during a screening review of domestic operating experience, foreign operating experience from the international Incident Reporting System (IRS) database, and NRC generic communications. The information reviewed included operating experience from January 2004 to approximately April 2009.

Domestic Operating Experience: The NRC, Office of Research (RES) provided a listing of Licensee Event Reports (LERs) related to failures, cracking, degradation, etc. of passive components. These results were reviewed by NRC staff. The operating experience elements of numerous AMPs were updated to reflect relevant operating experience identified by the review. In addition, the operating experience review identified a number of examples where vibration-induced fatigue caused cracking of plant components. The staff subsequently modified GALL AMP XI.M35, “One-time Inspection of ASME Code Class 1 Small-bore Piping,” to address these concerns.

Foreign Operating Experience: The international IRS, jointly operated by the International Atomic Energy Agency (IAEA) and the Nuclear Energy Agency (NEA), is used to compile and analyze information on NPP events and also promotes a systematic approach to collecting and disseminating the lessons learned from international operating experience. Events of safety significance and events from which lessons can be learned are reported to the IRS. The main objective of the IRS is to enhance the safety of NPPs by reducing the frequency and severity of safety significant unusual events at NPPs. NRC staff also reviewed international operating experience from: (a) the Organization for Economic Co-operation and Development (OECD) OECD/NEA Piping Failure Data Exchange database (including the data from 1970 to 2009) and (b) the OECD/NEA Stress Corrosion Cracking and Cable Aging database.

The foreign operating experience databases were queried for reports relating to aging effects in passive components. The identified reports were analyzed to determine if there were any revisions necessary for either AMR items or AMP content. Many of the reports identified MEAP combinations that were already addressed by the GALL Report. Some of the items were specific to foreign plants and not generically applicable to U.S. pressurized water reactors (PWRs) and boiling water reactors (BWRs). In addition, the IRS identified that stainless steel components are subject to chloride-induced stress corrosion cracking when they are exposed to the air-outdoor environment that involves a salt-laden atmospheric condition or salt water spray. Based on this review result, relevant SRP-LR sections were added and further evaluation is now recommended for those environmental conditions.

OVERVIEW OF THE GALL REPORT EVALUATION PROCESS

The GALL Report contains 11 chapters and an appendix. The majority of the chapters contain summary descriptions and tabulations of evaluations of AMPs for a large number of structures and components in major plant systems found in light-water reactor nuclear power plants. The major plant systems include the containment structures (Chapter II), structures and component supports (Chapter III), reactor vessel, internals and reactor coolant system (Chapter IV), engineered safety features (Chapter V), electrical components (Chapter VI), auxiliary systems (Chapter VII), and steam and power conversion system (Chapter VIII).

Chapter I of the GALL Report addresses the application of the ASME Code for license renewal. Chapter IX contains definitions of a selection of standard terms used within the GALL Report. Chapter X contains the time-limited aging analysis evaluation of AMPs under 10 CFR 54.21(c)(1)(iii). Chapter XI contains the AMPs for the structures and mechanical and electrical components. The Appendix of the GALL Report addresses quality assurance (QA) for AMPs.

The evaluation process for the AMPs and the application of the GALL Report is described in this document. The results of the GALL effort are presented in tabular format in the GALL Report.

<u>Table Column Headings</u>

The following describes the information presented in each column of the tables in Chapters II through VIII contained in this report.

Column Heading	**Description**
Item	Identifies a unique number for the item (i.e., VII.G.A-91). The first part of the number indicates the chapter and AMR system (e.g., VII.G is in the auxiliary systems, fire protection system), and the second part is a unique chapter-specific identifier within a chapter (e.g., A-91 for auxiliary systems).
Link	For each row in the subsystem tables, this item identifies the corresponding row identifier from GALL Volume 2, Rev. 1, if the row was derived from the earlier version of this report. Otherwise, blanks indicate a new row in this revision of the GALL Report.
Structure and/ or Component	Identifies the structure or components to which the row applies.
Material	Identifies the material of construction. See Chapter IX of this report for further information.
Environment	Identifies the environment applicable to this row. See Chapter IX of this report for further information.
Aging Effect/ Mechanism	Identifies the applicable aging effect and mechanism(s). See Chapter IX of the GALL Report for more information.
Aging Management Programs	Identifies the time-limited aging analysis or AMP found acceptable for adequately managing the effects of aging. See Chapters X and XI of the GALL Report.
Further Evaluation	Identifies whether further evaluation is needed.

The staff's evaluation of the adequacy of each generic AMP to manage certain aging effects for particular structures and components is based on its review of the following 10 program elements in each AMP.

AMP Element	Description
1. Scope of the Program	The scope of the program should include the specific structures and components subject to an AMR.
2. Preventive Actions	Preventive actions should mitigate or prevent the applicable aging effects.
3. Parameters Monitored or Inspected	Parameters monitored or inspected should be linked to the effects of aging on the intended functions of the particular structure and component.
4. Detection of Aging Effects	Detection of aging effects should occur before there is a loss of any structure and component intended function. This includes aspects such as method or technique (i.e., visual, volumetric, surface inspection), frequency, sample size, data collection, and timing of new/one-time inspections to ensure timely detection of aging effects.
5. Monitoring and Trending	Monitoring and trending should provide for prediction of the extent of the effects of aging and timely corrective or mitigative actions.
6. Acceptance Criteria	Acceptance criteria, against which the need for corrective action will be evaluated, should ensure that the particular structure and component's intended functions are maintained under all current licensing basis (CLB) design conditions during the period of extended operation.
7. Corrective Actions	Corrective actions, including root cause determination and prevention of recurrence, should be timely.
8. Confirmation Process	The confirmation process should ensure that preventive actions are adequate and that appropriate corrective actions have been completed and are effective.
9. Administrative Controls	Administrative controls should provide a formal review and approval process.
10. Operating Experience	Operating experience involving the AMP, including past corrective actions resulting in program enhancements or additional programs, should provide objective evidence to support a determination that the effects of aging will be adequately managed so that the structure and component intended functions will be maintained during the period of extended operation.

On the basis of its evaluation, if the staff determined that a program is adequate to manage certain aging effects for a particular structure or component without change, the "Further Evaluation" entry will indicate that no further evaluation is recommended for license renewal.

Chapter XI of the GALL Report contains the staff's evaluation of generic aging management programs that are relied on in the GALL Report, such as the ASME Section XI inservice inspection, water chemistry, or structures monitoring program.

APPLICATION OF THE GALL REPORT

The GALL Report is a technical basis document to the SRP-LR, which provides the staff with guidance in reviewing an LRA. The GALL Report should be treated in the same manner as an approved topical report that is generically applicable. An applicant may reference the GALL Report in an LRA to demonstrate that the programs at the applicant's facility correspond to those reviewed and approved in the GALL Report.

If an applicant takes credit for a program in GALL, it is incumbent on the applicant to ensure that the plant program contains all the elements of the referenced GALL program. In addition, the conditions and operating experience at the plant must be bound by the conditions and operating experience for which the GALL program was evaluated, otherwise it is incumbent on the applicant to augment the GALL program as appropriate to address the additional aging effects. The above verifications must be documented on-site in an auditable form. The applicant must include a certification in the LRA that the verifications have been completed.

The GALL Report contains one acceptable way to manage aging effects for license renewal. An applicant may propose alternatives for staff review in its plant-specific LRA. Use of the GALL Report is not required, but its use should facilitate both preparation of an LRA by an applicant and timely, uniform review by the NRC staff.

In addition, the GALL Report does not address scoping of structures and components for license renewal. Scoping is plant-specific, and the results depend on the plant design and CLB. The inclusion of a certain structure or component in the GALL Report does not mean that this particular structure or component is within the scope of license renewal for all plants. Conversely, the omission of a certain structure or component in the GALL Report does not mean that this particular structure or component is not within the scope of license renewal for any plants.

The GALL Report contains an evaluation of a large number of structures and components that may be in the scope of a typical LRA. The evaluation results documented in the GALL Report indicate that many existing, typical generic aging management programs are adequate to manage aging effects for particular structures or components for license renewal without change. The GALL Report also contains recommendations on specific areas for which existing generic programs should be augmented (require further evaluation) for license renewal and documents the technical basis for each such determination. In addition, the GALL Report identifies certain SSCs that may or may not be subject to particular aging effects, and those for which industry groups are developing generic aging management programs or investigating whether aging management is warranted.

The Appendix of the GALL Report addresses quality assurance (QA) for aging management programs. Those aspects of the aging management review process that affect the quality of safety-related structures, systems, and components are subject to the QA requirements of Appendix B to 10 CFR Part 50. For nonsafety-related structures and components subject to an AMR, the existing 10 CFR Part 50, Appendix B, QA program may be used by an applicant to address the elements of the corrective actions, confirmation process, and administrative controls for an aging management program for license renewal.

The GALL Report provides a technical basis for crediting existing plant programs and recommending areas for program augmentation and further evaluation. The incorporation of the

GALL Report information into the SRP-LR, as directed by the Commission, should improve the efficiency of the license renewal process and better focus staff resources.

CHAPTER I

APPLICATION OF THE ASME CODE

The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Division 1, Sections III (design) and XI (inservice inspection requirements) were developed and are revised periodically by industry code committees composed of representatives of utilities, reactor designers, architect-engineers, component manufacturers, insurance companies, the U.S. Nuclear Regulatory Commission (NRC), and others. In 1971, the Atomic Energy Commission (AEC), the predecessor of the NRC, incorporated the ASME Boiler and Pressure Vessel Code into the regulations in 10 CFR 50.55a through issuance of the Federal Register Notice (FRN) for the final rule (36 FR 11423 [June 12, 1971]).

The Statements of Consideration (SOCs) for the initial issuance of 10 CFR 50.55a provide the bases for AEC's endorsement and use of the ASME Code:

> "It has been generally recognized that, for boiling and pressurized water-cooled reactors, pressure vessels, piping, pumps, and valves which are part of the reactor coolant pressure boundary should, as a minimum, be designed, fabricated, inspected, and tested in accordance with the requirements of the applicable American Society of Mechanical Engineers (ASME) codes in effect at the time the equipment is purchased[.]"
>
> "Because of the safety significance of uniform early compliance by the nuclear industry with the requirements of these ASME codes and published code revisions, the Commission has adopted the following amendments to Part 50 and 115, which require that certain components and systems of water-cooled reactors important to safety comply with these codes and appropriate revisions to the codes at the earliest feasible time."
>
> "Compliance with the provisions of the amendments and the referenced codes is intended to insure a basic, sound quality level."

These ASME Code sections are based on the collective engineering judgment of the code committees and document the conditions that must be monitored, the inspection techniques to identify those conditions, the frequency of the inspections, and the acceptance criteria that the inspections' results must meet in order to assure the integrity of the structures and components considered in the code. The NRC has accepted this engineering judgment by endorsing the use of selected sections of the ASME Code, as incorporated in 10 CFR 50.55a.

In addition, the NRC periodically amends 10 CFR 50.55a and issues FRNs about this rule in order to endorse, by reference, newer editions and ASME Code Addenda subject to the modifications and limitations identified in 10 CFR 50.55a. At the time of this Standard Review Plan for License Renewal (SRP-LR) (NUREG-1800) and Generic Aging Lessons Learned (GALL) Report (NUREG-1801) update, the most recent editions of the ASME Code Sections III and XI were endorsed in 73 FR 52730-52750 (September 10, 2008). As stated in 65 FR 53050 (August 31, 2000):

> "To ensure that the GALL report conclusions will remain valid when future editions of the ASME Code are incorporated into the NRC regulations by the 10 CFR 50.55a rulemaking, the staff will perform an evaluation of these later editions for their adequacy for license renewal using the 10-element program evaluation described in the GALL Report as part of the 10 CFR 50.55a rulemaking."

The staff will document this evaluation in the SOC accompanying future 10 CFR 50.55a amendments, which will be published in a FRN.

To aid applicants in the development of their license renewal applications, the staff has developed a list of aging management programs (AMPs) in the GALL Report that are based on conformance with the 10-program element criteria defined in Section A.1.2.3 of the SRP-LR. Some of the AMPs referenced in the GALL Report are based entirely or in part on compliance with the requirements of ASME Section XI, as endorsed for use through reference in 10 CFR 50.55a., The staff has determined that the referenced ASME Section XI programs or requirements provide an acceptable basis for managing the effects of aging during the period of extended operation for these AMPs, except where noted and augmented in the GALL Report.

For aging management purposes, consideration of the acceptability for license renewal of ASME Section XI editions and addenda from the 1995 edition through the 2004 Addenda are discussed in FRNs 67 FR 60520 (September 26, 2002); 69 FR 58804 (October 1, 2004); and 73 FR 52730 (September 10, 2008) (via update of 10 CFR 50.55a). These FRNs provide that ASME Section XI editions and addenda from the 1995 edition through the 2004 edition, as modified and limited in the final rule, are acceptable and the conclusions in the current GALL Report at the time of the FRN issuance remain valid. Future FRNs that amend 10 CFR 50.55a will discuss the acceptability of editions and addenda more recent than the 2004 edition for their applicability for aging management for license renewal. Therefore, except where noted and augmented in the GALL Report, the following ASME Section XI editions and addenda are acceptable and should be treated as consistent with the GALL Report: (1) from the 1995 edition to the 2004 edition, as modified and limited in 10 CFR 50.55a, and (2) more recent editions, as evaluated for their adequacy for license renewal and discussed in the accompanying FRN for 10 CFR 50.55a rulemaking endorsing those specific editions. Hence, applicants for renewal should justify any exception to use an ASME Section XI edition or addenda that is (1) earlier than the 1995 edition, (2) not endorsed in 10 CFR 50.55a, or (3) not adequate for license renewal as discussed in the FRN issuing the 10 CFR 50.55a amendment.

In some cases, the staff has determined that specific requirements in ASME Section XI need to be augmented to ensure adequate aging management consistent with the license renewal rule. Thus, some of the AMPs in the GALL Report provide for additional augmented actions. For these situations, applicants for renewal should review the recommendations in the GALL Report and discuss proposed enhancements in their LRAs.

Pursuant to 10 CFR 50.55a(g)(4), a nuclear licensee is required to amend its current licensing basis (CLB) by updating its ASME Section XI edition and addenda of record to the most recently endorsed edition and addenda referenced in 10 CFR 50.55a one year prior to entering the next 10-year internal inservice inspection (ISI) for its unit. Pursuant to 10 CFR 54.21(b), an applicant for license renewal is required to periodically submit updates of its LRA to identify any changes in its CLB that materially affect the contents of the LRA. The rule requires an update of the LRA each year following the submittal of the application and an additional update 3 months prior to the completion of the NRC's review of the LRA. If an applicant's ASME Section XI edition of record is updated under the requirements of 10 CFR 50.55a(g)(4) during the NRC's review of the LRA, the applicant should update those AMPs in the LRA that are impacted by this change in the CLB when the applicant submits the next update of the LRA required by 10 CFR 54.21(b).

The current regulatory process, including 10 CFR 50.55a, continues into the period of extended operation. The NRC Director of the Office of Nuclear Reactor Regulation (NRR) may approve a licensee-proposed alternative to ASME Section XI if it is submitted as a relief request in accordance with 10 CFR 50.55a(a)(3). The staff's approval of an alternative program/relief request typically does not extend beyond the current 10-year interval for which the alternative was proposed. For cases in which this interval extends beyond the initial 40-year license period into the renewed license period, the approved relief remains in effect until the end of that interval, consistent with the specific approval (60 FR 22461, 22483).

Pursuant to 10 CFR 50.55a(b)(5), licensees may apply ASME Code cases listed in NRC Regulatory Guide (RG) 1.147, through the most recent endorsed revision, without NRC approval, subject to the limitations contained in the rule. The rule permits licensees to continue to apply the Code case, or a most recent version that is incorporated by the RG, until the end of the 10-year interval. For cases in which this interval extends beyond the initial 40-year license period into the renewal period, the Code case, or a more recent endorsed version, remains in effect until the end of that interval, consistent with 10 CFR 50.55a(b)(5) and the statements of consideration for the final license renewal rule 60 FR 22461.

CHAPTER II

CONTAINMENT STRUCTURES

A. Pressurized Water Reactor (PWR) Containments

B. Boiling Water Reactor (BWR) Containments

PRESSURIZED WATER REACTOR (PWR) CONTAINMENTS

A1. Concrete Containments (Reinforced and Prestressed)

A2. Steel Containments

A3. Common Components

A1. CONCRETE CONTAINMENTS (REINFORCED AND PRESTRESSED)

Systems, Structures, and Components

This section addresses the elements of pressurized water reactor (PWR) concrete containment structures. Concrete containment structures are divided into three elements: concrete, steel, and prestressing systems.

System Interfaces

Functional interfaces include the primary containment heating and ventilation system (VII.F3), containment isolation components (V.C), and containment spray system (V.A). Physical interfaces exist with any structure, system, or component that either penetrates the containment wall, such as the main steam system (VIII.B1) and feedwater system (VIII.D1), or is supported by the containment structure, such as the polar crane (VII.B). The containment structure basemat typically provides support to the nuclear steam supply system (NSSS) components and containment internal structures.

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
II.A1.CP-87	II.A1-4(C-03)	Concrete (accessible areas): dome; wall; basemat; ring girders; buttresses	Concrete	Air – indoor, uncontrolled or Air – outdoor	Increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive chemical attack	Chapter XI.S2, “ASME Section XI, Subsection IWL”	No
II.A1.CP-31	II.A1-2(C-01)	Concrete (accessible areas): dome; wall; basemat; ring girders; buttresses	Concrete	Air – outdoor	Loss of material (spalling, scaling) and cracking due to freeze-thaw	Chapter XI.S2, “ASME Section XI, Subsection IWL”	No
II.A1.CP-33	II.A1-3(C-04)	Concrete (accessible areas): dome; wall; basemat; ring girders; buttresses	Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Chapter XI.S2, “ASME Section XI, Subsection IWL”	No
II.A1.CP-32	II.A1-6(C-02)	Concrete (accessible areas): dome; wall; basemat; ring girders; buttresses	Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Chapter XI.S2, “ASME Section XI, Subsection IWL”	No
II.A1.CP-68	II.A1-7(C-05)	Concrete (accessible areas): 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) due to corrosion of embedded steel	Chapter XI.S2, “ASME Section XI, Subsection IWL”	No

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
II.A1.CP-100	II.A1-4(C-03)	Concrete (inaccessible areas): dome; wall; basemat; ring girders; buttresses	Concrete	Air – indoor, uncontrolled or Air – outdoor or Ground water/soil	Increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive chemical attack	Chapter XI.S2, "ASME Section XI, Subsection IWL," or Chapter XI.S6, " Structures Monitoring"	No
II.A1.CP-147	II.A1-2(C-01)	Concrete (inaccessible areas): dome; wall; basemat; ring girders; buttresses	Concrete	Air – outdoor or Ground water/soil	Loss of material (spalling, scaling) and cracking due to freeze-thaw	Further evaluation is required for plants that are located in moderate to severe weathering conditions (weathering index >100 day-inch/yr) (NUREG-1557) to determine if a plant-specific aging management program is needed. A plant-specific aging management program is not required if documented evidence confirms that the existing concrete had air entrainment content (as per Table CC-2231-2 of the ASME Section III Division 2), and subsequent inspections of accessible areas did not exhibit degradation related to freeze-thaw. Such inspections should be considered a part of the evaluation. If this condition is not satisfied, then a plant-specific aging management program is required to manage loss of material (spalling, scaling) and cracking due to freeze-thaw of concrete in inaccessible areas. The weathering index for the continental US is shown in ASTM C33-90, Fig. 1.	Yes, for plants located in moderate to severe weathering conditions

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
II.A1.CP-67	II.A1-3(C-04)	Concrete (inaccessible areas): dome; wall; basemat; ring girders; buttresses	Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Further evaluation is required to determine if a plant-specific aging management program is needed to manage cracking and expansion due to reaction with aggregate of concrete in Inaccessible Areas. A plant-specific aging management program is not required if (1) as described in NUREG-1557, investigations, tests, and petrographic examinations of aggregates performed in accordance with ASTM C295 and other ASTM reactivity tests, as required, can demonstrate that those aggregates do not adversely react within concrete, or (2) For potentially reactive aggregates, aggregate concrete reaction is not significant if it is demonstrated that the in-place concrete can perform its intended function.	Yes, if concrete is not constructed as stated
II.A1.CP-102	II.A1-6(C-02)	Concrete (inaccessible areas): dome; wall; basemat; ring girders; buttresses	Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Further evaluation is required to determine if a plant-specific aging management program is needed to manage increase in porosity, and permeability due to leaching of calcium hydroxide and carbonation of concrete in Inaccessible Areas. A plant-specific aging management program is not required if (1) There is evidence in the accessible areas that the flowing water has not caused leaching and carbonation, or (2) Evaluation determined that the observed leaching	Yes, if leaching is observed in accessible areas that impact intended function

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
						of calcium hydroxide and carbonation in accessible areas has no impact on the intended function of the concrete structure.	
II.A1.CP-97	II.A1-7(C-05)	Concrete (inaccessible areas): 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) due to corrosion of embedded steel	Chapter XI.S2, “ASME Section XI, Subsection IWL,” or Chapter XI.S6, " Structures Monitoring"	No
II.A1.CP-34	II.A1-1(C-08)	Concrete: dome; wall; basemat; ring girders; buttresses	Concrete	Air – indoor, uncontrolled or Air – outdoor	Reduction of strength and modulus due to elevated temperature (>150°F general; >200°F local)	Plant-specific aging management program The implementation of 10 CFR 50.55a and ASME Section XI, Subsection IWL would not be able to identify the reduction of strength and modulus of elasticity 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	Yes, if temperature limits are exceeded

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
						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 modulus of elasticity and these reductions are applied to the design calculations.	
II.A1.CP-101	II.A1-5(C-37)	Concrete: dome; wall; basemat; ring girders; buttresses	Concrete	Soil	Cracking and distortion due to increased stress levels from settlement	Chapter XI.S2, “ASME Section XI, Subsection IWL,” or Chapter XI.S6, " Structures Monitoring" 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.	Yes, if a de-watering system is relied upon to control settlement
II.A1.C-07	II.A1-8(C-07)	Concrete: foundation; subfoundation	Concrete; porous concrete	Water – flowing	Reduction of foundation strength and cracking due to differential settlement and erosion of porous concrete subfoundation	Chapter XI.S6, “Structures Monitoring” 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.	Yes, if a de-watering system is relied upon to control settlement
II.A1.C-11	II.A1-9(C-11)	Prestressing system: tendons	Steel	Air – indoor, uncontrolled or Air – outdoor	Loss of prestress due to relaxation; shrinkage; creep; elevated temperature	Loss of tendon prestress is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.5, “Concrete Containment Tendon Prestress” for	Yes, TLAA

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
						acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii). See Chapter X.S1 of this report for meeting the requirements of 10 CFR 54.21(c)(1)(iii). For periodic monitoring of prestress, see Chapter XI.S2.	
II.A1.C-10	II.A1-10(C-10)	Prestressing system: tendons; anchorage components	Steel	Air – indoor, uncontrolled or Air – outdoor	Loss of material due to corrosion	Chapter XI.S2, “ASME Section XI, Subsection IWL”	No
II.A1.CP-35	II.A1-11(C-09)	Steel elements (accessible areas): liner; liner anchors; integral attachments	Steel	Air – indoor, uncontrolled	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S1, “ASME Section XI, Subsection IWE,” and Chapter XI.S4, “10 CFR Part 50, Appendix J”	No
II.A1.CP-98	II.A1-11(C-09)	Steel elements (inaccessible areas): liner; liner anchors; integral attachments	Steel	Air – indoor, uncontrolled	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S1, “ASME Section XI, Subsection IWE” and Chapter XI.S4, “10 CFR Part 50, Appendix J” Additional plant-specific activities are warranted if loss of material due to corrosion is significant for inaccessible areas (embedded containment steel shell or liner). Loss of material due to corrosion is not significant if the following conditions are satisfied:	Yes, if corrosion is indicated from the IWE examinations

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
						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 moisture barrier, at the junction where the shell or liner becomes embedded, is subject to aging management activities in accordance with ASME Section XI, Subsection IWE requirements. 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. 4. Borated water spills and water ponding on the concrete floor are common and when detected are cleaned up or diverted to a sump in a timely manner. Operating experience has identified significant corrosion in some plants. If any of the above conditions cannot be satisfied, then a plant-specific aging management program for corrosion is necessary.	

A2. STEEL CONTAINMENTS

Systems, Structures, and Components

This section addresses the elements of pressurized water reactor (PWR) steel containment structures. Steel containment structures are divided into two elements: steel and concrete.

System Interfaces

Functional interfaces include the primary containment heating and ventilation system (VII.F3), containment isolation components (V.C), and containment spray system (V.A). Physical interfaces exist with any structure, system, or component that either penetrates the containment wall, such as the main steam system (VIII.B1) and feedwater system (VIII.D1), or is supported by the containment structure, such as the polar crane (VII.B). The containment structure basemat typically provides support to the nuclear steam supply system (NSSS) components and containment internal structures.

II CONTAINMENT STRUCTURES
A2 Steel Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
II.A2.CP-51	II.A2-2(C-28)	Concrete (accessible areas): basemat	Concrete	Air – outdoor	Loss of material (spalling, scaling) and cracking due to freeze-thaw	Chapter XI.S2, “ASME Section XI, Subsection IWL”	No
II.A2.CP-58	II.A2-3(C-38)	Concrete (accessible areas): basemat	Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Chapter XI.S2, “ASME Section XI, Subsection IWL”	No
II.A2.CP-72	II.A2-4(C-25)	Concrete (accessible areas): basemat	Concrete	Ground water/soil	Increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive chemical attack	Chapter XI.S2, “ASME Section XI, Subsection IWL,” or Chapter XI.S6, " Structures Monitoring"	No
II.A2.CP-155	II.A2-6(C-30)	Concrete (accessible areas): basemat	Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Chapter XI.S2, “ASME Section XI, Subsection IWL”	No
II.A2.CP-74	II.A2-7(C-43)	Concrete (accessible areas): basemat; reinforcing steel	Concrete; steel	Air – indoor, uncontrolled or Air – outdoor	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter XI.S2, “ASME Section XI, Subsection IWL”	No

II CONTAINMENT STRUCTURES
A2 Steel Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
II.A2.CP-70	II.A2-2(C-28)	Concrete (inaccessible areas): basemat	Concrete	Air – outdoor or Ground water/soil	Loss of material (spalling, scaling) and cracking due to freeze-thaw	Further evaluation is required for plants that are located in moderate to severe weathering conditions (weathering index >100 day-inch/yr) (NUREG-1557) to determine if a plant-specific aging management program is needed. A plant-specific aging management program is not required if documented evidence confirms that the existing concrete had air entrainment content (as per Table CC-2231-2 of the ASME Section III Division 2), and subsequent inspections of accessible areas did not exhibit degradation related to freeze-thaw. Such inspections should be considered a part of the evaluation. If this condition is not satisfied, then a plant-specific aging management program is required to manage loss of material (spalling, scaling) and cracking due to freeze-thaw of concrete in inaccessible areas. The weathering index for the continental US is shown in ASTM C33-90, Fig. 1.	Yes, for plants located in moderate to severe weathering conditions
II.A2.CP-104	II.A2-3(C-38)	Concrete (inaccessible areas): basemat	Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Further evaluation is required to determine if a plant-specific aging management program is needed to manage cracking and expansion due to reaction with aggregate of concrete in Inaccessible Areas. A plant-specific	Yes, if concrete is not constructed as stated

II CONTAINMENT STRUCTURES
A2 Steel Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
						aging management program is not required if (1) as described in NUREG-1557, investigations, tests, and petrographic examinations of aggregates performed in accordance with ASTM C295 and other ASTM reactivity tests, as required, can demonstrate that those aggregates do not adversely react within concrete, or (2) For potentially reactive aggregates, aggregate concrete reaction is not significant if it is demonstrated that the in-place concrete can perform its intended function.	
II.A2.CP-71	II.A2-4(C-25)	Concrete (inaccessible areas): basemat	Concrete	Ground water/soil	Increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive chemical attack	Chapter XI.S2, “ASME Section XI, Subsection IWL,” or Chapter XI.S6, " Structures Monitoring“	No
II.A2.CP-53	II.A2-6(C-30)	Concrete (inaccessible areas): basemat	Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Further evaluation is required to determine if a plant-specific aging management program is needed to manage increase in porosity, and permeability due to leaching of calcium hydroxide and carbonation of concrete in Inaccessible Areas. A plant-specific aging management program is not required if (1) There is evidence in the	Yes, if leaching is observed in accessible areas that impact intended function

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
						accessible areas of adjacent structures that the flowing water has not caused leaching and carbonation, or (2) Evaluation determined that the observed leaching of calcium hydroxide and carbonation in accessible areas has no impact on the intended function of the concrete structure.	
II.A2.CP-75	II.A2-7(C-43)	Concrete (inaccessible areas): basemat; reinforcing steel	Concrete; steel	Air – indoor, uncontrolled or Air – outdoor	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter XI.S2, “ASME Section XI, Subsection IWL,” or Chapter XI.S6, " Structures Monitoring"	No
II.A2.CP-69	II.A2-5(C-36)	Concrete: basemat	Concrete	Soil	Cracking and distortion due to increased stress levels from settlement	Chapter XI.S2, “ASME Section XI, Subsection IWL,” or Chapter XI.S6, " Structures Monitoring" 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.	Yes, if a de-watering system is relied upon to control settlement
II.A2.C-07	II.A2-8(C-07)	Concrete: foundation; subfoundation	Concrete; porous concrete	Water – flowing	Reduction of foundation strength and cracking due to differential settlement and	Chapter XI.S6, “ Structures Monitoring” 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	Yes, if a de-watering system is relied upon to control settlement

II CONTAINMENT STRUCTURES
A2 Steel Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
					erosion of porous concrete subfoundation	functioning of the de-watering system through the period of extended operation.	
II.A2.CP-35	II.A2-9(C-09)	Steel elements (accessible areas): liner; liner anchors; integral attachments	Steel	Air – indoor, uncontrolled	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S1, "ASME Section XI, Subsection IWE," and Chapter XI.S4, "10 CFR Part 50, Appendix J"	No
II.A2.CP-98	II.A2-9(C-09)	Steel elements (inaccessible areas): liner; liner anchors; integral attachments	Steel	Air – indoor, uncontrolled	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S1, "ASME Section XI, Subsection IWE" and Chapter XI.S4, "10 CFR Part 50, Appendix J" Additional plant-specific activities are warranted if loss of material due to corrosion is significant 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 moisture barrier, at the junction where the shell or liner becomes embedded, is subject to aging management activities in accordance with ASME Section XI, Subsection IWE requirements.	Yes, if corrosion is indicated from the IWE examinations

II CONTAINMENT STRUCTURES
A2 Steel Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
						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. 4. Borated water spills and water ponding on the concrete floor are common and when detected are cleaned up or diverted to a sump in a timely manner. Operating experience has identified significant corrosion in some plants. If any of the above conditions cannot be satisfied, then a plant-specific aging management program for corrosion is necessary.	

A3. COMMON COMPONENTS

Systems, Structures, and Components

This section addresses the common components of pressurized water reactor (PWR) containment structures. The common components include penetration sleeves and bellows; dissimilar metal welds; personnel airlock; equipment hatch; seals, gaskets, and moisture barriers.

System Interfaces

Functional interfaces include the primary containment heating and ventilation system (VII.F3), containment isolation components (V.C), and containment spray system (V.A). Physical interfaces exist with any structure, system, or component that either penetrates the containment wall, such as the main steam system (VIII.B1) and feedwater system (VIII.D1), or is supported by the containment structure, such as the polar crane (VII.B). The containment structure basemat typically provides support to the nuclear steam supply system (NSSS) components and containment internal structures.

II CONTAINMENT STRUCTURES
A3 Common Components

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
II.A3.CP-40	II.A3-7(C-18)	Moisture barriers (caulking, flashing, and other sealants)	Elastomers, rubber and other similar materials	Air – indoor, uncontrolled	Loss of sealing due to wear, damage, erosion, tear, surface cracks, or other defects	Chapter XI.S1, “ASME Section XI, Subsection IWE”	No
II.A3.CP-36	II.A3-1(C-12)	Penetration sleeves	Steel; dissimilar metal welds	Air – indoor, uncontrolled or Air – outdoor	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S1, “ASME Section XI, Subsection IWE,” and Chapter XI.S4, “10 CFR Part 50, Appendix J”	No
II.A3.CP-38	II.A3-2(C-15)	Penetration sleeves; penetration bellows	Stainless steel; dissimilar metal welds	Air – indoor, uncontrolled or Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.S1, “ASME Section XI, Subsection IWE,” and Chapter XI.S4, “10 CFR Part 50, Appendix J”	Yes, detection of aging effects is to be evaluated
II.A3.CP-37	II.A3-3(C-14)	penetration sleeves; penetration bellows	Steel; stainless steel; dissimilar metal welds	Air – indoor, uncontrolled or Air – outdoor	Cracking due to 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”	No
II.A3.C-13	II.A3-4(C-13)	Penetration sleeves; penetration bellows	Steel; stainless steel; dissimilar metal welds	Air – indoor, uncontrolled or Air – outdoor	Cumulative fatigue damage due to 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 SRP, 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
II.A3.C-16	II.A3-6(C-16)	Personnel airlock, equipment hatch, CRD hatch	Steel	Air – indoor, uncontrolled or Air – outdoor	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S1, “ASME Section XI, Subsection IWE,” and Chapter XI.S4, “10 CFR Part 50, Appendix J”	No

II CONTAINMENT STRUCTURES
A3 Common Components

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
II.A3.CP-39	II.A3-5(C-17)	Personnel airlock, equipment hatch, CRD hatch: locks, hinges, and closure mechanisms	Steel	Air – indoor, uncontrolled or Air – outdoor	Loss of leak tightness due to mechanical wear of locks, hinges and closure mechanisms	Chapter XI.S1, “ASME Section XI, Subsection IWE,” and Chapter XI.S4, “10 CFR Part 50, Appendix J”	No
II.A3.CP-150		Pressure-retaining bolting	Any	Any environment	Loss of preload due to self-loosening	Chapter XI.S1, “ASME Section XI, Subsection IWE,” and Chapter XI.S4, “10 CFR Part 50, Appendix J”	No
II.A3.CP-148		Pressure-retaining bolting	Steel	Air – indoor, uncontrolled or Air – outdoor	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S1, “ASME Section XI, Subsection IWE”	No
II.A3.CP-41	II.A3-7(C-18)	Seals and gaskets	Elastomers, rubber and other similar materials	Air – indoor, uncontrolled or Air – outdoor	Loss of sealing due to wear, damage, erosion, tear, surface cracks, or other defects	Chapter XI.S4, “10 CFR Part 50, Appendix J "	No
II.A3.CP-152		Service Level I coatings	Coatings	Air – indoor, uncontrolled	Loss of coating integrity due to blistering, cracking, flaking, peeling, or physical damage	Chapter XI.S8, "Protective Coating Monitoring and Maintenance"	No

BOILING WATER REACTOR (BWR) CONTAINMENTS

B1. Mark I Containments

B2. Mark II Containments

B3. Mark III Containments

B4. Common Components

B1. MARK I CONTAINMENTS

Systems, Structures, and Components

This section addresses the elements of boiling water reactor (BWR) Mark I containment structures. Steel containments are discussed in II.B1.1 and concrete containments are discussed in II.B1.2.

System Interfaces

Functional interfaces include the primary containment heating and ventilation system (VII.F3), containment isolation components (V.C), and standby gas treatment system (V.B). Physical interfaces exist with any structure, system, or component that either penetrates the containment wall, such as the main steam system (VIII.B2) and feedwater system (VIII.D2), or is supported by the containment structure. The containment structure basemat may provide support to the NSSS components and containment internal structures.

II CONTAINMENT STRUCTURES
B1.1 Mark I Steel Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
II.B1.1.CP-43	II.B1.1-2(C-19)	Steel elements (accessible areas): drywell shell; drywell head; drywell shell in sand pocket regions;	Steel	Air – indoor, uncontrolled	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S1, "ASME Section XI, Subsection IWE," and Chapter XI.S4, "10 CFR Part 50, Appendix J"	No
II.B1.1.C-23	II.B1.1-1(C-23)	Steel elements: drywell head; downcomers	Steel	Air – indoor, uncontrolled	Fretting or lockup due to mechanical wear	Chapter XI.S1, "ASME Section XI, Subsection IWE"	No
II.B1.1.CP-44		Steel elements: drywell support skirt	Steel	Concrete	None	None	No
II.B1.1.CP-109	II.B1.1-2(C-19)	Steel elements: torus ring girders; downcomers;	Steel	Air – indoor, uncontrolled or Treated water	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S1, "ASME Section XI, Subsection IWE" Plant specific aging management program is required if plant operating experience identified significant corrosion of the torus ring girders and downcomers. If protective coating is credited for preventing corrosion of the torus shell, the coating should be included in scope of license renewal and subject to aging management review.	Yes, if corrosion is significant
II.B1.1.CP-48	II.B1.1-2(C-19)	Steel elements: torus shell	Steel	Air – indoor, uncontrolled or Treated water	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S1, "ASME Section XI, Subsection IWE," and Chapter XI.S4, "10 CFR Part 50, Appendix J" Significant corrosion of the torus shell	Yes, if corrosion is significant Recoating of the torus is

II CONTAINMENT STRUCTURES
B1.1 Mark I Steel Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
						and degradation of its protective coating are identified in IN 88-82. Other industrywide operating indicates a number of incidences of torus corrosion. License renewal applicants are advised to address their plant specific operating experience related to the torus shell corrosion. If the identified corrosion is significant, a plant specific aging management is required. If protective coating is credited for preventing corrosion of the torus shell, the coating should be included in scope of license renewal and subject to aging management review .	recommended.
II.B1.1.CP-49	II.B1.1-3(C-20)	Steel elements: torus; vent line; vent header; vent line bellows; downcomers	Steel; stainless steel	Air – indoor, uncontrolled	Cracking due to 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"	No
II.B1.1.C-21	II.B1.1-4(C-21)	Steel elements: torus; vent line; vent header; vent line bellows; downcomers	Steel; stainless steel	Air – indoor, uncontrolled	Cumulative fatigue damage due to 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 SRP, 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
II.B1.1.CP-50	II.B1.1-5(C-22)	Steel elements: vent line bellows	Stainless steel	Air – indoor, uncontrolled	Cracking due to stress corrosion cracking	Chapter XI.S1, "ASME Section XI, Subsection IWE," and Chapter XI.S4, "10 CFR Part 50, Appendix J"	No

II CONTAINMENT STRUCTURES
B1.2 Mark I Concrete Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
II.B1.2.CP-79	II.B1.2-2(C-41)	Concrete (accessible areas): basemat; reinforcing steel	Concrete; steel	Air – indoor, uncontrolled or Air – outdoor	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter XI.S2, "ASME Section XI, Subsection IWL"	No
II.B1.2.CP-59	II.B1.2-4(C-39)	Concrete (accessible areas): containment; wall; basemat	Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Chapter XI.S2, "ASME Section XI, Subsection IWL"	No
II.B1.2.CP-54	II.B1.2-6(C-31)	Concrete (accessible areas): containment; wall; basemat	Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Chapter XI.S2, "ASME Section XI, Subsection IWL"	No
II.B1.2.CP-80	II.B1.2(C-41)	Concrete (inaccessible areas): basemat; reinforcing steel	Concrete; steel	Air – indoor, uncontrolled or Air – outdoor	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter XI.S6, "Structures Monitoring"	No
II.B1.2.CP-99	II.B1.2-4(C-39)	Concrete (inaccessible areas): containment; wall; basemat	Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Further evaluation is required to determine if a plant-specific aging management program is needed to manage cracking and expansion due to reaction with aggregate of concrete in Inaccessible Areas. A plant-specific	Yes, if concrete is not constructed as stated

II CONTAINMENT STRUCTURES
B1.2 Mark I Concrete Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
						aging management program is not required if (1) as described in NUREG-1557, investigations, tests, and petrographic examinations of aggregates performed in accordance with ASTM C295 and other ASTM reactivity tests, as required, can demonstrate that those aggregates do not adversely react within concrete, or (2) For potentially reactive aggregates, aggregate concrete reaction is not significant if it is demonstrated that the in-place concrete can perform its intended function.	
II.B1.2.CP-110	II.B1.2-6(C-31)	Concrete (inaccessible areas): containment; wall; basemat	Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Further evaluation is required to determine if a plant-specific aging management program is needed to manage increase in porosity, and permeability due to leaching of calcium hydroxide and carbonation of concrete in Inaccessible Areas. A plant-specific aging management program is not required if (1) There is evidence in the accessible areas that the flowing water has not caused leaching and carbonation, or (2) Evaluation determined that the observed leaching of calcium hydroxide and carbonation in accessible areas has no impact on the intended function of the concrete structure.	Yes, if leaching is observed in accessible areas that impact intended function

II CONTAINMENT STRUCTURES
B1.2 Mark I Concrete Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
II.B1.2.CP-105	II.B1.2-1(C-06)	Concrete elements, all	Concrete	Soil	Cracking and distortion due to increased stress levels from settlement	Chapter XI.S2, "ASME Section XI, Subsection IWL," or Chapter XI.S6, " Structures Monitoring" 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.	Yes, if a de-watering system is relied upon to control settlement
II.B1.2.CP-106	II.B1.2-5(C-26)	Concrete: containment; wall; basemat	Concrete	Air – indoor, uncontrolled or Air – outdoor	Increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive chemical attack	Chapter XI.S2, "ASME Section XI, Subsection IWL," or Chapter XI.S6, " Structures Monitoring"	No
II.B1.2.CP-57	II.B1.2-3(C-35)	Concrete: containment; wall; basemat	Concrete	Air – indoor, uncontrolled or Air – outdoor	Reduction of strength and modulus due to elevated temperature (>150°F general; >200°F local)	Plant-specific aging management program The implementation of 10 CFR 50.55a and ASME Code, Section XI, Subsection 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	Yes, if temperature limits are exceeded

II CONTAINMENT STRUCTURES
B1.2 Mark I Concrete Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
						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 modulus of elasticity, and these reductions are applied to the design calculations.	
II.B1.2.C-07	II.B1.2-7(C-07)	Concrete: foundation; subfoundation	Concrete; porous concrete	Water – flowing	Reduction of foundation strength and cracking due to differential settlement and erosion of porous concrete subfoundation	Chapter XI.S6, “Structures Monitoring” 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.	Yes, if a de-watering system is relied upon to control settlement
II.B1.2.CP-46	II.B1.2-8(C-46)	Steel elements (accessible areas): suppression chamber;	Steel	Air – indoor, uncontrolled or Treated water	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S1, “ASME Section XI, Subsection IWE,” and Chapter XI.S4, “10 CFR Part 50, Appendix J”	No

II CONTAINMENT STRUCTURES
B1.2 Mark I Concrete Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
		drywell; drywell head; embedded shell; region shielded by diaphragm floor (as applicable)					
II.B1.2.CP-114		Steel elements (inaccessible areas): support skirt	Steel	Concrete	None	None	No
II.B1.2.CP-63	II.B1.2-8(C-46)	Steel elements (inaccessible areas): suppression chamber; drywell; drywell head; embedded shell; region shielded by diaphragm floor (as applicable)	Steel	Air – indoor, uncontrolled or Treated water	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S1, "ASME Section XI, Subsection IWE" and Chapter XI.S4, "10 CFR Part 50, Appendix J" Additional plant-specific activities are warranted if loss of material due to corrosion is significant 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 concrete in contact with the embedded containment shell or liner. 2. The moisture barrier, at the junction where the shell or liner becomes embedded, is subject to aging management activities in accordance with ASME Section XI,	Yes, if corrosion is indicated from the IWE examinations

II CONTAINMENT STRUCTURES
B1.2 Mark I Concrete Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
						Subsection IWE requirements. 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. 4. Borated water spills and water ponding on the concrete floor are common and when detected are cleaned up or diverted to a sump in a timely manner. Operating experience has identified significant corrosion in some plants. If any of the above conditions cannot be satisfied, then a plant-specific aging management program for corrosion is necessary.	
II.B1.2.CP-117	II.B1.2-8(C-46)	Steel elements: downcomer pipes	Steel	Air – indoor, uncontrolled or Treated water	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S1, "ASME Section XI, Subsection IWE"	No
II.B1.2.C-23	II.B1.2-9(C-23)	Steel elements: drywell head; downcomers	Steel	Air – indoor, uncontrolled	Fretting or lockup due to mechanical wear	Chapter XI.S1, "ASME Section XI, Subsection IWE"	No
II.B1.2.C-49	II.B1.2-10(C-49)	Steel elements: suppression chamber (torus) liner (interior surface)	Steel; stainless steel	Air – indoor, uncontrolled or Treated water	Loss of material due to general (steel only), pitting, and crevice corrosion	Chapter XI.S1, "ASME Section XI, Subsection IWE," and Chapter XI.S4, "10 CFR Part 50, Appendix J"	No

B2. MARK II CONTAINMENTS

Systems, Structures, and Components

This section addresses the elements of boiling water reactor (BWR) Mark II containment structures. Mark II steel containments are discussed in II.B2.1. Mark II concrete containments are discussed in II.B2.2.

System Interfaces

Functional interfaces include the primary containment heating and ventilation system (VII.F3), containment isolation components (V.C), and standby gas treatment system (V.B). Physical interfaces exist with any structure, system, or component that either penetrates the containment wall, such as the main steam system (VIII.B2) and feedwater system (VIII.D2), or is supported by the containment structure. The containment structure basemat may provide support to the NSSS components and containment internal structures.

II CONTAINMENT STRUCTURES
B2.1 Mark II Steel Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
II.B2.1.CP-46	II.B2.1-1(C-46)	Steel elements (accessible areas): suppression chamber; drywell; drywell head; embedded shell; region shielded by diaphragm floor (as applicable)	Steel	Air – indoor, uncontrolled or Treated water	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S1, “ASME Section XI, Subsection IWE,” and Chapter XI.S4, “10 CFR Part 50, Appendix J”	No
II.B2.1.CP-114		Steel elements (inaccessible areas): support skirt	Steel	Concrete	None	None	No
II.B2.1.CP-63	II.B2.1-1(C-46)	Steel elements (inaccessible areas): suppression chamber; drywell; drywell head; embedded shell; region shielded by diaphragm floor (as applicable)	Steel	Air – indoor, uncontrolled or Treated water	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S1, “ASME Section XI, Subsection IWE” and Chapter XI.S4, “10 CFR Part 50, Appendix J” Additional plant-specific activities are warranted if loss of material due to corrosion is significant 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	Yes, if corrosion is indicated from the IWE examinations

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
						201.2R was used for the concrete in contact with the embedded containment shell or liner. 2. The moisture barrier, at the junction where the shell or liner becomes embedded, is subject to aging management activities in accordance with ASME Section XI, Subsection IWE requirements. 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. 4. Borated water spills and water ponding on the concrete floor are common and when detected are cleaned up or diverted to a sump in a timely manner. Operating experience has identified significant corrosion in some plants. If any of the above conditions cannot be satisfied, then a plant-specific aging management program for corrosion is necessary.	
II.B2.1.CP-117	II.B2.1-1(C-46)	Steel elements: downcomer pipes	Steel	Air – indoor, uncontrolled or Treated water	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S1, “ASME Section XI, Subsection IWE”	No
II.B2.1.C-23	II.B2.1-2(C-23)	Steel elements: drywell head; downcomers	Steel	Air – indoor, uncontrolled	Fretting or lockup due to	Chapter XI.S1, “ASME Section XI, Subsection IWE”	No

II CONTAINMENT STRUCTURES
B2.1 Mark II Steel Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
					mechanical wear		
II.B2.1.CP-107	II.B2.1-3(C-44)	Suppression pool shell	Steel; stainless steel; dissimilar metal welds	Air – indoor, uncontrolled or Treated Water	Cracking due to 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"	No
II.B2.1.C-45	II.B2.1-4(C-45)	Suppression pool shell; unbraced downcomers	Steel; stainless steel; dissimilar metal welds	Air – indoor, uncontrolled	Cumulative fatigue damage due to 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 SRP, 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
II.B2.1.CP-142	II.B2.1-3(C-44)	Unbraced downcomers	Steel; stainless steel; dissimilar metal welds	Air – indoor, uncontrolled or Treated water	Cracking due to cyclic loading (CLB fatigue analysis does not exist)	Chapter XI.S1, "ASME Section XI, Subsection IWE"	No

II CONTAINMENT STRUCTURES
B2.2 Mark II Concrete Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
II.B2.2.CP-79	II.B2.2-2(C-41)	Concrete (accessible areas): basemat; reinforcing steel	Concrete; steel	Air – indoor, uncontrolled or Air – outdoor	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter XI.S2, “ASME Section XI, Subsection IWL”	No
II.B2.2.CP-59	II.B2.2-4(C-39)	Concrete (accessible areas): containment; wall; basemat	Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Chapter XI.S2, “ASME Section XI, Subsection IWL”	No
II.B2.2.CP-54	II.B2.2-6(C-31)	Concrete (accessible areas): containment; wall; basemat	Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Chapter XI.S2, “ASME Section XI, Subsection IWL”	No
II.B2.2.CP-80	II.B2.2-2(C-41)	Concrete (inaccessible areas): basemat; reinforcing steel	Concrete; steel	Air – indoor, uncontrolled or Air – outdoor	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter XI.S6, "Structures Monitoring"	No
II.B2.2.CP-99	II.B2.2-4(C-39)	Concrete (inaccessible areas): containment; wall; basemat	Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Further evaluation is required to determine if a plant-specific aging management program is needed to manage cracking and expansion due to reaction with aggregate of concrete in Inaccessible Areas. A plant-specific aging management program is not	Yes, if concrete is not constructed as stated

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
						required if (1) as described in NUREG-1557, investigations, tests, and petrographic examinations of aggregates performed in accordance with ASTM C295 and other ASTM reactivity tests, as required, can demonstrate that those aggregates do not adversely react within concrete, or (2) For potentially reactive aggregates, aggregate concrete reaction is not significant if it is demonstrated that the in-place concrete can perform its intended function.	
II.B2.2.CP-110	II.B2.2-6(C-31)	Concrete (inaccessible areas): containment; wall; basemat	Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Further evaluation is required to determine if a plant-specific aging management program is needed to manage increase in porosity, and permeability due to leaching of calcium hydroxide and carbonation of concrete in Inaccessible Areas. A plant-specific aging management program is not required if (1) There is evidence in the accessible areas that the flowing water has not caused leaching and carbonation, or (2) Evaluation determined that the observed leaching of calcium hydroxide and carbonation in accessible areas has no impact on the intended function of the concrete structure.	Yes, if leaching is observed in accessible areas that impact intended function
II.B2.2.CP-105	II.B2.2-1(C-06)	Concrete elements, all	Concrete	Soil	Cracking and distortion due to increased	Chapter XI.S2, “ASME Section XI, Subsection IWL,” or Chapter XI.S6, " Structures Monitoring"	Yes, if a de-watering system is

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
					stress levels from settlement	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.	relied upon to control settlement
II.B2.2.CP-106	II.B2.2-5(C-26)	Concrete: containment; wall; basemat	Concrete	Air – indoor, uncontrolled or Air – outdoor	Increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive chemical attack	Chapter XI.S2, “ASME Section XI, Subsection IWL,” or Chapter XI.S6, " Structures Monitoring"	No
II.B2.2.CP-57	II.B2.2-3(C-35)	Concrete: containment; wall; basemat	Concrete	Air – indoor, uncontrolled or Air – outdoor	Reduction of strength and modulus due to elevated temperature (>150°F general; >200°F local)	Plant-specific aging management program The implementation of 10 CFR 50.55a and ASME Section XI, Subsection 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	Yes, if temperature limits are exceeded

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
						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 modulus of elasticity, and these reductions are applied to the design calculations.	
II.B2.2.C-07	II.B2.2-7(C-07)	Concrete: foundation; subfoundation	Concrete; porous concrete	Water – flowing	Reduction of foundation strength and cracking due to differential settlement and erosion of porous concrete subfoundation	Chapter XI.S6, “Structures Monitoring” 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.	Yes, if a de-watering system is relied upon to control settlement
II.B2.2.C-11	II.B2.2-8(C-11)	Prestressing system: tendons	Steel	Air – indoor, uncontrolled or Air – outdoor	Loss of prestress due to relaxation; shrinkage; creep; elevated temperature	Loss of tendon prestress is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.5, “Concrete Containment Tendon Prestress” for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii). See Chapter X.S1 of this report for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes, TLAA

II CONTAINMENT STRUCTURES
B2.2 Mark II Concrete Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
						For periodic monitoring of prestress, see Chapter XI.S2.	
II.B2.2.C-10	II.B2.2-9(C-10)	Prestressing system: tendons; anchorage components	Steel	Air – indoor, uncontrolled or Air – outdoor	Loss of material due to corrosion	Chapter XI.S2, "ASME Section XI, Subsection IWL"	No
II.B2.2.CP-46	II.B2.2-10(C-46)	Steel elements (accessible areas): suppression chamber; drywell; drywell head; embedded shell; region shielded by diaphragm floor (as applicable)	Steel	Air – indoor, uncontrolled or Treated water	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S1, "ASME Section XI, Subsection IWE," and Chapter XI.S4, "10 CFR Part 50, Appendix J"	No
II.B2.2.CP-114		Steel elements (inaccessible areas): support skirt	Steel	Concrete	None	None	No
II.B2.2.CP-63	II.B2.2-10(C-46)	Steel elements (inaccessible areas): suppression chamber; drywell; drywell head; embedded shell; region	Steel	Air – indoor, uncontrolled or Treated water	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S1, "ASME Section XI, Subsection IWE" and Chapter XI.S4, "10 CFR Part 50, Appendix J" Additional plant-specific activities are warranted if loss of material due to corrosion is significant for inaccessible	Yes, if corrosion is indicated from the IWE examinations

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
		shielded by diaphragm floor (as applicable)				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 concrete in contact with the embedded containment shell or liner. 2. The moisture barrier, at the junction where the shell or liner becomes embedded, is subject to aging management activities in accordance with ASME Section XI, Subsection IWE requirements. 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. 4. Borated water spills and water ponding on the concrete floor are common and when detected are cleaned up or diverted to a sump in a timely manner. Operating experience has identified significant corrosion in some plants. If any of the above conditions cannot be satisfied, then a plant-specific aging management program for corrosion is necessary.	

II CONTAINMENT STRUCTURES
B2.2 Mark II Concrete Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
II.B2.2.CP-117	II.B2.2-10(C-46)	Steel elements: downcomer pipes	Steel	Air – indoor, uncontrolled or Treated water	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S1, “ASME Section XI, Subsection IWE”	No
II.B2.2.C-23	II.B2.2-11(C-23)	Steel elements: drywell head; downcomers	Steel	Air – indoor, uncontrolled	Fretting or lockup due to mechanical wear	Chapter XI.S1, “ASME Section XI, Subsection IWE”	No
II.B2.2.C-49	II.B2.2-12(C-49)	Steel elements: suppression chamber (torus) liner (interior surface)	Steel; stainless steel	Air – indoor, uncontrolled or Treated water	Loss of material due to general (steel only), pitting, and crevice corrosion	Chapter XI.S1, “ASME Section XI, Subsection IWE,” and Chapter XI.S4, “10 CFR Part 50, Appendix J”	No
II.B2.2.CP-64	II.B2.2-13(C-47)	Steel elements: vent header; downcomers	Steel; stainless steel	Air – indoor, uncontrolled or Treated water	Cracking due to cyclic loading (CLB fatigue analysis does not exist)	Chapter XI.S1, “ASME Section XI, Subsection IWE”	No
II.B2.2.C-48	II.B2.2-14(C-48)	Steel elements: vent header; downcomers	Steel; stainless steel	Air – indoor, uncontrolled or Treated water	Cumulative fatigue damage due to 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 SRP, 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

B3. MARK III CONTAINMENTS

B3.1 Steel Containments

B3.2 Concrete Containments

B3. MARK III CONTAINMENTS

Systems, Structures, and Components

This section addresses the elements of boiling water reactor (BWR) Mark III containment structures. Mark III steel containments are discussed in II.B3.1. Mark III concrete containments are discussed in II.B3.2.

System Interfaces

Functional interfaces include the primary containment heating and ventilation system (VII.F3), containment isolation components (V.C), and standby gas treatment system (V.B). Physical interfaces exist with any structure, system, or component that either penetrates the containment wall, such as the main steam system (VIII.B2) and feedwater system (VIII.D2), or is supported by the containment structure. The containment structure basemat may provide support to the NSSS components and containment internal structures.

II CONTAINMENT STRUCTURES
B3.1 Mark III Steel Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
II.B3.1.CP-72	II.B3.1-1(C-25)	Concrete (accessible areas): basemat	Concrete	Ground water/soil	Increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive chemical attack	Chapter XI.S2, “ASME Section XI, Subsection IWL,” or Chapter XI.S6, " Structures Monitoring"	No
II.B3.1.CP-156	II.B3.1-3(C-30)	Concrete (accessible areas): basemat	Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Chapter XI.S2, “ASME Section XI, Subsection IWL”	No
II.B3.1.CP-66	II.B3.1-5(C-51)	Concrete (accessible areas): basemat, concrete fill-in annulus	Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Chapter XI.S2, “ASME Section XI, Subsection IWL”	No
II.B3.1.CP-74	II.B3.1-6(C-43)	Concrete (accessible areas): basemat; reinforcing steel	Concrete; steel	Air – indoor, uncontrolled or Air – outdoor	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter XI.S2, “ASME Section XI, Subsection IWL”	No
II.B3.1.CP-71	II.B3.1-1(C-25)	Concrete (inaccessible areas): basemat	Concrete	Ground water/soil	Increase in porosity and permeability; cracking; loss of material (spalling,	Chapter XI.S2, “ASME Section XI, Subsection IWL,” or Chapter XI.S6, " Structures Monitoring“	No

II CONTAINMENT STRUCTURES
B3.1 Mark III Steel Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
					scaling) due to aggressive chemical attack		
II.B3.1.CP-53	II.B3.1-3(C-30)	Concrete (inaccessible areas): basemat	Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Further evaluation is required to determine if a plant-specific aging management program is needed to manage increase in porosity, and permeability due to leaching of calcium hydroxide and carbonation of concrete in Inaccessible Areas. A plant-specific aging management program is not required if (1) There is evidence in the accessible areas of adjacent structures that the flowing water has not caused leaching and carbonation, or (2) Evaluation determined that the observed leaching of calcium hydroxide and carbonation in accessible areas has no impact on the intended function of the concrete structure.	Yes, if leaching is observed in accessible areas that impact intended function
II.B3.1.CP-83	II.B3.1-5(C-51)	Concrete (inaccessible areas): basemat, concrete fill-in annulus	Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Further evaluation is required to determine if a plant-specific aging management program is needed to manage cracking and expansion due to reaction with aggregate of concrete in Inaccessible Areas. A plant-specific aging management program is not required if (1) as described in NUREG-1557, investigations, tests, and petrographic examinations of	Yes, if concrete is not constructed as stated

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
						aggregates performed in accordance with ASTM C295 and other ASTM reactivity tests, as required, can demonstrate that those aggregates do not adversely react within concrete, or (2) For potentially reactive aggregates, aggregate concrete reaction is not significant if it is demonstrated that the in-place concrete can perform its intended function.	
II.B3.1.CP-75	II.B3.1-6(C-43)	Concrete (inaccessible areas): basemat; reinforcing steel	Concrete; steel	Air – indoor, uncontrolled or Air – outdoor	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter XI.S2, "ASME Section XI, Subsection IWL," or Chapter XI.S6, " Structures Monitoring"	No
II.B3.1.CP-69	II.B3.1-2(C-36)	Concrete: basemat	Concrete	Soil	Cracking and distortion due to increased stress levels from settlement	Chapter XI.S2, "ASME Section XI, Subsection IWL," or Chapter XI.S6, " Structures Monitoring" 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.	Yes, if a de-watering system is relied upon to control settlement
II.B3.1.CP-65	II.B3.1-4(C-50)	Concrete: basemat, concrete fill-in annulus	Concrete	Air – indoor, uncontrolled or Air – outdoor	Reduction of strength and modulus due to elevated temperature	Plant-specific aging management program The implementation of 10 CFR 50.55a and ASME Section	Yes, if temperature limits are exceeded

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
					(>150°F general; >200°F local)	XI, Subsection IWL would not be able to identify the reduction of strength and modulus of elasticity 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 modulus of elasticity and these reductions are applied to the design calculations.	
II.B3.1.C-07	II.B3.1-7(C-07)	Concrete: foundation; subfoundation	Concrete; porous concrete	Water – flowing	Reduction of foundation strength and cracking	Chapter XI.S6, “Structures Monitoring” If a de-watering system is relied upon for control of erosion of cement from	Yes, if a de-watering system is relied upon to

II CONTAINMENT STRUCTURES
B3.1 Mark III Steel Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
					due to differential settlement and erosion of porous concrete subfoundation	porous concrete subfoundations, then the licensee is to ensure proper functioning of the de-watering system through the period of extended operation.	control settlement
II.B3.1.CP-43	II.B3.1-8(C-19)	Steel elements (accessible areas): drywell shell; drywell head	Steel	Air – indoor, uncontrolled	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S1, “ASME Section XI, Subsection IWE,” and Chapter XI.S4, “10 CFR Part 50, Appendix J”	No
II.B3.1.CP-113	II.B3.1-8(C-19)	Steel elements (inaccessible areas): drywell shell; drywell head; and drywell shell	Steel	Air – indoor, uncontrolled or Concrete	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S1, “ASME Section XI, Subsection IWE,” and Chapter XI.S4, “10 CFR Part 50, Appendix J”	Yes, if corrosion is indicated from the IWE examinations
II.B3.1.C-24	II.B3.1-9(C-24)	Steel elements: suppression chamber shell (interior surface)	Stainless steel	Air – indoor, uncontrolled	Cracking due to stress corrosion cracking	Chapter XI.S1, “ASME Section XI, Subsection IWE,” and Chapter XI.S4, “10 CFR Part 50, Appendix J”	No
II.B3.1.CP-158	II.B3.1-8(C-19)	Steel elements: suppression chamber shell (interior surface)	Steel	Air – indoor, uncontrolled or Treated water	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S1, “ASME Section XI, Subsection IWE” Plant-specific aging management program is required if plant operating experience identified significant corrosion. If protective coating is credited for preventing corrosion, the coating should be included in scope of license renewal and subject to aging management review.	Yes, if corrosion is significant

II CONTAINMENT STRUCTURES
B3.2 Mark III Concrete Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
II.B3.2.CP-84	II.B3.2-5(C-27)	Concrete (accessible areas): dome; wall; basemat	Concrete	Air – indoor, uncontrolled or Air – outdoor or Ground water/soil	Increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive chemical attack	Chapter XI.S2, “ASME Section XI, Subsection IWL,” or Chapter XI.S6, " Structures Monitoring"	No
II.B3.2.CP-52	II.B3.2-3(C-29)	Concrete (accessible areas): dome; wall; basemat	Concrete	Air – outdoor or Ground water/soil	Loss of material (spalling, scaling) and cracking due to freeze-thaw	Chapter XI.S2, “ASME Section XI, Subsection IWL”	No
II.B3.2.CP-60	II.B3.2-4(C-40)	Concrete (accessible areas): dome; wall; basemat	Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Chapter XI.S2, “ASME Section XI, Subsection IWL”	No
II.B3.2.CP-55	II.B3.2-6(C-32)	Concrete (accessible areas): dome; wall; basemat	Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Chapter XI.S2, “ASME Section XI, Subsection IWL”	No
II.B3.2.CP-88	II.B3.2-7(C-42)	Concrete (accessible areas): dome; wall; basemat; reinforcing steel	Concrete; steel	Air – indoor, uncontrolled or Air – outdoor	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter XI.S2, “ASME Section XI, Subsection IWL”	No

II CONTAINMENT STRUCTURES
B3.2 Mark III Concrete Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
II.B3.2.CP-73	II.B3.2-5(C-27)	Concrete (inaccessible areas): dome; wall; basemat	Concrete	Air – indoor, uncontrolled or Air – outdoor or Ground water/soil	Increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive chemical attack	Chapter XI.S2, "ASME Section XI, Subsection IWL," or Chapter XI.S6, " Structures Monitoring"	No
II.B3.2.CP-135	II.B3.2-3(C-29)	Concrete (inaccessible areas): dome; wall; basemat	Concrete	Air – outdoor or Ground water/soil	Loss of material (spalling, scaling) and cracking due to freeze-thaw	Further evaluation is required for plants that are located in moderate to severe weathering conditions (weathering index >100 day-inch/yr) (NUREG-1557) to determine if a plant-specific aging management program is needed. A plant-specific aging management program is not required if documented evidence confirms that the existing concrete had air entrainment content (as per Table CC-2231-2 of the ASME Section III Division 2), and subsequent inspections of accessible areas did not exhibit degradation related to freeze-thaw. Such inspections should be considered a part of the evaluation. If this condition is not satisfied, then a plant-specific aging management program is required to manage loss of material (spalling, scaling) and cracking due to freeze-thaw of concrete in inaccessible areas. The weathering index for the	Yes, for plants located in moderate to severe weathering conditions

II CONTAINMENT STRUCTURES
B3.2 Mark III Concrete Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
						continental US is shown in ASTM C33-90, Fig. 1.	
II.B3.2.CP-121	II.B3.2-4(C-40)	Concrete (inaccessible areas): dome; wall; basemat	Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Further evaluation is required to determine if a plant-specific aging management program is needed to manage cracking and expansion due to reaction with aggregate of concrete in Inaccessible Areas. A plant-specific aging management program is not required if (1) as described in NUREG-1557, investigations, tests, and petrographic examinations of aggregates performed in accordance with ASTM C295 and other ASTM reactivity tests, as required, can demonstrate that those aggregates do not adversely react within concrete, or (2) For potentially reactive aggregates, aggregate concrete reaction is not significant if it is demonstrated that the in-place concrete can perform its intended function.	Yes, if concrete is not constructed as stated
II.B3.2.CP-122	II.B3.2-6(C-32)	Concrete (inaccessible areas): dome; wall; basemat	Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Further evaluation is required to determine if a plant-specific aging management program is needed to manage increase in porosity, and permeability due to leaching of calcium hydroxide and carbonation of concrete in Inaccessible Areas. A plant-specific aging management program is not required if (1) There is	Yes, if leaching is observed in accessible areas that impact intended function

II CONTAINMENT STRUCTURES
B3.2 Mark III Concrete Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
						evidence in the accessible areas that the flowing water has not caused leaching and carbonation, or (2) Evaluation determined that the observed leaching of calcium hydroxide and carbonation in accessible areas has no impact on the intended function of the concrete structure.	
II.B3.2.CP-89	II.B3.2-7(C-42)	Concrete (inaccessible areas): dome; wall; basemat; reinforcing steel	Concrete; steel	Air – indoor, uncontrolled or Air – outdoor	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter XI.S2, “ASME Section XI, Subsection IWL,” or Chapter XI.S6, " Structures Monitoring"	No
II.B3.2.CP-105	II.B3.2-1(C-06)	Concrete elements, all	Concrete	Soil	Cracking and distortion due to increased stress levels from settlement	Chapter XI.S2, “ASME Section XI, Subsection IWL,” or Chapter XI.S6, " Structures Monitoring" 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.	Yes, if a de-watering system is relied upon to control settlement
II.B3.2.CP-108	II.B3.2-2(C-33)	Concrete: dome; wall; basemat	Concrete	Air – indoor, uncontrolled or Air – outdoor	Reduction of strength and modulus due to elevated temperature (>150°F general; >200°F local)	Plant-specific aging management program The implementation of 10 CFR 50.55a and ASME Section XI, Subsection IWL would not be able to identify the reduction of strength and modulus of elasticity due to elevated temperature.	Yes, if temperature limits are exceeded

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
						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 modulus of elasticity and these reductions are applied to the design calculations.	
II.B3.2.C-07	II.B3.2-8(C-07)	Concrete: foundation; subfoundation	Concrete; porous concrete	Water – flowing	Reduction of foundation strength and cracking due to differential settlement and erosion of porous concrete subfoundation	Chapter XI.S6, “Structures Monitoring” 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.	Yes, if a de-watering system is relied upon to control settlement

II CONTAINMENT STRUCTURES
B3.2 Mark III Concrete Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
II.B3.2.CP-35	II.B3.2-9(C-09)	Steel elements (accessible areas): liner; liner anchors; integral attachments	Steel	Air – indoor, uncontrolled	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S1, “ASME Section XI, Subsection IWE,” and Chapter XI.S4, “10 CFR Part 50, Appendix J”	No
II.B3.2.CP-98	II.B3.2-9(C-09)	Steel elements (inaccessible areas): liner; liner anchors; integral attachments	Steel	Air – indoor, uncontrolled	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S1, “ASME Section XI, Subsection IWE” and Chapter XI.S4, “10 CFR Part 50, Appendix J” Additional plant-specific activities are warranted if loss of material due to corrosion is significant 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 moisture barrier, at the junction where the shell or liner becomes embedded, is subject to aging management activities in accordance with ASME Section XI, Subsection IWE requirements. 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	Yes, if corrosion is indicated from the IWE examinations

II CONTAINMENT STRUCTURES
B3.2 Mark III Concrete Containments

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
						containment shell or liner. 4. Borated water spills and water ponding on the concrete floor are common and when detected are cleaned up or diverted to a sump in a timely manner. Operating experience has identified significant corrosion in some plants. If any of the above conditions cannot be satisfied, then a plant-specific aging management program for corrosion is necessary.	
II.B3.2.C-24	II.B3.2-10(C-24)	Steel elements: suppression chamber shell (interior surface)	Stainless steel	Air – indoor, uncontrolled	Cracking due to stress corrosion cracking	Chapter XI.S1, “ASME Section XI, Subsection IWE,” and Chapter XI.S4, “10 CFR Part 50, Appendix J”	No

B4. COMMON COMPONENTS

Systems, Structures, and Components

This section addresses the common components of boiling water reactor (BWR) containments. The common components include penetration sleeves and bellows; dissimilar metal welds; personnel airlock; equipment hatch; CRD hatch; seals, gaskets, and moisture barriers.

System Interfaces

Functional interfaces include the primary containment heating and ventilation system (VII.F3), containment isolation components (V.C), and standby gas treatment system (V.B). Physical interfaces exist with any structure, system, or component that either penetrates the containment wall, such as the main steam system (VIII.B2) and feedwater system (VIII.D2), or is supported by the containment structure. The containment structure basemat may provide support to the NSSS components and containment internal structures.

II CONTAINMENT STRUCTURES
B4 Common Components

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
II.B4.CP-40	II.B4-7(C-18)	Moisture barriers (caulking, flashing, and other sealants)	Elastomers, rubber and other similar materials	Air – indoor, uncontrolled	Loss of sealing due to wear, damage, erosion, tear, surface cracks, or other defects	Chapter XI.S1, “ASME Section XI, Subsection IWE”	No
II.B4.CP-36	II.B4-1(C-12)	Penetration sleeves	Steel; dissimilar metal welds	Air – indoor, uncontrolled or Air – outdoor	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S1, “ASME Section XI, Subsection IWE,” and Chapter XI.S4, “10 CFR Part 50, Appendix J”	No
II.B4.CP-38	II.B4-2(C-15)	Penetration sleeves; penetration bellows	Stainless steel; dissimilar metal welds	Air – indoor, uncontrolled or Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.S1, “ASME Section XI, Subsection IWE,” and Chapter XI.S4, “10 CFR Part 50, Appendix J”	Yes, detection of aging effects is to be evaluated
II.B4.CP-37	II.B4-3(C-14)	penetration sleeves; penetration bellows	Steel; stainless steel; dissimilar metal welds	Air – indoor, uncontrolled or Air – outdoor	Cracking due to 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”	No
II.B4.C-13	II.B4-4(C-13)	Penetration sleeves; penetration bellows	Steel; stainless steel; dissimilar metal welds	Air – indoor, uncontrolled or Air – outdoor	Cumulative fatigue damage due to 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 SRP, 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
II.B4.C-16	II.B4-6(C-16)	Personnel airlock, equipment hatch, CRD hatch	Steel	Air – indoor, uncontrolled or Air – outdoor	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S1, “ASME Section XI, Subsection IWE,” and Chapter XI.S4, “10 CFR Part 50, Appendix J”	No

II CONTAINMENT STRUCTURES
B4 Common Components

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
II.B4.CP-39	II.B4-5(C-17)	Personnel airlock, equipment hatch, CRD hatch: locks, hinges, and closure mechanisms	Steel	Air – indoor, uncontrolled or Air – outdoor	Loss of leak tightness due to mechanical wear of locks, hinges and closure mechanisms	Chapter XI.S1, “ASME Section XI, Subsection IWE,” and Chapter XI.S4, “10 CFR Part 50, Appendix J”	No
II.B4.CP-150		Pressure-retaining bolting	Any	Any environment	Loss of preload due to self-loosening	Chapter XI.S1, “ASME Section XI, Subsection IWE,” and Chapter XI.S4, “10 CFR Part 50, Appendix J”	No
II.B4.CP-148		Pressure-retaining bolting	Steel	Air – indoor, uncontrolled or Air – outdoor	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S1, “ASME Section XI, Subsection IWE”	No
II.B4.CP-41	II.B4-7(C-18)	Seals and gaskets	Elastomers, rubber and other similar materials	Air – indoor, uncontrolled or Air – outdoor	Loss of sealing due to wear, damage, erosion, tear, surface cracks, or other defects	Chapter XI.S4, “10 CFR Part 50, Appendix J "	No
II.B4.CP-152		Service Level I coatings	Coatings	Air – indoor, uncontrolled	Loss of coating integrity due to blistering, cracking, flaking, peeling, or physical damage	Chapter XI.S8, "Protective Coating Monitoring and Maintenance"	No

CHAPTER III

STRUCTURES AND COMPONENT SUPPORTS

Chapter III A: Safety Related and Other Structures

Safety-related structures are those defined pursuant to 10 CFR 54.4(a)(1), and the other structures are those defined pursuant to 10 CFR 54.4(a)(2) and 10 CFR 54.4(a)(3). Structures in this section are organized into nine groups and are discussed separately under subheadings A1 through A9.

Chapter III B: Component Supports

Component supports include supports for ASME piping and components; supports for cable trays, conduit, HVAC ducts, TubeTrack®, instrument tubing, non-ASME piping and components; anchorage of racks, panels, cabinets, and enclosures for electrical equipment and instrumentation; supports for emergency diesel generator (EDG) and HVAC system components; and supports for platforms, pipe whip restraints, jet impingement shields, masonry walls, and other miscellaneous structures.

III.A SAFETY RELATED AND OTHER STRUCTURES

A1. Group 1 Structures (BWR Reactor Bldg., PWR Shield Bldg., Control Room/Bldg.)

A2. Group 2 Structures (BWR Reactor Bldg. with Steel Superstructure)

A3. Group 3 Structures (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)

A4. Group 4 Structures (Containment Internal Structures, excluding Refueling Canal)

A5. Group 5 Structures (Fuel Storage Facility, Refueling Canal)

A6. Group 6 Structures (Water-Control Structures)

A7. Group 7 Structures (Concrete Tanks and Missile Barriers)

A8. Group 8 Structures (Steel Tanks and Missile Barriers)

A9. Group 9 Structures (BWR Unit Vent Stack)

A1. GROUP 1 STRUCTURES (BWR REACTOR BLDG., PWR SHIELD BLDG., CONTROL ROOM/BLDG.)

Systems, Structures, and Components

This section addresses the elements of the boiling water reactor (BWR) reactor building, pressurized water reactor (PWR) shield building, and control room/building. 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.

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 Room/Bldg.)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
III.A1.TP-25	III.A1-2(T-03)	Concrete (accessible areas): all	Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Chapter XI.S6, "Structures Monitoring"	No
III.A1.TP-27	III.A1-4(T-05)	Concrete (accessible areas): below-grade exterior; foundation	Concrete	Ground water/soil	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter XI.S6, "Structures Monitoring"	No
III.A1.TP-23	III.A1-6(T-01)	Concrete (accessible areas): exterior above- and below-grade; foundation	Concrete	Air – outdoor	Loss of material (spalling, scaling) and cracking due to freeze-thaw	Chapter XI.S6, "Structures Monitoring"	No
III.A1.TP-24	III.A1-7(T-02)	Concrete (accessible areas): exterior above- and below-grade; foundation	Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Chapter XI.S6, "Structures Monitoring"	No
III.A1.TP-26	III.A1-9(T-04)	Concrete (accessible areas): interior and above-grade exterior	Concrete	Air – indoor, uncontrolled or Air – outdoor	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter XI.S6, "Structures Monitoring"	No
III.A1.TP-204	III.A1-2(T-03)	Concrete (inaccessible areas): all	Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Further evaluation is required to determine if a plant-specific aging management program is needed to manage cracking and expansion due to	Yes, if concrete is not constructed as stated

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
						reaction with aggregate of concrete in Inaccessible Areas. A plant-specific aging management program is not required if (1) as described in NUREG-1557, investigations, tests, and petrographic examinations of aggregates performed in accordance with ASTM C295 and other ASTM reactivity tests, as required, can demonstrate that those aggregates do not adversely react within concrete, or (2) For potentially reactive aggregates, aggregate concrete reaction is not significant if it is demonstrated that the in-place concrete can perform its intended function.	
III.A1.TP-212	III.A1-4(T-05)	Concrete (inaccessible areas): below-grade exterior; foundation	Concrete	Ground water/soil	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter XI.S6, "Structures Monitoring"	No
III.A1.TP-29	III.A1-5(T-07)	Concrete (inaccessible areas): below-grade exterior; foundation	Concrete	Ground water/soil	Increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive chemical attack	Chapter XI.S6, "Structures Monitoring"	No
III.A1.TP-67	III.A1-7(T-02)	Concrete (inaccessible areas): exterior	Concrete	Water – flowing	Increase in porosity and permeability; loss	Further evaluation is required to determine if a plant-specific aging management program is needed to	Yes, if leaching is observed in

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
		above- and below-grade; foundation			of strength due to leaching of calcium hydroxide and carbonation	manage increase in porosity, and permeability due to leaching of calcium hydroxide and carbonation of concrete in Inaccessible Areas. A plant-specific aging management program is not required if (1) There is evidence in the accessible areas that the flowing water has not caused leaching and carbonation, or (2) Evaluation determined that the observed leaching of calcium hydroxide and carbonation in accessible areas has no impact on the intended function of the concrete structure.	accessible areas that impact intended function
III.A1.TP-108	III.A1-6(T-01)	Concrete (inaccessible areas): foundation	Concrete	Air – outdoor	Loss of material (spalling, scaling) and cracking due to freeze-thaw	Further evaluation is required for plants that are located in moderate to severe weathering conditions (weathering index >100 day-inch/yr) (NUREG-1557) to determine if a plant-specific aging management program is needed. A plant-specific aging management program is not required if documented evidence confirms that the existing concrete had air entrainment content (as per Table CC-2231-2 of the ASME Section III Division 2), and subsequent inspections of accessible areas did not exhibit degradation related to freeze-thaw. Such inspections should be considered a part of the evaluation. If this condition is not satisfied, then a plant-specific aging management program is required to manage loss of	Yes, for plants located in moderate to severe weathering conditions

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
						material (spalling, scaling) and cracking due to freeze-thaw of concrete in inaccessible areas. The weathering index for the continental US is shown in ASTM C33-90, Fig. 1.	
III.A1.TP-114	III.A1-1(T-10)	Concrete: all	Concrete	Air – indoor, uncontrolled	Reduction of strength and modulus due to elevated temperature (>150°F general; >200°F local)	Plant-specific aging management program Subsection CC-3400 of ASME Section III, Division 2, and Appendix A of ACI 349 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, where the temperatures 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 those given above may be allowed in the concrete if tests and/or calculations are provided to evaluate the reduction in strength and modulus of elasticity and these reductions are applied to the design calculations.	Yes, if temperature limits are exceeded
III.A1.TP-30	III.A1-3(T-08)	Concrete: all	Concrete	Soil	Cracking and distortion due to increased	Chapter XI.S6, “Structures Monitoring” If a de-watering system is relied upon for control of settlement, then the licensee	Yes, if a de-watering system is

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
					stress levels from settlement	is to ensure proper functioning of the de-watering system through the period of extended operation.	relied upon to control settlement
III.A1.TP-31	III.A1-8(T-09)	Concrete: foundation; subfoundation	Concrete; porous concrete	Water – flowing under foundation	Reduction of foundation strength and cracking due to differential settlement and erosion of porous concrete subfoundation	Chapter XI.S6, “Structures Monitoring” 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.	Yes, if a de-watering system is relied upon to control settlement
III.A1.TP-28	III.A1-10(T-06)	Concrete: interior; above-grade exterior	Concrete	Air – indoor, uncontrolled or Air – outdoor	Increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive chemical attack	Chapter XI.S6, "Structures Monitoring"	No
III.A1.TP-300		High-strength structural bolting	Low-alloy steel, actual measured yield strength ≥ 150 ksi (1,034 MPa)	Air – indoor, uncontrolled or Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.S6, "Structures Monitoring“ Note: ASTM A 325, F 1852, and ASTM A 490 bolts used in civil structures have not shown to be prone to SCC. SCC potential need not be evaluated for these bolts.	No
III.A1.T-12	III.A1-11(T-12)	Masonry walls: all	Concrete block	Air – indoor, uncontrolled or Air – outdoor	Cracking due to restraint shrinkage, creep, and aggressive environment	Chapter XI.S5, “Masonry Walls”	No

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
III.A1.TP-302	III.A1-12(T-11)	Steel components: all structural steel	Steel	Air – indoor, uncontrolled or Air – outdoor	Loss of material due to corrosion	Chapter XI.S6, “Structures Monitoring” 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
III.A1.TP-261		Structural bolting	Any	Any environment	Loss of preload due to self-loosening	Chapter XI.S6, "Structures Monitoring"	No
III.A1.TP-248		Structural bolting	Steel	Air – indoor, uncontrolled	Loss of material due to general, pitting and crevice corrosion	Chapter XI.S6, "Structures Monitoring"	No
III.A1.TP-274		Structural bolting	Steel; galvanized steel	Air – outdoor	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S6, "Structures Monitoring"	No

A2. GROUP 2 STRUCTURES (BWR REACTOR BLDG. WITH STEEL SUPERSTRUCTURE)

Systems, Structures, and Components

This section addresses the elements of the boiling water reactor (BWR) reactor building with steel superstructure. For this group, the applicable structural elements are identified: concrete, steel, and masonry walls. 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
A2 Group 2 Structures (BWR Reactor Bldg. with Steel Superstructure)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
III.A2.TP-25	III.A2-2(T-03)	Concrete (accessible areas): all	Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Chapter XI.S6, "Structures Monitoring"	No
III.A2.TP-27	III.A2-4(T-05)	Concrete (accessible areas): below-grade exterior; foundation	Concrete	Ground water/soil	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter XI.S6, "Structures Monitoring"	No
III.A2.TP-23	III.A2-6(T-01)	Concrete (accessible areas): exterior above- and below-grade; foundation	Concrete	Air – outdoor	Loss of material (spalling, scaling) and cracking due to freeze-thaw	Chapter XI.S6, "Structures Monitoring"	No
III.A2.TP-24	III.A2-7(T-02)	Concrete (accessible areas): exterior above- and below-grade; foundation	Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Chapter XI.S6, "Structures Monitoring"	No
III.A2.TP-26	III.A2-9(T-04)	Concrete (accessible areas): interior and above-grade exterior	Concrete	Air – indoor, uncontrolled or Air – outdoor	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter XI.S6, "Structures Monitoring"	No

III STRUCTURES AND COMPONENT SUPPORTS
A2 Group 2 Structures (BWR Reactor Bldg. with Steel Superstructure)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
III.A2.TP-204	III.A2-2(T-03)	Concrete (inaccessible areas): all	Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Further evaluation is required to determine if a plant-specific aging management program is needed to manage cracking and expansion due to reaction with aggregate of concrete in Inaccessible Areas. A plant-specific aging management program is not required if (1) as described in NUREG-1557, investigations, tests, and petrographic examinations of aggregates performed in accordance with ASTM C295 and other ASTM reactivity tests, as required, can demonstrate that those aggregates do not adversely react within concrete, or (2) For potentially reactive aggregates, aggregate concrete reaction is not significant if it is demonstrated that the in-place concrete can perform its intended function.	Yes, if concrete is not constructed as stated
III.A2.TP-212	III.A2-4(T-05)	Concrete (inaccessible areas): below-grade exterior; foundation	Concrete	Ground water/soil	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter XI.S6, "Structures Monitoring"	No
III.A2.TP-29	III.A2-5(T-07)	Concrete (inaccessible areas): below-grade exterior; foundation	Concrete	Ground water/soil	Increase in porosity and permeability; cracking; loss of material (spalling, scaling)	Chapter XI.S6, "Structures Monitoring"	No

III STRUCTURES AND COMPONENT SUPPORTS
A2 Group 2 Structures (BWR Reactor Bldg. with Steel Superstructure)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
					due to aggressive chemical attack		
III.A2.TP-67	III.A2-7(T-02)	Concrete (inaccessible areas): exterior above- and below-grade; foundation	Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Further evaluation is required to determine if a plant-specific aging management program is needed to manage increase in porosity, and permeability due to leaching of calcium hydroxide and carbonation of concrete in Inaccessible Areas. A plant-specific aging management program is not required if (1) There is evidence in the accessible areas that the flowing water has not caused leaching and carbonation, or (2) Evaluation determined that the observed leaching of calcium hydroxide and carbonation in accessible areas has no impact on the intended function of the concrete structure.	Yes, if leaching is observed in accessible areas that impact intended function
III.A2.TP-108	III.A2-6(T-01)	Concrete (inaccessible areas): foundation	Concrete	Air – outdoor	Loss of material (spalling, scaling) and cracking due to freeze-thaw	Further evaluation is required for plants that are located in moderate to severe weathering conditions (weathering index >100 day-inch/yr) (NUREG-1557) to determine if a plant-specific aging management program is needed. A plant-specific aging management program is not required if documented evidence confirms that the existing concrete had air entrainment content (as per Table CC-2231-2 of the ASME Section III Division 2), and subsequent inspections of accessible areas did not	Yes, for plants located in moderate to severe weathering conditions

III STRUCTURES AND COMPONENT SUPPORTS
A2 Group 2 Structures (BWR Reactor Bldg. with Steel Superstructure)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
						exhibit degradation related to freeze-thaw. Such inspections should be considered a part of the evaluation. If this condition is not satisfied, then a plant-specific aging management program is required to manage loss of material (spalling, scaling) and cracking due to freeze-thaw of concrete in inaccessible areas. The weathering index for the continental US is shown in ASTM C33-90, Fig. 1.	
III.A2.TP-114	III.A2-1(T-10)	Concrete: all	Concrete	Air – indoor, uncontrolled	Reduction of strength and modulus due to elevated temperature (>150°F general; >200°F local)	Plant-specific aging management program Subsection CC-3400 of ASME Section III, Division 2, and Appendix A of ACI 349 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, where the temperatures 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 those given above may be allowed in the concrete if tests and/or calculations are provided	Yes, if temperature limits are exceeded

III STRUCTURES AND COMPONENT SUPPORTS
A2 Group 2 Structures (BWR Reactor Bldg. with Steel Superstructure)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
						to evaluate the reduction in strength and modulus of elasticity and these reductions are applied to the design calculations.	
III.A2.TP-30	III.A2-3(T-08)	Concrete: all	Concrete	Soil	Cracking and distortion due to increased stress levels from settlement	Chapter XI.S6, “Structures Monitoring” 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.	Yes, if a de-watering system is relied upon to control settlement
III.A2.TP-31	III.A2-8(T-09)	Concrete: foundation; subfoundation	Concrete; porous concrete	Water – flowing under foundation	Reduction of foundation strength and cracking due to differential settlement and erosion of porous concrete subfoundation	Chapter XI.S6, “Structures Monitoring” 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.	Yes, if a de-watering system is relied upon to control settlement
III.A2.TP-28	III.A2-10(T-06)	Concrete: interior; above-grade exterior	Concrete	Air – indoor, uncontrolled or Air – outdoor	Increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive chemical attack	Chapter XI.S6, "Structures Monitoring"	No
III.A2.TP-300		High-strength structural bolting	Low-alloy steel, actual measured yield strength ≥ 150 ksi	Air – indoor, uncontrolled or Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.S6, "Structures Monitoring“ Note: ASTM A 325, F 1852, and ASTM A 490 bolts used in civil structures have not shown to be prone to SCC. SCC potential need not be evaluated	No

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
			(1,034 MPa)			for these bolts.	
III.A2.T-12	III.A2-11(T-12)	Masonry walls: all	Concrete block	Air – indoor, uncontrolled or Air – outdoor	Cracking due to restraint shrinkage, creep, and aggressive environment	Chapter XI.S5, “Masonry Walls”	No
III.A2.TP-302	III.A2-12(T-11)	Steel components: all structural steel	Steel	Air – indoor, uncontrolled or Air – outdoor	Loss of material due to corrosion	Chapter XI.S6, “Structures Monitoring” 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
III.A2.TP-261		Structural bolting	Any	Any environment	Loss of preload due to self-loosening	Chapter XI.S6, "Structures Monitoring"	No
III.A2.TP-248		Structural bolting	Steel	Air – indoor, uncontrolled	Loss of material due to general, pitting and crevice corrosion	Chapter XI.S6, "Structures Monitoring"	No
III.A2.TP-274		Structural bolting	Steel; galvanized steel	Air – outdoor	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S6, "Structures Monitoring"	No

A3. GROUP 3 STRUCTURES (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)

Systems, Structures, and Components

This section addresses the elements of the auxiliary building, diesel generator building, radwaste building, turbine building, switchgear room, yard structures, and station blackout (SBO) structures. For this group, the applicable structural elements are identified: concrete, steel, and masonry walls. 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

A3 Group 3 Structures (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)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
III.A3.TP-25	III.A3-2(T-03)	Concrete (accessible areas): all	Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Chapter XI.S6, "Structures Monitoring"	No
III.A3.TP-27	III.A3-4(T-05)	Concrete (accessible areas): below-grade exterior; foundation	Concrete	Ground water/soil	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter XI.S6, "Structures Monitoring"	No
III.A3.TP-23	III.A3-6(T-01)	Concrete (accessible areas): exterior above- and below-grade; foundation	Concrete	Air – outdoor	Loss of material (spalling, scaling) and cracking due to freeze-thaw	Chapter XI.S6, "Structures Monitoring"	No
III.A3.TP-24	III.A3-7(T-02)	Concrete (accessible areas): exterior above- and below-grade; foundation	Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Chapter XI.S6, "Structures Monitoring"	No
III.A3.TP-26	III.A3-9(T-04)	Concrete (accessible areas): interior and above-grade exterior	Concrete	Air – indoor, uncontrolled or Air – outdoor	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter XI.S6, "Structures Monitoring"	No

III STRUCTURES AND COMPONENT SUPPORTS

A3 Group 3 Structures (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)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
III.A3.TP-204	III.A3-2(T-03)	Concrete (inaccessible areas): all	Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Further evaluation is required to determine if a plant-specific aging management program is needed to manage cracking and expansion due to reaction with aggregate of concrete in Inaccessible Areas. A plant-specific aging management program is not required if (1) as described in NUREG-1557, investigations, tests, and petrographic examinations of aggregates performed in accordance with ASTM C295 and other ASTM reactivity tests, as required, can demonstrate that those aggregates do not adversely react within concrete, or (2) For potentially reactive aggregates, aggregate concrete reaction is not significant if it is demonstrated that the in-place concrete can perform its intended function.	Yes, if concrete is not constructed as stated
III.A3.TP-212	III.A3-4(T-05)	Concrete (inaccessible areas): below-grade exterior; foundation	Concrete	Ground water/soil	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter XI.S6, "Structures Monitoring"	No
III.A3.TP-29	III.A3-5(T-07)	Concrete (inaccessible areas): below-grade exterior;	Concrete	Ground water/soil	Increase in porosity and permeability; cracking; loss of	Chapter XI.S6, "Structures Monitoring"	No

III STRUCTURES AND COMPONENT SUPPORTS

A3 Group 3 Structures (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)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
		foundation			material (spalling, scaling) due to aggressive chemical attack		
III.A3.TP-67	III.A3-7(T-02)	Concrete (inaccessible areas): exterior above- and below-grade; foundation	Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Further evaluation is required to determine if a plant-specific aging management program is needed to manage increase in porosity, and permeability due to leaching of calcium hydroxide and carbonation of concrete in Inaccessible Areas. A plant-specific aging management program is not required if (1) There is evidence in the accessible areas that the flowing water has not caused leaching and carbonation, or (2) Evaluation determined that the observed leaching of calcium hydroxide and carbonation in accessible areas has no impact on the intended function of the concrete structure.	Yes, if leaching is observed in accessible areas that impact intended function
III.A3.TP-108	III.A3-6(T-01)	Concrete (inaccessible areas): foundation	Concrete	Air – outdoor	Loss of material (spalling, scaling) and cracking due to freeze-thaw	Further evaluation is required for plants that are located in moderate to severe weathering conditions (weathering index >100 day-inch/yr) (NUREG-1557) to determine if a plant-specific aging management program is needed. A plant-specific aging management program is not required if documented evidence confirms that the existing concrete had air	Yes, for plants located in moderate to severe weathering conditions

III STRUCTURES AND COMPONENT SUPPORTS
A3 Group 3 Structures (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)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
						entrainment content (as per Table CC-2231-2 of the ASME Section III Division 2), and subsequent inspections of accessible areas did not exhibit degradation related to freeze-thaw. Such inspections should be considered a part of the evaluation. If this condition is not satisfied, then a plant-specific aging management program is required to manage loss of material (spalling, scaling) and cracking due to freeze-thaw of concrete in inaccessible areas. The weathering index for the continental US is shown in ASTM C33-90, Fig. 1.	
III.A3.TP-114	III.A3-1(T-10)	Concrete: all	Concrete	Air – indoor, uncontrolled	Reduction of strength and modulus due to elevated temperature (>150°F general; >200°F local)	Plant-specific aging management program Subsection CC-3400 of ASME Section III, Division 2, and Appendix A of ACI 349 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, where the temperatures are not allowed to exceed 200°F. If significant equipment loads are supported by concrete at temperatures exceeding 150°F, an evaluation of the	Yes, if temperature limits are exceeded

III STRUCTURES AND COMPONENT SUPPORTS
A3 Group 3 Structures (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)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
						ability to withstand the postulated design loads is to be made. Higher temperatures than those given above may be allowed in the concrete if tests and/or calculations are provided to evaluate the reduction in strength and modulus of elasticity and these reductions are applied to the design calculations.	
III.A3.TP-30	III.A3-3(T-08)	Concrete: all	Concrete	Soil	Cracking and distortion due to increased stress levels from settlement	Chapter XI.S6, “Structures Monitoring” 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.	Yes, if a de-watering system is relied upon to control settlement
III.A3.TP-31	III.A3-8(T-09)	Concrete: foundation; subfoundation	Concrete; porous concrete	Water – flowing under foundation	Reduction of foundation strength and cracking due to differential settlement and erosion of porous concrete subfoundation	Chapter XI.S6, “Structures Monitoring” 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.	Yes, if a de-watering system is relied upon to control settlement
III.A3.TP-28	III.A3-10(T-06)	Concrete: interior; above-grade exterior	Concrete	Air – indoor, uncontrolled or Air – outdoor	Increase in porosity and permeability; cracking; loss of material (spalling, scaling)	Chapter XI.S6, "Structures Monitoring"	No

III STRUCTURES AND COMPONENT SUPPORTS

A3 Group 3 Structures (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)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
					due to aggressive chemical attack		
III.A3.TP-300		High-strength structural bolting	Low-alloy steel, actual measured yield strength ≥ 150 ksi (1,034 MPa)	Air – indoor, uncontrolled or Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.S6, "Structures Monitoring“ Note: ASTM A 325, F 1852, and ASTM A 490 bolts used in civil structures have not shown to be prone to SCC. SCC potential need not be evaluated for these bolts.	No
III.A3.T-12	III.A3-11(T-12)	Masonry walls: all	Concrete block	Air – indoor, uncontrolled or Air – outdoor	Cracking due to restraint shrinkage, creep, and aggressive environment	Chapter XI.S5, “Masonry Walls”	No
III.A3.TP-302	III.A3-12(T-11)	Steel components: all structural steel	Steel	Air – indoor, uncontrolled or Air – outdoor	Loss of material due to corrosion	Chapter XI.S6, “Structures Monitoring” 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
III.A3.TP-219		Steel components: piles	Steel	Ground water/soil	Loss of material due to corrosion	Chapter XI.S6, "Structures Monitoring"	No
III.A3.TP-261		Structural bolting	Any	Any environment	Loss of preload due to self-loosening	Chapter XI.S6, "Structures Monitoring"	No

III STRUCTURES AND COMPONENT SUPPORTS

A3 Group 3 Structures (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)

Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
III.A3.TP-248		Structural bolting	Steel	Air – indoor, uncontrolled	Loss of material due to general, pitting and crevice corrosion	Chapter XI.S6, "Structures Monitoring"	No
III.A3.TP-274		Structural bolting	Steel; galvanized steel	Air – outdoor	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S6, "Structures Monitoring"	No

A4. GROUP 4 STRUCTURES (CONTAINMENT INTERNAL STRUCTURES, EXCLUDING REFUELING CANAL)

Systems, Structures, and Components

This section addresses the elements of the containment internal structures, excluding refueling canal. For this group, the applicable structural elements are identified: concrete and steel elements. 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
A4 Group 4 Structures (Containment Internal Structures, excluding Refueling Canal)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
III.A4.TP-25	III.A4-2(T-03)	Concrete (accessible areas): all	Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Chapter XI.S6, "Structures Monitoring"	No
III.A4.TP-26	III.A4-3(T-04)	Concrete (accessible areas): interior and above-grade exterior	Concrete	Air – indoor, uncontrolled or Air – outdoor	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter XI.S6, "Structures Monitoring"	No
III.A4.TP-204	III.A4-2(T-03)	Concrete (inaccessible areas): all	Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Further evaluation is required to determine if a plant-specific aging management program is needed to manage cracking and expansion due to reaction with aggregate of concrete in Inaccessible Areas. A plant-specific aging management program is not required if (1) as described in NUREG-1557, investigations, tests, and petrographic examinations of aggregates performed in accordance with ASTM C295 and other ASTM reactivity tests, as required, can demonstrate that those aggregates do not adversely react within concrete, or (2) For potentially reactive aggregates, aggregate concrete reaction is not significant if it is demonstrated that the in-place concrete can perform its intended function.	Yes, if concrete is not constructed as stated

III STRUCTURES AND COMPONENT SUPPORTS
A4 Group 4 Structures (Containment Internal Structures, excluding Refueling Canal)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
III.A4.TP-305		Concrete (inaccessible areas): exterior above- and below-grade; foundation	Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Further evaluation is required to determine if a plant-specific aging management program is needed to manage increase in porosity, and permeability due to leaching of calcium hydroxide and carbonation of concrete in Inaccessible Areas. A plant-specific aging management program is not required if (1) There is evidence in the accessible areas that the flowing water has not caused leaching and carbonation, or (2) Evaluation determined that the observed leaching of calcium hydroxide and carbonation in accessible areas has no impact on the intended function of the concrete structure.	Yes, if leaching is observed in accessible areas that impact intended function
III.A4.TP-114	III.A4-1(T-10)	Concrete: all	Concrete	Air – indoor, uncontrolled	Reduction of strength and modulus due to elevated temperature (>150°F general; >200°F local)	Plant-specific aging management program Subsection CC-3400 of ASME Section III, Division 2, and Appendix A of ACI 349 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, where the temperatures 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	Yes, if temperature limits are exceeded

III STRUCTURES AND COMPONENT SUPPORTS
A4 Group 4 Structures (Containment Internal Structures, excluding Refueling Canal)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
						loads is to be made. Higher temperatures than those given above may be allowed in the concrete if tests and/or calculations are provided to evaluate the reduction in strength and modulus of elasticity and these reductions are applied to the design calculations.	
III.A4.TP-304		Concrete: all	Concrete	Soil	Cracking and distortion due to increased stress levels from settlement	Chapter XI.S6, “Structures Monitoring” 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.	Yes, if a de-watering system is relied upon to control settlement
III.A4.TP-28	III.A4-4(T-06)	Concrete: interior; above-grade exterior	Concrete	Air – indoor, uncontrolled or Air – outdoor	Increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive chemical attack	Chapter XI.S6, "Structures Monitoring"	No
III.A4.TP-300		High-strength structural bolting	Low-alloy steel, actual measured yield strength ≥ 150 ksi (1,034 MPa)	Air – indoor, uncontrolled or Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.S6, "Structures Monitoring“ Note: ASTM A 325, F 1852, and ASTM A 490 bolts used in civil structures have not shown to be prone to SCC. SCC potential need not be evaluated for these bolts.	No

III STRUCTURES AND COMPONENT SUPPORTS
A4 Group 4 Structures (Containment Internal Structures, excluding Refueling Canal)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
III.A4.TP-301		Service Level I coatings	Coatings	Air – indoor, uncontrolled	Loss of coating integrity due to blistering, cracking, flaking, peeling, physical damage	Chapter XI.S8, "Protective Coating Monitoring and Maintenance"	No
III.A4.TP-35	III.A4-6(T-13)	Sliding surfaces: radial beam seats in BWR drywell	Lubrite; Fluorogold; Lubrofluor	Air – indoor, uncontrolled	Loss of mechanical function due to corrosion, distortion, dirt, overload, wear	Chapter XI.S6, "Structures Monitoring"	No
III.A4.TP-302	III.A4-5(T-11)	Steel components: all structural steel	Steel	Air – indoor, uncontrolled or Air – outdoor	Loss of material due to corrosion	Chapter XI.S6, “Structures Monitoring” 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
III.A4.TP-261		Structural bolting	Any	Any environment	Loss of preload due to self-loosening	Chapter XI.S6, "Structures Monitoring"	No
III.A4.TP-248		Structural bolting	Steel	Air – indoor, uncontrolled	Loss of material due to general, pitting and crevice corrosion	Chapter XI.S6, "Structures Monitoring"	No
III.A4.TP-274		Structural bolting	Steel; galvanized steel	Air – outdoor	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S6, "Structures Monitoring"	No

A5. GROUP 5 STRUCTURES (FUEL STORAGE FACILITY, REFUELING CANAL)

Systems, Structures, and Components

This section addresses the elements of the fuel storage facility and refueling canal. For this group, the applicable structural elements are identified: concrete, steel, and masonry walls. 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
A5 Group 5 Structures (Fuel Storage Facility, Refueling Canal)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
III.A5.TP-25	III.A5-2(T-03)	Concrete (accessible areas): all	Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Chapter XI.S6, "Structures Monitoring"	No
III.A5.TP-27	III.A5-4(T-05)	Concrete (accessible areas): below-grade exterior; foundation	Concrete	Ground water/soil	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter XI.S6, "Structures Monitoring"	No
III.A5.TP-23	III.A5-6(T-01)	Concrete (accessible areas): exterior above- and below-grade; foundation	Concrete	Air – outdoor	Loss of material (spalling, scaling) and cracking due to freeze-thaw	Chapter XI.S6, "Structures Monitoring"	No
III.A5.TP-24	III.A5-7(T-02)	Concrete (accessible areas): exterior above- and below-grade; foundation	Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Chapter XI.S6, "Structures Monitoring"	No
III.A5.TP-26	III.A5-9(T-04)	Concrete (accessible areas): interior and above-grade exterior	Concrete	Air – indoor, uncontrolled or Air – outdoor	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter XI.S6, "Structures Monitoring"	No

III STRUCTURES AND COMPONENT SUPPORTS
A5 Group 5 Structures (Fuel Storage Facility, Refueling Canal)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
III.A5.TP-204	III.A5-2(T-03)	Concrete (inaccessible areas): all	Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Further evaluation is required to determine if a plant-specific aging management program is needed to manage cracking and expansion due to reaction with aggregate of concrete in Inaccessible Areas. A plant-specific aging management program is not required if (1) as described in NUREG-1557, investigations, tests, and petrographic examinations of aggregates performed in accordance with ASTM C295 and other ASTM reactivity tests, as required, can demonstrate that those aggregates do not adversely react within concrete, or (2) For potentially reactive aggregates, aggregate concrete reaction is not significant if it is demonstrated that the in-place concrete can perform its intended function.	Yes, if concrete is not constructed as stated
III.A5.TP-212	III.A5-4(T-05)	Concrete (inaccessible areas): below-grade exterior; foundation	Concrete	Ground water/soil	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter XI.S6, "Structures Monitoring"	No
III.A5.TP-29	III.A5-5(T-07)	Concrete (inaccessible areas): below-grade exterior; foundation	Concrete	Ground water/soil	Increase in porosity and permeability; cracking; loss of material (spalling, scaling)	Chapter XI.S6, "Structures Monitoring"	No

III STRUCTURES AND COMPONENT SUPPORTS
A5 Group 5 Structures (Fuel Storage Facility, Refueling Canal)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
					due to aggressive chemical attack		
III.A5.TP-67	III.A5-7(T-02)	Concrete (inaccessible areas): exterior above- and below-grade; foundation	Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Further evaluation is required to determine if a plant-specific aging management program is needed to manage increase in porosity, and permeability due to leaching of calcium hydroxide and carbonation of concrete in Inaccessible Areas. A plant-specific aging management program is not required if (1) There is evidence in the accessible areas that the flowing water has not caused leaching and carbonation, or (2) Evaluation determined that the observed leaching of calcium hydroxide and carbonation in accessible areas has no impact on the intended function of the concrete structure.	Yes, if leaching is observed in accessible areas that impact intended function
III.A5.TP-108	III.A5-6(T-01)	Concrete (inaccessible areas): foundation	Concrete	Air – outdoor	Loss of material (spalling, scaling) and cracking due to freeze-thaw	Further evaluation is required for plants that are located in moderate to severe weathering conditions (weathering index >100 day-inch/yr) (NUREG-1557) to determine if a plant-specific aging management program is needed. A plant-specific aging management program is not required if documented evidence confirms that the existing concrete had air entrainment content (as per Table CC-2231-2 of the ASME Section III Division 2), and subsequent inspections of accessible areas did not	Yes, for plants located in moderate to severe weathering conditions

III STRUCTURES AND COMPONENT SUPPORTS
A5 Group 5 Structures (Fuel Storage Facility, Refueling Canal)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
						exhibit degradation related to freeze-thaw. Such inspections should be considered a part of the evaluation. If this condition is not satisfied, then a plant-specific aging management program is required to manage loss of material (spalling, scaling) and cracking due to freeze-thaw of concrete in inaccessible areas. The weathering index for the continental US is shown in ASTM C33-90, Fig. 1.	
III.A5.TP-114	III.A5-1(T-10)	Concrete: all	Concrete	Air – indoor, uncontrolled	Reduction of strength and modulus due to elevated temperature (>150°F general; >200°F local)	Plant-specific aging management program Subsection CC-3400 of ASME Section III, Division 2, and Appendix A of ACI 349 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, where the temperatures 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 those given above may be allowed in the concrete if tests and/or calculations are provided	Yes, if temperature limits are exceeded

III STRUCTURES AND COMPONENT SUPPORTS
A5 Group 5 Structures (Fuel Storage Facility, Refueling Canal)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
						to evaluate the reduction in strength and modulus of elasticity and these reductions are applied to the design calculations.	
III.A5.TP-30	III.A5-3(T-08)	Concrete: all	Concrete	Soil	Cracking and distortion due to increased stress levels from settlement	Chapter XI.S6, “Structures Monitoring” 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.	Yes, if a de-watering system is relied upon to control settlement
III.A5.TP-31	III.A5-8(T-09)	Concrete: foundation; subfoundation	Concrete; porous concrete	Water – flowing under foundation	Reduction of foundation strength and cracking due to differential settlement and erosion of porous concrete subfoundation	Chapter XI.S6, “Structures Monitoring” 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.	Yes, if a de-watering system is relied upon to control settlement
III.A5.TP-28	III.A5-10(T-06)	Concrete: interior; above-grade exterior	Concrete	Air – indoor, uncontrolled or Air – outdoor	Increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive chemical attack	Chapter XI.S6, "Structures Monitoring"	No
III.A5.TP-300		High-strength structural bolting	Low-alloy steel, actual measured yield strength ≥ 150 ksi	Air – indoor, uncontrolled or Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.S6, "Structures Monitoring“ Note: ASTM A 325, F 1852, and ASTM A 490 bolts used in civil structures have not shown to be prone to SCC. SCC potential need not be evaluated	No

III STRUCTURES AND COMPONENT SUPPORTS
A5 Group 5 Structures (Fuel Storage Facility, Refueling Canal)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
			(1,034 MPa)			for these bolts.	
III.A5.T-12	III.A5-11(T-12)	Masonry walls: all	Concrete block	Air – indoor, uncontrolled or Air – outdoor	Cracking due to restraint shrinkage, creep, and aggressive environment	Chapter XI.S5, “Masonry Walls”	No
III.A5.TP-34		Masonry walls: all	Concrete block	Air – outdoor	Loss of material (spalling, scaling) and cracking due to freeze-thaw	Chapter XI.S5, “Masonry Walls”	No
III.A5.TP-302	III.A5-12(T-11)	Steel components: all structural steel	Steel	Air – indoor, uncontrolled or Air – outdoor	Loss of material due to corrosion	Chapter XI.S6, “Structures Monitoring” 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
III.A5.T-14	III.A5-13(T-14)	Steel components: fuel pool liner	Stainless steel	Treated water or Treated borated water	Cracking due to stress corrosion cracking; Loss of material due to pitting and crevice corrosion	Chapter XI.M2, "Water Chemistry," and monitoring of the spent fuel pool water level in accordance with technical specifications and leakage from the leak chase channels.	No, unless leakages have been detected through the SFP liner that cannot be accounted for from the leak chase channels
III.A5.TP-261		Structural bolting	Any	Any environment	Loss of preload due to self-loosening	Chapter XI.S6, "Structures Monitoring"	No

III STRUCTURES AND COMPONENT SUPPORTS
A5 Group 5 Structures (Fuel Storage Facility, Refueling Canal)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
III.A5.TP-248		Structural bolting	Steel	Air – indoor, uncontrolled	Loss of material due to general, pitting and crevice corrosion	Chapter XI.S6, "Structures Monitoring"	No
III.A5.TP-274		Structural bolting	Steel; galvanized steel	Air – outdoor	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S6, "Structures Monitoring"	No

A6. GROUP 6 STRUCTURES (WATER-CONTROL STRUCTURES)

Systems, Structures, and Components

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
A6 Group 6 Structures (Water-Control Structures)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
III.A6.TP-38	III.A6-1(T-18)	Concrete (accessible areas): all	Concrete	Air – indoor, uncontrolled or Air – outdoor or Ground water/soil	Cracking; loss of bond; and loss of material (spalling, scaling) due to 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 programs.	No
III.A6.TP-36	III.A6-5(T-15)	Concrete (accessible areas): exterior above- and below-grade; foundation	Concrete	Air – outdoor	Loss of material (spalling, scaling) and cracking due to 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 programs.	No
III.A6.TP-37	III.A6-6(T-16)	Concrete (accessible areas): exterior above- and below-grade; foundation; interior slab	Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	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 programs.	No
III.A6.TP-104	III.A6-1(T-18)	Concrete (inaccessible areas): all	Concrete	Air – indoor, uncontrolled or Air – outdoor or Ground water/soil	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter XI.S6, "Structures Monitoring"	No
III.A6.TP-220	III.A6-2(T-17)	Concrete (inaccessible areas): all	Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Further evaluation is required to determine if a plant-specific aging management program is needed to manage cracking and expansion due to reaction with aggregate of concrete in Inaccessible Areas. A plant-specific	Yes, if concrete is not constructed as stated

III STRUCTURES AND COMPONENT SUPPORTS
A6 Group 6 Structures (Water-Control Structures)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
						aging management program is not required if (1) as described in NUREG-1557, investigations, tests, and petrographic examinations of aggregates performed in accordance with ASTM C295 and other ASTM reactivity tests, as required, can demonstrate that those aggregates do not adversely react within concrete, or (2) For potentially reactive aggregates, aggregate concrete reaction is not significant if it is demonstrated that the in-place concrete can perform its intended function.	
III.A6.TP-107	III.A6-3(T-19)	Concrete (inaccessible areas): all	Concrete	Ground water/soil	Increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive chemical attack	Chapter XI.S6, "Structures Monitoring"	No
III.A6.TP-110	III.A6-5(T-15)	Concrete (inaccessible areas): exterior above- and below-grade; foundation; interior slab	Concrete	Air – outdoor	Loss of material (spalling, scaling) and cracking due to freeze-thaw	Further evaluation is required for plants that are located in moderate to severe weathering conditions (weathering index >100 day-inch/yr) (NUREG-1557) to determine if a plant-specific aging management program is needed. A plant-specific aging management program is not required if documented evidence confirms that the existing concrete	Yes, for plants located in moderate to severe weathering conditions

III STRUCTURES AND COMPONENT SUPPORTS
A6 Group 6 Structures (Water-Control Structures)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
						had air entrainment content (as per Table CC-2231-2 of the ASME Section III Division 2), and subsequent inspections of accessible areas did not exhibit degradation related to freeze-thaw. Such inspections should be considered a part of the evaluation. If this condition is not satisfied, then a plant-specific aging management program is required to manage loss of material (spalling, scaling) and cracking due to freeze-thaw of concrete in inaccessible areas. The weathering index for the continental US is shown in ASTM C33-90, Fig. 1.	
III.A6.TP-109	III.A6-6(T-16)	Concrete (inaccessible areas): exterior above- and below-grade; foundation; interior slab	Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Further evaluation is required to determine if a plant-specific aging management program is needed to manage increase in porosity, and permeability due to leaching of calcium hydroxide and carbonation of concrete in Inaccessible Areas. A plant-specific aging management program is not required if (1) There is evidence in the accessible areas that the flowing water has not caused leaching and carbonation, or (2) Evaluation determined that the observed leaching of calcium hydroxide and carbonation in accessible areas has no impact on	Yes, if leaching is observed in accessible areas that impact intended function

III STRUCTURES AND COMPONENT SUPPORTS
A6 Group 6 Structures (Water-Control Structures)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
						the intended function of the concrete structure.	
III.A6.TP-30	III.A6-4(T-08)	Concrete: all	Concrete	Soil	Cracking and distortion due to increased stress levels from settlement	Chapter XI.S6, “Structures Monitoring” 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.	Yes, if a de-watering system is relied upon to control settlement
III.A6.T-20	III.A6-7(T-20)	Concrete: exterior above- and below-grade; foundation; interior slab	Concrete	Water – flowing	Loss of material due to 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 programs.	No
III.A6.TP-31	III.A6-8(T-09)	Concrete: foundation; subfoundation	Concrete; porous concrete	Water – flowing under foundation	Reduction of foundation strength and cracking due to differential settlement and erosion of porous concrete subfoundation	Chapter XI.S6, “Structures Monitoring” 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.	Yes, if a de-watering system is relied upon to control settlement
III.A6.T-22	III.A6-9(T-22)	Earthen water-control structures: dams; embankments; reservoirs; channels; canals and ponds	Various	Water – flowing or standing	Loss of material; loss of form due to erosion, settlement, sedimentation, frost action, waves, currents,	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 programs.	No

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
					surface runoff, seepage		
III.A6.TP-223		Group 6: Wooden Piles; sheeting	Wood	Air – outdoor or Water – flowing or standing or Ground water/soil	Loss of material; change in material properties due to weathering, chemical degradation, and insect infestation repeated wetting and drying, fungal decay	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 programs.	No
III.A6.T-12	III.A6-10(T-12)	Masonry walls: all	Concrete block	Air – indoor, uncontrolled or Air – outdoor	Cracking due to restraint shrinkage, creep, and aggressive environment	Chapter XI.S5, “Masonry Walls”	No
III.A6.TP-7	III.A6-12(TP-7)	Seals; gasket; moisture barriers (caulking, flashing, and other sealants)	Elastomers (such as EPDM rubber)	Various	Loss of sealing due to deterioration of seals, gaskets, and moisture barriers (caulking, flashing, and other sealants)	Chapter XI.S6, "Structures Monitoring"	No
III.A6.TP-261		Structural bolting	Any	Any environment	Loss of preload due to self-loosening	Chapter XI.S6, "Structures Monitoring"	No

III STRUCTURES AND COMPONENT SUPPORTS
A6 Group 6 Structures (Water-Control Structures)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
III.A6.TP-248		Structural bolting	Steel	Air – indoor, uncontrolled	Loss of material due to general, pitting and crevice corrosion	Chapter XI.S6, "Structures Monitoring"	No
III.A6.TP-221		Structural bolting	Steel	Air – indoor, uncontrolled or Air – outdoor or Water – flowing or standing	Loss of material due to general, 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 programs.	No

A7. GROUP 7 STRUCTURES (CONCRETE TANKS AND MISSILE BARRIERS)

Systems, Structures, and Components

This section addresses the elements of concrete tanks and missile barriers. 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
A7 Group 7 Structures (Concrete Tanks and Missile Barriers)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
III.A7.TP-25	III.A7-1(T-03)	Concrete (accessible areas): all	Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Chapter XI.S6, "Structures Monitoring"	No
III.A7.TP-27	III.A7-3(T-05)	Concrete (accessible areas): below-grade exterior; foundation	Concrete	Ground water/soil	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter XI.S6, "Structures Monitoring"	No
III.A7.TP-23	III.A7-5(T-01)	Concrete (accessible areas): exterior above- and below-grade; foundation	Concrete	Air – outdoor	Loss of material (spalling, scaling) and cracking due to freeze-thaw	Chapter XI.S6, "Structures Monitoring"	No
III.A7.TP-24	III.A7-6(T-02)	Concrete (accessible areas): exterior above- and below-grade; foundation	Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Chapter XI.S6, "Structures Monitoring"	No
III.A7.TP-26	III.A7-8(T-04)	Concrete (accessible areas): interior and above-grade exterior	Concrete	Air – indoor, uncontrolled or Air – outdoor	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter XI.S6, "Structures Monitoring"	No

III STRUCTURES AND COMPONENT SUPPORTS
A7 Group 7 Structures (Concrete Tanks and Missile Barriers)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
III.A7.TP-204	III.A7-1(T-03)	Concrete (inaccessible areas): all	Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Further evaluation is required to determine if a plant-specific aging management program is needed to manage cracking and expansion due to reaction with aggregate of concrete in Inaccessible Areas. A plant-specific aging management program is not required if (1) as described in NUREG-1557, investigations, tests, and petrographic examinations of aggregates performed in accordance with ASTM C295 and other ASTM reactivity tests, as required, can demonstrate that those aggregates do not adversely react within concrete, or (2) For potentially reactive aggregates, aggregate concrete reaction is not significant if it is demonstrated that the in-place concrete can perform its intended function.	Yes, if concrete is not constructed as stated
III.A7.TP-212	III.A7-3(T-05)	Concrete (inaccessible areas): below-grade exterior; foundation	Concrete	Ground water/soil	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter XI.S6, "Structures Monitoring"	No
III.A7.TP-29	III.A7-4(T-07)	Concrete (inaccessible areas): below-grade exterior; foundation	Concrete	Ground water/soil	Increase in porosity and permeability; cracking; loss of material (spalling, scaling)	Chapter XI.S6, "Structures Monitoring"	No

III STRUCTURES AND COMPONENT SUPPORTS
A7 Group 7 Structures (Concrete Tanks and Missile Barriers)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
					due to aggressive chemical attack		
III.A7.TP-67	III.A7-6(T-02)	Concrete (inaccessible areas): exterior above- and below-grade; foundation	Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Further evaluation is required to determine if a plant-specific aging management program is needed to manage increase in porosity, and permeability due to leaching of calcium hydroxide and carbonation of concrete in Inaccessible Areas. A plant-specific aging management program is not required if (1) There is evidence in the accessible areas that the flowing water has not caused leaching and carbonation, or (2) Evaluation determined that the observed leaching of calcium hydroxide and carbonation in accessible areas has no impact on the intended function of the concrete structure.	Yes, if leaching is observed in accessible areas that impact intended function
III.A7.TP-108	III.A7-5(T-01)	Concrete (inaccessible areas): foundation	Concrete	Air – outdoor	Loss of material (spalling, scaling) and cracking due to freeze-thaw	Further evaluation is required for plants that are located in moderate to severe weathering conditions (weathering index >100 day-inch/yr) (NUREG-1557) to determine if a plant-specific aging management program is needed. A plant-specific aging management program is not required if documented evidence confirms that the existing concrete had air entrainment content (as per Table CC-2231-2 of the ASME Section III Division 2), and subsequent inspections of accessible areas did not	Yes, for plants located in moderate to severe weathering conditions

III STRUCTURES AND COMPONENT SUPPORTS
A7 Group 7 Structures (Concrete Tanks and Missile Barriers)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
						exhibit degradation related to freeze-thaw. Such inspections should be considered a part of the evaluation. If this condition is not satisfied, then a plant-specific aging management program is required to manage loss of material (spalling, scaling) and cracking due to freeze-thaw of concrete in inaccessible areas. The weathering index for the continental US is shown in ASTM C33-90, Fig. 1.	
III.A7.TP-30	III.A7-2(T-08)	Concrete: all	Concrete	Soil	Cracking and distortion due to increased stress levels from settlement	Chapter XI.S6, “Structures Monitoring” 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.	Yes, if a de-watering system is relied upon to control settlement
III.A7.TP-31	III.A7-7(T-09)	Concrete: foundation; subfoundation	Concrete; porous concrete	Water – flowing under foundation	Reduction of foundation strength and cracking due to differential settlement and erosion of porous concrete subfoundation	Chapter XI.S6, “Structures Monitoring” 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.	Yes, if a de-watering system is relied upon to control settlement
III.A7.TP-28	III.A7-9(T-06)	Concrete: interior; above-grade exterior	Concrete	Air – indoor, uncontrolled or Air – outdoor	Increase in porosity and permeability; cracking; loss of material (spalling, scaling)	Chapter XI.S6, "Structures Monitoring"	No

III STRUCTURES AND COMPONENT SUPPORTS
A7 Group 7 Structures (Concrete Tanks and Missile Barriers)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
					due to aggressive chemical attack		
III.A7.TP-300		High-strength structural bolting	Low-alloy steel, actual measured yield strength ≥ 150 ksi (1,034 MPa)	Air – indoor, uncontrolled or Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.S6, "Structures Monitoring" Note: ASTM A 325, F 1852, and ASTM A 490 bolts used in civil structures have not shown to be prone to SCC. SCC potential need not be evaluated for these bolts.	No
III.A7.TP-302	III.A7-10(T-11)	Steel components: all structural steel	Steel	Air – indoor, uncontrolled or Air – outdoor	Loss of material due to corrosion	Chapter XI.S6, "Structures Monitoring" 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
III.A7.T-23	III.A7-11(T-23)	Steel components: tank liner	Stainless steel	Water – standing	Cracking due to stress corrosion cracking; Loss of material due to pitting and crevice corrosion	A plant-specific aging management program is to be evaluated.	Yes, plant-specific
III.A7.TP-261		Structural bolting	Any	Any environment	Loss of preload due to self-loosening	Chapter XI.S6, "Structures Monitoring"	No
III.A7.TP-248		Structural bolting	Steel	Air – indoor, uncontrolled	Loss of material due to general, pitting and crevice corrosion	Chapter XI.S6, "Structures Monitoring"	No

III STRUCTURES AND COMPONENT SUPPORTS
A7 Group 7 Structures (Concrete Tanks and Missile Barriers)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
III.A7.TP-274		Structural bolting	Steel; galvanized steel	Air – outdoor	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S6, "Structures Monitoring"	No

A8. GROUP 8 STRUCTURES (STEEL TANKS AND MISSILE BARRIERS)

Systems, Structures, and Components

This section addresses the elements of steel tanks and missile barriers. 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
A8 Group 8 Structures (Steel Tanks and Missile Barriers)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
III.A8.TP-25	III.A8-1(T-03)	Concrete (accessible areas): all	Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Chapter XI.S6, "Structures Monitoring"	No
III.A8.TP-27	III.A8-3(T-05)	Concrete (accessible areas): below-grade exterior; foundation	Concrete	Ground water/soil	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter XI.S6, "Structures Monitoring"	No
III.A8.TP-23	III.A8-5(T-01)	Concrete (accessible areas): exterior above- and below-grade; foundation	Concrete	Air – outdoor	Loss of material (spalling, scaling) and cracking due to freeze-thaw	Chapter XI.S6, "Structures Monitoring"	No
III.A8.TP-24	III.A8-6(T-02)	Concrete (accessible areas): exterior above- and below-grade; foundation	Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Chapter XI.S6, "Structures Monitoring"	No
III.A8.TP-204	III.A8-1(T-03)	Concrete (inaccessible areas): all	Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Further evaluation is required to determine if a plant-specific aging management program is needed to manage cracking and expansion due to reaction with aggregate of concrete in Inaccessible Areas. A plant-specific aging management program is not required if (1) as described in NUREG-	Yes, if concrete is not constructed as stated

III STRUCTURES AND COMPONENT SUPPORTS
A8 Group 8 Structures (Steel Tanks and Missile Barriers)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
						1557, investigations, tests, and petrographic examinations of aggregates performed in accordance with ASTM C295 and other ASTM reactivity tests, as required, can demonstrate that those aggregates do not adversely react within concrete, or (2) For potentially reactive aggregates, aggregate concrete reaction is not significant if it is demonstrated that the in-place concrete can perform its intended function.	
III.A8.TP-212	III.A8-3(T-05)	Concrete (inaccessible areas): below-grade exterior; foundation	Concrete	Ground water/soil	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter XI.S6, "Structures Monitoring"	No
III.A8.TP-29	III.A8-4(T-07)	Concrete (inaccessible areas): below-grade exterior; foundation	Concrete	Ground water/soil	Increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive chemical attack	Chapter XI.S6, "Structures Monitoring"	No
III.A8.TP-67	III.A8-6(T-02)	Concrete (inaccessible areas): exterior above- and below-grade; foundation	Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide	Further evaluation is required to determine if a plant-specific aging management program is needed to manage increase in porosity, and permeability due to leaching of calcium hydroxide and carbonation of concrete	Yes, if leaching is observed in accessible areas that impact

III STRUCTURES AND COMPONENT SUPPORTS
A8 Group 8 Structures (Steel Tanks and Missile Barriers)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
					and carbonation	in Inaccessible Areas. A plant-specific aging management program is not required if (1) There is evidence in the accessible areas that the flowing water has not caused leaching and carbonation, or (2) Evaluation determined that the observed leaching of calcium hydroxide and carbonation in accessible areas has no impact on the intended function of the concrete structure.	intended function
III.A8.TP-108	III.A8-5(T-01)	Concrete (inaccessible areas): foundation	Concrete	Air – outdoor	Loss of material (spalling, scaling) and cracking due to freeze-thaw	Further evaluation is required for plants that are located in moderate to severe weathering conditions (weathering index >100 day-inch/yr) (NUREG-1557) to determine if a plant-specific aging management program is needed. A plant-specific aging management program is not required if documented evidence confirms that the existing concrete had air entrainment content (as per Table CC-2231-2 of the ASME Section III Division 2), and subsequent inspections of accessible areas did not exhibit degradation related to freeze-thaw. Such inspections should be considered a part of the evaluation. If this condition is not satisfied, then a plant-specific aging management program is required to manage loss of material (spalling, scaling) and cracking due to freeze-thaw of concrete in inaccessible areas.	Yes, for plants located in moderate to severe weathering conditions

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
						The weathering index for the continental US is shown in ASTM C33-90, Fig. 1.	
III.A8.TP-30	III.A8-2(T-08)	Concrete: all	Concrete	Soil	Cracking and distortion due to increased stress levels from settlement	Chapter XI.S6, “Structures Monitoring” 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.	Yes, if a de-watering system is relied upon to control settlement
III.A8.TP-31	III.A8-7(T-09)	Concrete: foundation; subfoundation	Concrete; porous concrete	Water – flowing under foundation	Reduction of foundation strength and cracking due to differential settlement and erosion of porous concrete subfoundation	Chapter XI.S6, “Structures Monitoring” 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.	Yes, if a de-watering system is relied upon to control settlement
III.A8.TP-300		High-strength structural bolting	Low-alloy steel, actual measured yield strength ≥ 150 ksi (1,034 MPa)	Air – indoor, uncontrolled or Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.S6, "Structures Monitoring“ Note: ASTM A 325, F 1852, and ASTM A 490 bolts used in civil structures have not shown to be prone to SCC. SCC potential need not be evaluated for these bolts.	No
III.A8.TP-302	III.A8-8(T-11)	Steel components: all structural steel	Steel	Air – indoor, uncontrolled or Air – outdoor	Loss of material due to corrosion	Chapter XI.S6, “Structures Monitoring” 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

III STRUCTURES AND COMPONENT SUPPORTS
A8 Group 8 Structures (Steel Tanks and Missile Barriers)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
III.A8.T-23	III.A8-9(T-23)	Steel components: tank liner	Stainless steel	Water – standing	Cracking due to stress corrosion cracking; Loss of material due to pitting and crevice corrosion	A plant-specific aging management program is to be evaluated.	Yes, plant-specific
III.A8.TP-261		Structural bolting	Any	Any environment	Loss of preload due to self-loosening	Chapter XI.S6, "Structures Monitoring"	No
III.A8.TP-248		Structural bolting	Steel	Air – indoor, uncontrolled	Loss of material due to general, pitting and crevice corrosion	Chapter XI.S6, "Structures Monitoring"	No
III.A8.TP-274		Structural bolting	Steel; galvanized steel	Air – outdoor	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S6, "Structures Monitoring"	No

A9. GROUP 9 STRUCTURES (BWR UNIT VENT STACK)

Systems, Structures, and Components

This section addresses the elements of the boiling water reactor (BWR) unit vent stack. For this group, the applicable structural element is identified: concrete. 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
A9 Group 9 Structures (BWR Unit Vent Stack)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
III.A9.TP-25	III.A9-1(T-03)	Concrete (accessible areas): all	Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Chapter XI.S6, "Structures Monitoring"	No
III.A9.TP-27	III.A9-3(T-05)	Concrete (accessible areas): below-grade exterior; foundation	Concrete	Ground water/soil	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter XI.S6, "Structures Monitoring"	No
III.A9.TP-23	III.A9-5(T-01)	Concrete (accessible areas): exterior above- and below-grade; foundation	Concrete	Air – outdoor	Loss of material (spalling, scaling) and cracking due to freeze-thaw	Chapter XI.S6, "Structures Monitoring"	No
III.A9.TP-24	III.A9-6(T-02)	Concrete (accessible areas): exterior above- and below-grade; foundation	Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Chapter XI.S6, "Structures Monitoring"	No
III.A9.TP-26	III.A9-8(T-04)	Concrete (accessible areas): interior and above-grade exterior	Concrete	Air – indoor, uncontrolled or Air – outdoor	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter XI.S6, "Structures Monitoring"	No

III STRUCTURES AND COMPONENT SUPPORTS
A9 Group 9 Structures (BWR Unit Vent Stack)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
III.A9.TP-204	III.A9-1(T-03)	Concrete (inaccessible areas): all	Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Further evaluation is required to determine if a plant-specific aging management program is needed to manage cracking and expansion due to reaction with aggregate of concrete in Inaccessible Areas. A plant-specific aging management program is not required if (1) as described in NUREG-1557, investigations, tests, and petrographic examinations of aggregates performed in accordance with ASTM C295 and other ASTM reactivity tests, as required, can demonstrate that those aggregates do not adversely react within concrete, or (2) For potentially reactive aggregates, aggregate concrete reaction is not significant if it is demonstrated that the in-place concrete can perform its intended function.	Yes, if concrete is not constructed as stated
III.A9.TP-212	III.A9-3(T-05)	Concrete (inaccessible areas): below-grade exterior; foundation	Concrete	Ground water/soil	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter XI.S6, "Structures Monitoring"	No
III.A9.TP-29	III.A9-4(T-07)	Concrete (inaccessible areas): below-grade exterior; foundation	Concrete	Ground water/soil	Increase in porosity and permeability; cracking; loss of material (spalling, scaling)	Chapter XI.S6, "Structures Monitoring"	No

III A9 STRUCTURES AND COMPONENT SUPPORTS
Group 9 Structures (BWR Unit Vent Stack)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
					due to aggressive chemical attack		
III.A9.TP-67	III.A9-6(T-02)	Concrete (inaccessible areas): exterior above- and below-grade; foundation	Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Further evaluation is required to determine if a plant-specific aging management program is needed to manage increase in porosity, and permeability due to leaching of calcium hydroxide and carbonation of concrete in Inaccessible Areas. A plant-specific aging management program is not required if (1) There is evidence in the accessible areas that the flowing water has not caused leaching and carbonation, or (2) Evaluation determined that the observed leaching of calcium hydroxide and carbonation in accessible areas has no impact on the intended function of the concrete structure.	Yes, if leaching is observed in accessible areas that impact intended function
III.A9.TP-108	III.A9-5(T-01)	Concrete (inaccessible areas): foundation	Concrete	Air – outdoor	Loss of material (spalling, scaling) and cracking due to freeze-thaw	Further evaluation is required for plants that are located in moderate to severe weathering conditions (weathering index >100 day-inch/yr) (NUREG-1557) to determine if a plant-specific aging management program is needed. A plant-specific aging management program is not required if documented evidence confirms that the existing concrete had air entrainment content (as per Table CC-2231-2 of the ASME Section III Division 2), and subsequent inspections of accessible areas did not	Yes, for plants located in moderate to severe weathering conditions

III STRUCTURES AND COMPONENT SUPPORTS
A9 Group 9 Structures (BWR Unit Vent Stack)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
						exhibit degradation related to freeze-thaw. Such inspections should be considered a part of the evaluation. If this condition is not satisfied, then a plant-specific aging management program is required to manage loss of material (spalling, scaling) and cracking due to freeze-thaw of concrete in inaccessible areas. The weathering index for the continental US is shown in ASTM C33-90, Fig. 1.	
III.A9.TP-30	III.A9-2(T-08)	Concrete: all	Concrete	Soil	Cracking and distortion due to increased stress levels from settlement	Chapter XI.S6, “Structures Monitoring” 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.	Yes, if a de-watering system is relied upon to control settlement
III.A9.TP-31	III.A9-7(T-09)	Concrete: foundation; subfoundation	Concrete; porous concrete	Water – flowing under foundation	Reduction in foundation strength, cracking due to differential settlement, erosion of porous concrete subfoundation	Chapter XI.S6, “Structures Monitoring” 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.	Yes, if a de-watering system is relied upon to control settlement
III.A9.TP-28	III.A9-9(T-06)	Concrete: interior; above-grade exterior	Concrete	Air – indoor, uncontrolled or Air – outdoor	Increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive	Chapter XI.S6, "Structures Monitoring"	No

III STRUCTURES AND COMPONENT SUPPORTS
A9 Group 9 Structures (BWR Unit Vent Stack)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
					chemical attack		
III.A9.TP-300		High-strength structural bolting	Low-alloy steel, actual measured yield strength ≥ 150 ksi (1,034 MPa)	Air – indoor, uncontrolled or Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.S6, "Structures Monitoring“ Note: ASTM A 325, F 1852, and ASTM A 490 bolts used in civil structures have not shown to be prone to SCC. SCC potential need not be evaluated for these bolts.	No
III.A9.TP-261		Structural bolting	Any	Any environment	Loss of preload due to self-loosening	Chapter XI.S6, "Structures Monitoring"	No
III.A9.TP-248		Structural bolting	Steel	Air – indoor, uncontrolled	Loss of material due to general, pitting and crevice corrosion	Chapter XI.S6, "Structures Monitoring"	No
III.A9.TP-274		Structural bolting	Steel; galvanized steel	Air – outdoor	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S6, "Structures Monitoring"	No

III.B COMPONENT SUPPORTS

B1. Supports for ASME Piping and Components

B2. Supports for Cable Trays, Conduit, HVAC Ducts, TubeTrack®, Instrument Tubing, Non-ASME Piping and Components

B3. Anchorage of Racks, Panels, Cabinets, and Enclosures for Electrical Equipment and Instrumentation

B4. Supports for Emergency Diesel Generator (EDG), HVAC System Components, and Other Miscellaneous Mechanical Equipment

B5. Supports for Platforms, Pipe Whip Restraints, Jet Impingement Shields, Masonry Walls, and Other Miscellaneous Structures

B1. SUPPORTS FOR ASME PIPING AND COMPONENTS

B1.1 Class 1

B1.2 Class 2 and 3

B1.3 Class MC (BWR Containment Supports)

B1. SUPPORTS FOR ASME PIPING AND COMPONENTS

Systems, Structures, and Components

This section addresses supports and anchorage for ASME piping systems and components. It is subdivided into Class 1 (III.B1.1), Class 2 and 3 (III.B1.2), and Class MC (III.B1.3). 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
B1.1 Class 1

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
III.B1.1.TP-42	III.B1.1-1(T-29)	Building concrete at locations of expansion and grouted anchors; grout pads for support base plates	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"	No
III.B1.1.T-28	III.B1.1-2(T-28)	Constant and variable load spring hangers; guides; stops	Steel	Air – indoor, uncontrolled or Air – outdoor	Loss of mechanical function due to corrosion, distortion, dirt, overload, fatigue due to vibratory and cyclic thermal loads	Chapter XI.S3, “ASME Section XI, Subsection IWF”	No
III.B1.1.TP-41	III.B1.1-3(T-27)	High-strength structural bolting	Low-alloy steel, actual measured yield strength ≥ 150 ksi (1,034 MPa)	Air – indoor, uncontrolled	Cracking due to stress corrosion cracking	Chapter XI.S3, “ASME Section XI, Subsection IWF”	No
III.B1.1.TP-45	III.B1.1-5(T-32)	Sliding surfaces	Lubrite®; graphitic tool steel; Fluorogold; Lubrofluor	Air – indoor, uncontrolled or Air – outdoor	Loss of mechanical function due to corrosion, distortion, dirt, debris, overload, wear	Chapter XI.S3, “ASME Section XI, Subsection IWF”	No

III STRUCTURES AND COMPONENT SUPPORTS
B1.1 Class 1

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
III.B1.1.TP-229		Structural bolting	Any	Any environment	Loss of preload due to self-loosening	Chapter XI.S3, “ASME Section XI, Subsection IWF”	No
III.B1.1.TP-232		Structural bolting	Stainless steel	Treated water	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.S3, “ASME Section XI, Subsection IWF”	No
III.B1.1.TP-226		Structural Bolting	Steel	Air – indoor, uncontrolled	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S3, “ASME Section XI, Subsection IWF”	No
III.B1.1.TP-235		Structural bolting	Steel; galvanized steel	Air – outdoor	Loss of material due to pitting and crevice corrosion	Chapter XI.S3, “ASME Section XI, Subsection IWF”	No
III.B1.1.TP-8	III.B1.1-6(TP-8) III.B1.1-7(TP-11) III.B1.1-9(TP-5)	Support members; welds; bolted connections; support anchorage to building structure	Aluminum; galvanized steel; stainless steel	Air – indoor, uncontrolled	None	None	No
III.B1.1.TP-3	III.B1.1-8(TP-3)	Support members; welds; bolted connections; support anchorage to building structure	Galvanized steel; aluminum	Air with borated water leakage	Loss of material due to boric acid corrosion	Chapter XI.M10, “Boric Acid Corrosion”	No

III STRUCTURES AND COMPONENT SUPPORTS
B1.1 Class 1

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
III.B1.1.TP-4	III.B1.1-10(TP-4)	Support members; welds; bolted connections; support anchorage to building structure	Stainless steel	Air with borated water leakage	None	None	No
III.B1.1.T-26	III.B1.1-12(T-26)	Support members; welds; bolted connections; support anchorage to building structure	Steel	Air – indoor, uncontrolled	Cumulative fatigue damage due to 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 SRP, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
III.B1.1.T-24	III.B1.1-13(T-24)	Support members; welds; bolted connections; support anchorage to building structure	Steel	Air – indoor, uncontrolled or Air – outdoor	Loss of material due to general and pitting corrosion	Chapter XI.S3, “ASME Section XI, Subsection IWF”	No
III.B1.1.T-25	III.B1.1-14(T-25)	Support members; welds; bolted connections; support anchorage to building structure	Steel	Air with borated water leakage	Loss of material due to boric acid corrosion	Chapter XI.M10, “Boric Acid Corrosion”	No

III STRUCTURES AND COMPONENT SUPPORTS
B1.1 Class 1

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
III.B1.1.TP-10	III.B1.1-11(TP-10)	Support members; welds; bolted connections; support anchorage to building structure	Steel; stainless steel	Treated water <60C (<140 F)	Loss of material due to general (steel only), pitting, and crevice corrosion	Chapter XI.M2, "Water Chemistry," for BWR water, and Chapter XI.S3, "ASME Section XI, Subsection IWF"	No
III.B1.1.T-33	III.B1.1-15(T-33)	Vibration isolation elements	Non-metallic (e.g., rubber)	Air – indoor, uncontrolled or Air – outdoor	Reduction or loss of isolation function due to radiation hardening, temperature, humidity, sustained vibratory loading	Chapter XI.S3, "ASME Section XI, Subsection IWF"	No

III STRUCTURES AND COMPONENT SUPPORTS
B1.2 Class 2 and 3

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
III.B1.2.TP-42	III.B1.2-1(T-29)	Building concrete at locations of expansion and grouted anchors; grout pads for support base plates	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"	No
III.B1.2.T-28	III.B1.2-2(T-28)	Constant and variable load spring hangers; guides; stops	Steel	Air – indoor, uncontrolled or Air – outdoor	Loss of mechanical function due to corrosion, distortion, dirt, overload, fatigue due to vibratory and cyclic thermal loads	Chapter XI.S3, "ASME Section XI, Subsection IWF"	No
III.B1.2.TP-45	III.B1.2-3(T-32)	Sliding surfaces	Lubrite®; graphitic tool steel; Fluorogold; Lubrofluor	Air – indoor, uncontrolled or Air – outdoor	Loss of mechanical function due to corrosion, distortion, dirt, debris, overload, wear	Chapter XI.S3, "ASME Section XI, Subsection IWF"	No
III.B1.2.TP-229		Structural bolting	Any	Any environment	Loss of preload due to self-loosening	Chapter XI.S3, "ASME Section XI, Subsection IWF"	No

III STRUCTURES AND COMPONENT SUPPORTS
B1.2 Class 2 and 3

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
III.B1.2.TP-232		Structural bolting	Stainless steel	Treated water	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, "Water Chemistry," and Chapter XI.S3, "ASME Section XI, Subsection IWF"	No
III.B1.2.TP-226		Structural Bolting	Steel	Air – indoor, uncontrolled	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S3, "ASME Section XI, Subsection IWF"	No
III.B1.2.TP-235		Structural bolting	Steel; galvanized steel	Air – outdoor	Loss of material due to pitting and crevice corrosion	Chapter XI.S3, "ASME Section XI, Subsection IWF"	No
III.B1.2.TP-8	III.B1.2-4(TP-8) III.B1.2-5(TP-11) III.B1.2-7(TP-5)	Support members; welds; bolted connections; support anchorage to building structure	Aluminum; galvanized steel; stainless steel	Air – indoor, uncontrolled	None	None	No
III.B1-2.TP-3	III.B1.2-6(TP-3)	Support members; welds; bolted connections; support anchorage to building structure	Galvanized steel; aluminum	Air with borated water leakage	Loss of material due to boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No
III.B1.2.TP-4	III.B1.2-8(TP-4)	Support members; welds; bolted connections; support	Stainless steel	Air with borated water leakage	None	None	No

III STRUCTURES AND COMPONENT SUPPORTS
B1.2 Class 2 and 3

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
		anchorage to building structure					
III.B1.2.T-26	III.B1.2-9(T-26)	Support members; welds; bolted connections; support anchorage to building structure	Steel	Air – indoor, uncontrolled	Cumulative fatigue damage due to 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 SRP, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
III.B1.2.T-24	III.B1.2-10(T-24)	Support members; welds; bolted connections; support anchorage to building structure	Steel	Air – indoor, uncontrolled or Air – outdoor	Loss of material due to general and pitting corrosion	Chapter XI.S3, “ASME Section XI, Subsection IWF”	No
III.B1.2.T-25	III.B1.2-11(T-25)	Support members; welds; bolted connections; support anchorage to building structure	Steel	Air with borated water leakage	Loss of material due to boric acid corrosion	Chapter XI.M10, “Boric Acid Corrosion”	No
III.B1.2.T-33	III.B1.2-12(T-33)	Vibration isolation elements	Non-metallic (e.g., rubber)	Air – indoor, uncontrolled or Air – outdoor	Reduction or loss of isolation function due to radiation hardening, temperature,	Chapter XI.S3, “ASME Section XI, Subsection IWF”	No

III STRUCTURES AND COMPONENT SUPPORTS
B1.2 Class 2 and 3

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
					humidity, sustained vibratory loading		

III STRUCTURES AND COMPONENT SUPPORTS
B1.3 Class MC (BWR Containment Supports)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
III.B1.3.TP-42	III.B1.3-1(T-29)	Building concrete at locations of expansion and grouted anchors; grout pads for support base plates	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"	No
III.B1.3.T-28	III.B1.3-2(T-28)	Constant and variable load spring hangers; guides; stops	Steel	Air – indoor, uncontrolled or Air – outdoor	Loss of mechanical function due to corrosion, distortion, dirt, overload, fatigue due to vibratory and cyclic thermal loads	Chapter XI.S3, “ASME Section XI, Subsection IWF”	No
III.B1.3.TP-45	III.B1.3-3(T-32)	Sliding surfaces	Lubrite®; graphitic tool steel; Fluorogold; Lubrofluor	Air – indoor, uncontrolled or Air – outdoor	Loss of mechanical function due to corrosion, distortion, dirt, debris, overload, wear	Chapter XI.S3, “ASME Section XI, Subsection IWF”	No
III.B1.3.TP-229		Structural bolting	Any	Any environment	Loss of preload due to self-loosening	Chapter XI.S3, “ASME Section XI, Subsection IWF”	No
III.B1.3.TP-232		Structural bolting	Stainless steel	Treated water	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.S3, “ASME Section XI, Subsection IWF”	No

III STRUCTURES AND COMPONENT SUPPORTS
B1.3 Class MC (BWR Containment Supports)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
III.B1.3.TP-226		Structural bolting	Steel	Air – indoor, uncontrolled	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S3, “ASME Section XI, Subsection IWF”	No
III.B1.3.TP-235		Structural bolting	Steel; galvanized steel	Air – outdoor	Loss of material due to pitting and crevice corrosion	Chapter XI.S3, “ASME Section XI, Subsection IWF”	No
III.B1-3.TP-8	III.B1.3-4(TP-8) III.B1.3-5(TP-11) III.B1.3-7(TP-5)	Support members; welds; bolted connections; support anchorage to building structure	Aluminum; galvanized steel; stainless steel	Air – indoor, uncontrolled	None	None	No
III.B1.3.TP-3	III.B1.3-6(TP-3)	Support members; welds; bolted connections; support anchorage to building structure	Galvanized steel; aluminum	Air with borated water leakage	Loss of material due to boric acid corrosion	Chapter XI.M10, “Boric Acid Corrosion”	No
III.B1.3.TP-4	III.B1.3-8(TP-4)	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
B1.3 Class MC (BWR Containment Supports)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
III.B1.3.T-26	III.B1.3-9(T-26)	Support members; welds; bolted connections; support anchorage to building structure	Steel	Air – indoor, uncontrolled	Cumulative fatigue damage due to 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 SRP, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
III.B1.3.T-24	III.B1.3-10(T-24)	Support members; welds; bolted connections; support anchorage to building structure	Steel	Air – indoor, uncontrolled or Air – outdoor	Loss of material due to general and pitting corrosion	Chapter XI.S3, "ASME Section XI, Subsection IWF"	No
III.B1.3.T-33	III.B1.3-11(T-33)	Vibration isolation elements	Non-metallic (e.g., rubber)	Air – indoor, uncontrolled or Air – outdoor	Reduction or loss of isolation function due to radiation hardening, temperature, humidity, sustained vibratory loading	Chapter XI.S3, "ASME Section XI, Subsection IWF"	No

B2. SUPPORTS FOR CABLE TRAYS, CONDUIT, HVAC DUCTS, TUBETRACK®, INSTRUMENT TUBING, NON-ASME PIPING AND COMPONENTS

Systems, Structures, and Components

This section addresses supports and anchorage for cable trays, conduit, heating, ventilation, and air-conditioning (HVAC) ducts, TubeTrack®, instrument tubing, and non-ASME piping and components. 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
III.B2.TP-42	III.B2-1(T-29)	Building concrete at locations of expansion and grouted anchors; grout pads for support base plates	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"	No
III.B2.TP-300		High-strength structural bolting	Low-alloy steel, actual measured yield strength ≥ 150 ksi (1,034 MPa)	Air – indoor, uncontrolled or Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.S6, "Structures Monitoring“ Note: ASTM A 325, F 1852, and ASTM A 490 bolts used in civil structures have not shown to be prone to SCC. SCC potential need not be evaluated for these bolts.	No
III.B2.TP-46	III.B2-2(TP-1)	Sliding support bearings; sliding support surfaces	Lubrite®; graphitic tool steel; Fluorogold; Lubrofluor	Air – indoor, uncontrolled	Loss of mechanical function due to corrosion, distortion, dirt, debris, overload, wear	Chapter XI.S6, "Structures Monitoring"	No
III.B2.TP-47	III.B2-3(TP-2)	Sliding support bearings; sliding support surfaces	Lubrite®; graphitic tool steel; Fluorogold; Lubrofluor	Air – outdoor	Loss of mechanical function due to corrosion, distortion, dirt, debris, overload, wear	Chapter XI.S6, "Structures Monitoring"	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
III.B2.TP-261		Structural bolting	Any	Any environment	Loss of preload due to self-loosening	Chapter XI.S6, "Structures Monitoring"	No
III.B2.TP-248		Structural bolting	Steel	Air – indoor, uncontrolled	Loss of material due to general, pitting and crevice corrosion	Chapter XI.S6, "Structures Monitoring"	No
III.B2.TP-274		Structural bolting	Steel; galvanized steel	Air – outdoor	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S6, "Structures Monitoring"	No
III.B2.TP-8	III.B2-4(TP-8) III.B2-8(TP-5) III.B2-5(TP-11)	Support members; welds; bolted connections; support anchorage to building structure	Aluminum; galvanized steel; stainless steel	Air – indoor, uncontrolled	None	None	No
III.B2.TP-3	III.B2-6(TP-3)	Support members; welds; bolted connections; support anchorage to building structure	Galvanized steel; aluminum	Air with borated water leakage	Loss of material due to boric acid corrosion	Chapter XI.M10, “Boric Acid Corrosion”	No
III.B2.TP-6	III.B2-7(TP-6)	Support members; welds; bolted connections; support anchorage to	Galvanized steel; aluminum; stainless steel	Air – outdoor	Loss of material due to pitting and crevice corrosion	Chapter XI.S6, "Structures Monitoring"	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
		building structure					
III.B2.TP-4	III.B2-9(TP-4)	Support members; welds; bolted connections; support anchorage to building structure	Stainless steel	Air with borated water leakage	None	None	No
III.B2.TP-43	III.B2-10(T-30)	Support members; welds; bolted connections; support anchorage to building structure	Steel	Air – indoor, uncontrolled or Air – outdoor	Loss of material due to general and pitting corrosion	Chapter XI.S6, "Structures Monitoring"	No
III.B2.T-25	III.B2-11(T-25)	Support members; welds; bolted connections; support anchorage to building structure	Steel	Air with borated water leakage	Loss of material due to boric acid corrosion	Chapter XI.M10, “Boric Acid Corrosion”	No

B3. ANCHORAGE OF RACKS, PANELS, CABINETS, AND ENCLOSURES FOR ELECTRICAL EQUIPMENT AND INSTRUMENTATION

Systems, Structures, and Components

This section addresses supports and anchorage for racks, panels, cabinets, and enclosures for electrical equipment and instrumentation. 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
B3 Anchorage of Racks, Panels, Cabinets, and Enclosures for Electrical Equipment and Instrumentation

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
III.B3.TP-42	III.B3-1(T-29)	Building concrete at locations of expansion and grouted anchors; grout pads for support base plates	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"	No
III.B3.TP-300		High-strength structural bolting	Low-alloy steel, actual measured yield strength ≥ 150 ksi (1,034 MPa)	Air – indoor, uncontrolled or Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.S6, "Structures Monitoring“ Note: ASTM A 325, F 1852, and ASTM A 490 bolts used in civil structures have not shown to be prone to SCC. SCC potential need not be evaluated for these bolts.	No
III.B3.TP-261		Structural bolting	Any	Any environment	Loss of preload due to self-loosening	Chapter XI.S6, "Structures Monitoring"	No
III.B3.TP-248		Structural bolting	Steel	Air – indoor, uncontrolled	Loss of material due to general, pitting and crevice corrosion	Chapter XI.S6, "Structures Monitoring"	No
III.B3.TP-274		Structural bolting	Steel; galvanized steel	Air – outdoor	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S6, "Structures Monitoring"	No
III.B3.TP-8	III.B3-2(TP-8) III.B3-5(TP-5)	Support members; welds; bolted connections;	Aluminum; galvanized steel; stainless	Air – indoor, uncontrolled	None	None	No

III STRUCTURES AND COMPONENT SUPPORTS
B3 Anchorage of Racks, Panels, Cabinets, and Enclosures for Electrical Equipment and Instrumentation

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
	III.B3-3(TP-11)	support anchorage to building structure	steel				
III.B3.TP-3	III.B3-4(TP-3)	Support members; welds; bolted connections; support anchorage to building structure	Galvanized steel; aluminum	Air with borated water leakage	Loss of material due to boric acid corrosion	Chapter XI.M10, “Boric Acid Corrosion”	No
III.B3.TP-4	III.B3-6(TP-4)	Support members; welds; bolted connections; support anchorage to building structure	Stainless steel	Air with borated water leakage	None	None	No
III.B3.TP-43	III.B3-7(T-30)	Support members; welds; bolted connections; support anchorage to building structure	Steel	Air – indoor, uncontrolled or Air – outdoor	Loss of material due to general and pitting corrosion	Chapter XI.S6, "Structures Monitoring"	No
III.B3.T-25	III.B3-8(T-25)	Support members; welds; bolted connections; support	Steel	Air with borated water leakage	Loss of material due to boric acid corrosion	Chapter XI.M10, “Boric Acid Corrosion”	No

III STRUCTURES AND COMPONENT SUPPORTS
B3 Anchorage of Racks, Panels, Cabinets, and Enclosures for Electrical Equipment and Instrumentation

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
		anchorage to building structure					

B4. SUPPORTS FOR EMERGENCY DIESEL GENERATOR (EDG), HVAC SYSTEM COMPONENTS, AND OTHER MISCELLANEOUS MECHANICAL EQUIPMENT

Systems, Structures, and Components

This section addresses supports and anchorage for the emergency diesel generator (EDG) and HVAC system components, and other miscellaneous mechanical equipment. 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
B4 Supports for Emergency Diesel Generator (EDG), HVAC System Components, and Other Miscellaneous Mechanical Equipment

Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
III.B4.TP-42	III.B4-1(T-29)	Building concrete at locations of expansion and grouted anchors; grout pads for support base plates	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"	No
III.B4.TP-300		High-strength structural bolting	Low-alloy steel, actual measured yield strength ≥ 150 ksi (1,034 MPa)	Air – indoor, uncontrolled or Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.S6, "Structures Monitoring“ Note: ASTM A 325, F 1852, and ASTM A 490 bolts used in civil structures have not shown to be prone to SCC. SCC potential need not be evaluated for these bolts.	No
III.B4.TP-46	III.B4-2(TP-1)	Sliding support bearings; sliding support surfaces	Lubrite®; graphitic tool steel; Fluorogold; Lubrofluor	Air – indoor, uncontrolled	Loss of mechanical function due to corrosion, distortion, dirt, debris, overload, wear	Chapter XI.S6, "Structures Monitoring"	No
III.B4.TP-47	III.B4-3(TP-2)	Sliding support bearings; sliding support surfaces	Lubrite®; graphitic tool steel; Fluorogold; Lubrofluor	Air – outdoor	Loss of mechanical function due to corrosion, distortion, dirt, debris, overload, wear	Chapter XI.S6, "Structures Monitoring"	No

III STRUCTURES AND COMPONENT SUPPORTS
B4 Supports for Emergency Diesel Generator (EDG), HVAC System Components, and Other Miscellaneous Mechanical Equipment

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
III.B4.TP-261		Structural bolting	Any	Any environment	Loss of preload due to self-loosening	Chapter XI.S6, "Structures Monitoring"	No
III.B4.TP-248		Structural bolting	Steel	Air – indoor, uncontrolled	Loss of material due to general, pitting and crevice corrosion	Chapter XI.S6, "Structures Monitoring"	No
III.B4.TP-274		Structural bolting	Steel; galvanized steel	Air – outdoor	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S6, "Structures Monitoring"	No
III.B4.TP-8	III.B4-4(TP-8) III.B4-8(TP-5) III.B4-5(TP-11)	Support members; welds; bolted connections; support anchorage to building structure	Aluminum; galvanized steel; stainless steel	Air – indoor, uncontrolled	None	None	No
III.B4.TP-3	III.B4-6(TP-3)	Support members; welds; bolted connections; support anchorage to building structure	Galvanized steel; aluminum	Air with borated water leakage	Loss of material due to boric acid corrosion	Chapter XI.M10, “Boric Acid Corrosion”	No
III.B4.TP-6	III.B4-7(TP-6)	Support members; welds; bolted connections; support anchorage to	Galvanized steel; aluminum; stainless steel	Air – outdoor	Loss of material due to pitting and crevice corrosion	Chapter XI.S6, "Structures Monitoring"	No

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
		building structure					
III.B4.TP-4	III.B4-9(TP-4)	Support members; welds; bolted connections; support anchorage to building structure	Stainless steel	Air with borated water leakage	None	None	No
III.B4.TP-43	III.B4-10(T-30)	Support members; welds; bolted connections; support anchorage to building structure	Steel	Air – indoor, uncontrolled or Air – outdoor	Loss of material due to general and pitting corrosion	Chapter XI.S6, "Structures Monitoring"	No
III.B4.T-25	III.B4-11(T-25)	Support members; welds; bolted connections; support anchorage to building structure	Steel	Air with borated water leakage	Loss of material due to boric acid corrosion	Chapter XI.M10, “Boric Acid Corrosion”	No
III.B4.TP-44	III.B4-12(T-31)	Vibration isolation elements	Non-metallic (e.g., rubber)	Air – indoor, uncontrolled or Air – outdoor	Reduction or loss of isolation function due to radiation hardening, temperature,	Chapter XI.S3, “ASME Section XI, Subsection IWF”	No

III STRUCTURES AND COMPONENT SUPPORTS
B4 Supports for Emergency Diesel Generator (EDG), HVAC System Components, and Other Miscellaneous Mechanical Equipment

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
					humidity, sustained vibratory loading		

B5. SUPPORTS FOR PLATFORMS, PIPE WHIP RESTRAINTS, JET IMPINGEMENT SHIELDS, MASONRY WALLS, AND OTHER MISCELLANEOUS STRUCTURES

Systems, Structures, and Components

This section addresses supports and anchorage for platforms, pipe whip restraints, jet impingement shields, masonry walls, and other miscellaneous structures. 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
B5 Supports for Platforms, Pipe Whip Restraints, Jet Impingement Shields, Masonry Walls, and Other Miscellaneous Structures

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
III.B5.TP-42	III.B5-1(T-29)	Building concrete at locations of expansion and grouted anchors; grout pads for support base plates	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"	No
III.B5.TP-300		High-strength structural bolting	Low-alloy steel, actual measured yield strength ≥ 150 ksi (1,034 MPa)	Air – indoor, uncontrolled or Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.S6, "Structures Monitoring“ Note: ASTM A 325, F 1852, and ASTM A 490 bolts used in civil structures have not shown to be prone to SCC. SCC potential need not be evaluated for these bolts.	No
III.B5.TP-261		Structural bolting	Any	Any environment	Loss of preload due to self-loosening	Chapter XI.S6, "Structures Monitoring"	No
III.B5.TP-248		Structural bolting	Steel	Air – indoor, uncontrolled	Loss of material due to general, pitting and crevice corrosion	Chapter XI.S6, "Structures Monitoring"	No
III.B5.TP-274		Structural bolting	Steel; galvanized steel	Air – outdoor	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S6, "Structures Monitoring"	No
III.B5.TP-8	III.B5-2(TP-8) III.B5-5(TP-5)	Support members; welds; bolted connections;	Aluminum; galvanized steel; stainless	Air – indoor, uncontrolled	None	None	No

III STRUCTURES AND COMPONENT SUPPORTS
B5 Supports for Platforms, Pipe Whip Restraints, Jet Impingement Shields, Masonry Walls, and Other Miscellaneous Structures

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
	III.B5-3(TP-11)	support anchorage to building structure	steel				
III.B5.TP-3	III.B5-4(TP-3)	Support members; welds; bolted connections; support anchorage to building structure	Galvanized steel; aluminum	Air with borated water leakage	Loss of material due to boric acid corrosion	Chapter XI.M10, “Boric Acid Corrosion”	No
III.B5.TP-4	III.B5-6(TP-4)	Support members; welds; bolted connections; support anchorage to building structure	Stainless steel	Air with borated water leakage	None	None	No
III.B5.TP-43	III.B5-7(T-30)	Support members; welds; bolted connections; support anchorage to building structure	Steel	Air – indoor, uncontrolled or Air – outdoor	Loss of material due to general and pitting corrosion	Chapter XI.S6, "Structures Monitoring"	No
III.B5.T-25	III.B5-8(T-25)	Support members; welds; bolted connections; support	Steel	Air with borated water leakage	Loss of material due to boric acid corrosion	Chapter XI.M10, “Boric Acid Corrosion”	No

III STRUCTURES AND COMPONENT SUPPORTS
B5 Supports for Platforms, Pipe Whip Restraints, Jet Impingement Shields, Masonry Walls, and Other Miscellaneous Structures

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
		anchorage to building structure					

CHAPTER IV

REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

MAJOR PLANT SECTIONS

A1. Reactor Vessel (Boiling Water Reactor)

A2. Reactor Vessel (Pressurized Water Reactor)

B1. Reactor Vessel Internals (Boiling Water Reactor)

B2. Reactor Vessel Internals (PWR) - Westinghouse

B3. Reactor Vessel Internals (PWR) - Combustion Engineering

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

C1. Reactor Coolant Pressure Boundary (Boiling Water Reactor)

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

D1. Steam Generator (Recirculating)

D2. Steam Generator (Once-Through)

E. Common Miscellaneous Material/Environment Combinations

A1. REACTOR VESSEL (BOILING WATER REACTOR)

Systems, Structures, and Components

This section addresses the boiling water reactor (BWR) pressure vessel and consists of the vessel shell and flanges, attachment welds, top and bottom heads, nozzles (including safe ends) for the reactor coolant recirculating system and connected systems (such as high and low pressure core spray, high and low pressure coolant injection, main steam, and feedwater systems), penetrations for control rod drive (CRD) stub tubes, instrumentation, standby liquid control, flux monitor, drain lines, and control rod drive mechanism housings. The support skirt and attachment welds for vessel supports are also included in the following table for the BWR vessel. Based on Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” all structures and components that comprise the reactor vessel are governed by Group A Quality Standards.

Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in IV.E.

System Interfaces

The systems that interface with the reactor vessel include the reactor vessel internals (IV.B1), the reactor coolant pressure boundary (IV.C1), the emergency core cooling system (V.D2), and the standby liquid control system (VII.E2).

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.A1.R-68	IV.A1-1(R-68)	Nozzle safe ends and 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 due to stress corrosion cracking, intergranular stress corrosion cracking	Chapter XI.M7, “BWR Stress Corrosion Cracking,” and Chapter XI.M2, “Water Chemistry”	No
IV.A1.R-66	IV.A1-2(R-66)	Nozzles: control rod drive return line	Steel (with or without stainless steel cladding)	Reactor coolant	Cracking due to cyclic loading	Chapter XI.M6, “BWR Control Rod Drive Return Line Nozzle”	No
IV.A1.R-65	IV.A1-3(R-65)	Nozzles: feedwater	Steel (with or without stainless steel cladding)	Reactor coolant	Cracking due to cyclic loading	Chapter XI.M5, “BWR Feedwater Nozzle”	No

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.A1.R-67	IV.A1-4(R-67)	Nozzles: low-pressure coolant injection or RHR injection mode	Steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	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 1E17 n/cm^2 (E >1 MeV) at the end of the period of extended operation. In accordance with approved BWRVIP-74, the TLAA is to evaluate the impact of neutron embrittlement on: (a) the adjusted reference temperature values used for calculation of the plant's pressure-temperature limits, (b) the need for inservice inspection of circumferential welds, and (c) the Charpy upper shelf energy or the equivalent margins analyses performed in accordance with 10 CFR Part 50, Appendix G The applicant may choose to demonstrate that the materials of the nozzles are not controlling for the TLAA evaluations. See the Standard Review Plan, Section 4.2 "Reactor Vessel Neutron Embrittlement" for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes, TLAA

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.A1.RP-369	IV.A1-5(R-69)	Penetrations: control rod drive stub tubes; in core monitor housings; jet pump instrument; standby liquid control; flux monitor	Stainless steel; nickel alloy	Reactor coolant	Cracking due to stress corrosion cracking, intergranular stress corrosion cracking, cyclic loading	Chapter XI.M8, “BWR Penetrations,” and Chapter XI.M2, “Water Chemistry”	No
IV.A1.RP-371	IV.A1-5(R-69)	Penetrations: drain line	Stainless steel; nickel alloy	Reactor coolant	Cracking due to stress corrosion cracking, intergranular stress corrosion cracking, cyclic loading	Chapter XI.M1, “ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD,” and Chapter XI.M2, “Water Chemistry”	No
IV.A1.R-70	IV.A1-6(R-70)	Pressure vessel support skirt and attachment welds	Steel	Air – indoor, uncontrolled	Cumulative fatigue damage due to fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
IV.A1.R-04	IV.A1-7(R-04)	Reactor vessel components: flanges; nozzles; penetrations; safe ends; thermal sleeves; vessel shells, heads and welds	Steel (with or without nickel-alloy or stainless steel cladding); stainless steel; nickel alloy	Reactor coolant	Cumulative fatigue damage due to fatigue	Fatigue is a TLAA evaluated for the period of extended operation, and for Class 1 components environmental effects on fatigue are to be addressed. (See SRP, Sec 4.3 “Metal Fatigue,” for acceptable methods to comply with 10 CFR 54.21(c)(1))	Yes, TLAA

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.A1.RP-157	IV.A1-8(RP-25)	Reactor Vessel: flanges; nozzles; penetrations; safe ends; vessel shells, heads and welds	Steel (with stainless steel or nickel-alloy cladding); stainless steel; nickel alloy	Reactor coolant	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
IV.A1.RP-50	IV.A1-11(R-59)	Top head enclosure (without cladding): top head; nozzles (vent, top head spray or RCIC, and spare)	Steel	Reactor coolant	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
IV.A1.RP-51	IV.A1-9(R-60)	Top head enclosure: closure studs and nuts	High-strength, low-alloy steel	Air with reactor coolant leakage	Cracking due to stress corrosion cracking, intergranular stress corrosion cracking	Chapter XI.M3, “Reactor Head Closure Stud Bolting”	No
IV.A1.RP-201		Top head enclosure: closure studs and nuts	High-strength, low-alloy steel	Air with reactor coolant leakage	Cumulative fatigue damage due to fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.A1.RP-165		Top head enclosure: closure studs and nuts	High-strength, low-alloy steel	Air with reactor coolant leakage	Loss of material due to general, pitting, and crevice corrosion, or wear	Chapter XI.M3, “Reactor Head Closure Stud Bolting”	No
IV.A1.R-61	IV.A1-10(R-61)	Top head enclosure: vessel flange leak detection line	Stainless steel; nickel alloy	Air with reactor coolant leakage (Internal); or reactor coolant	Cracking due to stress corrosion cracking, 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
IV.A1.RP-227	IV.A1-14(R-63)	Vessel shell (including applicable beltline) components: shell; shell plates or forgings; shell welds; nozzle plates or forgings; nozzle welds	Steel (with or without cladding)	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M31, “Reactor Vessel Surveillance”	Yes, plant specific or integrated surveillance program
IV.A1.R-64	IV.A1-12(R-64)	Vessel shell: attachment welds	Stainless steel; nickel alloy	Reactor coolant	Cracking due to stress corrosion cracking, intergranular stress corrosion cracking	Chapter XI.M4, “BWR Vessel ID Attachment Welds,” and Chapter XI.M2, “Water Chemistry”	No

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.A1.R-62	IV.A1-13(R-62)	Vessel shell: intermediate beltline shell; beltline welds	Steel (with or without stainless steel cladding)	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Neutron irradiation embrittlement is a time-dependent aging mechanism evaluated for extended operation for all ferritic materials that have a neutron fluence >1E17 n/cm^2 (E >1 MeV) at the end of the period of extended operation. Aspects may involve a TLAA. In accordance with approved BWRVIP-74, the TLAA evaluates the impact of neutron embrittlement on: (a) adjusted reference temperature values used for calculation of the plant's pressure-temperature limits, (b) need for inservice inspection of circumferential welds, and (c) Charpy upper shelf energy or the equivalent margins analyses performed in accordance with 10 CFR Part 50, Appendix G. Additionally, the applicant is to monitor axial beltline weld embrittlement. One acceptable method is to determine that the mean RTNDT of the axial beltline welds at the end of the extended period of operation is less than the value specified by the staff in its March 7, 2000 letter (ADAMS ML031430372). See the Standard Review Plan, Section 4.2 "Reactor Vessel Neutron Embrittlement" for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes, TLAA

A2. REACTOR VESSEL (PRESSURIZED WATER REACTOR)

Systems, Structures, and Components

This section addresses the pressurized water reactor (PWR) vessel pressure boundary and consists of the vessel shell and flanges, the top closure head and bottom head, the control rod drive (CRD) mechanism housings, nozzles (including safe ends) for reactor coolant inlet and outlet lines and safety injection, and penetrations through either the closure head or bottom head domes for instrumentation and leakage monitoring tubes. Attachments to the vessel such as core support pads, as well as pressure vessel support and attachment welds, are also included in the table. Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all systems, structures, and components that comprise the reactor coolant system are governed by Group A Quality Standards.

Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in IV.E.

System Interfaces

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

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.A2.RP-154	IV.A2-1(RP-13)	Bottom-mounted instrument guide tube (external to bottom head)	Stainless steel	Reactor coolant	Cracking due to stress corrosion cracking	A plant-specific aging management program is to be evaluated	Yes, plant-specific
IV.A2.RP-52	IV.A2-2(R-71)	Closure head: stud assembly	High-strength, low-alloy steel	Air with reactor coolant leakage	Cracking due to stress corrosion cracking	Chapter XI.M3, “Reactor Head Closure Stud Bolting”	No
IV.A2.RP-54	IV.A2-4(R-73)	Closure head: stud assembly	High-strength, low-alloy steel	Air with reactor coolant leakage	Cumulative fatigue damage due to fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
IV.A2.RP-53	IV.A2-3(R-72)	Closure head: stud assembly	High-strength, low-alloy steel	Air with reactor coolant leakage	Loss of material due to general, pitting, and crevice corrosion, or wear	Chapter XI.M3, “Reactor Head Closure Stud Bolting”	No
IV.A2.R-74	IV.A2-5(R-74)	Closure head: vessel flange leak detection line	Stainless steel	Air with reactor coolant leakage (Internal); or reactor coolant	Cracking due to 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
IV.A2.R-80	IV.A2-8(R-80)	Control rod drive head penetration: Flange bolting	Stainless steel	Air (with reactor coolant leakage)	Loss of preload due to thermal effects, gasket creep, and self-loosening	Chapter XI.M18, "Bolting Integrity"	No

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.A2.R-78	IV.A2-6(R-78)	Control rod drive head penetration: flange bolting	Stainless steel	Air with reactor coolant leakage	Cracking due to stress corrosion cracking	Chapter XI.M18, "Bolting Integrity"	No
IV.A2.R-79	IV.A2-7(R-79)	Control rod drive head penetration: flange bolting	Stainless steel	Air with reactor coolant leakage	Loss of material due to wear	Chapter XI.M18, "Bolting Integrity"	No
IV.A2.RP-186	IV.A2-9(R-75)	Control rod drive head penetration: nozzle welds	Nickel alloy	Reactor coolant	Cracking due to 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,” and Chapter XI.M11B, “Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in RCPB Components (PWRs Only)"	No
IV.A2.R-77	IV.A2-10(R-77)	Control rod drive head penetration: pressure housing	Cast austenitic stainless steel	Reactor coolant >250°C (>482°F)	Loss of fracture toughness due to thermal aging embrittlement	Chapter XI.M12, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)"	No
IV.A2.RP-55	IV.A2-11(R-76)	Control rod drive head penetration: pressure housing	Stainless steel; nickel alloy	Reactor coolant	Cracking due to 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”	No

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.A2.RP-57	IV.A2-12(R-88)	Core support pads; core guide lugs	Nickel alloy	Reactor coolant	Cracking due to 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”	No
IV.A2.R-17	IV.A2-13(R-17)	External surfaces	Steel	Air with borated water leakage	Loss of material due to boric acid corrosion	Chapter XI.M10, “Boric Acid Corrosion”	No
IV.A2.RP-379	IV.A2-13(R-17)	External surfaces: reactor vessel top head and bottom head	Steel	Air with borated water leakage	Loss of material due to boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion," and Chapter XI.M11B, "Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in RCPB Components (PWRs Only)"	No
IV.A2.RP-28	IV.A2-14(RP-28)	Flanges; nozzles; penetrations; pressure housings; safe ends; vessel shells, heads welds	Steel (with stainless steel or nickel-alloy cladding); stainless steel; nickel alloy	Reactor coolant	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry”	No

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.A2.RP-234	IV.A2-15(R-83)	Nozzle safe ends and welds: inlet; outlet; safety injection	Stainless steel; nickel alloy welds and/or buttering	Reactor coolant	Cracking due to 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,” and Chapter XI.M11B, “Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in RCPB Components (PWRs Only)" for nickel alloy components	No
IV.A2.RP-228	IV.A2-17(R-82)	Nozzles: inlet; outlet; safety injection	Steel (with or without cladding)	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M31, “Reactor Vessel Surveillance”	Yes, plant specific or integrated surveillance program

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.A2.R-81	IV.A2-16(R-81)	Nozzles: inlet; outlet; safety injection	Steel (with stainless steel or nickel-alloy cladding)	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Neutron irradiation embrittlement is a TLAA evaluated for extended operation for all ferritic materials with a neutron fluence greater than 1E17 n/cm^2 (E >1 MeV) at the end of the period of extended operation. The TLAA is to evaluate the impact of neutron embrittlement on: (a) the RTPTS value based on the requirements in 10 CFR 50.61, (b) the adjusted reference temperature values used for calculation of the plant's pressure-temperature limits, and (c) the Charpy upper shelf energy or the equivalent margins analyses performed in accordance with 10 CFR Part 50, Appendix G requirements. The applicant may choose to demonstrate that the materials in the inlet, outlet, and safety injection nozzles are not controlling for the TLAA evaluations.	Yes, TLAA
IV.A2.R-90	IV.A2-18(R-90)	Penetrations: head vent pipe (top head); instrument tubes (top head)	Nickel alloy	Reactor coolant	Cracking due to 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," and Chapter XI.M11B, "Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in RCPB Components (PWRs Only)"	No

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.A2.RP-59	IV.A2-19(R-89)	Penetrations: instrument tubes (bottom head)	Nickel alloy	Reactor coolant	Cracking due to 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,” and Chapter XI.M11B, “Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in RCPB Components (PWRs Only)"	No
IV.A2.R-70	IV.A2-20(R-70)	Pressure vessel support skirt and attachment welds	Steel	Air – indoor, uncontrolled	Cumulative fatigue damage due to fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
IV.A2.R-219	IV.A2-21(R-219)	Reactor vessel components: flanges; nozzles; penetrations; pressure housings; safe ends; thermal sleeves; vessel shells, heads and welds	Steel (with or without nickel-alloy or stainless steel cladding); stainless steel; nickel alloy	Reactor coolant	Cumulative fatigue damage due to fatigue	Fatigue is a TLAA evaluated for the period of extended operation, and for Class 1 components environmental effects on fatigue are to be addressed. (See SRP, Sec 4.3 “Metal Fatigue,” for acceptable methods to comply with 10 CFR 54.21(c)(1))	Yes, TLAA

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.A2.R-85	IV.A2-22(R-85)	Vessel shell: upper shell; intermediate shell; lower shell (including beltline welds)	SA508-Cl 2 forgings clad (with stainless steel) using a high-heat-input welding process	Reactor coolant	Crack growth due to cyclic loading	Growth of intergranular separations (underclad cracks) in low-alloy steel forging 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 an underclad flaw is in accordance 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).	Yes, TLAA
IV.A2.RP-229	IV.A2-24(R-86)	Vessel shell: upper shell; intermediate shell; lower shell (including beltline welds)	Steel (with or without cladding)	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M31, "Reactor Vessel Surveillance"	Yes, plant specific or integrated surveillance program

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.A2.R-84	IV.A2-23(R-84)	Vessel shell: upper shell; intermediate shell; lower shell (including beltline welds)	Steel (with stainless steel or nickel-alloy cladding)	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Neutron irradiation embrittlement is a TLAA evaluated for extended operation for all ferritic materials with a neutron fluence greater than 1E17 n/cm^2 (E >1 MeV) at the end of the period of extended operation. The TLAA is to evaluate the impact of neutron embrittlement on: (a) the RTPTS value based on the requirements in 10 CFR 50.61, (b) the adjusted reference temperature values used for calculation of the plant's pressure-temperature limits, and (c) the Charpy upper shelf energy or the equivalent margins analyses performed in accordance with 10 CFR Part 50, Appendix G requirements. See the Standard Review Plan, Section 4.2 "Reactor Vessel Neutron Embrittlement" for acceptable methods for meeting the requirements of 10 CFR 54.21(c).	Yes, TLAA
IV.A2.R-87	IV.A2-25(R-87)	Vessel shell: vessel flange	Steel	Reactor coolant	Loss of material due to wear	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD" for Class 1 components	No

B1. REACTOR VESSEL INTERNALS (BOILING WATER REACTOR)

Systems, Structures, and Components

This section addresses the boiling water reactor (BWR) vessel internals and consists of the core shroud (including repairs) and core plate, the top guide, feedwater spargers, core spray lines and spargers, jet pump assemblies, fuel supports and control rod drive (CRD), and instrument housings, such as the intermediate range monitor (IRM) dry tubes, the low power range monitor (LPRM) dry tubes, and the source range monitor (SRM) dry tubes. Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all structures and components that comprise the reactor vessel are governed by Group A or B Quality Standards.

Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in IV.E.

System Interfaces

The systems that interface with the reactor vessel internals include the reactor pressure vessel (IV.A1) and the reactor coolant pressure boundary (IV.C1).

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B1.R-92	IV.B1-1(R-92)	Core shroud (including repairs) and core plate: core shroud (upper, central, lower)	Stainless steel	Reactor coolant and neutron flux	Cracking due to 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"	No
IV.B1.R-96	IV.B1-2(R-96)	Core shroud (including repairs) and core plate: shroud support structure (shroud support cylinder, shroud support plate, shroud support legs)	Nickel alloy	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking, intergranular stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M9, "BWR Vessel Internals" for shroud support, and Chapter XI.M2, "Water Chemistry"	No
IV.B1.R-95	IV.B1-4(R-95)	Core shroud and core plate: access hole cover (mechanical)	Nickel alloy	Reactor coolant and neutron flux	Cracking due to 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"	No

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B1.R-94	IV.B1-5(R-94)	Core shroud and core plate: access hole cover (welded)	Nickel alloy	Reactor coolant and neutron flux	Cracking due to 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" 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 cover welds.	No
IV.B1.R-93	IV.B1-6(R-93)	Core shroud and core plate: core plate and plate bolts (used in early BWRs)	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking, intergranular stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M9, "BWR Vessel Internals" for core plate, and Chapter XI.M2, "Water Chemistry"	No
IV.B1.R-97	IV.B1-3(R-97)	Core shroud and core plate: LPCI coupling	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking, intergranular stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M9, "BWR Vessel Internals" for the LPCI coupling, and Chapter XI.M2, "Water Chemistry"	No

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B1.R-99	IV.B1-7(R-99)	Core spray lines and spargers: core spray lines (headers); spray rings; spray nozzles; thermal sleeves	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking, intergranular stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M9, “BWR Vessel Internals” for core spray internals, and Chapter XI.M2, “Water Chemistry”	No
IV.B1.R-104	IV.B1-8(R-104)	Fuel supports and control rod drive assemblies: control rod drive housing	Stainless steel	Reactor coolant	Cracking due to stress corrosion cracking, intergranular stress corrosion cracking	Chapter XI.M9, “BWR Vessel Internals” for lower plenum, and Chapter XI.M2, “Water Chemistry”	No
IV.B1.RP-220	IV.B1-9(R-103)	Fuel supports and control rod drive assemblies: orificed fuel support	Cast austenitic stainless steel	Reactor coolant >250°C (>482°F) and neutron flux	Loss of fracture toughness due to thermal aging, neutron irradiation embrittlement	Chapter XI.M9, "BWR Vessel Internals"	No
IV.B1.R-105	IV.B1-10(R-105)	Instrumentation: Intermediate range monitor (IRM) dry tubes; source range monitor (SRM) dry tubes; incore neutron flux monitor guide tubes	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking, intergranular stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M9, “BWR Vessel Internals” for lower plenum, and Chapter XI.M2, “Water Chemistry”	No

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B1.RP-219	IV.B1-11(R-101)	Jet pump assemblies: castings	Cast austenitic stainless steel	Reactor coolant >250°C (>482°F) and neutron flux	Loss of fracture toughness due to thermal aging, neutron irradiation embrittlement	Chapter XI.M9, "BWR Vessel Internals"	No
IV.B1.R-100	IV.B1-13(R-100)	Jet pump assemblies: thermal sleeve; inlet header; riser brace arm; holddown beams; inlet elbow; mixing assembly; diffuser castings	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking, intergranular stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M9, “BWR Vessel Internals” for jet pump assembly, and Chapter XI.M2, “Water Chemistry”	No
IV.B1.R-53	IV.B1-14(R-53)	Reactor vessel internal components	Stainless steel; nickel alloy	Reactor coolant	Cumulative fatigue damage due to fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B1.RP-182		Reactor vessel internals components	PH martensitic stainless steel (17-4PH and 15-5PH); martensitic stainless steel (SS 403, 410, 431, etc.)	Reactor coolant >250°C (>482°F) and neutron flux	Loss of fracture toughness due to thermal aging, neutron irradiation embrittlement	Chapter XI.M9, "BWR Vessel Internals"	No
IV.B1.RP-26	IV.B1-15(RP-26)	Reactor vessel internals components	Stainless steel; nickel alloy	Reactor coolant	Loss of material due to 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”	No
IV.B1.RP-381		Reactor vessel internals components	X-750 alloy	Reactor coolant and neutron flux	Cracking due to intergranular stress corrosion cracking	Chapter XI.M9, “BWR Vessel Internals” for core plate, and Chapter XI.M2, “Water Chemistry”	No
IV.B1.RP-200		Reactor vessel internals components	X-750 alloy	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M9, "BWR Vessel Internals"	No
IV.B1.RP-377		Reactor vessel internals components: Jet pump wedge surface	Stainless steel	Reactor coolant	Loss of material due to wear	Chapter XI.M9, "BWR Vessel Internals"	No

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B1.RP-155	IV.B1-16(RP-18)	Steam dryers	Stainless steel	Reactor coolant	Cracking due to flow-induced vibration	Chapter XI.M9, "BWR Vessel Internals" for steam dryer	No
IV.B1.R-98	IV.B1-17(R-98)	Top guide	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking, intergranular stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M9, “BWR Vessel Internals” for top guide, and Chapter XI.M2, “Water Chemistry”	No

B2. REACTOR VESSEL INTERNALS (PWR) - WESTINGHOUSE

Systems, Structures, and Components

This section addresses the Westinghouse pressurized water reactor (PWR) vessel internals and consists of the upper internals assembly, the control rod guide tube assemblies, the core barrel, the baffle/former assembly, the lower internal assembly, and the instrumentation support structures. Based on Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” all structures and components that comprise the reactor vessel are governed by Group A or B Quality Standards.

Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in IV.E.

System Interfaces

The systems that interface with the reactor vessel internals include the reactor pressure vessel (IV.A2).

Inspection Plan

An applicant will submit an inspection plan for reactor internals to the NRC for review and approval with the application for license renewal in accordance with Chapter XI.M16A, “PWR Vessel Internals.”

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
IV.B2.RP-300	IV.B2-33(R-108)	Alignment and interfacing components: internals hold down spring	Stainless steel	Reactor coolant and neutron flux	Loss of preload due to thermal and irradiation enhanced stress relaxation; loss of material due to wear	Chapter XI.M16A, "PWR Vessel Internals" Primary components (identified in the "Structure and Components" column) no Expansion components	No
IV.B2.RP-301	IV.B2-40(R-112)	Alignment and interfacing components: upper core plate alignment pins	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking	'Chapter XI.M2, "Water Chemistry," and Chapter XI.M16A, "PWR Vessel Internals" Existing Program components (identified in the "Structure and Components" column) no Expansion components	No
IV.B2.RP-299	IV.B2-34(R-115)	Alignment and interfacing components: upper core plate alignment pins	Stainless steel	Reactor coolant and neutron flux	Loss of material due to wear	Chapter XI.M16A, "PWR Vessel Internals" Existing Program components (identified in the "Structure and Components" column) no Expansion components	No
IV.B2.RP-271	IV.B2-10(R-125)	Baffle-to-former assembly: accessible baffle-to-former bolts	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress corrosion cracking and fatigue	Chapter XI.M2, "Water Chemistry," and Chapter XI.M16A, "PWR Vessel Internals" Primary components (identified in the "Structure and Components" column) (for Expansion components see AMR Items IV.B2.RP-273 and IV.B2.RP-286)	No

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
IV.B2.RP-272	IV.B2-6(R-128)	Baffle-to-former assembly: accessible baffle-to-former bolts	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement; change in dimension due to void swelling; loss of preload due to thermal and irradiation enhanced stress relaxation	Chapter XI.M16A, “PWR Vessel Internals” Primary components (identified in the "Structure and Components" column) (for Expansion components see AMR Items IV.B2.RP-274 and IV.B2.RP-287)	No
IV.B2.RP-270	IV.B2-1(R-124)	Baffle-to-former assembly: baffle and former plates	Stainless steel	Reactor coolant and neutron flux	Change in dimension due to void swelling	Chapter XI.M16A, “PWR Vessel Internals” Primary components (identified in the "Structure and Components" column) no Expansion components	No
IV.B2.RP-275	IV.B2-6(R-128)	Baffle-to-former assembly: baffle-edge bolts (all plants with baffle-edge bolts)	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress corrosion cracking and fatigue	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” Primary components (identified in the "Structure and Components" column) no Expansion components	No

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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
IV.B2.RP-354		Baffle-to-former assembly: baffle-edge bolts (all plants with baffle-edge bolts)	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement; change in dimension due to void swelling; loss of preload due to thermal and irradiation enhanced stress relaxation	Chapter XI.M16A, "PWR Vessel Internals" Primary components (identified in the "Structure and Components" column) no Expansion components	No
IV.B2.RP-273	IV.B2-10(R-125)	Baffle-to-former assembly: barrel-to-former bolts	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress corrosion cracking and fatigue	Chapter XI.M2, "Water Chemistry," and Chapter XI.M16A, "PWR Vessel Internals" Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B2.RP-271)	No

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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
IV.B2.RP-274	IV.B2-6(R-128)	Baffle-to-former assembly: barrel-to-former bolts	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement; change in dimension due to void swelling; loss of preload due to thermal and irradiation enhanced stress relaxation	Chapter XI.M16A, “PWR Vessel Internals” Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B2.RP-272)	No
IV.B2.RP-284	IV.B2-12(R-143)	Bottom mounted instrument system: flux thimble tubes	Stainless steel (with or without chrome plating)	Reactor coolant and neutron flux	Loss of material due to wear	Chapter XI.M16A, “PWR Vessel Internals” Existing Program components (identified in the "Structure and Components" column) No expansion components; and Chapter XI.M37, "Flux Thimble Tube Inspection"	No
IV.B2.RP-293	IV.B2-24(R-138)	Bottom-mounted instrumentation system: bottom-mounted instrumentation (BMI) column bodies	Stainless steel	Reactor coolant and neutron flux	Cracking due to fatigue	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B2.RP-298)	No

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2.RP-292	IV.B2-21(R-140)	Bottom-mounted instrumentation system: bottom-mounted instrumentation (BMI) column bodies	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M16A, “PWR Vessel Internals” Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B2.RP-297)	No
IV.B2.RP-296		Control rod guide tube (CRGT) assemblies: CRGT guide plates (cards)	Stainless steel	Reactor coolant and neutron flux	Loss of material due to wear	Chapter XI.M16A, “PWR Vessel Internals” Primary Components (identified in the "Structure and Components" column) (for Expansion components see AMR Line Item IV.B2.RP-386)	No
IV.B2.RP-298	IV.B2-28(R-118)	Control rod guide tube (CRGT) assemblies: CRGT lower flange welds (accessible)	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking and fatigue	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” Primary components (identified in the "Structure and Components" column) (for Expansion components see AMR Items IV.B2.RP-291 and IV.B2.RP-293)	No
IV.B2.RP-297		Control rod guide tube (CRGT) assemblies: CRGT lower flange welds (accessible)	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to thermal aging and neutron irradiation embrittlement	Chapter XI.M16A, “PWR Vessel Internals” Primary components (identified in the "Structure and Components" column) (for Expansion components see AMR Items IV.B2.RP-290 and IV.B2.RP-292)	No

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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
IV.B2.RP-386		Control rod guide tube (CRGT) assemblies: C-tubes and sheaths	Stainless steel	Reactor coolant and neutron flux	Loss of material due to wear	Chapter XI.M16A, “PWR Vessel Internals” Expansion components (identified in the "Structure and Components" column) are only the components associated with a primary component that exceeded the acceptance limit. (for Primary components see AMR Item IV.B2.RP-296)	No
IV.B2.RP-355		Control rod guide tube assemblies: guide tube support pins	Nickel alloy	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking and fatigue	A plant-specific aging management program is to be evaluated	Yes, plant-specific
IV.B2.RP-356		Control rod guide tube assemblies: guide tube support pins	Nickel alloy	Reactor coolant and neutron flux	Loss of material due to wear	A plant-specific aging management program is to be evaluated	Yes, plant-specific
IV.B2.RP-387		Core barrel assembly: core barrel axial welds	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking, and irradiation-assisted stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B2.RP-276)	No

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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
IV.B2.RP-388		Core barrel assembly: core barrel axial welds	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M16A, “PWR Vessel Internals” Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B2.RP-276)	No
IV.B2.RP-282	IV.B2-8(R-120)	Core barrel assembly: core barrel flange	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking and fatigue	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B2.RP-276)	No
IV.B2.RP-345		Core barrel assembly: core barrel flange	Stainless steel	Reactor coolant and neutron flux	Loss of material due to wear	Chapter XI.M16A, “PWR Vessel Internals” Existing Program components (identified in the "Structure and Components" column) no Expansion components	No
IV.B2.RP-278	IV.B2-8(R-120)	Core barrel assembly: core barrel outlet nozzle welds	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking and fatigue	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” Expansion component (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B2.RP-276)	No

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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
IV.B2.RP-280	IV.B2-8(R-120)	Core barrel assembly: lower core barrel flange weld	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking and irradiation-assisted stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” Expansion component (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B2.RP-276)	No
IV.B2.RP-281	IV.B2-9(R-122)	Core barrel assembly: lower core barrel flange weld	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M16A, “PWR Vessel Internals” Expansion Components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B2.RP-276)	No
IV.B2.RP-276	IV.B2-8(R-120)	Core barrel assembly: upper core barrel flange weld	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking and irradiation-assisted stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” Primary components (identified in the "Structure and Components" column) (for Expansion components see AMR Items IV.B2.RP-278, IV.B2.RP-280, IV.B2.RP-282, IV.B2.RP-294, IV.B2.RP-295,IV.B2.RP-281, IV.B2.RP-387, and IV.B2.RP-388)	No
IV.B2.RP-285	IV.B2-14(R-137)	Lower internals assembly: clevis insert bolts	Nickel alloy	Reactor coolant and neutron flux	Loss of material due to wear	Chapter XI.M16A, “PWR Vessel Internals” Existing Program components (identified in the "Structure and Components" column) no Expansion components	No

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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
IV.B2.RP-289	IV.B2-20(R-130)	Lower internals assembly: lower core plate and extra-long (XL) lower core plate	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress corrosion cracking, and fatigue	'Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” Existing Program components (identified in the "Structure and Components" column) no Expansion components	No
IV.B2.RP-288	IV.B2-18(R-132)	Lower internals assembly: lower core plate and extra-long (XL) lower core plate	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement; loss of material due to wear	Chapter XI.M16A, “PWR Vessel Internals” Existing Program components (identified in the "Structure and Components" column) no Expansion components	No
IV.B2.RP-291	IV.B2-24(R-138)	Lower support assembly: lower support column bodies (cast)	Cast austenitic stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B2.RP-298)	No
IV.B2.RP-290	IV.B2-21(R-140)	Lower support assembly: lower support column bodies (cast)	Cast austenitic stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to thermal aging and neutron irradiation embrittlement	Chapter XI.M16A, “PWR Vessel Internals” Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B2.RP-297)	No

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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
IV.B2.RP-294	IV.B2-24(R-138)	Lower support assembly: lower support column bodies (non-cast)	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B2.RP-276)	No
IV.B2.RP-295	IV.B2-22(R-141)	Lower support assembly: lower support column bodies (non-cast)	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M16A, “PWR Vessel Internals” Expansion Components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B2.RP-276)	No
IV.B2.RP-286	IV.B2-16(R-133)	Lower support assembly: lower support column bolts	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress corrosion cracking and fatigue	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B2.RP-271)	No

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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
IV.B2.RP-287	IV.B2-17(R-135)	Lower support assembly: lower support column bolts	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement; loss of preload due to thermal and irradiation enhanced stress relaxation	Chapter XI.M16A, "PWR Vessel Internals" Expansion component (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B2.RP-272)	No
IV.B2.RP-303	IV.B2-31(R-53)	Reactor vessel internal components	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cumulative fatigue damage due to fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
IV.B2.RP-24	IV.B2-32(RP-24)	Reactor vessel internal components	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, "Water Chemistry"	No
IV.B2.RP-268		Reactor vessel internal components (inaccessible locations)	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking, and irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," and Chapter XI.M16A, "PWR Vessel Internals"	Yes, if accessible Primary, Expansion or Existing program components indicate aging effects that need management

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Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2.RP-269		Reactor vessel internal components (inaccessible locations)	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement; change in dimension due to void swelling; loss of preload due to thermal and irradiation enhanced stress relaxation; loss of material due to wear	Chapter XI.M16A, “PWR Vessel Internals”	Yes, if accessible Primary, Expansion or Existing program components indicate aging effects that need management
IV.B2.RP-265		Reactor vessel internal components with no additional measures	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking, and irradiation-assisted stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” Note: Components with no additional measures are not uniquely identified in GALL tables - Components with no additional measures are defined in Section 3.3.1 of MRP-227, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines"	No

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Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B2.RP-267		Reactor vessel internal components with no additional measures	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement; change in dimension due to void swelling; loss of preload due to thermal and irradiation enhanced stress relaxation; loss of material due to wear	Chapter XI.M16A, "PWR Vessel Internals" Note: Components with no additional measures are not uniquely identified in GALL tables - Components with no additional measures are defined in Section 3.3.1 of MRP-227, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines"	No
IV.B2.RP-382	IV.B2-26(R-142)	Reactor vessel internals: core support structure	Stainless steel; nickel alloy; cast austenitic stainless steel	Reactor coolant and neutron flux	Cracking, or Loss of material due to wear	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD"	No
IV.B2.RP-302		Thermal shield assembly: thermal shield flexures	Stainless steel	Reactor coolant and neutron flux	Cracking due to fatigue; loss of material due to wear	Chapter XI.M16A, "PWR Vessel Internals" Primary components (identified in the "Structure and Components" column) no Expansion components	No

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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
IV.B2.RP-346		Upper internals assembly: upper support ring or skirt	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking and fatigue	'Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” Existing Program components (identified in the "Structure and Components" column) no Expansion components	No

B3. REACTOR VESSEL INTERNALS (PWR) - COMBUSTION ENGINEERING

Systems, Structures, and Components

This section addresses the Combustion Engineering pressurized water reactor (PWR) vessel internals and consists of the upper internals assembly, the control element assembly (CEA) shrouds, the core support barrel, the core shroud assembly, and the lower internal assembly. Based on Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” all structures and components that comprise the reactor vessel are governed by Group A or B Quality Standards.

Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in IV.E.

System Interfaces

The systems that interface with the reactor vessel internals include the reactor pressure vessel (IV.A2).

Inspection Plan

An applicant will submit an inspection plan for reactor internals to the NRC for review and approval with the application for license renewal in accordance with Chapter XI.M16A, “PWR Vessel Internals.”

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
IV.B3.RP-312	IV.B3-2(R-149)	Control Element Assembly (CEA): shroud assemblies: instrument guide tubes in peripheral CEA assemblies	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking and fatigue	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” Primary components (identified in the "Structure and Components" column) (for Expansion components see AMR Item IV.B3.RP-313)	No
IV.B3.RP-313		Control Element Assembly (CEA): shroud assemblies: remaining instrument guide tubes in CEA assemblies	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking and fatigue	Chapter XI.M2, "Water Chemistry," and Chapter XI.M16A, "PWR Vessel Internals" Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B3.RP-312)	No
IV.B3.RP-320	IV.B3-9(R-162)	Core shroud assemblies (all plants): guide lugs and guide lug insert bolts	Stainless steel	Reactor coolant and neutron flux	Cracking due to fatigue	'Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” Existing Program components (identified in the "Structure and Components" column) no Expansion components	No

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B3 Reactor Vessel Internals (PWR) - Combustion Engineering

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B3.RP-319	IV.B3-9(R-162)	Core shroud assemblies (all plants): guide lugs and guide lug insert bolts	Stainless steel	Reactor coolant and neutron flux	Loss of material due to wear; Loss of preload due to thermal and irradiation enhanced stress relaxation	Chapter XI.M16A, "PWR Vessel Internals" Existing Program components (identified in the "Structure and Components" column) no Expansion components	No
IV.B3.RP-358		Core shroud assemblies (for bolted core shroud assemblies): (a) shroud plates and (b) former plates	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” Expansion components (identified in the "Structure and Components" column) (for Primary component see AMR Item IV.B3.RP-314)	No
IV.B3.RP-318	IV.B3-8(R-163)	Core shroud assemblies (for bolted core shroud assemblies): (a) shroud plates and (b) former plates	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement; change in dimension due to void swelling	Chapter XI.M16A, "PWR Vessel Internals" Primary components (identified in the "Structure and Components" column) no Expansion components	No

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B3 Reactor Vessel Internals (PWR) - Combustion Engineering

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B3.RP-316	IV.B3-9(R-162)	Core shroud assemblies (for bolted core shroud assemblies): barrel-shroud bolts with neutron exposures greater than 3 dpa	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," and Chapter XI.M16A, "PWR Vessel Internals" Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B3.RP-314)	No
IV.B3.RP-317	IV.B3-7(R-165)	Core shroud assemblies (for bolted core shroud assemblies): barrel-shroud bolts with neutron exposures greater than 3 dpa	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of preload due to thermal and irradiation enhanced stress relaxation; loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M16A, "PWR Vessel Internals" Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B3.RP-315)	No
IV.B3.RP-314	IV.B3-9(R-162)	Core shroud assemblies (for bolted core shroud assemblies): core shroud bolts (accessible)	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress corrosion cracking and fatigue	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” Primary components (identified in the "Structure and Components" column) (for Expansion components see AMR Items IV.B3.RP-316, IV.B3.RP-330, and IV.B3.RP-358)	No

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B3 Reactor Vessel Internals (PWR) - Combustion Engineering

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B3.RP-315	IV.B3-7(R-165)	Core shroud assemblies (for bolted core shroud assemblies): core shroud bolts (accessible)	Stainless steel	Reactor coolant and neutron flux	Loss of preload due to thermal and irradiation enhanced stress relaxation; loss of fracture toughness due to neutron irradiation embrittlement; change in dimension due to void swelling	Chapter XI.M16A, "PWR Vessel Internals," Primary components (identified in the "Structure and Components" column) (for Expansion components see AMR Items IV.B3.RP-317, and IV.B3.RP-331)	No
IV.B3.RP-359		Core shroud assemblies (welded): (shroud plates and (b) former plates	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement; change in dimension due to void swelling	Chapter XI.M16A, "PWR Vessel Internals," Primary components (identified in the "Structure and Components" column) no Expansion components	No

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B3 Reactor Vessel Internals (PWR) - Combustion Engineering

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B3.RP-322		Core shroud assembly (for welded core shrouds in two vertical sections): Core shroud plate-former plate weld (a) The axial and horizontal weld seams at the core shroud re-entrant corners as visible from the core side of the shroud, within six inches of the central flange and horizontal stiffeners, and (b) the horizontal stiffeners in shroud plate-to-former plate weld	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” Primary components (identified in the "Structure and Components" column) (for Expansion components see AMR Item IV.B3.RP-323)	No

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B3 Reactor Vessel Internals (PWR) - Combustion Engineering

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B3.RP-326		Core shroud assembly (for welded core shrouds in two vertical sections): gap between the upper and lower plates	Stainless steel	Reactor coolant and neutron flux	Change in dimension due to void swelling	Chapter XI.M16A, "PWR Vessel Internals" Primary components (identified in the "Structure and Components" column) no Expansion components	No
IV.B3.RP-323		Core shroud assembly (for welded core shrouds in two vertical sections): remaining axial welds in shroud plate-to-former plate	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B3.RP-322)	No

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B3 Reactor Vessel Internals (PWR) - Combustion Engineering

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B3.RP-324		Core shroud assembly (for welded core shrouds with full-height shroud plates): axial weld seams at the core shroud re-entrant corners, at the core mid-plane (+3 feet in height) as visible from the core side of the shroud	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” Primary components (identified in the "Structure and Components" column) (for Expansion components see AMR Item IV.B3.RP-325)	No

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
IV.B3.RP-360		Core shroud assembly (for welded core shrouds with full-height shroud plates): axial weld seams at the core shroud re-entrant corners, at the core mid-plane (+3 feet in height) as visible from the core side of the shroud	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M16A, "PWR Vessel Internals" Primary components (identified in the "Structure and Components" column) (for Expansion components see AMR Item IV.B3.RP-361)	No
IV.B3.RP-325		Core shroud assembly (for welded core shrouds with full-height shroud plates): remaining axial welds, ribs, and rings	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B3.RP-324)	No

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
IV.B3.RP-361		Core shroud assembly (for welded core shrouds with full-height shroud plates): remaining axial welds, ribs, and rings	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M16A, "PWR Vessel Internals" Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B3.RP-360)	No
IV.B3.RP-362		Core support barrel assembly: lower cylinder welds	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M16A, "PWR Vessel Internals" Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B3.RP-327)	No
IV.B3.RP-329	IV.B3-15(R-155)	Core support barrel assembly: lower cylinder welds and remaining core barrel assembly welds	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B3.RP-327)	No
IV.B3.RP-333		Core support barrel assembly: lower flange weld, if fatigue life cannot be demonstrated by TLAA	Stainless steel	Reactor coolant and neutron flux	Cracking due to fatigue	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” Primary components (identified in the "Structure and Components" column) no Expansion components	Yes, evaluate to determine the potential locations and extent of fatigue cracking

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B3.RP-389		Core support barrel assembly: lower flange weld (if fatigue analysis exists)	Stainless steel	Reactor coolant and neutron flux	Cumulative fatigue damage due to fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
IV.B3.RP-328	IV.B3-15(R-155)	Core support barrel assembly: surfaces of the lower core barrel flange weld (accessible surfaces)	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking and fatigue	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” Primary components (identified in the "Structure and Components" column) no Expansion components	No
IV.B3.RP-332	IV.B3-17(R-156)	Core support barrel assembly: upper core barrel flange	Stainless steel	Reactor coolant and neutron flux	Loss of material due to wear	Chapter XI.M16A, “PWR Vessel Internals” Existing Program components (identified in the "Structure and Components" column) no Expansion components	No
IV.B3.RP-327	IV.B3-15(R-155)	Core support barrel assembly: upper core support barrel flange weld (accessible surfaces)	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” Primary components (identified in the "Structure and Components" column) (for Expansion components see AMR Items IV.B3.RP-329, IV.B3.RP-335, IV.B3.RP-362, IV.B3.RP-363, IV.B3.RP-364)	No

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
IV.B3.RP-357		Incore instrumentation (ICI): ICI thimble tubes - lower	Zircaloy-4	Reactor coolant and neutron flux	Loss of material due to wear	A plant-specific aging management program is to be evaluated	Yes, plant-specific
IV.B3.RP-336	IV.B3-22(R-170)	Lower support structure: A286 fuel alignment pins (all plants with core shroud assembled in two vertical sections)	Stainless steel	Reactor coolant and neutron flux	Loss of material due to wear; loss of fracture toughness due to neutron irradiation embrittlement; loss of preload due to thermal and irradiation enhanced stress relaxation	Chapter XI.M16A, “PWR Vessel Internals” Existing Program components (identified in the "Structure and Components" column) no Expansion components	No
IV.B3.RP-334	IV.B3-23(R-167)	Lower support structure: A286 fuel alignment pins (all plants with core shroud assembled with full-height shroud plates)	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress corrosion cracking and fatigue	'Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” Existing Program components (identified in the "Structure and Components" column) no Expansion components	No
IV.B3.RP-364		Lower support structure: core support column	Cast austenitic stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation and thermal embrittlement	Chapter XI.M16A, "PWR Vessel Internals" Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B3RP-327)	No

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**
IV.B3.RP-363		Lower support structure: core support column	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M16A, "PWR Vessel Internals" Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B3RP-327)	No
IV.B3.RP-330	IV.B3-23(R-167)	Lower support structure: core support column bolts	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress corrosion cracking and fatigue	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item 'IV.B3.RP-314)	No
IV.B3.RP-331		Lower support structure: core support column bolts	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M16A, “PWR Vessel Internals” Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item 'IV.B3.RP-315)	No
IV.B3.RP-335	IV.B3-23(R-167)	Lower support structure: core support column welds, applicable to all plants except those assembled with full-height shroud plates	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking, and fatigue	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B3.RP-327)	No

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B3 Reactor Vessel Internals (PWR) - Combustion Engineering

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B3.RP-365		Lower support structure: core support plate	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M16A, "PWR Vessel Internals" Primary component (identified in the "Structure and Components" column) no Expansion components	No
IV.B3.RP-343		Lower support structure: core support plate (applicable to plants with a core support plate), if fatigue life cannot be demonstrated by TLAA	Stainless steel	Reactor coolant and neutron flux	Cracking due to fatigue	Chapter XI.M2, “Water Chemistry”, and Chapter XI.M16A, “PWR Vessel Internals” Primary components (identified in the "Structure and Components" column) no Expansion components	Yes, evaluate to determine the potential locations and extent of fatigue cracking
IV.B3.RP-390		Lower support structure: core support plate (applicable to plants with a core support plate), if fatigue analysis exists	Stainless steel	Reactor coolant and neutron flux	Cumulative fatigue damage due to fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
IV.B3.RP-342		Lower support structure: deep beams (applicable assemblies with full height shroud plates)	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking, and fatigue	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” Primary components (identified in the "Structure and Components" column) no Expansion components	No

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B3 Reactor Vessel Internals (PWR) - Combustion Engineering

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B3.RP-366		Lower support structure: deep beams (applicable assemblies with full height shroud plates)	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M16A, “PWR Vessel Internals” Primary components (identified in the "Structure and Components" column) no Expansion components	No
IV.B3.RP-339	IV.B3-24(R-53)	Reactor vessel internal components	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cumulative fatigue damage due to fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
IV.B3.RP-24	IV.B3-25(RP-24)	Reactor vessel internal components	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry”	No
IV.B3.RP-309		Reactor vessel internal components (inaccessible locations)	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking, and irradiation-assisted stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals”	Yes, if accessible Primary, Expansion or Existing program components indicate aging effects that need management

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B3 Reactor Vessel Internals (PWR) - Combustion Engineering

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B3.RP-311		Reactor vessel internal components (inaccessible locations)	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement; change in dimension due to void swelling; loss of preload due to thermal and irradiation enhanced stress relaxation; loss of material due to wear	Chapter XI.M16A, “PWR Vessel Internals”	Yes, if accessible Primary, Expansion or Existing program components indicate aging effects that need management
IV.B3.RP-306		Reactor vessel internal components with no additional measures	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking, and irradiation-assisted stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” Note: Components with no additional measures are not uniquely identified in GALL tables - Components with no additional measures are defined in Section 3.3.1 of MRP-227, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines"	No

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
IV.B3.RP-307		Reactor vessel internal components with no additional measures	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement; change in dimension due to void swelling; loss of preload due to thermal and irradiation enhanced stress relaxation; loss of material due to wear	Chapter XI.M16A, “PWR Vessel Internals” Note: Components with no additional measures are not uniquely identified in GALL tables - Components with no additional measures are defined in Section 3.3.1 of MRP-227, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines"	No
IV.B3.RP-382	IV.B3-22(R-170)	Reactor vessel internals: core support structure	Stainless steel; nickel alloy; cast austenitic stainless steel	Reactor coolant and neutron flux	Cracking, or Loss of material due to wear	Chapter XI.M1, “ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD”	No

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B3 Reactor Vessel Internals (PWR) - Combustion Engineering

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B3.RP-338		Upper internals assembly: fuel alignment plate (applicable to plants with core shrouds assembled with full height shroud plates), if fatigue life cannot be demonstrated by TLAA	Stainless steel	Reactor coolant and neutron flux	Cracking due to fatigue	'Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” Primary components (identified in the "Structure and Components" column) no Expansion components	Yes, evaluate to determine the potential locations and extent of fatigue cracking
IV.B3.RP-391		Upper internals assembly: fuel alignment plate (applicable to plants with core shrouds assembled with full height shroud plates), if fatigue analysis exists	Stainless steel	Reactor coolant and neutron flux	Cumulative fatigue damage due to fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA

B4. REACTOR VESSEL INTERNALS (PWR) - BABCOCK AND WILCOX

Systems, Structures, and Components

This section addresses the Babcock and Wilcox pressurized water reactor (PWR) vessel internals and consists of the plenum cover and plenum cylinder, the upper grid assembly, the control rod guide tube (CRGT) assembly, the core support shield assembly, the core barrel assembly, the lower grid assembly, and the flow distributor assembly. Based on Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” all structures and components that comprise the reactor vessel are governed by Group A or B Quality Standards.

Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in IV.E.

System Interfaces

The systems that interface with the reactor vessel internals include the reactor pressure vessel (IV.A2).

Inspection Plan

An applicant will submit an inspection plan for reactor internals to the NRC for review and approval with the application for license renewal in accordance with Chapter XI.M16A, “PWR Vessel Internals.”

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B4 Reactor Vessel Internals (PWR) – Babcock & Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B4.RP-242	IV.B4-4(R-183)	Control rod guide tube (CRGT) assembly: accessible surfaces at four screw locations (every 90 degrees) for CRGT spacer castings	Cast austenitic stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to thermal aging embrittlement	Chapter XI.M16A, "PWR Vessel Internals" Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Items IV.B4.RP-253 and IV.B4.RP-258)	No
IV.B4.RP-245	IV.B4-13(R-194)	Core barrel assembly: (a) upper thermal shield bolts; (b) surveillance specimen holder tube bolts (Davis-Besse, only); (c) surveillance specimen tube holder studs, and nuts (Crystal River Unit 3, only)	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking	Chapter XI.M2, "Water Chemistry," and Chapter XI.M16A, "PWR Vessel Internals" Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Items IV.B4.RP-247 and IV.B4.RP-248)	No

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B4 Reactor Vessel Internals (PWR) – Babcock & Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B4.RP-247	IV.B4-13(R-194)	Core barrel assembly: accessible lower core barrel (LCB) bolts and locking devices	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking	Chapter XI.M2, "Water Chemistry," and Chapter XI.M16A, "PWR Vessel Internals" Primary components (identified in the "Structure and Components" column) (for Expansion components see AMR Items IV.B4.RP-245, IV.B4.RP-246, IV.B4.RP-254, and IV.B4.RP-256)	No
IV.B4.RP-249	IV.B4-12(R-196)	Core barrel assembly: baffle plate accessible surfaces within one inch around each baffle plate flow and bolt hole	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M16A, "PWR Vessel Internals" Primary components (identified in the "Structure and Components" column) (for Expansion components see AMR Item IV.B4.RP-250)	No
IV.B4.RP-241	IV.B4-7(R-125)	Core barrel assembly: baffle/former assembly: (a) accessible baffle-to-former bolts and screws; (b) accessible locking devices (including welds) of baffle-to-former bolts	Stainless steel	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," and Chapter XI.M16A, "PWR Vessel Internals" Primary Components (identified in the "Structure and Components" column) (for Expansion components see AMR Items IV.B4.RP-244 and IV.B4.RP-375)	No

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B4.RP-240	IV.B4-1(R-128)	Core barrel assembly: baffle/former assembly: (a) accessible baffle-to-former bolts and screws; (b) accessible locking devices (including welds) of baffle-to-former bolts	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement; loss of preload due to thermal and irradiation enhanced stress relaxation; loss of material due to wear	Chapter XI.M16A, “PWR Vessel Internals.” Primary components (identified in the "Structure and Components" column) (for Expansion components see AMR Item IV.B4.RP-243.)	No
IV.B4.RP-250	IV.B4-12(R-196)	Core barrel assembly: core barrel cylinder (including vertical and circumferential seam welds); former plates	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M16A, “PWR Vessel Internals” Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B4.RP-249)	No
IV.B4.RP-375		Core barrel assembly: internal baffle-to-baffle bolts	Stainless steel	Reactor coolant and neutron flux	Cracking due to fatigue	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B4.RP-241)	No

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B4.RP-244	IV.B4-7(R-125)	Core barrel assembly; (a) external baffle-to-baffle bolts; (b) core barrel-to-former bolts; (c) locking devices (including welds) of external baffle-to-baffle bolts and core barrel-to-former bolts	Stainless steel	Reactor coolant and neutron flux	Cracking due to irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry," and Chapter XI.M16A, "PWR Vessel Internals" Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B4.RP-241)	No
IV.B4.RP-243	IV.B4-1(R-128)	Core barrel assembly; (a) external baffle-to-baffle bolts; (b) core barrel-to-former bolts; (c) locking devices (including welds) of external baffle-to-baffle bolts and core barrel-to-former bolts; (d) internal baffle-to-baffle bolts	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement; loss of preload due to thermal and irradiation enhanced stress relaxation; loss of material due to wear	Chapter XI.M16A, "PWR Vessel Internals" Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B4.RP-240)	No

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B4.RP-248	IV.B4-12(R-196)	Core support shield (CSS) assembly: accessible upper core barrel (UCB) bolts and locking devices	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” Primary components (identified in the "Structure and Components" column) (for Expansion components see AMR Items IV.B4.RP-245, IV.B4.RP-246, IV.B4.RP-254, IV.B4.RP-247, and IV.B4.RP-256)	No
IV.B4.RP-253	IV.B4-21(R-191)	Core support shield (CSS) assembly: (a) CSS cast outlet nozzles (Oconee Unit 3 and Davis-Besse, only); (b) CSS vent valve discs	Cast austenitic stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to thermal aging embrittlement	Chapter XI.M16A, “PWR Vessel Internals” Primary components (identified in the "Structure and Components" column) (for Expansion components see AMR Item IV.B4.RP-242)	No
IV.B4.RP-252	IV.B4-16(R-188)	Core support shield (CSS) assembly: (a) CSS vent valve disc shaft or hinge pin (b) CSS vent valve top retaining ring (c) CSS vent valve bottom retaining ring	Stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to thermal aging embrittlement	Chapter XI.M16A, “PWR Vessel Internals” Primary components (identified in the "Structure and Components" column) No Expansion components	No

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B4 Reactor Vessel Internals (PWR) – Babcock & Wilcox

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B4.RP-251	IV.B4-15(R-190)	Core support shield (CSS) assembly: CSS top flange; plenum cover assembly: plenum cover weldment rib pads and plenum cover support flange	Stainless steel	Reactor coolant and neutron flux	Loss of material due to wear	Chapter XI.M16A, “PWR Vessel Internals” Primary component (identified in the "Structure and Components" column) No Expansion components	No
IV.B4.RP-256	IV.B4-25(R-210)	Flow distributor assembly: flow distributor bolts and locking devices	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals,” Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Items IV.B4.RP-247 and IV.B4.RP-248)	No
IV.B4.RP-259	IV.B4-31(R-205)	Incore Monitoring Instrumentation (IMI) guide tube assembly: accessible top surfaces of IMI guide tube spider-to-lower grid rib sections welds	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness due to thermal aging, neutron irradiation embrittlement	Chapter XI.M16A, “PWR Vessel Internals” Primary components (identified in the "Structure and Components" column) (for Expansion components see Item IV.B4.RP-260)	No

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B4.RP-258	IV.B4-4(R-183)	Incore Monitoring Instrumentation (IMI) guide tube assembly: accessible top surfaces of IMI Incore guide tube spider castings	Cast austenitic stainless steel	Reactor coolant and neutron flux	Loss of fracture toughness due to thermal aging, neutron irradiation embrittlement	Chapter XI.M16A, “PWR Vessel Internals” Primary components (identified in the "Structure and Components" column) (for Expansion components see Item IV.B4.RP-242)	No
IV.B4.RP-254	IV.B4-25(R-210)	Lower grid assembly: alloy X-750 lower grid shock pad bolts and locking devices (TMI-1, only)	Nickel alloy	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals,” Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Items IV.B4.RP-247 and IV.B4.RP-248)	No
IV.B4.RP-246	IV.B4-12(R-196)	Lower grid assembly: lower thermal shield (LTS) bolts	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking	'Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Items IV.B4.RP-247 and IV.B4.RP-248)	No

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B4.RP-260	IV.B4-31(R-205)	Lower grid assembly: (a) accessible pads; (b) accessible pad-to-rib section welds; (c) accessible alloy X-750 dowels, cap screws and locking devices	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement	Chapter XI.M16A, “PWR Vessel Internals" Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B4.RP-259)	No
IV.B4.RP-262	IV.B4-32(R-203)	Lower grid assembly: accessible alloy X-750 dowel-to-lower fuel assembly support pad welds	Nickel alloy	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B4.RP-261)	No
IV.B4.RP-261	IV.B4-32(R-203)	Lower grid assembly: alloy X-750 dowel-to-guide block welds	Nickel alloy	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals” Primary components (identified in the "Structure and Components" column) (for Expansion components see AMR Items IV.B4.RP-262 and IV.B4.RP-352)	No

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B4.R-53	IV.B4-37(R-53)	Reactor vessel internal components	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cumulative fatigue damage due to fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
IV.B4.RP-24	IV.B4-38(RP-24)	Reactor vessel internal components	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry”	No
IV.B4.RP-376		Reactor vessel internal components	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Reduction in ductility and fracture toughness due to neutron irradiation	Ductility - Reduction in Fracture Toughness is a TLAA (BAW-2248A) to be evaluated for the period of extended operation. See the SRP, Section 4.7, "Other Plant-Specific TLAAs," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
IV.B4.RP-238		Reactor vessel internal components (inaccessible locations)	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking, and irradiation-assisted stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M16A, “PWR Vessel Internals”	Yes, if accessible Primary, Expansion or Existing program components indicate aging effects that need management

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B4.RP-239		Reactor vessel internal components (inaccessible locations)	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement; change in dimension due to void swelling; loss of preload due to thermal and irradiation enhanced stress relaxation; loss of material due to wear	Chapter XI.M16A, "PWR Vessel Internals"	Yes, if accessible Primary, Expansion or Existing program components indicate aging effects that need management
IV.B4.RP-236		Reactor vessel internal components with no additional measures	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking, and irradiation-assisted stress corrosion cracking	Chapter XI.M2, "Water Chemistry" and Chapter XI.M16A, "PWR Vessel Internals" Note: Components with no additional measures are not uniquely identifies in GALL tables - Components with no additional measures are defined in Section 3.3.1 of MRP-227, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines"	No

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.B4.RP-237		Reactor vessel internal components with no additional measures	Stainless steel; nickel alloy	Reactor coolant and neutron flux	Loss of fracture toughness due to neutron irradiation embrittlement; change in dimension due to void swelling; loss of preload due to thermal and irradiation enhanced stress relaxation; loss of material due to wear	Chapter XI.M16A, "PWR Vessel Internals" Note: Components with no additional measures are not uniquely identified in GALL tables - Components with no additional measures are defined in Section 3.3.1 of MRP-227, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines"	No
IV.B4.RP-382	IV.B4-42(R-179)	Reactor vessel internals: core support structure	Stainless steel; nickel alloy; cast austenitic stainless steel	Reactor coolant and neutron flux	Cracking, or Loss of material due to wear	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD"	No
IV.B4.RP-352		Upper grid assembly: alloy X-750 dowel-to-upper fuel assembly support pad welds (all plants except Davis-Besse)	Nickel alloy	Reactor coolant and neutron flux	Cracking due to stress corrosion cracking	Chapter XI.M2, "Water Chemistry," and Chapter XI.M16A, "PWR Vessel Internals" Expansion components (identified in the "Structure and Components" column) (for Primary components see AMR Item IV.B4.RP-261)	No

C1. REACTOR COOLANT PRESSURE BOUNDARY (BOILING WATER REACTOR)

Systems, Structures, and Components

This section addresses the boiling water reactor (BWR) primary coolant pressure boundary and consists of the reactor coolant recirculation system and portions of other systems connected to the pressure vessel extending to the second containment isolation valve or to the first anchor point outside containment. The connected systems include the residual heat removal (RHR), low–pressure core spray (LPCS), high–pressure core spray (HPCS), low–pressure coolant injection (LPCI), high–pressure coolant injection (HPCI), reactor core isolation cooling (RCIC), isolation condenser (IC), reactor water cleanup (RWC), standby liquid control (SLC), feedwater (FW), and main steam (MS) systems; and the steam line to the HPCI and RCIC pump turbines. Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all systems, structures, and components that comprise the reactor coolant pressure boundary are governed by Group A Quality Standards.

Pump and valve internals perform their intended functions with moving parts or with a change in configuration, or are subject to replacement based on qualified life or specified time period. Pursuant to 10 CFR 54.21(a)(1), therefore, they are not subject to an aging management review.

Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in IV.E.

System Interfaces

The systems that interface with the reactor coolant pressure boundary include the reactor pressure vessel (IV.A1), the emergency core cooling system (V.D2), the standby liquid control system (VII.E2), the reactor water cleanup system (VII.E3), the shutdown cooling system (older plants) (VII.E4), the main steam system (VIII.B2), and the feedwater system (VIII.D2).

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.C1.RP-230	IV.C1-1(R-03)	Class 1 piping, fittings and branch connections < NPS 4	Steel; stainless steel	Reactor coolant	Cracking due to stress corrosion cracking, intergranular stress corrosion cracking (for stainless steel only), and thermal, mechanical, and vibratory loading	Chapter XI.M1, “ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD” for Class 1 components, Chapter XI.M2, “Water Chemistry,” and XI.M35, "One-Time Inspection of ASME Code Class 1 Small-bore Piping"	No
IV.C1.R-52	IV.C1-2(R-52)	Class 1 piping, piping components, and piping elements	Cast austenitic stainless steel	Reactor coolant >250°C (>482°F)	Loss of fracture toughness due to thermal aging embrittlement	Chapter XI.M12, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)"	No
IV.C1.R-08	IV.C1-3(R-08)	Class 1 pump casings; valve bodies and bonnets	Cast austenitic stainless steel	Reactor coolant >250°C (>482°F)	Loss of fracture toughness due to thermal aging embrittlement	Chapter XI.M1, “ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD” for Class 1 components For pump casings and valve bodies, screening for susceptibility to thermal aging is not necessary. The ASME Section XI inspection requirements are sufficient for managing the effects of loss of fracture toughness due to thermal aging embrittlement of CASS pump casings and valve bodies.	No

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.C1.RP-43	IV.C1-10(R-27)	Closure bolting	Steel; stainless steel	Air	Loss of preload due to thermal effects, gasket creep, and self-loosening	Chapter XI.M18, "Bolting Integrity"	No
IV.C1.RP-42	IV.C1-12(R-26)	Closure bolting	Steel; stainless steel	Air with reactor coolant leakage	Loss of material due to general (steel only), pitting, and crevice corrosion or wear	Chapter XI.M18, "Bolting Integrity"	No
IV.C1.R-15	IV.C1-4(R-15)	Isolation condenser components	Stainless steel	Reactor coolant	Cracking due to stress corrosion cracking, 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” The AMP in Chapter XI.M1 is to be augmented to detect cracking due to stress corrosion cracking and verification of the program's effectiveness 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 includes temperature and radioactivity monitoring of the shell side water, and eddy current testing of tubes.	Yes, detection of aging effects is to be evaluated

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.C1.R-225	IV.C1-5(R-225)	Isolation condenser components	Steel; stainless steel	Reactor coolant	Cracking due to cyclic loading	Chapter XI.M1, “ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD” for Class 1 components The AMP in Chapter XI.M1 is to be augmented to detect cracking due to cyclic loading and verification of the program's effectiveness 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 includes temperature and radioactivity monitoring of the shell side water, and eddy current testing of tubes.	Yes, detection of aging effects is to be evaluated
IV.C1.RP-39	IV.C1-6(R-16)	Isolation condenser components	Steel; stainless steel	Reactor coolant	Loss of material due to general (steel only), pitting, and crevice corrosion	Chapter XI.M1, “ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD,” and Chapter XI.M2, “Water Chemistry”	No
IV.C1.R-23	IV.C1-7(R-23)	Piping, piping components, and piping elements	Steel	Reactor coolant	Wall thinning due to flow-accelerated corrosion	Chapter XI.M17, “Flow-Accelerated Corrosion”	No
IV.C1.R-21	IV.C1-8(R-21)	Piping, piping components, and piping elements greater than or equal to 4 NPS	Nickel alloy	Reactor coolant	Cracking due to stress corrosion cracking, intergranular stress corrosion cracking	Chapter XI.M7, “BWR Stress Corrosion Cracking,” and Chapter XI.M2, “Water Chemistry”	No

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.C1.R-20	IV.C1-9(R-20)	Piping, piping components, and piping elements greater than or equal to 4 NPS	Stainless steel	Reactor coolant	Cracking due to stress corrosion cracking, intergranular stress corrosion cracking	Chapter XI.M7, “BWR Stress Corrosion Cracking,” and Chapter XI.M2, “Water Chemistry”	No
IV.C1.RP-44	IV.C1-11(R-28)	Pump and valve closure bolting	Steel; stainless steel	System temperature up to 288°C (550°F)	Cumulative fatigue damage due to fatigue	Fatigue is a TLAA evaluated for the period of extended operation; check ASME Code limits for allowable cycles (less than 7000 cycles) of thermal stress range. (SRP Sec 4.3 “Metal Fatigue,” for acceptable methods to comply with 10 CFR 54.21(c)(1))	Yes, TLAA
IV.C1.RP-158	IV.C1-14(RP-27)	Reactor coolant pressure boundary components	Steel (with stainless steel or nickel-alloy cladding); stainless steel; nickel alloy	Reactor coolant	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
IV.C1.R-220	IV.C1-15(R-220)	Reactor coolant pressure boundary components: piping, piping components, and piping elements	Steel (with or without nickel-alloy or stainless steel cladding); stainless steel; nickel alloy	Reactor coolant	Cumulative fatigue damage due to fatigue	Fatigue is a TLAA evaluated for the period of extended operation, and for Class 1 components environmental effects on fatigue are to be addressed. (See SRP, Sec 4.3 “Metal Fatigue,” for acceptable methods to comply with 10 CFR 54.21(c)(1))	Yes, TLAA

C2. REACTOR COOLANT SYSTEM AND CONNECTED LINES (PRESSURIZED WATER REACTOR)

Systems, Structures, and Components

This section addresses the pressurized water reactor (PWR) primary coolant pressure boundary and consists of the reactor coolant system and portions of other connected systems generally extending up to and including the second containment isolation valve or to the first anchor point and including the containment isolation valves, the reactor coolant pump, valves, pressurizer, and the pressurizer relief tank. The connected systems include the residual heat removal (RHR) or low pressure injection system, high pressure injection system, sampling system, and the small-bore piping. With respect to other systems such as the core flood system (CFS) or the safety injection tank (SIT) and the chemical and volume control system (CVCS), the isolation valves associated with the boundary between ASME Code class 1 and 2 are located inside the containment. Based on Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” and with the exception of the pressurizer relief tank, which is governed by Group B Quality Standards, all systems, structures, and components that comprise the reactor coolant system are governed by Group A Quality Standards. The recirculating pump seal water heat exchanger is discussed in V.D1.

Pump and valve internals perform their intended functions with moving parts or with a change in configuration, or are subject to replacement based on qualified life or specified time period. Pursuant to 10 CFR 54.21(a)(1), therefore, they are not subject to an aging management review.

Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in IV.E.

System Interfaces

The systems that interface with the reactor coolant pressure boundary include the reactor pressure vessel (IV.A2), the steam generators (IV.D1 and IV.D2), the emergency core cooling system (V.D1), and the chemical and volume control system (VII.E1).

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.C2.RP-235	IV.C2-1(R-02)	Class 1 piping, fittings and branch connections < NPS 4	Stainless steel; steel with stainless steel cladding	Reactor coolant	Cracking due to stress corrosion cracking, intergranular stress corrosion cracking (for stainless steel only), and thermal, mechanical, and vibratory loading	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD" for Class 1 components, Chapter XI.M2, "Water Chemistry," and XI.M35, "One-Time Inspection of ASME Code Class 1 Small-bore Piping"	No
IV.C2.R-05	IV.C2-3(R-05)	Class 1 piping, piping components, and piping elements	Cast austenitic stainless steel	Reactor coolant	Cracking due to stress corrosion cracking	Monitoring and control of primary water chemistry in accordance with EPRI 1014986 minimize the potential for SCC. Material selection according to 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, a plant-specific aging management program is 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.	Yes, plant-specific

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.C2.R-52	IV.C2-4(R-52)	Class 1 piping, piping components, and piping elements	Cast austenitic stainless steel	Reactor coolant >250°C (>482°F)	Loss of fracture toughness due to thermal aging embrittlement	Chapter XI.M12, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)"	No
IV.C2.RP-344	IV.C2-2(R-07)	Class 1 piping, piping components, and piping elements	Stainless steel; steel with stainless steel cladding	Reactor coolant	Cracking due to 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”	No
IV.C2.R-09	IV.C2-5(R-09)	Class 1 pump casings; valve bodies	Steel (with stainless steel cladding); stainless steel	Reactor coolant	Cracking due to 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”	No
IV.C2.R-08	IV.C2-6(R-08)	Class 1 pump casings; valve bodies and bonnets	Cast austenitic stainless steel	Reactor coolant >250°C (>482°F)	Loss of fracture toughness due to thermal aging embrittlement	Chapter XI.M1, “ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD” for Class 1 components For pump casings and valve bodies, screening for susceptibility to thermal aging is not necessary. The ASME Section XI inspection requirements are sufficient for managing the effects of loss of fracture toughness due to thermal aging embrittlement of CASS pump casings and valve bodies.	No

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.C2.R-11	IV.C2-7(R-11)	Closure bolting	High-strength, low-alloy steel; stainless steel	Air with reactor coolant leakage	Cracking due to stress corrosion cracking	Chapter XI.M18, "Bolting Integrity"	No
IV.C2.R-12	IV.C2-8(R-12)	Closure bolting	Low-alloy steel, stainless steel	Air (with reactor coolant leakage)	Loss of preload due to thermal effects, gasket creep, and self-loosening	Chapter XI.M18, "Bolting Integrity"	No
IV.C2.RP-166		Closure bolting	Steel	Air – indoor, uncontrolled	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M18, "Bolting Integrity"	No
IV.C2.RP-167		Closure bolting	Steel	Air with borated water leakage	Loss of material due to boric acid corrosion	Chapter XI.M10, “Boric Acid Corrosion”	No
IV.C2.R-17	IV.C2-9(R-17)	External surfaces	Steel	Air with borated water leakage	Loss of material due to boric acid corrosion	Chapter XI.M10, “Boric Acid Corrosion”	No
IV.C2.RP-380	IV.C2-9(R-17)	External surfaces: reactor coolant pressure boundary piping or components adjacent to dissimilar metal (Alloy 82/182) welds	Steel	Air with borated water leakage	Loss of material due to boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion," and Chapter XI.M11B, "Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in RCPB Components (PWRs Only)"	No

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.C2.R-18	IV.C2-10(R-18)	Piping and components (External surfaces); bolting	Steel; stainless steel	System temperature up to 340°C (644°F)	Cumulative fatigue damage due to fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
IV.C2.RP-222	IV.C2-11(RP-11)	Piping, piping components, and piping elements	Copper alloy	Closed-cycle cooling water	Loss of material due to pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
IV.C2.RP-12	IV.C2-12(RP-12)	Piping, piping components, and piping elements	Copper alloy (>15% Zn or >8% Al)	Closed-cycle cooling water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
IV.C2.RP-159	IV.C2-13(RP-31)	Piping, piping components, and piping elements	Nickel alloy	Reactor coolant or steam	Cracking due to 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,” and Chapter XI.M11B, “Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in RCPB Components (PWRs Only)"	No
IV.C2.RP-221	IV.C2-14(RP-10)	Piping, piping components, and piping elements	Steel	Closed-cycle cooling water	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.C2.RP-23	IV.C2-15(RP-23)	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 due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry”	No
IV.C2.R-58	IV.C2-18(R-58)	Pressurizer components	Steel (with stainless steel or nickel-alloy cladding); stainless steel	Reactor coolant	Cracking due to 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” Cracks in the pressurizer cladding could propagate from cyclic loading into the ferrite base metal and weld metal. However, because the weld metal between the surge nozzle and the vessel lower head is subjected to the maximum stress cycles and the area is periodically inspected as part of the ISI program, the existing AMP is adequate for managing the effect of pressurizer clad cracking.	No

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.C2.R-25	IV.C2-19(R-25)	Pressurizer components	Steel (with stainless steel or nickel-alloy cladding); stainless steel	Reactor coolant	Cracking due to 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”	No
IV.C2.R-217	IV.C2-20(R-217)	Pressurizer heater sheaths and sleeves; heater bundle diaphragm plate	Stainless steel	Reactor coolant	Cracking due to 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”	No
IV.C2.RP-37	IV.C2-21(R-06)	Pressurizer instrumentation penetrations; heater sheaths and sleeves; heater bundle diaphragm plate; manways and flanges	Nickel alloy; nickel-alloy cladding	Reactor coolant	Cracking due to 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,” and Chapter XI.M11B, “Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in RCPB Components (PWRs Only)"	No
IV.C2.RP-231	IV.C2-22(R-14)	Pressurizer relief tank: tank shell and heads; flanges; nozzles	Stainless steel; steel with stainless steel cladding	Treated borated water >60°C (>140°F)	Cracking due to stress corrosion cracking	Chapter XI.M1, “ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD” for ASME Code components, and Chapter XI.M2, “Water Chemistry”	No

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.C2.R-13	IV.C2-23(R-13)	Pressurizer relief tank: tank shell and heads; flanges; nozzles	Steel (with stainless steel or nickel-alloy cladding)	Treated borated water	Cumulative fatigue damage due to fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
IV.C2.RP-383		Pressurizer relief tank: tank shell and heads; flanges; nozzles (non-ASME Section XI components)	Stainless steel; steel with stainless steel cladding	Treated borated water >60°C (>140°F)	Cracking due to stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
IV.C2.RP-156	IV.C2-24(RP-22)	Pressurizer surge and steam space nozzles; welds	Nickel alloy	Reactor coolant or steam	Cracking due to 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,” and Chapter XI.M11B, “Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in RCPB Components (PWRs Only)"	No
IV.C2.R-19	IV.C2-16(R-19)	Pressurizer: integral support	Steel; stainless steel	Air with metal temperature up to 288°C (550°F)	Cracking due to cyclic loading	Chapter XI.M1, “ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD” for Class 1 components	No

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.C2.RP-40	IV.C2-17(R-24)	Pressurizer: spray head	Nickel alloy	Reactor coolant	Cracking due to stress corrosion cracking, primary water stress corrosion cracking	Chapter XI.M2, "Water Chemistry," and Chapter XI.M32, "One-Time Inspection"	No
IV.C2.RP-41	IV.C2-17(R-24)	Pressurizer: spray head	Stainless steel	Reactor coolant	Cracking due to stress corrosion cracking	Chapter XI.M2, "Water Chemistry," and Chapter XI.M32, "One-Time Inspection"	No
IV.C2.R-223	IV.C2-25(R-223)	Reactor coolant pressure boundary components: piping, piping components, and piping elements; flanges; nozzles and safe ends; pressurizer vessel shell heads and welds; heater sheaths and sleeves; penetrations; thermal sleeves	Steel (with or without nickel-alloy or stainless steel cladding); stainless steel; nickel alloy	Reactor coolant	Cumulative fatigue damage due to fatigue	Fatigue is a TLAA evaluated for the period of extended operation, and for Class 1 components environmental effects on fatigue are to be addressed. (See SRP, Sec 4.3 "Metal Fatigue," for acceptable methods to comply with 10 CFR 54.21(c)(1))	Yes, TLAA

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

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.C2.R-56	IV.C2-26(R-56)	Reactor coolant system piping and fittings: cold leg; hot leg; surge line; spray line	Steel (with stainless steel cladding); stainless steel	Reactor coolant	Cracking due to cyclic loading	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD" for Class 1 components	No
IV.C2.R-30	IV.C2-27(R-30)	Reactor coolant system piping and fittings: cold leg; hot leg; surge line; spray line	Steel (with stainless steel cladding); stainless steel	Reactor coolant	Cracking due to 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"	No

D1. STEAM GENERATOR (RECIRCULATING)

Systems, Structures, and Components

This section addresses the recirculating-type steam generators, as found in Westinghouse and Combustion Engineering pressurized water reactors (PWRs), including all internal components and water/steam nozzles and safe ends. Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," the primary water side (tube side) of the steam generator is governed by Group A Quality Standards, and the secondary water side is governed by Group B Quality Standards.

Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in IV.E.

System Interfaces

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

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
IV.D1.R-10	IV.D1-2(R-10)	Closure bolting	Steel	Air with reactor coolant leakage	Cracking due to stress corrosion cracking	Chapter XI.M18, "Bolting Integrity"	No
IV.D1.RP-46	IV.D1-10(R-32)	Closure bolting	Steel; stainless steel	Air – indoor, uncontrolled (External)	Loss of preload due to thermal effects, gasket creep, and self-loosening	Chapter XI.M18, "Bolting Integrity"	No
IV.D1.R-17	IV.D1-3(R-17)	External surfaces	Steel	Air with borated water leakage	Loss of material due to boric acid corrosion	Chapter XI.M10, “Boric Acid Corrosion”	No
IV.D1.RP-36	IV.D1-4(R-01)	Instrument penetrations and primary side nozzles; safe ends; welds	Steel (with nickel-alloy cladding); nickel alloy	Reactor coolant	Cracking due to 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,” and Chapter XI.M11B, “Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in RCPB Components (PWRs Only)"	No
IV.D1.R-37	IV.D1-5(R-37)	Pressure boundary and structural: steam nozzle and safe end; feedwater nozzle and safe end	Steel	Secondary feedwater or steam	Wall thinning due to flow-accelerated corrosion	Chapter XI.M17, “Flow-Accelerated Corrosion”	No

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
IV.D1.RP-17	IV.D1-7(RP-17)	Primary side components: divider plate	Stainless steel	Reactor coolant	Cracking due to stress corrosion cracking	Chapter XI.M2, “Water Chemistry”	No
IV.D1.RP-367	IV.D1-6(RP-21)	Primary side components: divider plate	Steel (with nickel-alloy cladding); nickel alloy	Reactor coolant	Cracking due to primary water stress corrosion cracking	Chapter XI.M2, “Water Chemistry” For nickel alloy divider plate assemblies and associated welds made of Alloy 600, effectiveness of the chemistry control program should be verified to ensure that cracking due to PWSCC is not occurring.	Yes, detection of aging effects is to be evaluated
IV.D1.R-221	IV.D1-8(R-221)	Recirculating steam generator components: flanges; penetrations; nozzles; safe ends; lower heads and welds	Steel (with or without nickel-alloy or stainless steel cladding); stainless steel; nickel alloy	Reactor coolant	Cumulative fatigue damage due to fatigue	Fatigue is a TLAA evaluated for the period of extended operation, and for Class 1 components environmental effects on fatigue are to be addressed. (See SRP, Sec 4.3 “Metal Fatigue,” for acceptable methods to comply with 10 CFR 54.21(c)(1))	Yes, TLAA
IV.D1.RP-372		Steam generator components: shell assembly	Steel	Secondary feedwater or steam	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M2, “Water Chemistry” and Chapter XI.M32, “One-Time Inspection”	No

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
IV.D1.R-33	IV.D1-11(R-33)	Steam generator components: top head; steam nozzle and safe end; upper and lower shell; feedwater (FW) and auxiliary FW nozzle and safe end; FW impingement plate and support	Steel	Secondary feedwater or steam	Cumulative fatigue damage due to fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA

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
IV.D1.RP-368	IV.D1-12(R-34)	Steam generator components: upper and lower shell; transition cone; new transition cone closure weld	Steel	Secondary feedwater or steam	Loss of material due to 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” As noted in NRC IN 90-04, if general and pitting corrosion of the shell exists, Chapter XI.M1 methods 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. The new transition is only applicable to replacement recirculating steam generators.	Yes, detection of aging effects is to be evaluated
IV.D1.R-39	IV.D1-13(R-39)	Steam generator feedwater impingement plate and support	Steel	Secondary feedwater	Loss of material due to erosion	A plant-specific aging management program is to be evaluated	Yes, plant-specific
IV.D1.RP-48	IV.D1-16(R-41)	Steam generator structural: tube support lattice bars	Steel	Secondary feedwater or steam	Wall thinning due to flow-accelerated corrosion and general corrosion	Chapter XI.M19, “Steam Generators,” and Chapter XI.M2, “Water Chemistry”	No

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
IV.D1.R-42	IV.D1-17(R-42)	Steam generator structural: tube support plates	Steel	Secondary feedwater or steam	Ligament cracking due to corrosion	Chapter XI.M19, “Steam Generators,” and Chapter XI.M2, “Water Chemistry”	No
IV.D1.RP-384	IV.D1-14(RP-14)	Steam generator structural: U-bend supports including anti-vibration bars	Steel; chrome plated steel; stainless steel; nickel alloy	Secondary feedwater or steam	Cracking due to stress corrosion cracking or other mechanism(s)	Chapter XI.M19, “Steam Generators,” and Chapter XI.M2, “Water Chemistry”	No
IV.D1.RP-225	IV.D1-15(RP-15)	Steam generator structural: U-bend supports including anti-vibration bars	Steel; chrome plated steel; stainless steel; nickel alloy	Secondary feedwater or steam	Loss of material due to fretting	Chapter XI.M19, “Steam Generators”	No
IV.D1.RP-226	IV.D1-15(RP-15)	Steam generator structural: U-bend supports including anti-vibration bars	Steel; chrome plated steel; stainless steel; nickel alloy	Secondary feedwater or steam	Loss of material due to general (steel only), pitting, and crevice corrosion	Chapter XI.M19, “Steam Generators,” and Chapter XI.M2, “Water Chemistry”	No
IV.D1.RP-232	IV.D1-1(R-07)	Steam generator: primary nozzles; nozzle to safe end welds; manways; flanges	Stainless steel; steel with stainless steel cladding	Reactor coolant	Cracking due to 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”	No

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
IV.D1.RP-161	IV.D1-9(RP-16)	Steam generator: Tube bundle wrapper and associated supports and mounting hardware	Steel	Secondary feedwater or steam	Loss of material due to erosion, general, pitting, and crevice corrosion	Chapter XI.M19, “Steam Generators,” and Chapter XI.M2, “Water Chemistry”	No
IV.D1.R-40	IV.D1-18(R-40)	Tube plugs	Nickel alloy	Reactor coolant	Cracking due to primary water stress corrosion cracking	Chapter XI.M19, “Steam Generators,” and Chapter XI.M2, “Water Chemistry”	No
IV.D1.R-43	IV.D1-19(R-43)	Tubes	Nickel alloy	Secondary feedwater or steam	Changes in dimension ("denting") due to corrosion of carbon steel tube support plate	Chapter XI.M19, “Steam Generators,” and Chapter XI.M2, “Water Chemistry”	No
IV.D1.R-44	IV.D1-20(R-44)	Tubes and sleeves	Nickel alloy	Reactor coolant	Cracking due to primary water stress corrosion cracking	Chapter XI.M19, “Steam Generators,” and Chapter XI.M2, “Water Chemistry”	No
IV.D1.R-46	IV.D1-21(R-46)	Tubes and sleeves	Nickel alloy	Reactor coolant and secondary feedwater/steam	Cumulative fatigue damage due to fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA

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
IV.D1.R-48	IV.D1-22(R-48)	Tubes and sleeves	Nickel alloy	Secondary feedwater or steam	Cracking due to intergranular attack	Chapter XI.M19, “Steam Generators,” and Chapter XI.M2, “Water Chemistry”	No
IV.D1.R-47	IV.D1-23(R-47)	Tubes and sleeves	Nickel alloy	Secondary feedwater or steam	Cracking due to outer diameter stress corrosion cracking	Chapter XI.M19, “Steam Generators,” and Chapter XI.M2, “Water Chemistry”	No
IV.D1.RP-233	IV.D1-24(R-49)	Tubes and sleeves	Nickel alloy	Secondary feedwater or steam	Loss of material due to fretting and wear	Chapter XI.M19, “Steam Generators”	No
IV.D1.R-50	IV.D1-25(R-50)	Tubes and sleeves (exposed to phosphate chemistry)	Nickel alloy	Secondary feedwater or steam	Loss of material due to wastage and pitting corrosion	Chapter XI.M19, “Steam Generators,” and Chapter XI.M2, “Water Chemistry”	No
IV.D1.RP-385		Tube-to-tube sheet welds	Nickel alloy	Reactor coolant	Cracking due to primary water stress corrosion cracking	Chapter XI.M2, “Water Chemistry” A plant-specific program is to be evaluated; the effectiveness of the water chemistry program should be verified to ensure cracking is not occurring.	Yes, plant-specific.
IV.D1.RP-49	IV.D1-26(R-51)	Upper assembly and separators including: feedwater inlet ring and support	Steel	Secondary feedwater or steam	Wall thinning due to flow-accelerated corrosion	Chapter XI.M19, “Steam Generators,” and Chapter XI.M2, “Water Chemistry”	No

D2. STEAM GENERATOR (ONCE-THROUGH)

Systems, Structures, and Components

This section addresses the once-through type steam generators, as found in Babcock & Wilcox pressurized water reactors (PWRs), including all internal components and water/steam nozzles and safe ends. Based on Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” the primary water side (tube side) of the steam generator is governed by Group A Quality Standards, and the secondary water side is governed by Group B Quality Standards.

Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in IV.E.

System Interfaces

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

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
IV.D2.RP-46	IV.D2-6(R-32)	Closure bolting	Steel; stainless steel	Air – indoor, uncontrolled (External)	Loss of preload due to thermal effects, gasket creep, and self-loosening	Chapter XI.M18, "Bolting Integrity"	No
IV.D2.R-17	IV.D2-1(R-17)	External surfaces	Steel	Air with borated water leakage	Loss of material due to boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No
IV.D2.RP-36	IV.D2-2(R-01)	Instrument penetrations and primary side nozzles; safe ends; welds	Steel (with nickel-alloy cladding); nickel alloy	Reactor coolant	Cracking due to 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," and Chapter XI.M11B, "Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in RCPB Components (PWRs Only)"	No
IV.D2.R-222	IV.D2-3(R-222)	Once-through steam generator components: primary side nozzles, safe ends, and welds	Steel (with or without nickel-alloy or stainless steel cladding); stainless steel; nickel alloy	Reactor coolant	Cumulative fatigue damage due to fatigue	Fatigue is a TLAA evaluated for the period of extended operation, and for Class 1 components environmental effects on fatigue are to be addressed. (See SRP, Sec 4.3 "Metal Fatigue," for acceptable methods to comply with 10 CFR 54.21(c)(1))	Yes, TLAA

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
IV.D2.RP-47	IV.D2-4(R-35)	Primary side components: upper and lower heads, and tube sheet welds exposed to reactor coolant	Steel (with stainless steel or nickel-alloy cladding)	Reactor coolant	Cracking due to 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”	No
IV.D2.R-31	IV.D2-5(R-31)	Secondary manway covers; handhole covers	Steel	Air with leaking secondary-side water and/or steam	Loss of material due to erosion	Chapter XI.M1, “ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD” for Class 2 components	No
IV.D2.R-36	IV.D2-9(R-36)	Steam generator components: secondary side nozzles (vent, drain, and instrumentation)	Nickel alloy	Secondary feedwater or steam	Cracking due to 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.”	No
IV.D2.R-38	IV.D2-7(R-38)	Steam generator components: feedwater (FW) and auxiliary FW nozzles and safe ends; steam nozzles and safe ends	Steel	Secondary feedwater or steam	Wall thinning due to flow-accelerated corrosion	Chapter XI.M17, “Flow-Accelerated Corrosion”	No
IV.D2.RP-153	IV.D2-8(R-224)	Steam generator components: shell assembly	Steel	Secondary feedwater or steam	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No

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
IV.D2.R-33	IV.D2-10(R-33)	Steam generator components: top head; steam nozzle and safe end; upper and lower shell; feedwater (FW) and auxiliary FW nozzle and safe end; FW impingement plate and support	Steel	Secondary feedwater or steam	Cumulative fatigue damage due to fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
IV.D2.R-42	IV.D2-11(R-42)	Steam generator structural: tube support plates	Steel	Secondary feedwater or steam	Ligament cracking due to corrosion	Chapter XI.M19, “Steam Generators,” and Chapter XI.M2, “Water Chemistry”	No
IV.D2.RP-162		Steam generator: tube bundle wrapper and associated supports and mounting hardware	Steel	Secondary feedwater or steam	Loss of material due to erosion, general, pitting, and crevice corrosion	Chapter XI.M19, “Steam Generators,” and Chapter XI.M2, “Water Chemistry”	No
IV.D2.R-40	IV.D2-12(R-40)	Tube plugs	Nickel alloy	Reactor coolant	Cracking due to primary water stress corrosion cracking	Chapter XI.M19, “Steam Generators,” and Chapter XI.M2, “Water Chemistry”	No

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
IV.D2.R-226	IV.D2-13(R-226)	Tubes	Nickel alloy	Secondary feedwater or steam	Changes in dimension ("denting") due to corrosion of carbon steel tube support plate	Chapter XI.M19, "Steam Generators," and Chapter XI.M2, "Water Chemistry"	No
IV.D2.R-44	IV.D2-14(R-44)	Tubes and sleeves	Nickel alloy	Reactor coolant	Cracking due to primary water stress corrosion cracking	Chapter XI.M19, "Steam Generators," and Chapter XI.M2, "Water Chemistry"	No
IV.D2.R-46	IV.D2-15(R-46)	Tubes and sleeves	Nickel alloy	Reactor coolant and secondary feedwater/steam	Cumulative fatigue damage due to fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
IV.D2.R-48	IV.D2-16(R-48)	Tubes and sleeves	Nickel alloy	Secondary feedwater or steam	Cracking due to intergranular attack	Chapter XI.M19, "Steam Generators," and Chapter XI.M2, "Water Chemistry"	No
IV.D2.R-47	IV.D2-17(R-47)	Tubes and sleeves	Nickel alloy	Secondary feedwater or steam	Cracking due to outer diameter stress corrosion cracking	Chapter XI.M19, "Steam Generators," and Chapter XI.M2, "Water Chemistry"	No
IV.D2.RP-233	IV.D2-18(R-49)	Tubes and sleeves	Nickel alloy	Secondary feedwater or steam	Loss of material due to fretting and wear	Chapter XI.M19, "Steam Generators"	No

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
IV.D2.RP-185	IV.D2-4(R-35)	Tube-to-tube sheet welds	Nickel alloy	Reactor coolant	Cracking due to primary water stress corrosion cracking	Chapter XI.M2, “Water Chemistry” A plant-specific program is to be evaluated; the effectiveness of the water chemistry program should be verified to ensure cracking is not occurring	Yes, plant-specific

E. COMMON MISCELLANEOUS MATERIAL/ENVIRONMENT COMBINATIONS

Systems, Structures, and Components

This section addresses the aging management programs for miscellaneous material/environment combinations which may be found throughout the reactor vessel, internals and reactor coolant system's structures and components. For the material/environment combinations in this part, aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation, therefore, no resulting aging management programs for these structures and components are required.

System Interfaces

The structures and components covered in this section belong to the engineered safety features in PWRs and BWRs. (For example, see System Interfaces in V.A to V.D2 for details.)

IV REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM
E Common Miscellaneous Material Environment Combinations

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
IV.E.RP-03	IV.E-1(RP-03)	Piping, piping components, and piping elements	Nickel alloy	Air – indoor, uncontrolled (External)	None	None	No
IV.E.RP-378		Piping, piping components, and piping elements	Nickel alloy	Air with borated water leakage	None	None	No
IV.E.RP-04	IV.E-2(RP-04)	Piping, piping components, and piping elements	Stainless steel	Air – indoor, uncontrolled (External)	None	None	No
IV.E.RP-05	IV.E-3(RP-05)	Piping, piping components, and piping elements	Stainless steel	Air with borated water leakage	None	None	No
IV.E.RP-06	IV.E-4(RP-06)	Piping, piping components, and piping elements	Stainless steel	Concrete	None	None	No
IV.E.RP-07	IV.E-5(RP-07)	Piping, piping components, and piping elements	Stainless steel	Gas	None	None	No
IV.E.RP-353	IV.E-6(RP-01)	Piping, piping components, and piping elements	Steel	Concrete	None	None, provided: 1) attributes of the concrete are consistent with ACI 318 or ACI 349 (low water-to-cement ratio, low permeability, and adequate air entrainment) as cited in NUREG-1557, and 2) plant OE indicates no degradation of the concrete	No, if conditions are met.

CHAPTER V

ENGINEERED SAFETY FEATURES

MAJOR PLANT SECTIONS

A. Containment Spray System (Pressurized Water Reactors)

B. Standby Gas Treatment System (Boiling Water Reactors)

C. Containment Isolation Components

D1. Emergency Core Cooling System (Pressurized Water Reactors)

D2. Emergency Core Cooling System (Boiling Water Reactors)

E. External Surfaces of Components and Miscellaneous Bolting

F. Common Miscellaneous Material/Environment Combinations

A. CONTAINMENT SPRAY SYSTEM (PRESSURIZED WATER REACTORS)

Systems, Structures, and Components

This section addresses the containment spray system for pressurized water reactors (PWRs) designed to lower the pressure, temperature, and gaseous radioactivity (iodine) content of the containment atmosphere following a design basis event. Spray systems using chemically treated borated water are reviewed. The system consists of piping and valves, including containment isolation valves, flow elements, orifices, pumps, spray nozzles, eductors, and the containment spray system heat exchanger (for some plants).

Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all components that comprise the containment spray system outside or inside the containment are governed by Group B Quality Standards.

Pumps and valve internals perform their intended functions with moving parts or with a change in configuration. They are also subject to replacement based on qualified life or a specified time period. Pursuant to 10 CFR 54.21(a)(1), therefore, they are not subject to an aging management review.

Aging management programs for the degradation of external surfaces of components and miscellaneous bolting are included in V.E. Common miscellaneous material/environment combinations, where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation, are included in V.F.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The systems that interface with the containment spray system are the PWR emergency core cooling (V.D1), and open- or closed-cycle cooling water systems (VII.C1 or VII.C2).

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
V.A.E-26	V.A-1(E-26)	Ducting, piping, and components (External surfaces)	Steel	Air – indoor, uncontrolled (External)	Loss of material due to general corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
V.A.EP-42	V.A-2(EP-42)	Encapsulation components	Steel	Air – indoor, uncontrolled (Internal)	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
V.A.EP-43	V.A-3(EP-43)	Encapsulation components	Steel	Air with borated water leakage (Internal)	Loss of material due to general, pitting, crevice, and boric acid corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
V.A.E-28	V.A-4(E-28)	External surfaces	Steel	Air with borated water leakage	Loss of material due to boric acid corrosion	Chapter XI.M10, “Boric Acid Corrosion”	No
V.A.EP-94	V.A-5(EP-13)	Heat exchanger components	Copper alloy	Closed-cycle cooling water	Loss of material due to pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
V.A.EP-37	V.A-6(EP-37)	Heat exchanger components	Copper alloy (>15% Zn or >8% Al)	Closed-cycle cooling water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
V.A.EP-93	V.A-7(E-19)	Heat exchanger components	Stainless steel	Closed-cycle cooling water	Loss of material due to pitting and crevice corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
V.A.EP-91	V.A-8(E-20)	Heat exchanger components	Stainless steel	Raw water	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion; fouling that leads to corrosion	Chapter XI.M20, "Open-Cycle Cooling Water System"	No

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
V.A.EP-92	V.A-9(E-17)	Heat exchanger components	Steel	Closed-cycle cooling water	Loss of material due to general, pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
V.A.EP-90	V.A-10(E-18)	Heat exchanger components	Steel	Raw water	Loss of material due to general, pitting, crevice, and microbiologically-influenced corrosion; fouling that leads to corrosion	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
V.A.EP-100	V.A-11(EP-39)	Heat exchanger tubes	Copper alloy	Closed-cycle cooling water	Reduction of heat transfer due to fouling	Chapter XI.M21A, "Closed Treated Water Systems"	No
V.A.EP-78	V.A-12(EP-47)	Heat exchanger tubes	Copper alloy	Lubricating oil	Reduction of heat transfer due to fouling	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
V.A.EP-96	V.A-13(EP-35)	Heat exchanger tubes	Stainless steel	Closed-cycle cooling water	Reduction of heat transfer due to fouling	Chapter XI.M21A, "Closed Treated Water Systems"	No
V.A.EP-79	V.A-14(EP-50)	Heat exchanger tubes	Stainless steel	Lubricating oil	Reduction of heat transfer due to fouling	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
V.A.E-21	V.A-15(E-21)	Heat exchanger tubes	Stainless steel	Raw water	Reduction of heat transfer due to fouling	Chapter XI.M20, "Open-Cycle Cooling Water System"	No

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
V.A.EP-74	V.A-16(EP-34)	Heat exchanger tubes	Stainless steel	Treated water	Reduction of heat transfer due to fouling	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
V.A.EP-75	V.A-17(EP-40)	Heat exchanger tubes	Steel	Lubricating oil	Reduction of heat transfer due to fouling	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
V.A.E-43	V.A-18(E-43)	Motor cooler	Gray cast iron	Treated water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
V.A.E-29	V.A-19(E-29)	Piping and components (Internal surfaces)	Steel	Air – indoor, uncontrolled (Internal)	Loss of material due to general corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
V.A.EP-97	V.A-20(EP-36)	Piping, piping components, and piping elements	Copper alloy	Closed-cycle cooling water	Loss of material due to pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
V.A.EP-76	V.A-21(EP-45)	Piping, piping components, and piping elements	Copper alloy	Lubricating oil	Loss of material due to pitting and crevice corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
V.A.EP-27	V.A-22(EP-27)	Piping, piping components, and piping elements	Copper alloy (>15% Zn or >8% Al)	Closed-cycle cooling water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
V.A.EP-95	V.A-23(EP-33)	Piping, piping components, and piping elements	Stainless steel	Closed-cycle cooling water	Loss of material due to pitting and crevice corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No

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
V.A.EP-98	V.A-24(EP-44)	Piping, piping components, and piping elements	Stainless steel	Closed-cycle cooling water >60°C (>140°F)	Cracking due to stress corrosion cracking	Chapter XI.M21A, "Closed Treated Water Systems"	No
V.A.EP-77	V.A-25(EP-46)	Piping, piping components, and piping elements	Steel	Lubricating oil	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
V.A.EP-81	V.A-26(EP-53)	Piping, piping components, and piping elements (Internal surfaces); tanks	Stainless steel	Condensation (Internal)	Loss of material due to pitting and crevice corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
V.A.EP-41	V.A-27(EP-41)	Piping, piping components, and piping elements; tanks	Stainless steel	Treated water (borated)	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry”	No
V.A.E-12	V.A-28(E-12)	Piping, piping components, and piping elements; tanks	Stainless steel	Treated water (borated) >60°C (>140°F)	Cracking due to stress corrosion cracking	Chapter XI.M2, “Water Chemistry”	No

B. STANDBY GAS TREATMENT SYSTEM (BOILING WATER REACTORS)

Systems, Structures, and Components

This section addresses the standby gas treatment system found in boiling water reactors (BWRs) and consists of ductwork, filters, and fans. Based on Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” all components that comprise the standby gas treatment system are governed by Group B Quality Standards.

Specifically, charcoal absorber filters are to be addressed consistent with the NRC position on consumables, provided in the NRC letter from Christopher I. Grimes to Douglas J. Walters of NEI, dated March 10, 2000. Components that function as system filters 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, from an aging management review (on a plant-specific basis), under 10 CFR 54.21(a)(1)(ii). As part of the methodology description, the application should identify the standards that are relied on for replacement, for example, National Fire Protection Association (NFPA) standards for fire protection equipment.

Aging management programs for the degradation of external surfaces of components and miscellaneous bolting are included in V.E. Common miscellaneous material/environment combinations, where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation, are included in V.F.

System Interfaces

There are no system interfaces with the standby gas treatment system addressed in this section.

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
V.B.E-25	V.B-1(E-25)	Ducting and components (Internal surfaces)	Steel	Air – indoor, uncontrolled (Internal)	Loss of material due to general corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
V.B.E-40	V.B-2(E-40)	Ducting, closure bolting	Steel	Air – indoor, uncontrolled (External)	Loss of material due to general corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
V.B.E-26	V.B-3(E-26)	Ducting, piping, and components (External surfaces)	Steel	Air – indoor, uncontrolled (External)	Loss of material due to general corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
V.B.EP-59	V.B-4(E-06)	Elastomer seals and components	Elastomers	Air – indoor, uncontrolled (External)	Hardening and loss of strength due to elastomer degradation	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
V.B.EP-58	V.B-4(E-06)	Elastomer seals and components	Elastomers	Air – indoor, uncontrolled (Internal)	Hardening and loss of strength due to elastomer degradation	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
V.B.EP-37	V.B-5(EP-37)	Heat exchanger components	Copper alloy (>15% Zn or >8% Al)	Closed-cycle cooling water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
V.B.EP-97	V.B-6(EP-36)	Piping, piping components, and piping elements	Copper alloy	Closed-cycle cooling water	Loss of material due to pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
V.B.EP-27	V.B-7(EP-27)	Piping, piping components, and piping elements	Copper alloy (>15% Zn or >8% Al)	Closed-cycle cooling water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No

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
V.B.EP-54	V.B-8(EP-54)	Piping, piping components, and piping elements	Gray cast iron	Soil	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
V.B.EP-111	V.B-9(E-42)	Piping, piping components, and piping elements	Steel (with coating or wrapping)	Soil or concrete	Loss of material due to general, pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M41, "Buried and Underground Piping and Tanks"	No
V.B.EP-103		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated
V.B.EP-107		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Loss of material due to pitting and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated

C. CONTAINMENT ISOLATION COMPONENTS

Systems, Structures, and Components

This section addresses the containment isolation components found in all designs of boiling water reactors (BWR) and pressurized water reactors (PWR) in the United States. The system consists of isolation barriers in lines for BWR and PWR nonsafety systems, such as the plant heating, waste gas, plant drain, liquid waste, and cooling water systems. Based on Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” all components that comprise the containment isolation components are governed by Group A or B Quality Standards.

The aging management programs for hatchways, hatch doors, penetration sleeves, penetration bellows, seals, gaskets, and anchors are addressed in II.A and II.B. The containment isolation valves for in-scope systems are addressed in the appropriate sections in IV, VII, and VIII.

Aging management programs for the degradation of external surfaces of components and miscellaneous bolting are included in V.E. Common miscellaneous material/environment combinations, where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation, are included in V.F.

System Interfaces

There are no system interfaces with the containment isolation components addressed in this section.

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
V.C.E-35	V.C-1(E-35)	Containment isolation piping and components (External surfaces)	Steel	Air – indoor, uncontrolled (External)	Loss of material due to general corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
V.C.E-30	V.C-2(E-30)	Containment isolation piping and components (External surfaces)	Steel	Condensation (External)	Loss of material due to general corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
V.C.E-34	V.C-3(E-34)	Containment isolation piping and components (Internal surfaces)	Stainless steel	Raw water	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion; fouling that leads to corrosion	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
V.C.EP-63	V.C-4(E-33)	Containment isolation piping and components (Internal surfaces)	Stainless steel	Treated water	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
V.C.E-22	V.C-5(E-22)	Containment isolation piping and components (Internal surfaces)	Steel	Raw water	Loss of material due to general, pitting, crevice, and microbiologically-influenced corrosion; fouling that leads to corrosion	Chapter XI.M20, "Open-Cycle Cooling Water System"	No

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
V.C.EP-62	V.C-6(E-31)	Containment isolation piping and components (Internal surfaces)	Steel	Treated water	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
V.C.EP-95	V.C-7(EP-33)	Piping, piping components, and piping elements	Stainless steel	Closed-cycle cooling water	Loss of material due to pitting and crevice corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
V.C.EP-98	V.C-8(EP-44)	Piping, piping components, and piping elements	Stainless steel	Closed-cycle cooling water >60°C (>140°F)	Cracking due to stress corrosion cracking	Chapter XI.M21A, "Closed Treated Water Systems"	No
V.C.EP-99	V.C-9(EP-48)	Piping, piping components, and piping elements	Steel	Closed-cycle cooling water	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
V.C.EP-103		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated
V.C.EP-107		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Loss of material due to pitting and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated

D1. EMERGENCY CORE COOLING SYSTEM (PRESSURIZED WATER REACTORS)

Systems, Structures, and Components

This section addresses the emergency core cooling systems for pressurized water reactors (PWRs) designed to cool the reactor core and provide safe shutdown following a design basis accident. The core cooling systems consist of the core flood system (CFS), residual heat removal (RHR) (or shutdown cooling (SDC)), high-pressure safety injection (HPSI), low-pressure safety injection (LPSI), and spent fuel pool (SFP) cooling systems, the lines to the chemical and volume control system (CVCS), the emergency sump, the HPSI and LPSI pumps, the pump seal coolers, the RHR heat exchanger, and the refueling water tank (RWT).

Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all components that comprise the emergency core cooling system are governed by Group B Quality Standards. Portions of the RHR, HPSI, and LPSI systems and the CVCS extending from the reactor coolant system up to and including the second containment isolation valve are governed by Group A Quality Standards and covered in IV.C2.

Pumps and valve internals perform their intended functions with moving parts or with a change in configuration. They are also subject to replacement based on qualified life or a specified time period. Pursuant to 10 CFR 54.21(a)(1), therefore, they are not subject to an aging management review.

Aging management programs for the degradation of external surfaces of components and miscellaneous bolting are included in V.E. Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in VI.F.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The systems that interface with the emergency core cooling system include the reactor coolant system and connected lines (IV.C2), the containment spray system (V.A), the spent fuel pool cooling and cleanup system (VII.A3), the closed-cycle cooling water system (VII.C2), the ultimate heat sink (VII.C3), the chemical and volume control system (VII.E1), and the open-cycle cooling water system (service water system) (VII.C1).

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
V.D1.E-28	V.D1-1(E-28)	External surfaces	Steel	Air with borated water leakage	Loss of material due to boric acid corrosion	Chapter XI.M10, “Boric Acid Corrosion”	No
V.D1.EP-94	V.D1-2(EP-13)	Heat exchanger components	Copper alloy	Closed-cycle cooling water	Loss of material due to pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
V.D1.EP-37	V.D1-3(EP-37)	Heat exchanger components	Copper alloy (>15% Zn or >8% Al)	Closed-cycle cooling water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
V.D1.EP-93	V.D1-4(E-19)	Heat exchanger components	Stainless steel	Closed-cycle cooling water	Loss of material due to pitting and crevice corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
V.D1.EP-91	V.D1-5(E-20)	Heat exchanger components	Stainless steel	Raw water	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion; fouling that leads to corrosion	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
V.D1.EP-92	V.D1-6(E-17)	Heat exchanger components	Steel	Closed-cycle cooling water	Loss of material due to general, pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
V.D1.EP-90	V.D1-7(E-18)	Heat exchanger components	Steel	Raw water	Loss of material due to general, pitting, crevice, and microbiologically-influenced corrosion; fouling that leads to corrosion	Chapter XI.M20, "Open-Cycle Cooling Water System"	No

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
V.D1.EP-78	V.D1-8(EP-47)	Heat exchanger tubes	Copper alloy	Lubricating oil	Reduction of heat transfer due to fouling	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
V.D1.EP-96	V.D1-19(EP-35)	Heat exchanger tubes	Stainless steel	Closed-cycle cooling water	Reduction of heat transfer due to fouling	Chapter XI.M21A, "Closed Treated Water Systems"	No
V.D1.EP-79	V.D1-10(EP-50)	Heat exchanger tubes	Stainless steel	Lubricating oil	Reduction of heat transfer due to fouling	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
V.D1.E-21	V.D1-11(E-21)	Heat exchanger tubes	Stainless steel	Raw water	Reduction of heat transfer due to fouling	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
V.D1.EP-75	V.D1-12(EP-40)	Heat exchanger tubes	Steel	Lubricating oil	Reduction of heat transfer due to fouling	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
V.D1.E-43	V.D1-13(E-43)	Motor cooler	Gray cast iron	Treated water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No

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
V.D1.E-24	V.D1-14(E-24)	Orifice (miniflow recirculation)	Stainless steel	Treated water (borated)	Loss of material due to erosion	A plant-specific aging management program is to be evaluated for erosion of the orifice due to extended use of the centrifugal HPSI pump for normal charging. See LER 50-275/94-023 for evidence of erosion.	Yes, plant-specific
V.D1.E-01	V.D1-15(E-01)	Partially-encased tanks with breached moisture barrier	Stainless steel	Raw water	Loss of material due to 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
V.D1.EP-101	V.D2-18(EP-2)	Piping, piping components, and piping elements	Aluminum	Air with borated water leakage	Loss of material due to boric acid corrosion	Chapter XI.M10, “Boric Acid Corrosion”	No
V.D1.E-47	V.D1-16(E-47)	Piping, piping components, and piping elements	Cast austenitic stainless steel	Treated water (borated) >250°C (>482°F)	Loss of fracture toughness due to thermal aging embrittlement	Chapter XI.M12, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)"	No
V.D1.EP-97	V.D1-17(EP-36)	Piping, piping components, and piping elements	Copper alloy	Closed-cycle cooling water	Loss of material due to pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No

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
V.D1.EP-76	V.D1-19(EP-45)	Piping, piping components, and piping elements	Copper alloy	Lubricating oil	Loss of material due to pitting and crevice corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
V.D1.EP-27	V.D1-19(EP-27)	Piping, piping components, and piping elements	Copper alloy (>15% Zn or >8% Al)	Closed-cycle cooling water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
V.D1.EP-52	V.D1-20(EP-52)	Piping, piping components, and piping elements	Gray cast iron	Closed-cycle cooling water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
V.D1.EP-54	V.D1-21(EP-54)	Piping, piping components, and piping elements	Gray cast iron	Soil	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
V.D1.EP-95	V.D1-22(EP-33)	Piping, piping components, and piping elements	Stainless steel	Closed-cycle cooling water	Loss of material due to pitting and crevice corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
V.D1.EP-98	V.D1-23(EP-44)	Piping, piping components, and piping elements	Stainless steel	Closed-cycle cooling water >60°C (>140°F)	Cracking due to stress corrosion cracking	Chapter XI.M21A, "Closed Treated Water Systems"	No
V.D1.EP-80	V.D1-24(EP-51)	Piping, piping components, and piping elements	Stainless steel	Lubricating oil	Loss of material due to pitting and crevice corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
V.D1.EP-55	V.D1-25(EP-55)	Piping, piping components, and piping elements	Stainless steel	Raw water	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M20, "Open-Cycle Cooling Water System"	No

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
V.D1.EP-72	V.D1-26(EP-31)	Piping, piping components, and piping elements	Stainless steel	Soil or concrete	Loss of material due to pitting and crevice corrosion	Chapter XI.M41, "Buried and Underground Piping and Tanks"	No
V.D1.E-13	V.D1-27(E-13)	Piping, piping components, and piping elements	Stainless steel	Treated water (borated)	Cumulative fatigue damage due to fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
V.D1.EP-77	V.D1-28(EP-46)	Piping, piping components, and piping elements	Steel	Lubricating oil	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
V.D1.EP-81	V.D1-29(EP-53)	Piping, piping components, and piping elements (Internal surfaces); tanks	Stainless steel	Condensation (Internal)	Loss of material due to pitting and crevice corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
V.D1.EP-103		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated

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**
V.D1.EP-107		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Loss of material due to pitting and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated
V.D1.EP-41	V.D1-30(EP-41)	Piping, piping components, and piping elements; tanks	Stainless steel	Treated water (borated)	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry”	No
V.D1.E-12	V.D1-31(E-12)	Piping, piping components, and piping elements; tanks	Stainless steel	Treated water (borated) >60°C (>140°F)	Cracking due to stress corrosion cracking	Chapter XI.M2, “Water Chemistry”	No
V.D1.EP-49	V.D1-32(EP-49)	Pump casings	Steel (with stainless steel cladding)	Treated water (borated)	Loss of material due to cladding breach	A plant-specific aging management program is to be evaluated Reference NRC Information Notice 94-63, “Boric Acid Corrosion of Charging Pump Casings Caused by Cladding Cracks."	Yes, verify that plant-specific program addresses clad breach
V.D1.E-38	V.D1-33(E-38)	Safety injection tank (accumulator)	Steel (with stainless steel or nickel-alloy cladding)	Treated water (borated) >60°C (>140°F)	Cracking due to stress corrosion cracking	Chapter XI.M2, “Water Chemistry”	No

D2. EMERGENCY CORE COOLING SYSTEM (BOILING WATER REACTORS)

Systems, Structures, and Components

This section addresses the emergency core cooling systems for boiling water reactors (BWRs) designed to cool the reactor core and provide safe shutdown following a design basis accident. The cooling systems consist of the high-pressure coolant injection (HPCI), reactor core isolation cooling (RCIC), high-pressure core spray (HPCS), automatic depressurization (ADS), low-pressure core spray (LPCS), low-pressure coolant injection (LPCI), and residual heat removal (RHR) systems, including various pumps and valves, the RHR heat exchangers, and the drywell and suppression chamber spray system (DSCSS). The auxiliary area ventilation system includes RCIC, HPCI, RHR, and core spray pump room cooling.

Based on Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” all components that comprise the emergency core cooling system outside the containment are governed by Group B Quality Standards and the portion of the DSCSS inside the containment up to the isolation valve is governed by Group A Quality Standards. Portions of the HPCI, RCIC, HPCS, LPCS, and LPCI (or RHR) systems extending from the reactor vessel up to and including the second containment isolation valve are governed by Group A Quality Standards and covered in IV.C1.

Pumps and valve internals perform their intended functions with moving parts or with a change in configuration. They are also subject to replacement based on qualified life or a specified time period. Pursuant to 10 CFR 54.21(a)(1), therefore, they are not subject to an aging management review.

The system piping includes all pipe sizes, including instrument piping.

Aging management programs for the degradation of external surfaces of components and miscellaneous bolting are included in V.E. Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in VI.F.

System Interfaces

The systems that interface with the emergency core cooling system include the reactor vessel (IV.A1), the reactor coolant pressure boundary (IV.C1), the feedwater system (VIII.D2), the condensate system (VIII.E), the closed-cycle cooling water system (VII.C2), the open-cycle cooling water system (VII.C1), and the ultimate heat sink (VII.C3).

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
V.D2.EP-113	V.D2-1(E-04)	Drywell and suppression chamber spray system (internal surfaces): flow orifice; spray nozzles	Steel	Air – indoor, uncontrolled (Internal)	Loss of material due to general corrosion; fouling that leads to corrosion	A plant-specific aging management program is to be evaluated	Yes, plant-specific
V.D2.E-26	V.D2-2(E-26)	Ducting, piping, and components (External surfaces)	Steel	Air – indoor, uncontrolled (External)	Loss of material due to general corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
V.D2.EP-94	V.D2-3(EP-13)	Heat exchanger components	Copper alloy	Closed-cycle cooling water	Loss of material due to pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
V.D2.EP-37	V.D2-4(EP-37)	Heat exchanger components	Copper alloy (>15% Zn or >8% Al)	Closed-cycle cooling water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
V.D2.EP-93	V.D2-5(E-19)	Heat exchanger components	Stainless steel	Closed-cycle cooling water	Loss of material due to pitting and crevice corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
V.D2.EP-91	V.D2-6(E-20)	Heat exchanger components	Stainless steel	Raw water	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion; fouling that leads to corrosion	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
V.D2.EP-92	V.D2-7(E-17)	Heat exchanger components	Steel	Closed-cycle cooling water	Loss of material due to general, pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
V.D2.EP-90	V.D2-8(E-18)	Heat exchanger components	Steel	Raw water	Loss of material due to general, pitting, crevice, and microbiologically-influenced corrosion; fouling that leads to corrosion	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
V.D2.EP-78	V.D2-9(EP-47)	Heat exchanger tubes	Copper alloy	Lubricating oil	Reduction of heat transfer due to fouling	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
V.D2.EP-96	V.D2-10(EP-35)	Heat exchanger tubes	Stainless steel	Closed-cycle cooling water	Reduction of heat transfer due to fouling	Chapter XI.M21A, "Closed Treated Water Systems"	No
V.D2.EP-79	V.D2-11(EP-50)	Heat exchanger tubes	Stainless steel	Lubricating oil	Reduction of heat transfer due to fouling	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
V.D2.E-21	V.D2-12(E-21)	Heat exchanger tubes	Stainless steel	Raw water	Reduction of heat transfer due to fouling	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
V.D2.EP-74	V.D2-13(EP-34)	Heat exchanger tubes	Stainless steel	Treated water	Reduction of heat transfer due to fouling	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
V.D2.EP-75	V.D2-14(EP-40)	Heat exchanger tubes	Steel	Lubricating oil	Reduction of heat transfer due to fouling	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
V.D2.E-23	V.D2-15(E-23)	Heat exchanger tubes	Steel	Raw water	Reduction of heat transfer due to fouling	Chapter XI.M20, "Open-Cycle Cooling Water System"	No

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
V.D2.E-29	V.D2-16(E-29)	Piping and components (Internal surfaces)	Steel	Air – indoor, uncontrolled (Internal)	Loss of material due to general corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
V.D2.E-27	V.D2-17(E-27)	Piping and components (Internal surfaces)	Steel	Condensation (Internal)	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
V.D2.EP-71	V.D2-19(EP-26)	Piping, piping components, and piping elements	Aluminum	Treated water	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
V.D2.E-11	V.D2-20(E-11)	Piping, piping components, and piping elements	Cast austenitic stainless steel	Treated water >250°C (>482°F)	Loss of fracture toughness due to thermal aging embrittlement	Chapter XI.M12, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)"	No
V.D2.EP-97	V.D2-21(EP-36)	Piping, piping components, and piping elements	Copper alloy	Closed-cycle cooling water	Loss of material due to pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
V.D2.EP-76	V.D2-22(EP-45)	Piping, piping components, and piping elements	Copper alloy	Lubricating oil	Loss of material due to pitting and crevice corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
V.D2.EP-27	V.D2-23(EP-27)	Piping, piping components, and piping elements	Copper alloy (>15% Zn or >8% Al)	Closed-cycle cooling water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
V.D2.EP-54	V.D2-24(EP-54)	Piping, piping components, and piping elements	Gray cast iron	Soil	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No

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
V.D2.EP-95	V.D2-25(EP-33)	Piping, piping components, and piping elements	Stainless steel	Closed-cycle cooling water	Loss of material due to pitting and crevice corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
V.D2.EP-98	V.D2-26(EP-44)	Piping, piping components, and piping elements	Stainless steel	Closed-cycle cooling water >60°C (>140°F)	Cracking due to stress corrosion cracking	Chapter XI.M21A, "Closed Treated Water Systems"	No
V.D2.EP-72	V.D2-27(EP-31)	Piping, piping components, and piping elements	Stainless steel	Soil or concrete	Loss of material due to pitting and crevice corrosion	Chapter XI.M41, "Buried and Underground Piping and Tanks"	No
V.D2.EP-73	V.D2-28(EP-32)	Piping, piping components, and piping elements	Stainless steel	Treated water	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
V.D2.E-37	V.D2-29(E-37)	Piping, piping components, and piping elements	Stainless steel	Treated water >60°C (>140°F)	Cracking due to stress corrosion cracking, intergranular stress corrosion cracking	Chapter XI.M7, “BWR Stress Corrosion Cracking,” and Chapter XI.M2, “Water Chemistry”	No
V.D2.EP-77	V.D2-30(EP-46)	Piping, piping components, and piping elements	Steel	Lubricating oil	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
V.D2.E-07	V.D2-31(E-07)	Piping, piping components, and piping elements	Steel	Steam	Wall thinning due to flow-accelerated corrosion	Chapter XI.M17, “Flow-Accelerated Corrosion”	No

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
V.D2.E-10	V.D2-32(E-10)	Piping, piping components, and piping elements	Steel	Treated water	Cumulative fatigue damage due to fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
V.D2.EP-60	V.D2-33(E-08)	Piping, piping components, and piping elements	Steel	Treated water	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
V.D2.E-09	V.D2-34(E-09)	Piping, piping components, and piping elements	Steel	Treated water	Wall thinning due to flow-accelerated corrosion	Chapter XI.M17, “Flow-Accelerated Corrosion”	No
V.D2.EP-61	V.D2-35(E-14)	Piping, piping components, and piping elements (Internal surfaces)	Stainless steel	Condensation (Internal)	Loss of material due to pitting and crevice corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
V.D2.EP-103		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated
V.D2.EP-107		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Loss of material due to pitting and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated

E. EXTERNAL SURFACES OF COMPONENTS AND MISCELLANEOUS BOLTING

Systems, Structures, and Components

This section addresses the aging management programs for the degradation of external surfaces of all steel structures and components, including closure boltings in the engineered safety features in pressurized water reactors (PWRs) and boiling water reactors (BWRs). For the steel components in PWRs, this section addresses only boric acid corrosion of external surfaces as a result of dripping borated water leaking from an adjacent PWR component. Boric acid corrosion can also occur for 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.

System Interfaces

The structures and components covered in this section belong to the engineered safety features in PWRs and BWRs. (For example, see System Interfaces in V.A to V.D2 for details.)

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
V.E.EP-116		Bolting	Copper alloy	Any environment	Loss of preload due to thermal effects, gasket creep, and self-loosening	Chapter XI.M18, "Bolting Integrity"	No
V.E.EP-117		Bolting	Nickel alloy	Any environment	Loss of preload due to thermal effects, gasket creep, and self-loosening	Chapter XI.M18, "Bolting Integrity"	No
V.E.EP-120		Bolting	Stainless steel	Treated borated water	Loss of preload due to thermal effects, gasket creep, and self-loosening	Chapter XI.M18, "Bolting Integrity"	No
V.E.E-41	V.E-2(E-41)	Bolting	Steel	Air with borated water leakage	Loss of material due to boric acid corrosion	Chapter XI.M10, “Boric Acid Corrosion”	No
V.E.EP-64	V.E-1(EP-1)	Bolting	Steel; stainless steel	Air – outdoor (External)	Loss of material due to general (steel only), pitting, and crevice corrosion	Chapter XI.M18, "Bolting Integrity"	No
V.E.EP-118		Bolting	Steel; stainless steel	Air – outdoor (External)	Loss of preload due to thermal effects, gasket creep, and self-loosening	Chapter XI.M18, "Bolting Integrity"	No

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
V.E.EP-121		Bolting	Steel; stainless steel	Fuel oil	Loss of preload due to thermal effects, gasket creep, and self-loosening	Chapter XI.M18, "Bolting Integrity"	No
V.E.EP-119		Bolting	Steel; stainless steel	Raw water	Loss of preload due to thermal effects, gasket creep, and self-loosening	Chapter XI.M18, "Bolting Integrity"	No
V.E.EP-122		Bolting	Steel; stainless steel	Treated water	Loss of preload due to thermal effects, gasket creep, and self-loosening	Chapter XI.M18, "Bolting Integrity"	No
V.E.E-02	V.E-6(E-02)	Closure bolting	Steel	Air with steam or water leakage	Loss of material due to general corrosion	Chapter XI.M18, "Bolting Integrity"	No
V.E.E-03	V.E-3(E-03)	Closure bolting	Steel, high-strength	Air with steam or water leakage	Cracking due to cyclic loading, stress corrosion cracking	Chapter XI.M18, "Bolting Integrity"	No
V.E.EP-70	V.E-4(EP-25)	Closure bolting	Steel; stainless steel	Air – indoor, uncontrolled (External)	Loss of material due to general (steel only), pitting, and crevice corrosion	Chapter XI.M18, "Bolting Integrity"	No

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
V.E.EP-69	V.E-5(EP-24)	Closure bolting	Steel; stainless steel	Air – indoor, uncontrolled (External)	Loss of preload due to thermal effects, gasket creep, and self-loosening	Chapter XI.M18, "Bolting Integrity"	No
V.E.E-44	V.E-7(E-44)	External surfaces	Steel	Air – indoor, uncontrolled (External)	Loss of material due to general corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
V.E.E-45	V.E-8(E-45)	External surfaces	Steel	Air – outdoor (External)	Loss of material due to general corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
V.E.E-28	V.E-9(E-28)	External surfaces	Steel	Air with borated water leakage	Loss of material due to boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No
V.E.E-46	V.E-10(E-46)	External surfaces	Steel	Condensation (External)	Loss of material due to general corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
V.E.EP-114		Piping, piping components, and piping elements	Aluminum	Air - outdoor	Loss of material due to pitting and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
V.E.EP-38	V.E-11(EP-38)	Piping, piping components, and piping elements	Copper alloy (>15% Zn or >8% Al)	Air with borated water leakage	Loss of material due to boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No
V.E.EP-123		Underground piping, piping components, and piping elements	Steel; stainless steel	Air-indoor, uncontrolled (External) or condensation (External)	Loss of material due to general (steel only), pitting and crevice corrosion	Chapter XI.M41, "Buried and Underground Piping and Tanks"	No

F. COMMON MISCELLANEOUS MATERIAL/ENVIRONMENT COMBINATIONS

Systems, Structures, and Components

This section addresses the aging management programs for miscellaneous material/environment combinations which may be found throughout the emergency safety feature system's structures and components. For the material/environment combinations in this part, aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation and, therefore, no resulting aging management programs for these structures and components are required.

System Interfaces

The structures and components covered in this section belong to the engineered safety features in pressurized water reactors (PWRs) and boiling water reactors (BWRs). (For example, see System Interfaces in V.A to V.D2 for details.)

V ENGINEERED SAFETY FEATURES
F Common Miscellaneous Material/Environment Combinations

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
V.F.EP-14	V.F-1(EP-14)	Ducting, piping, and components	Galvanized steel	Air – indoor, controlled (External)	None	None	No
V.F.EP-15	V.F-6(EP-15)	Piping elements	Glass	Air – indoor, uncontrolled (External)	None	None	No
V.F.EP-87		Piping elements	Glass	Air – outdoor	None	None	No
V.F.EP-65		Piping elements	Glass	Air with borated water leakage	None	None	No
V.F.EP-68		Piping elements	Glass	Closed-cycle cooling water	None	None	No
V.F.EP-66		Piping elements	Glass	Condensation (Internal/External)	None	None	No
V.F.EP-67		Piping elements	Glass	Gas	None	None	No
V.F.EP-16	V.F-7(EP-16)	Piping elements	Glass	Lubricating oil	None	None	No
V.F.EP-28	V.F-8(EP-28)	Piping elements	Glass	Raw water	None	None	No

V ENGINEERED SAFETY FEATURES
F Common Miscellaneous Material/Environment Combinations

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
V.F.EP-29	V.F-10(EP-29)	Piping elements	Glass	Treated water	None	None	No
V.F.EP-30	V.F-9(EP-30)	Piping elements	Glass	Treated water (borated)	None	None	No
V.F.EP-3	V.F-2(EP-3)	Piping, piping components, and piping elements	Aluminum	Air – indoor, uncontrolled (Internal/External)	None	None	No
V.F.EP-10	V.F-3(EP-10)	Piping, piping components, and piping elements	Copper alloy	Air – indoor, uncontrolled (External)	None	None	No
V.F.EP-9	V.F-4(EP-9)	Piping, piping components, and piping elements	Copper alloy	Gas	None	None	No
V.F.EP-12	V.F-5(EP-12)	Piping, piping components, and piping elements	Copper alloy (≤15% Zn and ≤8% Al)	Air with borated water leakage	None	None	No
V.F.EP-17	V.F-11(EP-17)	Piping, piping components, and piping elements	Nickel alloy	Air – indoor, uncontrolled (External)	None	None	No
V.F.EP-115		Piping, piping components, and piping elements	Nickel alloy	Air with borated water leakage	None	None	No

V ENGINEERED SAFETY FEATURES
F Common Miscellaneous Material/Environment Combinations

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
V.F.EP-18	V.F-12(EP-18)	Piping, piping components, and piping elements	Stainless steel	Air – indoor, uncontrolled (External)	None	None	No
V.F.EP-82		Piping, piping components, and piping elements	Stainless steel	Air – indoor, uncontrolled (Internal)	None	None	No
V.F.EP-19	V.F-13(EP-19)	Piping, piping components, and piping elements	Stainless steel	Air with borated water leakage	None	None	No
V.F.EP-20	V.F-14(EP-20)	Piping, piping components, and piping elements	Stainless steel	Concrete	None	None	No
V.F.EP-22	V.F-15(EP-22)	Piping, piping components, and piping elements	Stainless steel	Gas	None	None	No
V.F.EP-4	V.F-16(EP-4)	Piping, piping components, and piping elements	Steel	Air – indoor, controlled (External)	None	None	No
V.F.EP-112	V.F-17(EP-5)	Piping, piping components, and piping elements	Steel	Concrete	None	None, provided 1) attributes of the concrete are consistent with ACI 318 or ACI 349 (low water-to-cement ratio, low permeability, and adequate air entrainment) as cited in NUREG-1557, and 2) plant OE indicates no degradation of the concrete	No, if conditions are met.

V ENGINEERED SAFETY FEATURES
F Common Miscellaneous Material/Environment Combinations

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
V.F.EP-7	V.F-18(EP-7)	Piping, piping components, and piping elements	Steel	Gas	None	None	No

CHAPTER VI

ELECTRICAL COMPONENTS

ELECTRICAL COMPONENTS

A. Equipment Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

B. Equipment Subject to 10 CFR 50.49 Environmental Qualification Requirements

A. EQUIPMENT NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS

Systems, Structures, and Components

This section addresses electrical cables and connections that are not subject to the environmental qualification requirements of 10 CFR 50.49 and that are installed in power and instrumentation and control (I&C) applications. The power cables and connections addressed are low-voltage (<1000 volts) and medium-voltage (2 kilovolts [kV] to 35 kV). High voltage (> 35 kV) power cables and connections have unique, specialized constructions and must be evaluated on a plant-specific basis.

This section also addresses components that are relied upon to meet the station blackout (SBO) requirements for restoration of offsite power. The offsite power system relied upon in the plant-specific current licensing basis for compliance with 10 CFR 50.63, that is used to connect the plant to the offsite power source, is included in the SBO restoration equipment scope. The electrical distribution equipment out to the first circuit breaker with the offsite distribution system (i.e., equipment in the switchyard) should be included within the SBO restoration equipment scope. This path typically includes the circuit breakers that connect to the offsite system power transformers (startup transformers), the transformers themselves, the intervening overhead or underground circuits between circuit breaker and transformer and transformer and onsite electrical distribution system, and associated control circuits and structures. However, the staff's review is based on the plant-specific current licensing basis, regulatory requirements, and offsite power design configurations.

Electrical cables and their required terminations (i.e., connections) are typically reviewed as a single commodity. The types of connections included in this review are splices, mechanical connectors, fuse holders, and terminal blocks. This common review is translated into program actions, which treat cables and connections in the same manner.

Electrical cables and connections that are in the plant’s environmental qualification (EQ) program are addressed in VI.B.

System Interfaces

Electrical cables and connections functionally interface with all plant systems that rely on electric power or instrumentation and control. Electrical cables and connections also interface with and are supported by structural commodities (e.g., cable trays, conduit, cable trenches, cable troughs, duct banks, cable vaults, and manholes) that are reviewed, as appropriate, in the Systems, Structures, and Components section.

VI ELECTRICAL COMPONENTS
A Equipment Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VI.A.LP-30	VI.A-1(LP-12)	Cable connections (metallic parts)	Various metals used for electrical contacts	Air – indoor, controlled or uncontrolled or Air – outdoor	Increased resistance of connection due to thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination, corrosion, and oxidation	Chapter XI.E6, "Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements"	No
VI.A.LP-35	VI.A-4(L-03)	Conductor insulation for inaccessible power cables greater than or equal to 400 volts (e.g., installed in conduit or direct buried)	Various organic polymers (e.g., EPR, SR, EPDM, XLPE)	Adverse localized environment caused by significant moisture	Reduced insulation resistance due to moisture	Chapter XI.E3, "Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements"	No
VI.A.LP-36	VI.A-5(L-04)	Connector contacts for electrical connectors exposed to borated water leakage	Various metals used for electrical contacts	Air with borated water leakage	Increased resistance of connection due to corrosion of connector contact surfaces caused by intrusion of borated water	Chapter XI.M10, “Boric Acid Corrosion”	No
VI.A.LP-24	VI.A-7(LP-02)	Fuse holders (not part of active equipment): insulation material	Insulation material: bakelite; phenolic melamine or ceramic; molded polycarbonate; other	Air – indoor, controlled or uncontrolled	None	None	No

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VI.A.LP-31	VI.A-8(LP-01)	Fuse holders (not part of active equipment): metallic clamps	Various metals used for electrical connections	Air – indoor, controlled or uncontrolled	Increased resistance of connection due to fatigue caused by frequent manipulation or vibration	Chapter XI.E5, "Fuse Holders" No aging management program is required for those applicants who can demonstrate these fuse holders are located in an environment that does not subject them to environmental aging mechanisms or fatigue caused by frequent manipulation or vibration	No
VI.A.LP-23	VI.A-8(LP-01)	Fuse holders (not part of active equipment): metallic clamps	Various metals used for electrical connections	Air – indoor, uncontrolled	Increased resistance of connection due to chemical contamination, corrosion, and oxidation (in an air, indoor controlled environment, increased resistance of connection due to chemical contamination, corrosion and oxidation do not apply); fatigue due to ohmic heating, thermal cycling, electrical transients	Chapter XI.E5, "Fuse Holders"	No
VI.A.LP-32	VI.A-10(LP-11)	High-voltage insulators	Porcelain; malleable iron; aluminum; galvanized steel; cement	Air – outdoor	Loss of material due to mechanical wear caused by wind blowing on transmission conductors	A plant-specific aging management program is to be evaluated	Yes, plant-specific

VI ELECTRICAL COMPONENTS
A Equipment Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VI.A.LP-28	VI.A-9(LP-07)	High-voltage insulators	Porcelain; malleable iron; aluminum; galvanized steel; cement	Air – outdoor	Reduced insulation resistance due to presence of salt deposits or surface contamination	A plant-specific aging management program is to be evaluated for plants located such that the potential exists for salt deposits or surface contamination (e.g., in the vicinity of salt water bodies or industrial pollution)	Yes, plant-specific
VI.A.LP-33	VI.A-2(L-01)	Insulation material for electrical cables and connections (including terminal blocks, fuse holders, etc.)	Various organic polymers (e.g., EPR, SR, EPDM, XLPE)	Adverse localized environment caused by heat, radiation, or moisture	Reduced insulation resistance due to thermal/thermoxidative degradation of organics, radiolysis, and photolysis (UV sensitive materials only) of organics; radiation-induced oxidation; moisture intrusion	Chapter XI.E1, "Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements"	No
VI.A.LP-34	VI.A-3(L-02)	Insulation material for electrical cables and connections used in instrumentation circuits that are sensitive to reduction in conductor insulation resistance (IR)	Various organic polymers (e.g., EPR, SR, EPDM, XLPE)	Adverse localized environment caused by heat, radiation, or moisture	Reduced insulation resistance due to thermal/thermoxidative degradation of organics, radiolysis, and photolysis (UV sensitive materials only) of organics; radiation-induced oxidation; moisture intrusion	Chapter XI.E2, "Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits"	No

VI ELECTRICAL COMPONENTS
A Equipment Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VI.A.LP-25	VI.A-11(LP-04)	Metal enclosed bus: bus/connections	Various metals used for electrical bus and connections	Air – indoor, controlled or uncontrolled or Air – outdoor	Increased resistance of connection due to the loosening of bolts caused by thermal cycling and ohmic heating	Chapter XI.E4, "Metal Enclosed Bus"	No
VI.A.LP-29	VI.A-12(LP-10)	Metal enclosed bus: enclosure assemblies	Elastomers	Air – indoor, controlled or uncontrolled or Air – outdoor	Surface cracking, crazing, scuffing, dimensional change (e.g. “ballooning” and “necking”), shrinkage, discoloration, hardening and loss of strength due to elastomer degradation	Chapter XI.E4, "Metal Enclosed Bus," or Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
VI.A.LP-41	VI.A-13(LP-06)	Metal enclosed bus: external surface of enclosure assemblies	Galvanized steel; aluminum	Air – indoor, controlled or uncontrolled	None	None	No
VI.A.LP-42	VI.A-13(LP-06)	Metal enclosed bus: external surface of enclosure assemblies	Galvanized steel; aluminum	Air – outdoor	Loss of material due to pitting and crevice corrosion	Chapter XI.E4, "Metal Enclosed Bus," or Chapter XI.S6, "Structures Monitoring"	No
VI.A.LP-44	VI.A-13(LP-06)	Metal enclosed bus: external surface of enclosure assemblies	Steel	Air – indoor, controlled	None	None	No
VI.A.LP-43	VI.A-13(LP-06)	Metal enclosed bus: external surface of enclosure assemblies	Steel	Air – indoor, uncontrolled or Air – outdoor	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.E4, "Metal Enclosed Bus," or Chapter XI.S6, "Structures Monitoring"	No

VI ELECTRICAL COMPONENTS
A Equipment Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VI.A.LP-26	VI.A-14(LP-05)	Metal enclosed bus: insulation; insulators	Porcelain; xenoy; thermo-plastic organic polymers	Air – indoor, controlled or uncontrolled or Air – outdoor	Reduced insulation resistance due to thermal/ thermoxidative degradation of organics/ thermoplastics, radiation-induced oxidation, moisture/debris intrusion, and ohmic heating	Chapter XI.E4, "Metal Enclosed Bus"	No
VI.A.LP-39	VI.A-15(LP-09)	Switchyard bus and connections	Aluminum; copper; bronze; stainless steel; galvanized steel	Air – outdoor	Loss of material due to wind-induced abrasion; Increased resistance of connection due to oxidation or loss of pre-load	A plant-specific aging management program is to be evaluated	Yes, plant-specific
VI.A.LP-46	VI.A-16(LP-08)	Transmission conductors	Aluminum	Air – outdoor	Loss of conductor strength due to corrosion	None - for Aluminum Conductor Aluminum Alloy Reinforced (ACAR)	None
VI.A.LP-38	VI.A-16(LP-08)	Transmission conductors	Aluminum; steel	Air – outdoor	Loss of conductor strength due to corrosion	A plant-specific aging management program is to be evaluated for Aluminum Conductor Steel Reinforced (ACSR)	Yes, plant-specific
VI.A.LP-47	VI.A-16(LP-08)	Transmission conductors	Aluminum; Steel	Air – outdoor	Loss of material due to wind-induced abrasion	A plant-specific aging management program is to be evaluated for ACAR and ACSR	Yes, plant-specific
VI.A.LP-48	VI.A-16(LP-08)	Transmission connectors	Aluminum; steel	Air – outdoor	Increased resistance of connection due to oxidation or loss of pre-load	A plant-specific aging management program is to be evaluated	Yes, plant-specific

B. EQUIPMENT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS

Systems, Structures, and Components

The Nuclear Regulatory Commission (NRC) has established nuclear station environmental qualification (EQ) requirements in 10 CFR Part 50 Appendix A, Criterion 4, and in 10 CFR 50.49. 10 CFR 50.49 specifically requires that an EQ program be established to demonstrate that certain electrical components located in harsh plant environments (i.e., those areas of the plant that could be subject to the harsh environmental effects of a loss of coolant accident [LOCA], high energy line breaks [HELBs] or post-LOCA radiation) are qualified to perform their safety function in those harsh environments after the effects of inservice aging. 10 CFR 50.49 requires that the effects of significant aging mechanisms be addressed as part of environmental qualification. Components in the EQ program have a qualified life, and the components are replaced at the end of that qualified life if it is shorter than the current operating term. The qualified life may be extended by methods such as refurbishment or reanalysis, but the licensee is required by the EQ regulation (10 CFR 50.49) to replace the component when its qualified life has expired.

Similarly, some nuclear power plants have mechanical equipment that was qualified in accordance with the provisions of Criterion 4 of Appendix A to 10 CFR Part 50.

System Interfaces

Equipment subject to 10 CFR 50.49 environmental qualification requirements functionally interfaces with all plant systems that rely on electric power or instrumentation and control.

VI ELECTRICAL COMPONENTS
B Equipment Subject to 10 CFR 50.49 Environmental Qualification Requirements

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VI.B.L-05	VI.B-1(L-05)	Electrical equipment subject to 10 CFR 50.49 EQ requirements	Various polymeric and metallic materials	Adverse localized environment caused by heat, radiation, oxygen, moisture, or voltage	Various aging effects due to various mechanisms in accordance with 10 CFR 50.49	EQ is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the Standard Review Plan, Section 4.4, "Environmental Qualification (EQ) of Electrical Equipment," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1)(i) and (ii). See Chapter X.E1, "Environmental Qualification (EQ) of Electric Components," of this report for meeting the requirements of 10 CFR 54.21(c)(1)(iii).	Yes, TLAA

CHAPTER VII

AUXILIARY SYSTEMS

MAJOR PLANT SECTIONS

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A2.	Spent Fuel Storage
A3.	Spent Fuel Pool Cooling and Cleanup (PWR)
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E4.	Shutdown Cooling System (Older BWR)
E5.	Waste Water Systems
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I.	External Surfaces of Components and Miscellaneous Bolting
J.	Common Miscellaneous Material/Environment Combinations

A1. NEW FUEL STORAGE

Systems, Structures, and Components

This section discusses those structures and components used for new fuel storage which include carbon steel new fuel storage racks located in the auxiliary building or the fuel handling building. The racks are exposed to the temperature and humidity in the auxiliary building. The racks are generally painted with a protective coating. Based on Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” all components used for new fuel storage are governed by Group C Quality Standards.

Aging management programs for the degradation of external surfaces of components and miscellaneous bolting are included in VII.I. Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in VII.J.

System Interfaces

No other systems discussed in this report interface with those used for new fuel storage.

VII AUXILIARY SYSTEMS
A1 New Fuel Storage

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.A1.A-94	VII.A1-1(A-94)	Structural steel	Steel	Air – indoor, uncontrolled (External)	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.S6, "Structures Monitoring"	No

A2. SPENT FUEL STORAGE

Systems, Structures, and Components

This section discusses those structures and components used for spent fuel storage and includes stainless steel spent fuel storage racks and neutron-absorbing materials (e.g., Boraflex, Boral®, or boron-steel sheets, if used) submerged in chemically treated oxygenated boiling water reactor (BWR) or borated pressurized water reactor (PWR) water. The intended function of a spent fuel rack is to separate spent fuel assemblies. Boraflex sheets fastened to the storage cells provide for neutron absorption and help maintain subcriticality of spent fuel assemblies in the spent fuel pool.

Based on Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” all components used for spent fuel storage are governed by Group C Quality Standards. In some plants, the Boraflex has been replaced by Boral® or boron steel.

Aging management programs for the degradation of external surfaces of components and miscellaneous bolting are included in VII.I. Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in VII.J.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

No other systems discussed in this report interface with those used for spent fuel storage.

VII AUXILIARY SYSTEMS
A2 Spent Fuel Storage

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.A2.AP-79	VII.A2-1(AP-79)	Piping, piping components, and piping elements	Steel (with stainless steel cladding); stainless steel	Treated borated water	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry”	No
VII.A2.A-96	VII.A2-6(A-96)	Spent fuel storage racks (BWR)	Stainless steel	Treated water >60°C (>140°F)	Cracking due to stress corrosion cracking	Chapter XI.M2, “Water Chemistry”	No
VII.A2.A-97	VII.A2-7(A-97)	Spent fuel storage racks (PWR)	Stainless steel	Treated borated water >60°C (>140°F)	Cracking due to stress corrosion cracking	Chapter XI.M2, “Water Chemistry”	No
VII.A2.A-87	VII.A2-2(A-87)	Spent fuel storage racks: neutron-absorbing sheets (BWR)	Boraflex	Treated water	Reduction of neutron-absorbing capacity due to boraflex degradation	Chapter XI.M22, “Boraflex Monitoring”	No
VII.A2.AP-236	VII.A2-3(A-89)	Spent fuel storage racks: neutron-absorbing sheets (BWR)	Boral®; boron steel, and other materials (excluding Boraflex)	Treated water	Reduction of neutron-absorbing capacity; change in dimensions and loss of material due to effects of SFP environment	Chapter XI.M40, "Monitoring of Neutron-Absorbing Materials other than Boraflex"	No
VII.A2.A-86	VII.A2-4(A-86)	Spent fuel storage racks: neutron-absorbing sheets (PWR)	Boraflex	Treated borated water	Reduction of neutron-absorbing capacity due to boraflex degradation	Chapter XI.M22, “Boraflex Monitoring”	No

VII AUXILIARY SYSTEMS
A2 Spent Fuel Storage

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.A2.AP-235	VII.A2-5(A-88)	Spent fuel storage racks: neutron-absorbing sheets (PWR)	Boral®; boron steel, and other materials (excluding Boraflex)	Treated borated water	Reduction of neutron-absorbing capacity; change in dimensions and loss of material due to effects of SFP environment	Chapter XI.M40, "Monitoring of Neutron-Absorbing Materials other than Boraflex"	No

A3. SPENT FUEL POOL COOLING AND CLEANUP (PRESSURIZED WATER REACTOR)

Systems, Structures, and Components

This section discusses the pressurized water reactor (PWR) spent fuel pool cooling and cleanup system and consists of piping, valves, heat exchangers, filters, linings, demineralizers, and pumps. The system contains borated water. The system removes heat from the spent fuel pool and transfers heat to the closed-cycle cooling water system, which in turn transfers heat to the open-cycle cooling water system. Based on Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” all components that comprise the PWR spent fuel pool cooling and cleanup system are governed by Group C Quality Standards.

With respect to filters, these items are to be addressed consistent with the Nuclear Regulatory Commission (NRC) position on consumables, provided in the NRC letter from Christopher I. Grimes to Douglas J. Walters of the Nuclear Energy Institute (NEI), dated March 10, 2000. Specifically, components that function as system filters 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 an aging management review under 10 CFR 54.21(a)(1)(ii). As part of the methodology description, the application should identify the standards that are relied on for replacement, for example, National Fire Protection Association (NFPA) standards for fire protection equipment.

Pump and valve internals perform their intended functions with moving parts or with a change in configuration. They are also subject to replacement based on qualified life or specified time period. Pursuant to 10 CFR 54.21(a)(1), therefore, they are not subject to an aging management review.

Aging management programs for the degradation of external surfaces of components and miscellaneous bolting are included in VII.I. Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in VII.J.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The systems that interface with the PWR spent fuel cooling and cleanup system are the PWR emergency core cooling system (V.D1), the closed-cycle cooling water system (VII.C2), and the PWR chemical and volume control system (VII.E1).

VII AUXILIARY SYSTEMS
A3 Spent Fuel Pool Cooling and Cleanup (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.A3.AP-100	VII.A3-1(A-15)	Elastomers, linings	Elastomers	Treated borated water	Hardening and loss of strength due to elastomer degradation	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
VII.A3.A-79	VII.A3-2(A-79)	External surfaces	Steel	Air with borated water leakage	Loss of material due to boric acid corrosion	Chapter XI.M10, “Boric Acid Corrosion”	No
VII.A3.AP-189	VII.A3-3(A-63)	Heat exchanger components	Steel	Closed-cycle cooling water	Loss of material due to general, pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.A3.AP-1	VII.A3-4(AP-1)	Piping, piping components, and piping elements	Aluminum	Air with borated water leakage	Loss of material due to boric acid corrosion	Chapter XI.M10, “Boric Acid Corrosion”	No
VII.A3.AP-199	VII.A3-5(AP-12)	Piping, piping components, and piping elements	Copper alloy	Closed-cycle cooling water	Loss of material due to general, pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.A3.AP-43	VII.A3-6(AP-43)	Piping, piping components, and piping elements	Copper alloy (>15% Zn or >8% Al)	Closed-cycle cooling water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VII.A3.AP-31	VII.A3-7(AP-31)	Piping, piping components, and piping elements	Gray cast iron	Treated water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No

VII AUXILIARY SYSTEMS
A3 Spent Fuel Pool Cooling and Cleanup (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.A3.AP-107	VII.A3-9(A-39)	Piping, piping components, and piping elements	Steel (with elastomer lining)	Treated water	Loss of material due to pitting and crevice corrosion (only for steel after lining/cladding degradation)	Chapter XI.M2, "Water Chemistry," and Chapter XI.M32, "One-Time Inspection"	No
VII.A3.AP-79	VII.A3-8(AP-79)	Piping, piping components, and piping elements	Steel (with stainless steel cladding); stainless steel	Treated borated water	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, "Water Chemistry"	No
VII.A3.A-56	VII.A3-10(A-56)	Piping, piping components, and piping elements	Steel (with stainless steel or nickel-alloy cladding)	Treated borated water >60°C (>140°F)	Cracking due to stress corrosion cracking	Chapter XI.M2, "Water Chemistry"	No

A4. SPENT FUEL POOL COOLING AND CLEANUP (BOILING WATER REACTOR)

Systems, Structures, and Components

This section discusses the boiling water reactor (BWR) spent fuel pool cooling and cleanup system and consists of piping, valves, heat exchangers, filters, linings, demineralizers, and pumps. The system contains chemically treated oxygenated water. The system removes heat from the spent fuel pool and transfers the heat to the closed-cycle cooling water system, which in turn transfers the heat to the open-cycle cooling water system. Based on Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” all components that comprise the BWR spent fuel pool cooling and cleanup system are governed by Group C Quality Standards.

With respect to filters, these items are to be addressed consistent with the Nuclear Regulatory Commission (NRC) position on consumables, provided in the NRC letter from Christopher I. Grimes to Douglas J. Walters of the Nuclear Energy Institute (NEI), dated March 10, 2000. Specifically, components that function as system filters 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 an aging management review under 10 CFR 54.21(a)(1)(ii). As part of the methodology description, the application should identify the standards that are relied on for replacement, for example, National Fire Protection Association (NFPA) standards for fire protection equipment.

Pump and valve internals perform their intended functions with moving parts or with a change in configuration. They are also subject to replacement based on qualified life or a specified time period. Pursuant to 10 CFR 54.21(a)(1), therefore, they are not subject to an aging management review.

Aging management programs for the degradation of external surfaces of components and miscellaneous bolting are included in VII.I. Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in VII.J.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The systems that interface with the BWR spent fuel cooling and cleanup system are the closed-cycle cooling water system (VII.C2) and the condensate system (VIII.E).

VII AUXILIARY SYSTEMS
A4 Spent Fuel Pool Cooling and Cleanup (BWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.A4.AP-101	VII.A4-1(A-16)	Elastomers, linings	Elastomers	Treated water	Hardening and loss of strength due to elastomer degradation	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
VII.A4.AP-111	VII.A4-2(A-70)	Heat exchanger components	Stainless steel; steel with stainless steel cladding	Treated water	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VII.A4.AP-189	VII.A4-3(A-63)	Heat exchanger components	Steel	Closed-cycle cooling water	Loss of material due to general, pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.A4.AP-139	VII.A4-4(AP-62)	Heat exchanger tubes	Stainless steel	Treated water	Reduction of heat transfer due to fouling	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection"	No
VII.A4.AP-130	VII.A4-5(AP-38)	Piping, piping components, and piping elements	Aluminum	Treated water	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VII.A4.AP-199	VII.A4-6(AP-12)	Piping, piping components, and piping elements	Copper alloy	Closed-cycle cooling water	Loss of material due to general, pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.A4.AP-140	VII.A4-7(AP-64)	Piping, piping components, and piping elements	Copper alloy	Treated water	Loss of material due to general, pitting, crevice, and galvanic corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No

VII AUXILIARY SYSTEMS
A4 Spent Fuel Pool Cooling and Cleanup (BWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.A4.AP-43	VII.A4-8(AP-43)	Piping, piping components, and piping elements	Copper alloy (>15% Zn or >8% Al)	Closed-cycle cooling water	Loss of material due to selective leaching	Chapter XI.M33, "Selective Leaching"	No
VII.A4.AP-32	VII.A4-9(AP-32)	Piping, piping components, and piping elements	Copper alloy (>15% Zn or >8% Al)	Treated water	Loss of material due to selective leaching	Chapter XI.M33, "Selective Leaching"	No
VII.A4.AP-31	VII.A4-10(AP-31)	Piping, piping components, and piping elements	Gray cast iron	Treated water	Loss of material due to selective leaching	Chapter XI.M33, "Selective Leaching"	No
VII.A4.AP-110	VII.A4-11(A-58)	Piping, piping components, and piping elements	Stainless steel	Treated water	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, "Water Chemistry," and Chapter XI.M32, "One-Time Inspection"	No
VII.A4.AP-108	VII.A4-12(A-40)	Piping, piping components, and piping elements	Steel (with elastomer lining or stainless steel cladding)	Treated water	Loss of material due to pitting and crevice corrosion (only for steel after lining/cladding degradation)	Chapter XI.M2, "Water Chemistry," and Chapter XI.M32, "One-Time Inspection"	No

A5. SUPPRESSION POOL CLEANUP SYSTEM (BOILING WATER REACTOR)

Systems, Structures, and Components

This section discusses the suppression pool cleanup system, which maintains water quality in the suppression pool in boiling water reactors (BWRs). The components of this system include piping, filters, valves, and pumps. These components are fabricated of carbon, low-alloy, or austenitic stainless steel. Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," the components that comprise the suppression pool cleanup system are governed by the same Group C Quality Standards Group as the corresponding components in the spent fuel pool cooling and cleanup system (VII.A4).

Pump and valve internals perform their intended functions with moving parts or with a change in configuration. They are also subject to replacement based on qualified life or specified time period. Pursuant to 10 CFR 54.21(a)(1), therefore, they are not subject to an aging management review.

Aging management programs for the degradation of external surfaces of components and miscellaneous bolting are included in VII.I. Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in VII.J.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The system that interfaces with the suppression pool cleanup system is the BWR containments (II.B), or BWR emergency core cooling system (V.D2).

Evaluation Summary

There are no tables associated with this section because the suppression pool cleanup system in BWRs is similar to the spent fuel pool cooling and cleanup system (VII.A4), and the components in the two systems are identical or very similar. Therefore, the reader is referred to the section for the spent fuel storage pool system for a listing of aging effects, aging mechanisms, and aging management programs that are to be applied to the suppression pool cleanup system components. (The only component in VII.A4 that may not be applicable to the suppression pool cleanup system is the heat exchanger [AMR line-items VII.A4.AP-111, VII.A4.4AP-139, VII.A4.AP-189].)

B. OVERHEAD HEAVY LOAD AND LIGHT LOAD (RELATED TO REFUELING) HANDLING SYSTEMS

Systems, Structures, and Components

Most commercial nuclear facilities have between fifty and one hundred cranes. Many of these cranes are industrial grade cranes that must meet the requirements of 29 CFR Volume XVII, Part 1910, and Section 1910.179. They do not fall within the scope of 10 CFR Part 54.4 and therefore are not required to be part of the integrated plant assessment (IPA). Normally fewer than ten cranes fall within the scope of 10 CFR Part 54.4. These cranes must comply with the requirements provided in 10 CFR Part 50.65 and Reg. Guide 1.160 for monitoring the effectiveness of maintenance at nuclear power plants.

The Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems (the Program) must demonstrate that the testing and the monitoring of the maintenance programs have been completed to ensure that the structures, systems, and components of these cranes are capable of sustaining their rated loads during the period of extended operation. The inspection is also to evaluate whether the usage of the cranes or hoists has been sufficient to warrant additional fatigue analysis. It should be noted that many of the systems and components of these cranes can be classified as moving parts or as components which change configuration, or they may be subject to replacement based on a qualified life. In any of these cases, they will not fall within the scope of this Aging Management Review (AMR). The primary components that this program is concerned with are the structural girders and beams that make up the bridge and the trolley.

Based on Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” all components that comprise the overhead heavy load and light load handling systems are governed by Group C Quality Standards.

Aging management programs for the degradation of external surfaces of components and miscellaneous bolting are included in VII.I. Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in VII.J.

System Interfaces

No other systems discussed in this report interface with the overhead heavy load and light load (related to refueling) handling systems. Physical interfaces exist with the supporting structure. The direct interface is at the connection to the structure.

VII AUXILIARY SYSTEMS
B Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.B.A-05	VII.B-1(A-05)	Cranes - rails	Steel	Air – indoor, uncontrolled (External)	Loss of material due to wear	Chapter XI.M23, "Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems"	No
VII.B.A-07	VII.B-3(A-07)	Cranes: rails and structural girders	Steel	Air – indoor, uncontrolled (External)	Loss of material due to general corrosion	Chapter XI.M23, "Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems"	No
VII.B.A-06	VII.B-2(A-06)	Cranes: structural girders	Steel	Air – indoor, uncontrolled (External)	Cumulative fatigue damage due to fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation for structural girders of cranes that fall within the scope of 10 CFR 54. See SRP-LR Sec. 4.7, "Other Plant-Specific Time-Limited Aging Analyses," for generic guidance for meeting the requirements of 10 CFR 54.21(c)(1)).	Yes, TLAA

C1. OPEN-CYCLE COOLING WATER SYSTEM (SERVICE WATER SYSTEM)

Systems, Structures, and Components

This section discusses the open-cycle cooling water (OCCW) (or service water) system, which consists of piping, heat exchangers, pumps, flow orifices, basket strainers, and valves, including containment isolation valves. Because the characteristics of an OCCW system may be unique to each facility, the OCCW system is defined as a system or systems that transfer heat from safety-related systems, structures, and components (SSCs) to the ultimate heat sink (UHS), such as a lake, ocean, river, spray pond, or cooling tower. The AMPs described in this section apply to any such system, provided the service conditions and materials of construction are identical to those identified in the section. The system removes heat from the closed-cycle cooling water system, and, in some plants, other auxiliary systems and components, such as steam turbine bearing oil coolers or miscellaneous coolers in the condensate system. The only heat exchangers addressed in this section are those removing heat from the closed-cycle cooling system. Heat exchangers for removing heat from other auxiliary systems and components are addressed in their respective systems, such as those for the steam turbine bearing oil coolers (VIII.A) and for the condensate system coolers (VIII.E).

Based on Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” all components that comprise the open-cycle cooling water system are governed by Group C Quality Standards, with the exception of those forming part of the containment penetration boundary which are governed by Group B Quality Standards.

Pump and valve internals perform their intended functions with moving parts or with a change in configuration. They are also subject to replacement based on qualified life or a specified time period. Pursuant to 10 CFR 54.21(a)(1), therefore, they are not subject to an aging management review.

Aging management programs for the degradation of external surfaces of components and miscellaneous bolting are included in VII.I. Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in VII.J.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The systems that may interface with the open-cycle cooling water system include the closed-cycle cooling water system (VII.C2), the ultimate heat sink (VII.C3), the emergency diesel generator system (VII.H2), the containment spray system (V.A), the PWR steam generator blowdown system (VIII.F), the condensate system (VIII.E), the auxiliary feedwater system (PWR) (VIII.G), the emergency core cooling system (PWR) (V.D1), and the emergency core cooling system (BWR) (V.D2).

VII AUXILIARY SYSTEMS
C1 Open-Cycle Cooling Water System (Service Water System)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.C1.AP-75	VII.C1-1(AP-75)	Elastomer: seals and components	Elastomers	Raw water	Hardening and loss of strength due to elastomer degradation	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VII.C1.AP-76	VII.C1-2(AP-76)	Elastomer: seals and components	Elastomers	Raw water	Loss of material due to erosion	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VII.C1.AP-179	VII.C1-3(A-65)	Heat exchanger components	Copper alloy	Raw water	Loss of material due to general, pitting, crevice, galvanic, and microbiologically-influenced corrosion; fouling that leads to corrosion	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VII.C1.A-66	VII.C1-4(A-66)	Heat exchanger components	Copper alloy (>15% Zn or >8% Al)	Raw water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VII.C1.AP-183	VII.C1-5(A-64)	Heat exchanger components	Steel	Raw water	Loss of material due to general, pitting, crevice, galvanic, and microbiologically-influenced corrosion; fouling that leads to corrosion	Chapter XI.M20, "Open-Cycle Cooling Water System"	No

VII AUXILIARY SYSTEMS
C1 Open-Cycle Cooling Water System (Service Water System)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.C1.AP-152		Heat exchanger components other than tubes	Titanium (ASTM Grades 1,2, 7, 11, or 12 that contains > 5% aluminum or more than 0.20% oxygen or any amount of tin)	Raw water	None	None	No
VII.C1.A-72	VII.C1-6(A-72)	Heat exchanger tubes	Copper alloy	Raw water	Reduction of heat transfer due to fouling	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VII.C1.AP-187	VII.C1-7(AP-61)	Heat exchanger tubes	Stainless steel	Raw water	Reduction of heat transfer due to fouling	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VII.C1.AP-153		Heat exchanger tubes	Titanium	Raw water	Reduction of heat transfer due to fouling	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VII.C1.AP-173		Piping, piping components, and piping elements	Aluminum	Soil or concrete	Loss of material due to pitting and crevice corrosion	Chapter XI.M41, "Buried and Underground Piping and Tanks"	No
VII.C1.AP-237		Piping, piping components, and piping elements	Asbestos cement pipe	Soil or concrete	Cracking, spalling, corrosion of rebar due to exposure of rebar	Chapter XI.M41, "Buried and Underground Piping and Tanks"	No
VII.C1.AP-178		Piping, piping components, and piping elements	Concrete	Soil or concrete	Cracking, spalling, corrosion of rebar due to exposure of rebar	Chapter XI.M41, "Buried and Underground Piping and Tanks"	No
VII.C1.AP-177		Piping, piping components, and piping elements	Concrete cylinder piping	Soil or concrete	Cracking, spalling, corrosion of rebar due to exposure of rebar	Chapter XI.M41, "Buried and Underground Piping and Tanks"	No

VII AUXILIARY SYSTEMS
C1 Open-Cycle Cooling Water System (Service Water System)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.C1.AP-253		Piping, piping components, and piping elements	Concrete; cementitious material	Air - outdoor	Changes in material properties due to aggressive chemical attack	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
VII.C1.AP-251		Piping, piping components, and piping elements	Concrete; cementitious material	Air - outdoor	Cracking due to settling	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
VII.C1.AP-252		Piping, piping components, and piping elements	Concrete; cementitious material	Air - outdoor	Loss of material due to abrasion, cavitation, aggressive chemical attack, and leaching	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
VII.C1.AP-250		Piping, piping components, and piping elements	Concrete; cementitious material	Raw Water	Changes in material properties due to aggressive chemical attack	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VII.C1.AP-248		Piping, piping components, and piping elements	Concrete; cementitious material	Raw Water	Cracking due to settling	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VII.C1.AP-249		Piping, piping components, and piping elements	Concrete; cementitious material	Raw Water	Loss of material due to abrasion, cavitation, aggressive chemical attack, and leaching	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VII.C1.AP-133	VII.C1-8(AP-47)	Piping, piping components, and piping elements	Copper alloy	Lubricating oil	Loss of material due to pitting and crevice corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No

VII AUXILIARY SYSTEMS
C1 Open-Cycle Cooling Water System (Service Water System)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.C1.AP-196	VII.C1-9(A-44)	Piping, piping components, and piping elements	Copper alloy	Raw water	Loss of material due to general, pitting, crevice, and microbiologically-influenced corrosion; fouling that leads to corrosion	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VII.C1.AP-174		Piping, piping components, and piping elements	Copper Alloy	Soil or concrete	Loss of material due to pitting and crevice corrosion	Chapter XI.M41, "Buried and Underground Piping and Tanks"	No
VII.C1.A-47	VII.C1-10(A-47)	Piping, piping components, and piping elements	Copper alloy (>15% Zn or >8% Al)	Raw water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VII.C1.AP-238		Piping, piping components, and piping elements	Fiberglass	Raw water (internal)	Cracking, blistering, change in color due to water absorption	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VII.C1.AP-176		Piping, piping components, and piping elements	Fiberglass	Soil or concrete	Cracking, blistering, change in color due to water absorption	Chapter XI.M41, "Buried and Underground Piping and Tanks"	No
VII.C1.A-51	VII.C1-11(A-51)	Piping, piping components, and piping elements	Gray cast iron	Raw water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No

VII AUXILIARY SYSTEMS
C1 Open-Cycle Cooling Water System (Service Water System)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.C1.A-02	VII.C1-12(A-02)	Piping, piping components, and piping elements	Gray cast iron	Soil	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VII.C1.AP-239		Piping, piping components, and piping elements	HDPE	Raw water (internal)	Cracking, blistering, change in color due to water absorption	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VII.C1.AP-175		Piping, piping components, and piping elements	HDPE	Soil or concrete	Cracking, blistering, change in color due to water absorption	Chapter XI.M41, "Buried and Underground Piping and Tanks"	No
VII.C1.AP-206	VII.C1-13(AP-53)	Piping, piping components, and piping elements	Nickel alloy	Raw water	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VII.C1.AP-156		Piping, piping components, and piping elements	Reinforced concrete, asbestos cement	Air – outdoor	Cracking due to aggressive chemical attack and leaching; Changes in material properties due to aggressive chemical attack	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No

VII AUXILIARY SYSTEMS
C1 Open-Cycle Cooling Water System (Service Water System)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.C1.AP-155		Piping, piping components, and piping elements	Reinforced concrete, asbestos cement	Raw water	Cracking due to aggressive chemical attack and leaching; Changes in material properties due to aggressive chemical attack	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VII.C1.AP-157		Piping, piping components, and piping elements	Reinforced concrete, asbestos cement	Soil or concrete	Cracking due to aggressive chemical attack and leaching; Changes in material properties due to aggressive chemical attack	Chapter XI.M41, "Buried and Underground Piping and Tanks"	No
VII.C1.AP-138	VII.C1-14(AP-59)	Piping, piping components, and piping elements	Stainless steel	Lubricating oil	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
VII.C1.A-54	VII.C1-15(A-54)	Piping, piping components, and piping elements	Stainless steel	Raw water	Loss of material due to pitting and crevice corrosion; fouling that leads to corrosion	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VII.C1.AP-137	VII.C1-16(AP-56)	Piping, piping components, and piping elements	Stainless steel	Soil or concrete	Loss of material due to pitting and crevice corrosion	Chapter XI.M41, "Buried and Underground Piping and Tanks"	No

VII AUXILIARY SYSTEMS
C1 Open-Cycle Cooling Water System (Service Water System)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.C1.AP-127	VII.C1-17(AP-30)	Piping, piping components, and piping elements	Steel	Lubricating oil	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
VII.C1.AP-194	VII.C1-19(A-38)	Piping, piping components, and piping elements	Steel (with coating or lining)	Raw water	Loss of material due to general, pitting, crevice, and microbiologically-influenced corrosion; fouling that leads to corrosion; lining/coating degradation	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VII.C1.AP-198	VII.C1-18(A-01)	Piping, piping components, and piping elements	Steel (with coating or wrapping)	Soil or concrete	Loss of material due to general, pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M41, "Buried and Underground Piping and Tanks"	No
VII.C1.AP-172		Piping, piping components, and piping elements	Super austenitic	Soil or concrete	Loss of material due to pitting and crevice corrosion	Chapter XI.M41, "Buried and Underground Piping and Tanks"	No
VII.C1.AP-171		Piping, piping components, and piping elements	Titanium	Soil or concrete	Loss of material due to pitting and crevice corrosion	Chapter XI.M41, "Buried and Underground Piping and Tanks"	No

VII AUXILIARY SYSTEMS
C1 Open-Cycle Cooling Water System (Service Water System)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.C1.AP-161		Piping, piping components, and piping elements	Titanium (ASTM Grades 1,2, 7, 11, or 12 that contains > 5% aluminum or more than 0.20% oxygen or any amount of tin)	Raw water	None	None	No
VII.C1.AP-209		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated
VII.C1.AP-221		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Loss of material due to pitting and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated

C2. CLOSED-CYCLE COOLING WATER SYSTEM

Systems, Structures, and Components

This section discusses the closed-cycle cooling water (CCCW) system, which consists of piping, radiation elements, temperature elements, heat exchangers, pumps, tanks, flow orifices, and valves, including containment isolation valves. The system contains chemically treated demineralized water. The closed-cycle cooling water system is designed to remove heat from various auxiliary systems and components such as the chemical and volume control system and the spent fuel cooling system to the open-cycle cooling water system (VII.C1). A CCCW system is defined as part of the service water system that does not reject heat directly to a heat sink, has water chemistry control, and is not subject to significant sources of contamination.

Based on RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all components in the closed-cycle cooling water system are classified as Group C Quality Standards, with the exception of those forming part of the containment penetration boundary, which are Group B.

The aging management programs (AMPs) for the heat exchanger between the closed-cycle and the open-cycle cooling water systems are addressed in the open-cycle cooling water system (VII.C1). The AMPs for the heat exchangers between the closed-cycle cooling water system and the interfacing auxiliary systems are included in the evaluations of their respective systems, such as those for the pressurized water reactor (PWR) and boiling water reactor (BWR) spent fuel pool cooling and cleanup systems (VII.A3 and VII.A4, respectively) and the PWR chemical and volume control system (VII.E1).

Pump and valve internals perform their intended functions with moving parts or with a change in configuration. They are also subject to replacement based on qualified life or a specified time period. Pursuant to 10 CFR 54.21(a)(1), therefore, they are not subject to an aging management review.

Aging management programs for the degradation of external surfaces of components and miscellaneous bolting are included in VII.I. Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in VII.J.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The systems that interface with the closed-cycle cooling water system include the open-cycle cooling water system (VII.C1), the PWR spent fuel pool cooling and cleanup system (VII.A3), the BWR spent fuel pool cooling and cleanup system (VII.A4), the PWR chemical and volume control system (VII.E1), the BWR reactor water cleanup system (VII.E3), the shutdown cooling system (older BWR, VII.E4), the primary containment heating and ventilation system (VII.F3), fire protection (VII.G), the emergency diesel generator system (VII.H2), the PWR containment spray system (V.A), the PWR and BWR emergency core cooling systems (V.D1 and V.D2), the PWR steam generator blowdown system (VIII.F), the condensate system (VIII.E), and the PWR auxiliary feedwater system (VIII.G).

VII AUXILIARY SYSTEMS
C2 Closed-Cycle Cooling Water System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.C2.AP-259		Elastomer seals and components	Elastomers	Closed-cycle cooling water	Hardening and loss of strength due to elastomer degradation	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
VII.C2.AP-189	VII.C2-1(A-63)	Heat exchanger components	Steel	Closed-cycle cooling water	Loss of material due to general, pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.C2.AP-205	VII.C2-2(AP-80)	Heat exchanger tubes	Copper Alloy	Closed-cycle cooling water	Reduction of heat transfer due to fouling	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.C2.AP-188	VII.C2-3(AP-63)	Heat exchanger tubes	Stainless steel	Closed-cycle cooling water	Reduction of heat transfer due to fouling	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.C2.AP-254		Piping, piping components, and piping elements	Aluminum	Closed-cycle cooling water	Loss of material due to pitting and crevice corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.C2.AP-257		Piping, piping components, and piping elements	Aluminum	Treated water	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VII.C2.AP-199	VII.C2-4(AP-12)	Piping, piping components, and piping elements	Copper alloy	Closed-cycle cooling water	Loss of material due to general, pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.C2.AP-133	VII.C2-5(AP-47)	Piping, piping components, and piping elements	Copper alloy	Lubricating oil	Loss of material due to pitting and crevice corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No

VII AUXILIARY SYSTEMS
C2 Closed-Cycle Cooling Water System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.C2.AP-43	VII.C2-6(AP-43)	Piping, piping components, and piping elements	Copper alloy (>15% Zn or >8% Al)	Closed-cycle cooling water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VII.C2.AP-32	VII.C2-7(AP-32)	Piping, piping components, and piping elements	Copper alloy (>15% Zn or >8% Al)	Treated water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VII.C2.A-50	VII.C2-8(A-50)	Piping, piping components, and piping elements	Gray cast iron	Closed-cycle cooling water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VII.C2.AP-31	VII.C2-9(AP-31)	Piping, piping components, and piping elements	Gray cast iron	Treated water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VII.C2.A-52	VII.C2-10(A-52)	Piping, piping components, and piping elements	Stainless steel	Closed-cycle cooling water	Loss of material due to pitting and crevice corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.C2.AP-186	VII.C2-11(AP-60)	Piping, piping components, and piping elements	Stainless steel	Closed-cycle cooling water >60°C (>140°F)	Cracking due to stress corrosion cracking	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.C2.AP-138	VII.C2-12(AP-59)	Piping, piping components, and piping elements	Stainless steel	Lubricating oil	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No

VII AUXILIARY SYSTEMS
C2 Closed-Cycle Cooling Water System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.C2.AP-127	VII.C2-13(AP-30)	Piping, piping components, and piping elements	Steel	Lubricating oil	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, "One-Time Inspection"	No
VII.C2.AP-209		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated
VII.C2.AP-221		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Loss of material due to pitting and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated
VII.C2.AP-202	VII.C2-14(A-25)	Piping, piping components, and piping elements; tanks	Steel	Closed-cycle cooling water	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No

C3. ULTIMATE HEAT SINK

Systems, Structures, and Components

The ultimate heat sink (UHS) consists of a lake, ocean, river, spray pond, or cooling tower. The UHS provides sufficient cooling water for safe reactor shutdown and reactor cooldown via the residual heat removal system or other similar system. Due to the varying configurations of connections to lakes, oceans, and rivers, a plant-specific aging management program (AMP) is required. Appropriate AMPs shall be provided to trend and project (1) deterioration of earthen dams and impoundments; (2) rate of silt deposition; (3) meteorological, climatological, and oceanic data since obtaining the Final Safety Analysis Report (FSAR) data; (4) water level extremes for plants located on rivers; and (5) aging degradation of all upstream and downstream dams affecting the UHS.

The systems, structures, and components included in this section consist of piping, valves, and pumps. The cooling tower is addressed in this report on water-control structures (III.A6). The ultimate heat sink absorbs heat from the residual heat removal system or other similar system. Based on Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” the piping and valves used for the ultimate heat sink are governed by Group C Quality Standards.

Pump and valve internals perform their intended functions with moving parts or with a change in configuration. They are also subject to replacement based on qualified life or a specified time period. Pursuant to 10 CFR 54.21(a)(1), therefore, they are not subject to an aging management review.

Aging management programs for the degradation of external surfaces of components and miscellaneous bolting are included in VII.I. Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in VII.J.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The systems that interface with the ultimate heat sink include the open-cycle cooling water system (VII.C1) and the PWR and BWR emergency core cooling systems (V.D1 and V.D2).

VII AUXILIARY SYSTEMS
C3 Ultimate Heat Sink

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.C3.AP-187	VII.C3-1(AP-61)	Heat exchanger tubes	Stainless steel	Raw water	Reduction of heat transfer due to fouling	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VII.C3.AP-195	VII.C3-2(A-43)	Piping, piping components, and piping elements	Copper alloy	Raw water	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VII.C3.A-47	VII.C3-3(A-47)	Piping, piping components, and piping elements	Copper alloy (>15% Zn or >8% Al)	Raw water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VII.C3.A-51	VII.C3-4(A-51)	Piping, piping components, and piping elements	Gray cast iron	Raw water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VII.C3.A-02	VII.C3-5(A-02)	Piping, piping components, and piping elements	Gray cast iron	Soil	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VII.C3.AP-206	VII.C3-6(AP-53)	Piping, piping components, and piping elements	Nickel alloy	Raw water	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VII.C3.A-53	VII.C3-7(A-53)	Piping, piping components, and piping elements	Stainless steel	Raw water	Loss of material due to pitting and crevice corrosion	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VII.C3.AP-137	VII.C3-8(AP-56)	Piping, piping components, and piping elements	Stainless steel	Soil	Loss of material due to pitting and crevice corrosion	Chapter XI.M41, "Buried and Underground Piping and Tanks"	No

VII AUXILIARY SYSTEMS
C3 Ultimate Heat Sink

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.C3.AP-194	VII.C3-10(A-38)	Piping, piping components, and piping elements	Steel (with coating or lining)	Raw water	Loss of material due to general, pitting, crevice, and microbiologically-influenced corrosion; fouling that leads to corrosion; lining/coating degradation	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VII.C3.AP-198	VII.C3-9(A-01)	Piping, piping components, and piping elements	Steel (with coating or wrapping)	Soil	Loss of material due to general, pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M41, "Buried and Underground Piping and Tanks"	No
VII.C3.AP-209		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated
VII.C3.AP-221		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Loss of material due to pitting and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated

D. COMPRESSED AIR SYSTEM

Systems, Structures, and Components

This section discusses the compressed air system, which consists of piping, valves (including containment isolation valves), air receivers, pressure regulators, filters, and dryers. The system components and piping are located in various buildings at most nuclear power plants. Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all components of the compressed air system are classified as Group D Quality Standards, with the exception of those forming part of the containment penetration boundary, which are Group B. However, the cleanliness of these components and high air quality is to be maintained because the air provides the motive power for instruments and active components (some of them safety-related) that may not function properly if nonsafety Group D equipment is contaminated.

With respect to filters, these items are to be addressed consistent with the Nuclear Regulatory Commission (NRC) position on consumables, provided in the NRC letter from Christopher I. Grimes to Douglas J. Walters of the Nuclear Energy Institute (NEI), dated March 10, 2000. Specifically, components that function as system filters 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 an aging management review under 10 CFR 54.21(a)(1)(ii). As part of the methodology description, the application should identify the standards that are relied on for replacement, for example, National Fire Protection Association (NFPA) standards for fire protection equipment.

Pump and valve internals perform their intended functions with moving parts or with a change in configuration. They are also subject to replacement based on qualified life or a specified time period. Pursuant to 10 CFR 54.21(a)(1), therefore, they are not subject to an aging management review.

Aging management programs for the degradation of external surfaces of components and miscellaneous bolting are included in VII.I. Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in VII.J.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

Various other systems discussed in this report may interface with the compressed air system.

VII AUXILIARY SYSTEMS
D Compressed Air System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.D.AP-121	VII.D-1(A-103)	Closure bolting	Steel; stainless steel	Condensation	Loss of material due to general (steel only), pitting, and crevice corrosion	Chapter XI.M18, "Bolting Integrity"	No
VII.D.A-80	VII.D-3(A-80)	Piping and components (External surfaces)	Steel	Air – indoor, uncontrolled (External)	Loss of material due to general corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
VII.D.AP-240		Piping, piping components, and piping elements	Copper alloy	Condensation	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M24, "Compressed Air Monitoring"	No
VII.D.AP-81	VII.D-4(AP-81)	Piping, piping components, and piping elements	Stainless steel	Condensation (Internal)	Loss of material due to pitting and crevice corrosion	Chapter XI.M24, "Compressed Air Monitoring"	No
VII.D.A-26	VII.D-2(A-26)	Piping, piping components, and piping elements: compressed air system	Steel	Condensation (Internal)	Loss of material due to general and pitting corrosion	Chapter XI.M24, "Compressed Air Monitoring"	No
VII.D.AP-209		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated
VII.D.AP-221		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Loss of material due to pitting and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated

E1. CHEMICAL AND VOLUME CONTROL SYSTEM (PRESSURIZED WATER REACTOR)

Systems, Structures, and Components

This section discusses a portion of the pressurized water reactor (PWR) chemical and volume control system (CVCS). The portion of the PWR CVCS covered in this section extends from the isolation valves associated with the reactor coolant pressure boundary (and Code change as discussed below) to the volume control tank. This portion of the PWR CVCS consists of high- and low-pressure piping and valves (including the containment isolation valves), regenerative and letdown heat exchangers, pumps, basket strainers, and the volume control tank. The system contains chemically treated borated water; the shell side of the letdown heat exchanger contains closed-cycle cooling water (treated water).

Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all components that comprise the CVCS are governed by Group C Quality Standards. Portions of the CVCS extending from the reactor coolant system up to and including the isolation valves associated with reactor coolant pressure boundary are governed by Group A Quality Standards and covered in IV.C2.

Pump and valve internals perform their intended functions with moving parts or with a change in configuration. They are also subject to replacement based on qualified life or a specified time period. Pursuant to 10 CFR 54.21(a)(1), therefore, they are not subject to an aging management review.

Aging management programs for the degradation of external surfaces of components and miscellaneous bolting are included in VII.I. Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in VII.J.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The systems that interface with the chemical and volume control system include the reactor coolant system (IV.C2), the emergency core cooling system (V.D1), the spent fuel pool cooling system (VII.A3), and the closed-cycle cooling water system (VII.C2).

VII AUXILIARY SYSTEMS
E1 Chemical and Volume Control System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.E1.A-79	VII.E1-1(A-79)	External surfaces	Steel	Air with borated water leakage	Loss of material due to boric acid corrosion	Chapter XI.M10, "Boric Acid Corrosion"	No
VII.E1.AP-203	VII.E1-2(AP-34)	Heat exchanger components	Copper alloy	Closed-cycle cooling water	Loss of material due to general, pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.E1.AP-65	VII.E1-3(AP-65)	Heat exchanger components	Copper alloy (>15% Zn or >8% Al)	Treated water	Loss of material due to selective leaching	Chapter XI.M33, "Selective Leaching"	No
VII.E1.AP-118	VII.E1-5(A-84)	Heat exchanger components	Stainless steel	Treated borated water >60°C (>140°F)	Cracking due to stress corrosion cracking	Chapter XI.M2, "Water Chemistry," and Chapter XI.M32, "One-Time Inspection"	No
VII.E1.AP-189	VII.E1-6(A-63)	Heat exchanger components	Steel	Closed-cycle cooling water	Loss of material due to general, pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.E1.A-100	VII.E1-4(A-100)	Heat exchanger components and tubes	Stainless steel	Treated borated water	Cumulative fatigue damage due to fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
VII.E1.AP-119	VII.E1-5(A-84)	Heat exchanger components and tubes	Stainless steel	Treated borated water >60°C (>140°F)	Cracking due to cyclic loading	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD"	No

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.E1.A-69	VII.E1-9(A-69)	Heat exchanger components, non-regenerative	Stainless steel	Treated borated water >60°C (>140°F)	Cracking due to stress corrosion cracking; cyclic loading	Chapter XI.M2, “Water Chemistry.” The AMP is to be augmented by verifying the absence of cracking due to stress corrosion cracking and cyclic loading. An acceptable verification program is to include temperature and radioactivity monitoring of the shell side water, and eddy current testing of tubes.	Yes, plant-specific
VII.E1.AP-115	VII.E1-7(A-76)	High-pressure pump, casing	Stainless steel	Treated borated water	Cracking due to cyclic loading	Chapter XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD"	No
VII.E1.AP-114	VII.E1-7(A-76)	High-pressure pump, casing	Stainless steel	Treated borated water >60°C (>140°F)	Cracking due to stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VII.E1.AP-122	VII.E1-8(A-104)	High-pressure pump, closure bolting	Steel, high-strength	Air with steam or water leakage	Cracking due to stress corrosion cracking; cyclic loading	Chapter XI.M18, "Bolting Integrity"	No
VII.E1.AP-1	VII.E1-10(AP-1)	Piping, piping components, and piping elements	Aluminum	Air with borated water leakage	Loss of material due to boric acid corrosion	Chapter XI.M10, “Boric Acid Corrosion”	No
VII.E1.AP-199	VII.E1-11(AP-12)	Piping, piping components, and piping elements	Copper alloy	Closed-cycle cooling water	Loss of material due to general, pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No

VII AUXILIARY SYSTEMS
E1 Chemical and Volume Control System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.E1.AP-133	VII.E1-12(AP-47)	Piping, piping components, and piping elements	Copper alloy	Lubricating oil	Loss of material due to pitting and crevice corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, "One-Time Inspection"	No
VII.E1.AP-43	VII.E1-13(AP-43)	Piping, piping components, and piping elements	Copper alloy (>15% Zn or >8% Al)	Closed-cycle cooling water	Loss of material due to selective leaching	Chapter XI.M33, "Selective Leaching"	No
VII.E1.AP-31	VII.E1-14(AP-31)	Piping, piping components, and piping elements	Gray cast iron	Treated water	Loss of material due to selective leaching	Chapter XI.M33, "Selective Leaching"	No
VII.E1.AP-138	VII.E1-15(AP-59)	Piping, piping components, and piping elements	Stainless steel	Lubricating oil	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, "One-Time Inspection"	No
VII.E1.A-57	VII.E1-16(A-57)	Piping, piping components, and piping elements	Stainless steel	Treated borated water	Cumulative fatigue damage due to fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA

VII AUXILIARY SYSTEMS
E1 Chemical and Volume Control System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.E1.A-34	VII.E1-18(A-34)	Piping, piping components, and piping elements	Steel	Air - indoor, uncontrolled	Cumulative fatigue damage due to fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
VII.E1.AP-127	VII.E1-19(AP-30)	Piping, piping components, and piping elements	Steel	Lubricating oil	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
VII.E1.AP-79	VII.E1-17(AP-79)	Piping, piping components, and piping elements	Steel (with stainless steel cladding); stainless steel	Treated borated water	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry”	No
VII.E1.AP-209		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated
VII.E1.AP-221		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Loss of material due to pitting and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated
VII.E1.AP-82	VII.E1-20(AP-82)	Piping, piping components, and piping elements; tanks	Stainless steel	Treated borated water >60°C (>140°F)	Cracking due to stress corrosion cracking	Chapter XI.M2, “Water Chemistry”	No

VII AUXILIARY SYSTEMS
E1 Chemical and Volume Control System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.E1.AP-85	VII.E1-21(AP-85)	Pump Casings	Steel (with stainless steel or nickel-alloy cladding)	Treated borated water	Loss of material due to cladding breach	A plant-specific aging management program is to be evaluated. Reference NRC Information Notice 94-63, "Boric Acid Corrosion of Charging Pump Casings Caused by Cladding Cracks."	Yes, verify that plant-specific program addresses clad cracking

E2. STANDBY LIQUID CONTROL SYSTEM (BOILING WATER REACTOR)

Systems, Structures, and Components

This section discusses the portion of the standby liquid control (SLC) system extending from the containment isolation valve to the solution storage tank. The system serves as a backup reactivity control system in all boiling water reactors (BWRs). The major components of this system are the piping, the solution storage tank, the solution storage tank heaters, valves, and pumps. All of the components from the storage tank to the explosive actuated discharge valve operate in contact with a sodium pentaborate ($Na_2B_{10}O_{16} \cdot 10H_2O$) solution.

Based on Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” all components that comprise the standby liquid control system are governed by Group B Quality Standards. The portions of the standby liquid control system extending from the reactor coolant pressure boundary up to and including the containment isolation valves are governed by Group A Quality Standards and are covered in IV.C1.

Pump and valve internals perform their intended functions with moving parts or with a change in configuration. They are also subject to replacement based on qualified life or a specified time period. Pursuant to 10 CFR 54.21(a)(1), therefore, they are not subject to an aging management review.

Aging management programs for the degradation of external surfaces of components and miscellaneous bolting are included in VII.I. Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in VII.J.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The system that interfaces with the SLC system is the BWR reactor pressure vessel (IV.A1). If used, the SLC system would inject sodium pentaborate solution into the pressure vessel near the bottom of the reactor core.

VII AUXILIARY SYSTEMS
E2 Standby Liquid Control System (BWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.E2.AP-141	VII.E2-1(AP-73)	Piping, piping components, and piping elements	Stainless steel	Sodium pentaborate solution	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VII.E2.AP-181	VII.E2-2(A-59)	Piping, piping components, and piping elements	Stainless steel	Sodium pentaborate solution >60°C (>140°F)	Cracking due to stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No

E3. REACTOR WATER CLEANUP SYSTEM (BOILING WATER REACTOR)

Systems, Structures, and Components

This section discusses the reactor water cleanup (RWCU) system, which provides for cleanup and particulate removal from the recirculating reactor coolant in all boiling water reactors (BWRs). Some plants may not include the RWCU system in the scope of license renewal, while other plants may include the RWCU system because it is associated with safety-related functions.

Based on Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” the portion of the RWCU system extending from the reactor coolant recirculation system up to and including the containment isolation valves are covered in IV.C1. The remainder of the system outboard of the isolation valves is governed by Group C Quality Standards. In this table, only aging management programs for RWCU-related piping and components outboard of the isolation valves are evaluated. The aging management program for containment isolation valves in the RWCU system is evaluated in IV.C1, which concerns the reactor coolant pressure boundary in BWRs.

Pump and valve internals perform their intended functions with moving parts or with a change in configuration. They are also subject to replacement based on qualified life or specified time period. Pursuant to 10 CFR 54.21(a)(1), therefore, they are not subject to an aging management review.

Aging management programs for the degradation of external surfaces of components and miscellaneous bolting are included in VII.I. Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in VII.J.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The systems that interface with the BWR reactor water cleanup system include the reactor coolant pressure boundary (IV.C1), the closed-cycle cooling water system (VII.C2), and the condensate system (VIII.E).

VII AUXILIARY SYSTEMS
E3 Reactor Water Cleanup System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.E3.AP-191	VII.E3-1(A-67)	Heat exchanger components	Stainless steel; steel with stainless steel cladding	Closed-cycle cooling water	Loss of material due to microbiologically-influenced corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.E3.AP-192	VII.E3-2(A-68)	Heat exchanger components	Stainless steel; steel with stainless steel cladding	Closed-cycle cooling water >60°C (>140°F)	Cracking due to stress corrosion cracking	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.E3.AP-112	VII.E3-3(A-71)	Heat exchanger components	Stainless steel; steel with stainless steel cladding	Treated water >60°C (>140°F)	Cracking due to stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VII.E3.AP-189	VII.E3-4(A-63)	Heat exchanger components	Steel	Closed-cycle cooling water	Loss of material due to general, pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.E3.AP-188	VII.E3-5(AP-63)	Heat exchanger tubes	Stainless steel	Closed-cycle cooling water	Reduction of heat transfer due to fouling	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.E3.AP-139	VII.E3-6(AP-62)	Heat exchanger tubes	Stainless steel	Treated water	Reduction of heat transfer due to fouling	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VII.E3.AP-130	VII.E3-7(AP-38)	Piping, piping components, and piping elements	Aluminum	Treated water	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No

VII AUXILIARY SYSTEMS
E3 Reactor Water Cleanup System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.E3.AP-199	VII.E3-8(AP-12)	Piping, piping components, and piping elements	Copper alloy	Closed-cycle cooling water	Loss of material due to general, pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.E3.AP-140	VII.E3-9(AP-64)	Piping, piping components, and piping elements	Copper alloy	Treated water	Loss of material due to general, pitting, crevice, and galvanic corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VII.E3.AP-43	VII.E3-10(AP-43)	Piping, piping components, and piping elements	Copper alloy (>15% Zn or >8% Al)	Closed-cycle cooling water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VII.E3.AP-32	VII.E3-11(AP-32)	Piping, piping components, and piping elements	Copper alloy (>15% Zn or >8% Al)	Treated water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VII.E3.AP-31	VII.E3-12(AP-31)	Piping, piping components, and piping elements	Gray cast iron	Treated water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VII.E3.AP-186	VII.E3-13(AP-60)	Piping, piping components, and piping elements	Stainless steel	Closed-cycle cooling water >60°C (>140°F)	Cracking due to stress corrosion cracking	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.E3.A-62	VII.E3-14(A-62)	Piping, piping components, and piping elements	Stainless steel	Treated water	Cumulative fatigue damage due to fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA

VII AUXILIARY SYSTEMS
E3 Reactor Water Cleanup System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.E3.AP-110	VII.E3-15(A-58)	Piping, piping components, and piping elements	Stainless steel	Treated water	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VII.E3.AP-283	VII.E3-16(A-60)	Piping, piping components, and piping elements	Stainless steel	Treated water >60°C (>140°F)	Cracking due to stress corrosion cracking, intergranular stress corrosion cracking	Chapter XI.M2, "Water Chemistry," and Chapter XI.M25, “BWR Reactor Water Cleanup System”	No
VII.E3.A-34	VII.E3-17(A-34)	Piping, piping components, and piping elements	Steel	Air - indoor, uncontrolled	Cumulative fatigue damage due to fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
VII.E3.AP-106	VII.E3-18(A-35)	Piping, piping components, and piping elements	Steel	Treated water	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VII.E3.AP-120	VII.E3-19(A-85)	Regenerative heat exchanger components	Stainless steel	Treated water >60°C (>140°F)	Cracking due to stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No

E4. SHUTDOWN COOLING SYSTEM (OLDER BWR)

Systems, Structures, and Components

This section discusses the shutdown cooling (SDC) system for older vintage boiling water reactors (BWRs) and consists of piping and fittings, the SDC system pump, the heat exchanger, and valves.

Based on Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” all components that comprise the SDC system are governed by Group B Quality Standards. Portions of the SDC system extending from the reactor coolant pressure boundary up to and including the containment isolation valves are governed by Group A Quality Standards and are covered in IV.C1.

Pump and valve internals perform their intended functions with moving parts or with a change in configuration. They are also subject to replacement based on qualified life or a specified time period. Pursuant to 10 CFR 54.21(a)(1), therefore, they are not subject to an aging management review.

Aging management programs for the degradation of external surfaces of components and miscellaneous bolting are included in VII.I. Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in VII.J.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The systems that interface with the SDC system include the reactor coolant pressure boundary (IV.C1) and the closed-cycle cooling water system (VII.C2).

VII AUXILIARY SYSTEMS
E4 Shutdown Cooling System (Older BWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.E4.AP-191	VII.E4-1(A-67)	Heat exchanger components	Stainless steel; steel with stainless steel cladding	Closed-cycle cooling water	Loss of material due to microbiologically-influenced corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.E4.AP-189	VII.E4-2(A-63)	Heat exchanger components	Steel	Closed-cycle cooling water	Loss of material due to general, pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.E4.AP-188	VII.E4-3(AP-63)	Heat exchanger tubes	Stainless steel	Closed-cycle cooling water	Reduction of heat transfer due to fouling	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.E4.AP-130	VII.E4-4(AP-38)	Piping, piping components, and piping elements	Aluminum	Treated water	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VII.E4.AP-199	VII.E4-5(AP-12)	Piping, piping components, and piping elements	Copper alloy	Closed-cycle cooling water	Loss of material due to general, pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.E4.AP-133	VII.E4-6(AP-47)	Piping, piping components, and piping elements	Copper alloy	Lubricating oil	Loss of material due to pitting and crevice corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
VII.E4.AP-140	VII.E4-7(AP-64)	Piping, piping components, and piping elements	Copper alloy	Treated water	Loss of material due to general, pitting, crevice, and galvanic corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No

VII AUXILIARY SYSTEMS
E4 Shutdown Cooling System (Older BWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.E4.AP-43	VII.E4-8(AP-43)	Piping, piping components, and piping elements	Copper alloy (>15% Zn or >8% Al)	Closed-cycle cooling water	Loss of material due to selective leaching	Chapter XI.M33, "Selective Leaching"	No
VII.E4.AP-32	VII.E4-9(AP-32)	Piping, piping components, and piping elements	Copper alloy (>15% Zn or >8% Al)	Treated water	Loss of material due to selective leaching	Chapter XI.M33, "Selective Leaching"	No
VII.E4.AP-31	VII.E4-10(AP-31)	Piping, piping components, and piping elements	Gray cast iron	Treated water	Loss of material due to selective leaching	Chapter XI.M33, "Selective Leaching"	No
VII.E4.AP-186	VII.E4-11(AP-60)	Piping, piping components, and piping elements	Stainless steel	Closed-cycle cooling water >60°C (>140°F)	Cracking due to stress corrosion cracking	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.E4.AP-138	VII.E4-12(AP-59)	Piping, piping components, and piping elements	Stainless steel	Lubricating oil	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, "One-Time Inspection"	No
VII.E4.A-62	VII.E4-13(A-62)	Piping, piping components, and piping elements	Stainless steel	Treated water	Cumulative fatigue damage due to fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
VII.E4.AP-110	VII.E4-14(A-58)	Piping, piping components, and piping elements	Stainless steel	Treated water	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, "Water Chemistry," and Chapter XI.M32, "One-Time Inspection"	No

VII AUXILIARY SYSTEMS
E4 Shutdown Cooling System (Older BWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.E4.A-61	VII.E4-15(A-61)	Piping, piping components, and piping elements	Stainless steel	Treated water >60°C (>140°F)	Cracking due to stress corrosion cracking	Chapter XI.M7, “BWR Stress Corrosion Cracking,” and Chapter XI.M2, “Water Chemistry”	No
VII.E4.AP-127	VII.E4-16(AP-30)	Piping, piping components, and piping elements	Steel	Lubricating oil	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
VII.E4.AP-106	VII.E4-17(A-35)	Piping, piping components, and piping elements	Steel	Treated water	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VII.E4.AP-209		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated
VII.E4.AP-221		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Loss of material due to pitting and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated

E5. WASTE WATER SYSTEMS

Systems, Structures, and Components

This section discusses liquid waste systems such as liquid radioactive waste systems, oily waste systems, floor drainage systems, chemical waste water systems, and secondary waste water systems. Plants may include portions of waste water systems within the scope of license renewal based on the criterion of 10CFR 54.4.(a)(2).

Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," radioactive-waste-containing portions of waste water systems are classified as Group C Quality Standards, with the exception of those forming part of the containment pressure boundary, which is classified as Group B. Waste water systems that do not contain radioactive waste or form a part of the containment pressure boundary are classified as Group D.

Pump and valve internals perform their intended functions with moving parts or with a change in configuration. They are also subject to replacement based on qualified life or specified time period. Pursuant to 10 CFR 54.21(a)(1), therefore, they are not subject to an aging management review.

Aging management programs for the degradation of external surfaces of components and miscellaneous bolting are included in VII.I. Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in VII.J.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

Various other systems discussed in this report may interface with waste water systems.

VII AUXILIARY SYSTEMS
E5 Wastewater Systems

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.E5.AP-276		Heat exchanger components	Nickel alloy	Waste water	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
VII.E5.AP-275		Heat exchanger components	Stainless steel	Waste water	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
VII.E5.AP-271		Piping, piping components, and piping elements	Copper alloy	Raw water (potable)	Loss of material due to pitting and crevice corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
VII.E5.AP-272		Piping, piping components, and piping elements	Copper alloy	Waste water	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
VII.E5.AP-274		Piping, piping components, and piping elements	Nickel alloy	Condensation (Internal)	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.E5.AP-273		Piping, piping components, and piping elements	Stainless steel	Condensation (Internal)	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
VII.E5.AP-270		Piping, piping components, and piping elements	Steel; stainless steel	Raw water (potable)	Loss of material due to general (steel only), pitting, and crevice corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
VII.E5.AP-279		Piping, piping components, and piping elements; tanks	Nickel alloy	Waste water	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
VII.E5.AP-278		Piping, piping components, and piping elements; tanks	Stainless steel	Waste water	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
VII.E5.AP-280		Piping, piping components, and piping elements; tanks	Steel	Condensation (Internal)	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No

VII AUXILIARY SYSTEMS
E5 Wastewater Systems

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.E5.AP-281		Piping, piping components, and piping elements; tanks	Steel	Waste water	Loss of material due to general, pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No

F1. CONTROL ROOM AREA VENTILATION SYSTEM

Systems, Structures, and Components

This section discusses the control room area ventilation system (with warm moist air as the normal environment), which contains ducts, piping and fittings, equipment frames and housings, flexible collars and seals, filters, and heating and cooling air handlers. Based on Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” all components that comprise the control room area ventilation system are governed by Group B Quality Standards.

With respect to filters and seals, these items are to be addressed consistent with the Nuclear Regulatory Commission (NRC) position on consumables, provided in the NRC letter from Christopher I. Grimes to Douglas J. Walters of the Nuclear Energy Institute (NEI), dated March 10, 2000. Specifically, components that function as system filters and seals 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 an aging management review under 10 CFR 54.21(a)(1)(ii). As part of the methodology description, the application should identify the standards that are relied on for replacement, for example, National Fire Protection Association (NFPA) standards for fire protection equipment.

Aging management programs for the degradation of external surfaces of components and miscellaneous bolting are included in VII.I. Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in VII.J.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The system that interfaces with the control room area ventilation system is the auxiliary and radwaste area ventilation system (VII.F2). The cooling coils receive their cooling water from other systems, such as the hot water heating system or the chilled water cooling system.

VII AUXILIARY SYSTEMS
F1 Control Room Area Ventilation System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.F1.AP-99	VII.F1-1(A-09)	Ducting and components	Stainless steel	Condensation	Loss of material due to pitting and crevice corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
VII.F1.A-10	VII.F1-2(A-10)	Ducting and components (External surfaces)	Steel	Air – indoor, uncontrolled (External)	Loss of material due to general corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
VII.F1.A-08	VII.F1-3(A-08)	Ducting and components (Internal surfaces)	Steel	Condensation (Internal)	Loss of material due to general, pitting, crevice, and (for drip pans and drain lines) microbiologically-influenced corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
VII.F1.A-105	VII.F1-4(A-105)	Ducting; closure bolting	Steel	Air – indoor, uncontrolled (External)	Loss of material due to general corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
VII.F1.AP-113	VII.F1-5(A-73)	Elastomer: seals and components	Elastomers	Air – indoor, uncontrolled (External)	Loss of material due to wear	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
VII.F1.AP-103	VII.F1-6(A-18)	Elastomer: seals and components	Elastomers	Air – indoor, uncontrolled (Internal)	Loss of material due to wear	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
VII.F1.AP-102	VII.F1-7(A-17)	Elastomer: seals and components	Elastomers	Air – indoor, uncontrolled (Internal/External)	Hardening and loss of strength due to elastomer degradation	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No

VII AUXILIARY SYSTEMS
F1 Control Room Area Ventilation System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.F1.AP-203	VII.F1-8(AP-34)	Heat exchanger components	Copper alloy	Closed-cycle cooling water	Loss of material due to general, pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.F1.AP-65	VII.F1-9(AP-65)	Heat exchanger components	Copper alloy (>15% Zn or >8% Al)	Treated water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VII.F1.AP-41	VII.F1-10(AP-41)	Heat exchanger components	Steel	Air – indoor, uncontrolled (External)	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
VII.F1.AP-189	VII.F1-11(A-63)	Heat exchanger components	Steel	Closed-cycle cooling water	Loss of material due to general, pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.F1.AP-205	VII.F1-12(AP-80)	Heat exchanger tubes	Copper alloy	Closed-cycle cooling water	Reduction of heat transfer due to fouling	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.F1.AP-204	VII.F1-13(AP-77)	Heat exchanger tubes	Steel	Closed-cycle cooling water	Reduction of heat transfer due to fouling	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.F1.AP-142	VII.F1-14(AP-74)	Piping, piping components, and piping elements	Aluminum	Condensation (Internal)	Loss of material due to pitting and crevice corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
VII.F1.AP-199	VII.F1-15(AP-12)	Piping, piping components, and piping elements	Copper alloy	Closed-cycle cooling water	Loss of material due to general, pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No

VII AUXILIARY SYSTEMS
F1 Control Room Area Ventilation System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.F1.AP-109	VII.F1-16(A-46)	Piping, piping components, and piping elements	Copper alloy	Condensation (External)	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
VII.F1.AP-43	VII.F1-17(AP-43)	Piping, piping components, and piping elements	Copper alloy (>15% Zn or >8% Al)	Closed-cycle cooling water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VII.F1.AP-31	VII.F1-18(AP-31)	Piping, piping components, and piping elements	Gray cast iron	Treated water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VII.F1.AP-127	VII.F1-19(AP-30)	Piping, piping components, and piping elements	Steel	Lubricating oil	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
VII.F1.AP-209		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated
VII.F1.AP-221		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Loss of material due to pitting and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated
VII.F1.AP-202	VII.F1-20(A-25)	Piping, piping components, and piping elements; tanks	Steel	Closed-cycle cooling water	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No

F2. Auxiliary and Radwaste Area Ventilation System

Systems, Structures, and Components

This section discusses the auxiliary and radwaste area ventilation systems (with warm moist air as the normal environment) and contains ducts, piping and fittings, equipment frames and housings, flexible collars and seals, filters, and heating and cooling air handlers. Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all components that comprise the auxiliary and radwaste area ventilation system are governed by Group B Quality Standards.

With respect to filters and seals, these items are to be addressed consistent with the NRC position on consumables, provided in the NRC letter from Christopher I. Grimes to Douglas J. Walters of the Nuclear Energy Institute (NEI), dated March 10, 2000. Specifically, components that function as system filters and seals 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 an aging management review under 10 CFR 54.21(a)(1)(ii). As part of the methodology description, the application should identify the standards that are relied on for replacement, for example, National Fire Protection Association (NFPA) standards for fire protection equipment.

Aging management programs for the degradation of external surfaces of components and miscellaneous bolting are included in VII.I. Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in VII.J.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The systems that interface with the auxiliary and radwaste area ventilation system are the control room area ventilation system (VII.F1) and the diesel generator building ventilation system (VII.F4). The cooling coils receive their cooling water from other systems, such as the hot water heating system or the chilled water cooling system.

VII AUXILIARY SYSTEMS
F2 Auxiliary and Radwaste Area Ventilation System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.F2.AP-99	VII.F2-1(A-09)	Ducting and components	Stainless steel	Condensation	Loss of material due to pitting and crevice corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
VII.F2.A-10	VII.F2-2(A-10)	Ducting and components (External surfaces)	Steel	Air – indoor, uncontrolled (External)	Loss of material due to general corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
VII.F2.A-08	VII.F2-3(A-08)	Ducting and components (Internal surfaces)	Steel	Condensation (Internal)	Loss of material due to general, pitting, crevice, and (for drip pans and drain lines) microbiologically-influenced corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
VII.F2.A-105	VII.F2-4(A-105)	Ducting; closure bolting	Steel	Air – indoor, uncontrolled (External)	Loss of material due to general corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
VII.F2.AP-113	VII.F2-5(A-73)	Elastomer: seals and components	Elastomers	Air – indoor, uncontrolled (External)	Loss of material due to wear	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
VII.F2.AP-103	VII.F2-6(A-18)	Elastomer: seals and components	Elastomers	Air – indoor, uncontrolled (Internal)	Loss of material due to wear	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
VII.F2.AP-102	VII.F2-7(A-17)	Elastomer: seals and components	Elastomers	Air – indoor, uncontrolled (Internal/External)	Hardening and loss of strength due to elastomer degradation	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No

VII AUXILIARY SYSTEMS
F2 Auxiliary and Radwaste Area Ventilation System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.F2.AP-41	VII.F2-8(AP-41)	Heat exchanger components	Steel	Air – indoor, uncontrolled (External)	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
VII.F2.AP-189	VII.F2-9(A-63)	Heat exchanger components	Steel	Closed-cycle cooling water	Loss of material due to general, pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.F2.AP-205	VII.F2-10(AP-80)	Heat exchanger tubes	Copper Alloy	Closed-cycle cooling water	Reduction of heat transfer due to fouling	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.F2.AP-204	VII.F2-11(AP-77)	Heat exchanger tubes	Steel	Closed-cycle cooling water	Reduction of heat transfer due to fouling	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.F2.AP-142	VII.F2-12(AP-74)	Piping, piping components, and piping elements	Aluminum	Condensation (Internal)	Loss of material due to pitting and crevice corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
VII.F2.AP-199	VII.F2-13(AP-12)	Piping, piping components, and piping elements	Copper alloy	Closed-cycle cooling water	Loss of material due to general, pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.F2.AP-109	VII.F2-14(A-46)	Piping, piping components, and piping elements	Copper alloy	Condensation (External)	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
VII.F2.AP-43	VII.F2-15(AP-43)	Piping, piping components, and piping elements	Copper alloy (>15% Zn or >8% Al)	Closed-cycle cooling water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No

VII AUXILIARY SYSTEMS
F2 Auxiliary and Radwaste Area Ventilation System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.F2.AP-31	VII.F2-16(AP-31)	Piping, piping components, and piping elements	Gray cast iron	Treated water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VII.F2.AP-127	VII.F2-17(AP-30)	Piping, piping components, and piping elements	Steel	Lubricating oil	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
VII.F2.AP-209		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated
VII.F2.AP-221		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Loss of material due to pitting and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated
VII.F2.AP-202	VII.F2-18(A-25)	Piping, piping components, and piping elements; tanks	Steel	Closed-cycle cooling water	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No

F3. PRIMARY CONTAINMENT HEATING AND VENTILATION SYSTEM

Systems, Structures, and Components

This section discusses the primary containment heating and ventilation system (with warm moist air as the normal environment), which contains ducts, piping and fittings, equipment frames and housings, flexible collars and seals, filters, and heating and cooling air handlers. Based on Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” all components that comprise the primary containment heating and ventilation system are governed by Group C Quality Standards.

With respect to filters and seals, these items are to be addressed consistent with the Nuclear Regulatory Commission (NRC) position on consumables, provided in the NRC letter from Christopher I. Grimes to Douglas J. Walters of the Nuclear Energy Institute (NEI), dated March 10, 2000. Specifically, components that function as system filters and seals 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 an aging management review under 10 CFR 54.21(a)(1)(ii). As part of the methodology description, the application should identify the standards that are relied on for replacement, for example, National Fire Protection Association (NFPA) standards for fire protection equipment.

Aging management programs for the degradation of external surfaces of components and miscellaneous bolting are included in VII.I. Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in VII.J.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The systems that interface with the primary containment heating and ventilation system are the closed-cycle cooling water system (VII.C2) and the PWR and BWR containments (II.A and II.B, respectively). The cooling coils receive their cooling water from other systems, such as the hot water heating system or the chilled water cooling system.

VII AUXILIARY SYSTEMS
F3 Primary Containment Heating and Ventilation System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.F3.AP-99	VII.F3-1(A-09)	Ducting and components	Stainless steel	Condensation	Loss of material due to pitting and crevice corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
VII.F3.A-10	VII.F3-2(A-10)	Ducting and components (External surfaces)	Steel	Air – indoor, uncontrolled (External)	Loss of material due to general corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
VII.F3.A-08	VII.F3-3(A-08)	Ducting and components (Internal surfaces)	Steel	Condensation (Internal)	Loss of material due to general, pitting, crevice, and (for drip pans and drain lines) microbiologically-influenced corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
VII.F3.A-105	VII.F3-4(A-105)	Ducting; closure bolting	Steel	Air – indoor, uncontrolled (External)	Loss of material due to general corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
VII.F3.AP-113	VII.F3-5(A-73)	Elastomer: seals and components	Elastomers	Air – indoor, uncontrolled (External)	Loss of material due to wear	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
VII.F3.AP-103	VII.F3-6(A-18)	Elastomer: seals and components	Elastomers	Air – indoor, uncontrolled (Internal)	Loss of material due to wear	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
VII.F3.AP-102	VII.F3-7(A-17)	Elastomer: seals and components	Elastomers	Air – indoor, uncontrolled (Internal/External)	Hardening and loss of strength due to elastomer degradation	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No

VII AUXILIARY SYSTEMS
F3 Primary Containment Heating and Ventilation System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.F3.AP-203	VII.F3-8(AP-34)	Heat exchanger components	Copper alloy	Closed-cycle cooling water	Loss of material due to general, pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.F3.AP-65	VII.F3-9(AP-65)	Heat exchanger components	Copper alloy (>15% Zn or >8% Al)	Treated water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VII.F3.AP-41	VII.F3-10(AP-41)	Heat exchanger components	Steel	Air – indoor, uncontrolled (External)	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
VII.F3.AP-189	VII.F3-11(A-63)	Heat exchanger components	Steel	Closed-cycle cooling water	Loss of material due to general, pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.F3.AP-205	VII.F3-12(AP-80)	Heat exchanger tubes	Copper Alloy	Closed-cycle cooling water	Reduction of heat transfer due to fouling	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.F3.AP-204	VII.F3-13(AP-77)	Heat exchanger tubes	Steel	Closed-cycle cooling water	Reduction of heat transfer due to fouling	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.F3.AP-142	VII.F3-14(AP-74)	Piping, piping components, and piping elements	Aluminum	Condensation (Internal)	Loss of material due to pitting and crevice corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
VII.F3.AP-199	VII.F3-15(AP-12)	Piping, piping components, and piping elements	Copper alloy	Closed-cycle cooling water	Loss of material due to general, pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No

VII AUXILIARY SYSTEMS
F3 Primary Containment Heating and Ventilation System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.F3.AP-109	VII.F3-16(A-46)	Piping, piping components, and piping elements	Copper alloy	Condensation (External)	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
VII.F3.AP-43	VII.F3-17(AP-43)	Piping, piping components, and piping elements	Copper alloy (>15% Zn or >8% Al)	Closed-cycle cooling water	Loss of material due to selective leaching	Chapter XI.M33, "Selective Leaching"	No
VII.F3.A-50	VII.F3-18(A-50)	Piping, piping components, and piping elements	Gray cast iron	Closed-cycle cooling water	Loss of material due to selective leaching	Chapter XI.M33, "Selective Leaching"	No
VII.F3.AP-127	VII.F3-19(AP-30)	Piping, piping components, and piping elements	Steel	Lubricating oil	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, "One-Time Inspection"	No
VII.F3.AP-202	VII.F3-20(A-25)	Piping, piping components, and piping elements; tanks	Steel	Closed-cycle cooling water	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No

F4. DIESEL GENERATOR BUILDING VENTILATION SYSTEM

Systems, Structures, and Components

This section discusses the diesel generator building ventilation system (with warm moist air as the normal environment), which contains ducts, piping and fittings, equipment frames and housings, flexible collars and seals, and heating and cooling air handlers. Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all components that comprise the diesel generator building ventilation system are governed by Group C Quality Standards.

With respect to filters and seals, these items are to be addressed consistent with the Nuclear Regulatory Commission (NRC) position on consumables, provided in the NRC letter from Christopher I. Grimes to Douglas J. Walters of the Nuclear Energy Institute (NEI), dated March 10, 2000. Specifically, components that function as system seals 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 an aging management review under 10 CFR 54.21(a)(1)(ii). As part of the methodology description, the application should identify the standards that are relied on for replacement, for example, National Fire Protection Association (NFPA) standards for fire protection equipment.

Aging management programs for the degradation of external surfaces of components and miscellaneous bolting are included in VII.I. Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in VII.J.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The system that interfaces with the diesel generator building system is the auxiliary and radwaste area ventilation system (VII.F2). The cooling coils receive their cooling water from other systems, such as the hot water heating system or the chilled water cooling system.

VII AUXILIARY SYSTEMS
F4 Diesel Generator Building Ventilation System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.F4.A-10	VII.F4-1(A-10)	Ducting and components (External surfaces)	Steel	Air – indoor, uncontrolled (External)	Loss of material due to general corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
VII.F4.A-08	VII.F4-2(A-08)	Ducting and components (Internal surfaces)	Steel	Condensation (Internal)	Loss of material due to general, pitting, crevice, and (for drip pans and drain lines) microbiologically-influenced corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
VII.F4.A-105	VII.F4-3(A-105)	Ducting; closure bolting	Steel	Air – indoor, uncontrolled (External)	Loss of material due to general corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
VII.F4.AP-113	VII.F4-4(A-73)	Elastomer: seals and components	Elastomers	Air – indoor, uncontrolled (External)	Loss of material due to wear	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
VII.F4.AP-103	VII.F4-5(A-18)	Elastomer: seals and components	Elastomers	Air – indoor, uncontrolled (Internal)	Loss of material due to wear	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
VII.F4.AP-102	VII.F4-6(A-17)	Elastomer: seals and components	Elastomers	Air – indoor, uncontrolled (Internal/External)	Hardening and loss of strength due to elastomer degradation	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
VII.F4.AP-41	VII.F4-7(AP-41)	Heat exchanger components	Steel	Air – indoor, uncontrolled (External)	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.F4.AP-189	VII.F4-8(A-63)	Heat exchanger components	Steel	Closed-cycle cooling water	Loss of material due to general, pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.F4.AP-204	VII.F4-9(AP-77)	Heat exchanger tubes	Steel	Closed-cycle cooling water	Reduction of heat transfer due to fouling	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.F4.AP-142	VII.F4-10(AP-74)	Piping, piping components, and piping elements	Aluminum	Condensation (Internal)	Loss of material due to pitting and crevice corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
VII.F4.AP-199	VII.F4-11(AP-12)	Piping, piping components, and piping elements	Copper alloy	Closed-cycle cooling water	Loss of material due to general, pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.F4.AP-109	VII.F4-12(A-46)	Piping, piping components, and piping elements	Copper alloy	Condensation (External)	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
VII.F4.AP-43	VII.F4-13(AP-43)	Piping, piping components, and piping elements	Copper alloy (>15% Zn or >8% Al)	Closed-cycle cooling water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VII.F4.AP-31	VII.F4-14(AP-31)	Piping, piping components, and piping elements	Gray cast iron	Treated water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VII.F4.AP-127	VII.F4-15(AP-30)	Piping, piping components, and piping elements	Steel	Lubricating oil	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No

VII AUXILIARY SYSTEMS
F4 Diesel Generator Building Ventilation System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.F4.AP-209		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated
VII.F4.AP-221		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Loss of material due to pitting and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated
VII.F4.AP-202	VII.F4-16(A-25)	Piping, piping components, and piping elements; tanks	Steel	Closed-cycle cooling water	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No

G. FIRE PROTECTION

Systems, Structures, and Components

This section discusses the fire protection systems for both boiling water reactors (BWRs) and pressurized water reactors (PWRs), which consist of several Class 1 structures, mechanical systems, and electrical components. The Class 1 structures include the intake structure, the turbine building, the auxiliary building, the diesel generator building, and the primary containment. Structural components include fire barrier walls, ceilings, floors, fire doors, and penetration seals. Mechanical systems include the high pressure service water system, the reactor coolant pump oil collect system, and the diesel fire system. Mechanical components include piping and fittings, filters, fire hydrants, mulsifiers, pumps, sprinklers, strainers, and valves (including containment isolation valves). Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all components that comprise the fire protection system are governed by Group C Quality Standards.

With respect to filters, seals, portable fire extinguishers, and fire hoses, these items are to be addressed consistent with the NRC position on consumables, provided in the NRC letter from Christopher I. Grimes to Douglas J. Walters of the Nuclear Energy Institute (NEI), dated March 10, 2000. Specifically, components that function as system filters, seals, portable fire extinguishers, and fire hoses 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 an aging management review under 10 CFR 54.21(a)(1)(ii). As part of the methodology description, the application should identify the standards that are relied on for replacement, for example, National Fire Protection Association (NFPA) standards for fire protection equipment.

Pump and valve internals perform their intended functions with moving parts or with a change in configuration. They are also subject to replacement based on qualified life or a specified time period. Pursuant to 10 CFR 54.21(a)(1), therefore, they are not subject to an aging management review.

Aging management programs for the degradation of external surfaces of components and miscellaneous bolting are included in VII.I. Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in VII.J.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The systems and structures that interface with the fire protection system include various Class 1 structures and component supports (III.A and III.B), the electrical components (VI.A and VI.B), the closed-cycle cooling water system (VII.C2), and the diesel fuel oil system (VII.H1).

VII AUXILIARY SYSTEMS
G Fire Protection

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.G.A-19	VII.G-1(A-19)	Fire barrier penetration seals	Elastomers	Air - indoor, uncontrolled	Increased hardness; shrinkage; loss of strength due to weathering	Chapter XI.M26, "Fire Protection"	No
VII.G.A-20	VII.G-2(A-20)	Fire barrier penetration seals	Elastomers	Air – outdoor	Increased hardness; shrinkage; loss of strength due to weathering	Chapter XI.M26, "Fire Protection"	No
VII.G.AP-149		Fire Hydrants	Steel	Air – outdoor	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M27, "Fire Water System"	No
VII.G.A-21	VII.G-3(A-21)	Fire rated doors	Steel	Air - indoor, uncontrolled	Loss of material due to wear	Chapter XI.M26, "Fire Protection"	No
VII.G.A-22	VII.G-4(A-22)	Fire rated doors	Steel	Air – outdoor	Loss of material due to wear	Chapter XI.M26, "Fire Protection"	No
VII.G.AP-150		Halon/carbon dioxide fire suppression system piping, piping components, and piping elements	Steel	Air – indoor, uncontrolled (External)	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M26, "Fire Protection"	No
VII.G.AP-41	VII.G-5(AP-41)	Heat exchanger components	Steel	Air – indoor, uncontrolled (External)	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No

VII AUXILIARY SYSTEMS
G Fire Protection

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.G.AP-40	VII.G-6(AP-40)	Heat exchanger components	Steel	Air – outdoor (External)	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
VII.G.AP-187	VII.G-7(AP-61)	Heat exchanger tubes	Stainless steel	Raw water	Reduction of heat transfer due to fouling	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VII.G.AP-180	VII.G-8(AP-83)	Piping, piping components, and piping elements	Aluminum	Raw water	Loss of material due to pitting and crevice corrosion	Chapter XI.M27, "Fire Water System"	No
VII.G.AP-143	VII.G-9(AP-78)	Piping, piping components, and piping elements	Copper alloy	Condensation (Internal)	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
VII.G.AP-132	VII.G-10(AP-44)	Piping, piping components, and piping elements	Copper alloy	Fuel oil	Loss of material due to general, pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M30, "Fuel Oil Chemistry," and Chapter XI.M32, "One-Time Inspection"	No
VII.G.AP-133	VII.G-11(AP-47)	Piping, piping components, and piping elements	Copper alloy	Lubricating oil	Loss of material due to pitting and crevice corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, "One-Time Inspection"	No
VII.G.AP-197	VII.G-12(A-45)	Piping, piping components, and piping elements	Copper alloy	Raw water	Loss of material due to general, pitting, crevice, and microbiologically-influenced corrosion; fouling that leads to corrosion	Chapter XI.M27, "Fire Water System"	No

VII AUXILIARY SYSTEMS
G Fire Protection

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.G.A-47	VII.G-13(A-47)	Piping, piping components, and piping elements	Copper alloy (>15% Zn or >8% Al)	Raw water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VII.G.A-51	VII.G-14(A-51)	Piping, piping components, and piping elements	Gray cast iron	Raw water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VII.G.A-02	VII.G-15(A-02)	Piping, piping components, and piping elements	Gray cast iron	Soil	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VII.G.AP-31	VII.G-16(AP-31)	Piping, piping components, and piping elements	Gray cast iron	Treated water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VII.G.AP-136	VII.G-17(AP-54)	Piping, piping components, and piping elements	Stainless steel	Fuel oil	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M30, “Fuel Oil Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VII.G.AP-138	VII.G-18(AP-59)	Piping, piping components, and piping elements	Stainless steel	Lubricating oil	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
VII.G.A-55	VII.G-19(A-55)	Piping, piping components, and piping elements	Stainless steel	Raw water	Loss of material due to pitting and crevice corrosion; fouling that leads to corrosion	Chapter XI.M27, "Fire Water System"	No
VII.G.AP-137	VII.G-20(AP-56)	Piping, piping components, and piping elements	Stainless steel	Soil	Loss of material due to pitting and crevice corrosion	Chapter XI.M41, "Buried and Underground Piping and Tanks"	No

VII AUXILIARY SYSTEMS
G Fire Protection

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.G.AP-234	VII.G-21(A-28)	Piping, piping components, and piping elements	Steel	Fuel oil	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M30, "Fuel Oil Chemistry", and Chapter XI.M32, "One-Time Inspection"	No
VII.G.AP-127	VII.G-22(AP-30)	Piping, piping components, and piping elements	Steel	Lubricating oil	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
VII.G.A-23	VII.G-23(A-23)	Piping, piping components, and piping elements	Steel	Moist air or condensation (Internal)	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
VII.G.A-33	VII.G-24(A-33)	Piping, piping components, and piping elements	Steel	Raw water	Loss of material due to general, pitting, crevice, and microbiologically-influenced corrosion; fouling that leads to corrosion	Chapter XI.M27, "Fire Water System"	No
VII.G.AP-198	VII.G-25(A-01)	Piping, piping components, and piping elements	Steel (with coating or wrapping)	Soil	Loss of material due to general, pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M41, "Buried and Underground Piping and Tanks"	No
VII.G.AP-209		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated

VII AUXILIARY SYSTEMS
G Fire Protection

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.G.AP-221		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Loss of material due to pitting and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated
VII.G.AP-117	VII.G-26(A-83)	Reactor coolant pump oil collection system: piping, tubing, valve bodies	Steel	Lubricating oil	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
VII.G.AP-116	VII.G-27(A-82)	Reactor coolant pump oil collection system: tanks	Steel	Lubricating oil	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
VII.G.A-90	VII.G-28(A-90)	Structural fire barriers: walls, ceilings and floors	Reinforced concrete	Air - indoor, uncontrolled	Concrete cracking and spalling due to aggressive chemical attack, and reaction with aggregates	Chapter XI.M26, “Fire Protection,” and Chapter XI.S6, “Structures Monitoring”	No
VII.G.A-91	VII.G-29(A-91)	Structural fire barriers: walls, ceilings and floors	Reinforced concrete	Air - indoor, uncontrolled	Loss of material due to corrosion of embedded steel	Chapter XI.M26, “Fire Protection,” and Chapter XI.S6, “Structures Monitoring”	No
VII.G.A-92	VII.G-30(A-92)	Structural fire barriers: walls, ceilings and floors	Reinforced concrete	Air – outdoor	Cracking, loss of material due to freeze-thaw, aggressive chemical attack, and reaction with aggregates	Chapter XI.M26, “Fire Protection,” and Chapter XI.S6, “Structures Monitoring”	No

VII AUXILIARY SYSTEMS
G Fire Protection

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.G.A-93	VII.G-31(A-93)	Structural fire barriers: walls, ceilings and floors	Reinforced concrete	Air – outdoor	Loss of material due to corrosion of embedded steel	Chapter XI.M26, “Fire Protection,” and Chapter XI.S6, “Structures Monitoring”	No

H1. DIESEL FUEL OIL SYSTEM

Systems, Structures, and Components

This section discusses the diesel fuel oil system, which consists of aboveground and underground piping, valves, pumps, and tanks. Based on Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” all components that comprise the diesel fuel oil system are governed by Group C Quality Standards.

Aging management programs for the degradation of external surfaces of components and miscellaneous bolting are included in VII.I. Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in VII.J.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The systems that interface with the diesel fuel oil system are the fire protection (VII.G) and emergency diesel generator systems (VII.H2).

VII AUXILIARY SYSTEMS
H1 Diesel Fuel Oil System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.H1.AP-129	VII.H1-1(AP-35)	Piping, piping components, and piping elements	Aluminum	Fuel oil	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M30, “Fuel Oil Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VII.H1.AP-199	VII.H1-2(AP-12)	Piping, piping components, and piping elements	Copper alloy	Closed-cycle cooling water	Loss of material due to general, pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.H1.AP-132	VII.H1-3(AP-44)	Piping, piping components, and piping elements	Copper alloy	Fuel oil	Loss of material due to general, pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M30, “Fuel Oil Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VII.H1.AP-43	VII.H1-4(AP-43)	Piping, piping components, and piping elements	Copper alloy (>15% Zn or >8% Al)	Closed-cycle cooling water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VII.H1.A-02	VII.H1-5(A-02)	Piping, piping components, and piping elements	Gray cast iron	Soil	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VII.H1.AP-136	VII.H1-6(AP-54)	Piping, piping components, and piping elements	Stainless steel	Fuel oil	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M30, “Fuel Oil Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No

VII AUXILIARY SYSTEMS
H1 Diesel Fuel Oil System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.H1.AP-137	VII.H1-7(AP-56)	Piping, piping components, and piping elements	Stainless steel	Soil	Loss of material due to pitting and crevice corrosion	Chapter XI.M41, "Buried and Underground Piping and Tanks"	No
VII.H1.A-24	VII.H1-8(A-24)	Piping, piping components, and piping elements	Steel	Air – outdoor (External)	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
VII.H1.AP-198	VII.H1-9(A-01)	Piping, piping components, and piping elements	Steel (with coating or wrapping)	Soil	Loss of material due to general, pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M41, "Buried and Underground Piping and Tanks"	No
VII.H1.AP-209		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated
VII.H1.AP-221		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Loss of material due to pitting and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated
VII.H1.AP-105	VII.H1-10(A-30)	Piping, piping components, and piping elements; tanks	Steel	Fuel oil	Loss of material due to general, pitting, crevice, and microbiologically-influenced corrosion; fouling that leads to corrosion	Chapter XI.M30, “Fuel Oil Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No

VII AUXILIARY SYSTEMS
H1 Diesel Fuel Oil System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.H1.A-95	VII.H1-11(A-95)	Tanks	Steel	Air – outdoor (External)	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M29, "Aboveground Metallic Tanks"	No

H2. EMERGENCY DIESEL GENERATOR SYSTEM

Systems, Structures, and Components

This section discusses the emergency diesel generator system, which contains piping, valves, filters, mufflers, strainers, and tanks. Based on Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” all components that comprise the emergency diesel generator system are governed by Group C Quality Standards.

With respect to filters and seals, these items are to be addressed consistent with the NRC position on consumables, provided in the NRC letter from Christopher I. Grimes to Douglas J. Walters of the Nuclear Energy Institute (NEI), dated March 10, 2000. Specifically, components that function as system filters 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 an aging management review under 10 CFR 54.21(a)(1)(ii). As part of the methodology description, the application should identify the standards that are relied on for replacement, for example, National Fire Protection Association (NFPA) standards for fire protection equipment.

Aging management programs for the degradation of external surfaces of components and miscellaneous bolting are included in VII.I. Common miscellaneous material/environment combinations where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation are included in VII.J.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The systems that interface with the emergency diesel generator system include the diesel fuel oil system (VII.H1), the closed-cycle cooling water system (VII.C2) and the open-cycle cooling water system (VII.C1) for some plants.

VII AUXILIARY SYSTEMS
H2 Emergency Diesel Generator System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.H2.AP-128	VII.H2-1(AP-33)	Diesel engine exhaust piping, piping components, and piping elements	Stainless steel	Diesel exhaust	Cracking due to stress corrosion cracking	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
VII.H2.AP-41	VII.H2-3(AP-41)	Heat exchanger components	Steel	Air – indoor, uncontrolled (External)	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
VII.H2.AP-40	VII.H2-4(AP-40)	Heat exchanger components	Steel	Air – outdoor (External)	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
VII.H2.AP-131	VII.H2-5(AP-39)	Heat exchanger components	Steel	Lubricating oil	Loss of material due to general, pitting, crevice, and microbiologically-influenced corrosion; fouling that leads to corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
VII.H2.AP-154		Heat exchanger tubes	Aluminum	Lubricating oil	Reduction of heat transfer due to fouling	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
VII.H2.AP-187	VII.H2-6(AP-61)	Heat exchanger tubes	Stainless steel	Raw water	Reduction of heat transfer due to fouling	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VII.H2.AP-255		Piping, piping components, and piping elements	Aluminum	Closed-cycle cooling water	Loss of material due to pitting and crevice corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No

VII AUXILIARY SYSTEMS
H2 Emergency Diesel Generator System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.H2.AP-129	VII.H2-7(AP-35)	Piping, piping components, and piping elements	Aluminum	Fuel oil	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M30, “Fuel Oil Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VII.H2.AP-162		Piping, piping components, and piping elements	Aluminum	Lubricating oil	Loss of material due to pitting and crevice corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
VII.H2.AP-258		Piping, piping components, and piping elements	Aluminum	Treated water	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VII.H2.AP-199	VII.H2-8(AP-12)	Piping, piping components, and piping elements	Copper alloy	Closed-cycle cooling water	Loss of material due to general, pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.H2.AP-132	VII.H2-9(AP-44)	Piping, piping components, and piping elements	Copper alloy	Fuel oil	Loss of material due to general, pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M30, “Fuel Oil Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VII.H2.AP-133	VII.H2-10(AP-47)	Piping, piping components, and piping elements	Copper alloy	Lubricating oil	Loss of material due to pitting and crevice corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No

VII AUXILIARY SYSTEMS
H2 Emergency Diesel Generator System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.H2.AP-193	VII.H2-11(AP-45)	Piping, piping components, and piping elements	Copper alloy	Raw water	Loss of material due to general, pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VII.H2.AP-43	VII.H2-12(AP-43)	Piping, piping components, and piping elements	Copper alloy (>15% Zn or >8% Al)	Closed-cycle cooling water	Loss of material due to selective leaching	Chapter XI.M33, "Selective Leaching"	No
VII.H2.A-47	VII.H2-13(A-47)	Piping, piping components, and piping elements	Copper alloy (>15% Zn or >8% Al)	Raw water	Loss of material due to selective leaching	Chapter XI.M33, "Selective Leaching"	No
VII.H2.A-51	VII.H2-14(A-51)	Piping, piping components, and piping elements	Gray cast iron	Raw water	Loss of material due to selective leaching	Chapter XI.M33, "Selective Leaching"	No
VII.H2.A-02	VII.H2-15(A-02)	Piping, piping components, and piping elements	Gray cast iron	Soil	Loss of material due to selective leaching	Chapter XI.M33, "Selective Leaching"	No
VII.H2.AP-136	VII.H2-16(AP-54)	Piping, piping components, and piping elements	Stainless steel	Fuel oil	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M30, "Fuel Oil Chemistry," and Chapter XI.M32, "One-Time Inspection"	No
VII.H2.AP-138	VII.H2-17(AP-59)	Piping, piping components, and piping elements	Stainless steel	Lubricating oil	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, "One-Time Inspection"	No

VII AUXILIARY SYSTEMS
H2 Emergency Diesel Generator System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.H2.AP-55	VII.H2-18(AP-55)	Piping, piping components, and piping elements	Stainless steel	Raw water	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VII.H2.AP-137	VII.H2-19(AP-56)	Piping, piping components, and piping elements	Stainless steel	Soil	Loss of material due to pitting and crevice corrosion	Chapter XI.M41, "Buried and Underground Piping and Tanks"	No
VII.H2.AP-127	VII.H2-20(AP-30)	Piping, piping components, and piping elements	Steel	Lubricating oil	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
VII.H2.A-23	VII.H2-21(A-23)	Piping, piping components, and piping elements	Steel	Moist air or condensation (Internal)	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
VII.H2.AP-194	VII.H2-22(A-38)	Piping, piping components, and piping elements	Steel (with coating or lining)	Raw water	Loss of material due to general, pitting, crevice, and microbiologically-influenced corrosion; fouling that leads to corrosion; lining/coating degradation	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VII.H2.AP-104	VII.H2-2(A-27)	Piping, piping components, and piping elements, diesel engine exhaust	Steel; stainless steel	Diesel exhaust	Loss of material due to general (steel only), pitting, and crevice corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No

VII AUXILIARY SYSTEMS
H2 Emergency Diesel Generator System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.H2.AP-209		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated
VII.H2.AP-221		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Loss of material due to pitting and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated
VII.H2.AP-202	VII.H2-23(A-25)	Piping, piping components, and piping elements; tanks	Steel	Closed-cycle cooling water	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VII.H2.AP-105	VII.H2-24(A-30)	Piping, piping components, and piping elements; tanks	Steel	Fuel oil	Loss of material due to general, pitting, crevice, and microbiologically-influenced corrosion; fouling that leads to corrosion	Chapter XI.M30, “Fuel Oil Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No

I. EXTERNAL SURFACES OF COMPONENTS AND MISCELLANEOUS BOLTING

Systems, Structures, and Components

This section addresses the aging management programs for the external surfaces of all steel structures and components, including closure bolting in the Auxiliary Systems in pressurized water reactors (PWRs) and boiling water reactors (BWRs). For the steel components in PWRs, this section addresses only boric acid corrosion of external surface as a result of dripping borated water that is leaking from an adjacent PWR component. Boric acid corrosion can also occur for 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 VII.

System Interfaces

The structures and components covered in this section belong to the Auxiliary Systems in PWRs and BWRs. (For example, see System Interfaces in VII.A1 to VII.H2 for details.)

VII AUXILIARY SYSTEMS
I External Surfaces of Components and Miscellaneous Bolting

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.I.AP-261		Bolting	Copper alloy	Any environment	Loss of preload due to thermal effects, gasket creep, and self-loosening	Chapter XI.M18, "Bolting Integrity"	No
VII.I.AP-262		Bolting	Nickel alloy	Any environment	Loss of preload due to thermal effects, gasket creep, and self-loosening	Chapter XI.M18, "Bolting Integrity"	No
VII.I.AP-243		Bolting	Stainless steel	Soil or concrete	Loss of material due to pitting and crevice corrosion	Chapter XI.M41, "Buried and Underground Piping and Tanks"	No
VII.I.AP-244		Bolting	Stainless steel	Soil	Loss of preload	Chapter XI.M18, "Bolting Integrity"	No
VII.I.AP-265		Bolting	Stainless steel	Treated borated water	Loss of preload due to thermal effects, gasket creep, and self-loosening	Chapter XI.M18, "Bolting Integrity"	No
VII.I.A-102	VII.I-2(A-102)	Bolting	Steel	Air with borated water leakage	Loss of material due to boric acid corrosion	Chapter XI.M10, “Boric Acid Corrosion”	No
VII.I.AP-241		Bolting	Steel	Soil or concrete	Loss of material due to general, pitting and crevice corrosion	Chapter XI.M41, "Buried and Underground Piping and Tanks"	No
VII.I.AP-242		Bolting	Steel	Soil	Loss of preload	Chapter XI.M18, "Bolting Integrity"	No

VII AUXILIARY SYSTEMS
I External Surfaces of Components and Miscellaneous Bolting

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.I.AP-126	VII.I-1(AP-28)	Bolting	Steel; stainless steel	Air – outdoor (External)	Loss of material due to general (steel only), pitting, and crevice corrosion	Chapter XI.M18, "Bolting Integrity"	No
VII.I.AP-263		Bolting	Steel; stainless steel	Air – outdoor (External)	Loss of preload due to thermal effects, gasket creep, and self-loosening	Chapter XI.M18, "Bolting Integrity"	No
VII.I.AP-266		Bolting	Steel; stainless steel	Fuel oil	Loss of preload due to thermal effects, gasket creep, and self-loosening	Chapter XI.M18, "Bolting Integrity"	No
VII.I.AP-264		Bolting	Steel; stainless steel	Raw water	Loss of preload due to thermal effects, gasket creep, and self-loosening	Chapter XI.M18, "Bolting Integrity"	No
VII.I.AP-267		Bolting	Steel; stainless steel	Treated water	Loss of preload due to thermal effects, gasket creep, and self-loosening	Chapter XI.M18, "Bolting Integrity"	No
VII.I.A-03	VII.I-6(A-03)	Closure bolting	Steel	Air with steam or water leakage	Loss of material due to general corrosion	Chapter XI.M18, "Bolting Integrity"	No
VII.I.A-04	VII.I-3(A-04)	Closure bolting	Steel, high-strength	Air with steam or water leakage	Cracking due to stress corrosion cracking; cyclic loading	Chapter XI.M18, "Bolting Integrity"	No

VII AUXILIARY SYSTEMS
I External Surfaces of Components and Miscellaneous Bolting

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.I.AP-125	VII.I-4(AP-27)	Closure bolting	Steel; stainless steel	Air – indoor, uncontrolled (External)	Loss of material due to general (steel only), pitting, and crevice corrosion	Chapter XI.M18, "Bolting Integrity"	No
VII.I.AP-124	VII.I-5(AP-26)	Closure bolting	Steel; stainless steel	Air – indoor, uncontrolled (External)	Loss of preload due to thermal effects, gasket creep, and self-loosening	Chapter XI.M18, "Bolting Integrity"	No
VII.I.A-105	VII.I-7(A-105)	Ducting; closure bolting	Steel	Air – indoor, uncontrolled (External)	Loss of material due to general corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
VII.I.A-77	VII.I-8(A-77)	External surfaces	Steel	Air – indoor, uncontrolled (External)	Loss of material due to general corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
VII.I.A-78	VII.I-9(A-78)	External surfaces	Steel	Air – outdoor (External)	Loss of material due to general corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
VII.I.A-79	VII.I-10(A-79)	External surfaces	Steel	Air with borated water leakage	Loss of material due to boric acid corrosion	Chapter XI.M10, “Boric Acid Corrosion”	No
VII.I.A-81	VII.I-11(A-81)	External surfaces	Steel	Condensation (External)	Loss of material due to general corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
VII.I.AP-256		Piping, piping components, and piping elements	Aluminum	Air - outdoor	Loss of material due to pitting and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No

VII AUXILIARY SYSTEMS
I External Surfaces of Components and Miscellaneous Bolting

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.I.AP-159		Piping, piping components, and piping elements	Copper alloy	Air – outdoor (External)	Loss of material due to pitting and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
VII.I.AP-66	VII.I-12(AP-66)	Piping, piping components, and piping elements	Copper alloy (>15% Zn or >8% Al)	Air with borated water leakage	Loss of material due to boric acid corrosion	Chapter XI.M10, “Boric Acid Corrosion”	No
VII.I.AP-284		Underground piping, piping components, and piping elements	Steel; stainless steel; copper alloy; aluminum	Air-indoor uncontrolled or condensation (external)	Loss of material due to general (steel only), pitting and crevice corrosion	Chapter XI.M41, “Buried and Underground Piping and Tanks”	No

J. COMMON MISCELLANEOUS MATERIAL/ENVIRONMENT COMBINATIONS

Systems, Structures, and Components

This section addresses the aging management programs for miscellaneous material/environment combinations which may be found throughout structures and components for auxiliary systems. For the material/environment combinations in this part, aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation. Therefore, no resulting aging management programs for these structures and components are required.

System Interfaces

The structures and components covered in this section belong to the auxiliary systems in pressurized water reactors (PWRs) and boiling water reactors (BWRs). (For example, see System Interfaces in VII.A to VII.I for details.)

VII AUXILIARY SYSTEMS
J Common Miscellaneous Material/Environment Combinations

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.J.AP-151		Heat exchanger components	Titanium	Air – indoor, uncontrolled or Air – outdoor	None	None	No
VII.J.AP-48	VII.J-7(AP-48)	Piping elements	Glass	Air	None	None	No
VII.J.AP-14	VII.J-8(AP-14)	Piping elements	Glass	Air – indoor, uncontrolled (External)	None	None	No
VII.J.AP-167		Piping elements	Glass	Air – outdoor	None	None	No
VII.J.AP-96		Piping elements	Glass	Air with borated water leakage	None	None	No
VII.J.AP-166		Piping elements	Glass	Closed-cycle cooling water	None	None	No
VII.J.AP-97		Piping elements	Glass	Condensation (Internal/External)	None	None	No
VII.J.AP-49	VII.J-9(AP-49)	Piping elements	Glass	Fuel oil	None	None	No
VII.J.AP-98		Piping elements	Glass	Gas	None	None	No

VII AUXILIARY SYSTEMS
J Common Miscellaneous Material/Environment Combinations

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.J.AP-15	VII.J-10(AP-15)	Piping elements	Glass	Lubricating oil	None	None	No
VII.J.AP-50	VII.J-11(AP-50)	Piping elements	Glass	Raw water	None	None	No
VII.J.AP-52	VII.J-12(AP-52)	Piping elements	Glass	Treated borated water	None	None	No
VII.J.AP-51	VII.J-13(AP-51)	Piping elements	Glass	Treated water	None	None	No
VII.J.AP-36	VII.J-1(AP-36)	Piping, piping components, and piping elements	Aluminum	Air – indoor, controlled (External)	None	None	No
VII.J.AP-134		Piping, piping components, and piping elements	Aluminum	Air – dry (Internal/External)	None	None	No
VII.J.AP-135		Piping, piping components, and piping elements	Aluminum	Air – indoor, uncontrolled (Internal/External)	None	None	No
VII.J.AP-37	VII.J-2(AP-37)	Piping, piping components, and piping elements	Aluminum	Gas	None	None	No

VII AUXILIARY SYSTEMS
J Common Miscellaneous Material/Environment Combinations

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.J.AP-8	VII.J-3(AP-8)	Piping, piping components, and piping elements	Copper alloy	Air – dry	None	None	No
VII.J.AP-144		Piping, piping components, and piping elements	Copper alloy	Air – indoor, uncontrolled (Internal/External)	None	None	No
VII.J.AP-9	VII.J-4(AP-9)	Piping, piping components, and piping elements	Copper alloy	Gas	None	None	No
VII.J.AP-11	VII.J-5(AP-11)	Piping, piping components, and piping elements	Copper alloy (≤15% Zn and ≤8% Al)	Air with borated water leakage	None	None	No
VII.J.AP-13	VII.J-6(AP-13)	Piping, piping components, and piping elements	Galvanized steel	Air - indoor, uncontrolled	None	None	No
VII.J.AP-277		Piping, piping components, and piping elements	Glass	Waste water	None	None	No
VII.J.AP-16	VII.J-14(AP-16)	Piping, piping components, and piping elements	Nickel alloy	Air – indoor, uncontrolled (External)	None	None	No
VII.J.AP-260		Piping, piping components, and piping elements	Nickel alloy	Air with borated water leakage	None	None	No

VII AUXILIARY SYSTEMS
J Common Miscellaneous Material/Environment Combinations

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.J.AP-268		Piping, piping components, and piping elements	PVC	Air – indoor, uncontrolled	None	None	No
VII.J.AP-269		Piping, piping components, and piping elements	PVC	Condensation (Internal)	None	None	No
VII.J.AP-20	VII.J-18(AP-20)	Piping, piping components, and piping elements	Stainless steel	Air – dry	None	None	No
VII.J.AP-17	VII.J-15(AP-17)	Piping, piping components, and piping elements	Stainless steel	Air – indoor, uncontrolled (External)	None	None	No
VII.J.AP-123		Piping, piping components, and piping elements	Stainless steel	Air – indoor, uncontrolled (Internal/External)	None	None	No
VII.J.AP-18	VII.J-16(AP-18)	Piping, piping components, and piping elements	Stainless steel	Air with borated water leakage	None	None	No
VII.J.AP-19	VII.J-17(AP-19)	Piping, piping components, and piping elements	Stainless steel	Concrete	None	None	No
VII.J.AP-22	VII.J-19(AP-22)	Piping, piping components, and piping elements	Stainless steel	Gas	None	None	No

VII AUXILIARY SYSTEMS
J Common Miscellaneous Material/Environment Combinations

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VII.J.AP-4	VII.J-22(AP-4)	Piping, piping components, and piping elements	Steel	Air – dry	None	None	No
VII.J.AP-2	VII.J-20(AP-2)	Piping, piping components, and piping elements	Steel	Air – indoor, controlled (External)	None	None	No
VII.J.AP-282	VII.J-21(AP-3)	Piping, piping components, and piping elements	Steel	Concrete	None	None, provided 1) attributes of the concrete are consistent with ACI 318 or ACI 349 (low water-to-cement ratio, low permeability, and adequate air entrainment) as cited in NUREG-1557, and 2) plant OE indicates no degradation of the concrete	No, if conditions are met.
VII.J.AP-6	VII.J-23(AP-6)	Piping, piping components, and piping elements	Steel	Gas	None	None	No
VII.J.AP-160		Piping, piping components, and piping elements	Titanium	Air – indoor, uncontrolled or Air – outdoor	None	None	No

CHAPTER VIII

STEAM AND POWER CONVERSION SYSTEM

MAJOR PLANT SECTIONS

A. Steam Turbine System

B1. Main Steam System (PWR)

B2. Main Steam System (BWR)

C. Extraction Steam System

D1. Feedwater System (PWR)

D2. Feedwater System (BWR)

E. Condensate System

F. Steam Generator Blowdown System (PWR)

G. Auxiliary Feedwater System (PWR)

H. External Surfaces of Components and Miscellaneous Bolting

I. Common Miscellaneous Material/Environment Combinations

A. STEAM TURBINE SYSTEM

Systems, Structures, and Components

This section addresses the piping and fittings in the steam turbine system for both pressurized water reactors (PWRs) and boiling water reactors (BWRs) and consists of the lines from the high-pressure (HP) turbine to the moisture separator/reheater (MSR) and the lines from the MSR to the low-pressure (LP) turbine. Based on Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” all components that comprise the steam turbine system are governed by Group D Quality Standards.

The steam turbine performs its intended functions with moving parts. They are subject to replacement based on qualified life or a specified time period. Pursuant to 10 CFR 54.2(a)(1), therefore, they are not subject to an aging management review.

Aging management programs for the degradation of external surfaces of components and miscellaneous bolting are included in VIII.H. Common miscellaneous material/environment combinations, where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation, are included in VIII.I.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The systems that interface with the steam turbine system include the PWR and BWR main steam system (VIII.B1 and VIII.B2), the extraction steam system (VIII.C), and the condensate system (VIII.E).

VIII STEAM AND POWER CONVERSION SYSTEM
A Steam Turbine System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VIII.A.S-23	VIII.A-1(S-23)	Heat exchanger components	Steel	Closed-cycle cooling water	Loss of material due to general, pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VIII.A.SP-64	VIII.A-2(SP-64)	Heat exchanger components and tubes	Steel	Closed-cycle cooling water	Reduction of heat transfer due to fouling	Chapter XI.M21A, "Closed Treated Water Systems"	No
VIII.A.SP-92	VIII.A-3(SP-32)	Piping, piping components, and piping elements	Copper alloy	Lubricating oil	Loss of material due to pitting and crevice corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
VIII.A.SP-31	VIII.A-4(SP-31)	Piping, piping components, and piping elements	Copper alloy	Raw water	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VIII.A.SP-101	VIII.A-5(SP-61)	Piping, piping components, and piping elements	Copper alloy	Treated water	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.A.SP-30	VIII.A-6(SP-30)	Piping, piping components, and piping elements	Copper alloy (>15% Zn or >8% Al)	Raw water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VIII.A.SP-28	VIII.A-7(SP-28)	Piping, piping components, and piping elements	Gray cast iron	Raw water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No

VIII STEAM AND POWER CONVERSION SYSTEM
A Steam Turbine System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VIII.A.SP-27	VIII.A-8(SP-27)	Piping, piping components, and piping elements	Gray cast iron	Treated water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VIII.A.SP-95	VIII.A-9(SP-38)	Piping, piping components, and piping elements	Stainless steel	Lubricating oil	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
VIII.A.SP-98	VIII.A-11(SP-45)	Piping, piping components, and piping elements	Stainless steel	Steam	Cracking due to stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.A.SP-155	VIII.A-12(SP-43)	Piping, piping components, and piping elements	Stainless steel	Steam	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.A.SP-91	VIII.A-14(SP-25)	Piping, piping components, and piping elements	Steel	Lubricating oil	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
VIII.A.SP-71	VIII.A-15(S-04)	Piping, piping components, and piping elements	Steel	Steam	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.A.S-15	VIII.A-17(S-15)	Piping, piping components, and piping elements	Steel	Steam	Wall thinning due to flow-accelerated corrosion	Chapter XI.M17, “Flow-Accelerated Corrosion”	No

VIII STEAM AND POWER CONVERSION SYSTEM
A Steam Turbine System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VIII.A.SP-118		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated
VIII.A.SP-127		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Loss of material due to pitting and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated

B1. MAIN STEAM SYSTEM (PWR)

Systems, Structures, and Components

This section addresses the main steam system for pressurized water reactors (PWRs). The section includes the main steam lines from the steam generator to the steam turbine and the turbine bypass lines from the main steam lines to the condenser. Also included are the lines to the main feedwater (FW) and auxiliary feedwater (AFW) pump turbines, steam drains, and valves, including the containment isolation valves on the main steam lines and the lines to the AFW pump turbines.

Based on Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” the portion of the main steam system extending from the steam generator up to the second containment isolation valve is governed by Group B or C Quality Standards, and all other components that comprise the main steam system located downstream of these isolation valves are governed by Group D Quality Standards.

The internals of the valves perform their intended functions with moving parts or with a change in configuration. They are subject to replacement based on qualified life or a specified time period. Pursuant to 10 CFR 54.21(a)(1), therefore, they are not subject to an aging management review.

Aging management programs for the degradation of the external surfaces of components and miscellaneous bolting are included in VIII.H. Common miscellaneous material/environment combinations, where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation, are included in VIII.I.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The systems and structures that interface with the main steam system include PWR concrete or steel containment structures (II.A1 and II.A2), common components (II.A3), the steam generator (IV.D1 and IV.D2), the steam turbine system (VIII.A), the feedwater system (VIII.D1), the condensate system (VIII.E), and the auxiliary feedwater system (VIII.G).

VIII STEAM AND POWER CONVERSION SYSTEM
B1 Main Steam System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VIII.B1.SP-157	VIII.B1-1(SP-18)	Piping, piping components, and piping elements	Nickel alloy	Steam	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.B1.SP-110		Piping, piping components, and piping elements	Stainless steel	Condensation (Internal)	Loss of material due to pitting and crevice corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
VIII.B1.SP-98	VIII.B1-2(SP-44)	Piping, piping components, and piping elements	Stainless steel	Steam	Cracking due to stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.B1.SP-155	VIII.B1-3(SP-43)	Piping, piping components, and piping elements	Stainless steel	Steam	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.B1.SP-87	VIII.B1-4(SP-16)	Piping, piping components, and piping elements	Stainless steel	Treated water	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.B1.SP-88	VIII.B1-5(SP-17)	Piping, piping components, and piping elements	Stainless steel	Treated water >60°C (>140°F)	Cracking due to stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.B1.SP-59	VIII.B1-6(SP-59)	Piping, piping components, and piping elements	Steel	Air – outdoor (Internal)	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
VIII.B1.SP-60	VIII.B1-7(SP-60)	Piping, piping components, and piping elements	Steel	Condensation (Internal)	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No

VIII STEAM AND POWER CONVERSION SYSTEM
B1 Main Steam System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VIII.B1.SP-71	VIII.B1-8(S-07)	Piping, piping components, and piping elements	Steel	Steam	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.B1.S-15	VIII.B1-9(S-15)	Piping, piping components, and piping elements	Steel	Steam	Wall thinning due to flow-accelerated corrosion	Chapter XI.M17, “Flow-Accelerated Corrosion”	No
VIII.B1.S-08	VIII.B1-10(S-08)	Piping, piping components, and piping elements	Steel	Steam or Treated water	Cumulative fatigue damage due to fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
VIII.B1.SP-74	VIII.B1-11(S-10)	Piping, piping components, and piping elements	Steel	Treated water	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.B1.SP-118		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated
VIII.B1.SP-127		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Loss of material due to pitting and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated

B2. MAIN STEAM SYSTEM (BWR)

Systems, Structures, and Components

This section addresses the main steam system for boiling water reactors (BWRs). The section includes the main steam lines from the outermost containment isolation valve to the steam turbines and the turbine bypass lines from the main steam lines to the condenser. Also included are steam drains, lines to the main feedwater (FW), high-pressure coolant injection (HPCI), and reactor core isolation cooling (RCIC) turbines.

Based on Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” portions of the main steam system extending from the outermost containment isolation valve up to and including the turbine stop and bypass valves, as well as connected piping up to and including the first valve that is either normally closed or capable of automatic closure during all modes of normal reactor operation, are governed by Group B Quality Standards. The remaining portions of the main steam system consist of components governed by Group D Quality Standards. For BWRs containing a shutoff valve in addition to the two containment isolation valves in the main steam line, Group B Quality Standards apply only to those portions of the system extending from the outermost containment isolation valves up to and including the shutoff valve. The portion of the main steam system extending from the reactor pressure vessel up to the second isolation valve and including the containment isolation valves is governed by Group A Quality Standards, and is covered in IV.C1.

The internal of the valves perform their intended functions with moving parts or with a change in configuration. They are subject to replacement based on qualified life or a specified time period. Pursuant to 10 CFR 54.21(a)(1), therefore, they are not subject to an aging management review.

Aging management programs for the degradation of the external surfaces of components and miscellaneous bolting are included in VIII.H. Common miscellaneous material/environment combinations, where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation, are included in VIII.I.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The systems that interface with the main steam system include the BWR Mark 1, Mark 2, or Mark 3 containment structures (II.B1, II.B2, and II.B3, respectively) and common components (II.B4), the reactor coolant pressure boundary (IV.C1), the steam turbine system (VIII.A), the feedwater system (VIII.D2), and the condensate system (VIII.E).

VIII STEAM AND POWER CONVERSION SYSTEM
B2 Main Steam System (BWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VIII.B2.SP-110		Piping, piping components, and piping elements	Stainless steel	Condensation (Internal)	Loss of material due to pitting and crevice corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
VIII.B2.SP-98	VIII.B2-1(SP-45)	Piping, piping components, and piping elements	Stainless steel	Steam	Cracking due to stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.B2.SP-155	VIII.B2-2(SP43)	Piping, piping components, and piping elements	Stainless steel	Steam	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.B2.SP-160	VIII.B2-3(S-05)	Piping, piping components, and piping elements	Steel	Steam	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.B2.S-15	VIII.B2-4(S-15)	Piping, piping components, and piping elements	Steel	Steam	Wall thinning due to flow-accelerated corrosion	Chapter XI.M17, “Flow-Accelerated Corrosion”	No
VIII.B2.S-08	VIII.B2-5(S-08)	Piping, piping components, and piping elements	Steel	Steam or Treated water	Cumulative fatigue damage due to fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
VIII.B2.SP-73	VIII.B2-6(S-09)	Piping, piping components, and piping elements	Steel	Treated water	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No

VIII STEAM AND POWER CONVERSION SYSTEM
B2 Main Steam System (BWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VIII.B2.SP-118		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated
VIII.B2.SP-127		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Loss of material due to pitting and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated

C. EXTRACTION STEAM SYSTEM

Systems, Structures, and Components

This section addresses the extraction steam lines for both pressurized water reactors (PWRs) and boiling water reactors (BWRs), which extend from the steam turbine to the feedwater heaters, including the drain lines. Based on Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” all components that comprise the extraction steam system are governed by Group D Quality Standards.

The internals of the valves perform their intended functions with moving parts or with a change in configuration. They are subject to replacement based on qualified life or a specified time period. Pursuant to 10 CFR 54.21(a)(1), therefore, they are not subject to an aging management review.

Aging management programs for the degradation of the external surfaces of components and miscellaneous bolting are included in VIII.H. Common miscellaneous material/environment combinations, where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation, are included in VIII.I.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The systems that interface with the extraction steam system include the steam turbine system (VIII.A), the PWR and BWR feedwater system (VIII.D1 and VIII.D2), and the condensate system (VIII.E).

VIII C STEAM AND POWER CONVERSION SYSTEM
Extraction Steam System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VIII.C.SP-87	VIII.C-1(SP-16)	Piping, piping components, and piping elements	Stainless steel	Treated water	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.C.SP-88	VIII.C-2(SP-17)	Piping, piping components, and piping elements	Stainless steel	Treated water >60°C (>140°F)	Cracking due to stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.C.SP-71	VIII.C-3(S-04)	Piping, piping components, and piping elements	Steel	Steam	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.C.S-15	VIII.C-5(S-15)	Piping, piping components, and piping elements	Steel	Steam	Wall thinning due to flow-accelerated corrosion	Chapter XI.M17, “Flow-Accelerated Corrosion”	No
VIII.C.SP-73	VIII.C-6(S-09)	Piping, piping components, and piping elements	Steel	Treated water	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.C.SP-118		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated
VIII.C.SP-127		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Loss of material due to pitting and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated

D1. FEEDWATER SYSTEM (PWR)

Systems, Structures, and Components

This section addresses the main feedwater system for pressurized water reactors (PWRs), which extends from the condensate system to the steam generator. It consists of the main feedwater lines, feedwater pumps, and valves, including the containment isolation valves. Based on Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” the portion of the feedwater system extending from the secondary side of the steam generator up to the second containment isolation valve is governed by Group B or C Quality Standards. All other components in the feedwater system located downstream from these isolation valves are governed by Group D Quality Standards.

Pump and valve internals perform their intended functions with moving parts or with a change in configuration. They are subject to replacement based on qualified life or specified time period. Pursuant to 10 CFR 54.21(a)(1), therefore, they not subject to an aging management review.

Aging management programs for the degradation of the external surfaces of components and miscellaneous bolting are included in VIII.H. Common miscellaneous material/environment combinations, where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation, are included in VIII.I.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The systems and structures that interface with the feedwater system include PWR concrete or steel containment structures (II.A1 and II.A2) and common components (II.A3), the steam generators (IV.D1 and IV.D2), the main steam system (VIII.B1), the extraction steam system (VIII.C), the condensate system (VIII.E), and the auxiliary feedwater system (VIII.G).

VIII STEAM AND POWER CONVERSION SYSTEM
D1 Feedwater Systems (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VIII.D1.SP-90	VIII.D1-1(SP-24)	Piping, piping components, and piping elements	Aluminum	Treated water	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.D1.SP-92	VIII.D1-2(SP-32)	Piping, piping components, and piping elements	Copper alloy	Lubricating oil	Loss of material due to pitting and crevice corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
VIII.D1.SP-95	VIII.D1-3(SP-38)	Piping, piping components, and piping elements	Stainless steel	Lubricating oil	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
VIII.D1.SP-87	VIII.D1-4(SP-16)	Piping, piping components, and piping elements	Stainless steel	Treated water	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.D1.SP-88	VIII.D1-5(SP-17)	Piping, piping components, and piping elements	Stainless steel	Treated water >60°C (>140°F)	Cracking due to stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.D1.SP-91	VIII.D1-6(SP-25)	Piping, piping components, and piping elements	Steel	Lubricating oil	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
VIII.D1.S-11	VIII.D1-7(S-11)	Piping, piping components, and piping elements	Steel	Treated water	Cumulative fatigue damage due to fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA

VIII STEAM AND POWER CONVERSION SYSTEM
D1 Feedwater Systems (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VIII.D1.SP-74	VIII.D1-8(S-10)	Piping, piping components, and piping elements	Steel	Treated water	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.D1.S-16	VIII.D1-9(S-16)	Piping, piping components, and piping elements	Steel	Treated water	Wall thinning due to flow-accelerated corrosion	Chapter XI.M17, “Flow-Accelerated Corrosion”	No
VIII.D1.SP-118		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated
VIII.D1.SP-127		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Loss of material due to pitting and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated

D2. FEEDWATER SYSTEM (BWR)

Systems, Structures, and Components

This section addresses the main feedwater system for boiling water reactors (BWRs), which extends from the condensate and condensate booster system to the outermost feedwater isolation valve on the feedwater lines to the reactor vessel. It consists of the main feedwater lines, feedwater pumps, and valves.

Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," the portions of the feedwater system extending from the outermost containment isolation valves up to and including the shutoff valve, or the first valve that is either normally closed or capable of closure during all modes of normal reactor operation, are governed by Group B Quality Standards. The remaining portions of the feedwater system consist of components governed by Group D Quality Standards. The portion of the feedwater system extending from the reactor vessel up to the second containment isolation valve, including the isolation valves, is governed by Group A Quality Standards and is covered in IV.C1.

Pump and valve internals perform their intended functions with moving parts or with a change in configuration. They are subject to replacement based on qualified life or a specified time period. Pursuant to 10 CFR 54.21(a)(1), therefore, they are not subject to an aging management review.

Aging management programs for the degradation of the external surfaces of components and miscellaneous bolting are included in VIII.H. Common miscellaneous material/environment combinations, where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation, are included in VIII.I.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The systems that interface with the feedwater system include the BWR Mark 1, Mark 2, or Mark 3 containment structures (II.B1, II.B2, and II.B3, respectively) and common components (II.B4), the reactor coolant pressure boundary (IV.C1), the main steam system (VIII.B2), the extraction steam system (VIII.C), and the condensate system (VIII.E).

VIII STEAM AND POWER CONVERSION SYSTEM
D2 Feedwater Systems (BWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VIII.D2.SP-90	VIII.D2-1(SP-24)	Piping, piping components, and piping elements	Aluminum	Treated water	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.D2.SP-92	VIII.D2-2(SP-32)	Piping, piping components, and piping elements	Copper alloy	Lubricating oil	Loss of material due to pitting and crevice corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
VIII.D2.SP-95	VIII.D2-3(SP-38)	Piping, piping components, and piping elements	Stainless steel	Lubricating oil	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
VIII.D2.SP-87	VIII.D2-4(SP-16)	Piping, piping components, and piping elements	Stainless steel	Treated water	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.D2.SP-91	VIII.D2-5(SP-25)	Piping, piping components, and piping elements	Steel	Lubricating oil	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
VIII.D2.S-11	VIII.D2-6(S-11)	Piping, piping components, and piping elements	Steel	Treated water	Cumulative fatigue damage due to fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 “Metal Fatigue,” for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
VIII.D2.SP-73	VIII.D2-7(S-09)	Piping, piping components, and piping elements	Steel	Treated water	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No

VIII STEAM AND POWER CONVERSION SYSTEM
D2 Feedwater Systems (BWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VIII.D2.S-16	VIII.D2-8(S-16)	Piping, piping components, and piping elements	Steel	Treated water	Wall thinning due to flow-accelerated corrosion	Chapter XI.M17, “Flow-Accelerated Corrosion”	No
VIII.D2.SP-118		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated
VIII.D2.SP-127		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Loss of material due to pitting and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated

E. CONDENSATE SYSTEM

Systems, Structures, and Components

This section addresses the condensate system for both pressurized water reactors (PWRs) and boiling water reactors (BWRs), which extend from the condenser hotwells to the suction of feedwater pumps, including condensate and condensate booster pumps, condensate coolers, condensate cleanup system, and condensate storage tanks. Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," all components that comprise the condensate system are governed by Group D Quality Standards.

Pump and valve internals perform their intended functions with moving parts or with a change in configuration. They are subject to replacement based on qualified life or a specified time period. Pursuant to 10 CFR 54.21(a)(1), therefore, they are not subject to an aging management review.

Aging management programs for the degradation of the external surfaces of components and miscellaneous bolting are included in VIII.H. Common miscellaneous material/environment combinations, where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation, are included in VIII.I.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The systems that interface with the condensate system include the steam turbine system (VIII.A), the PWR and BWR main steam system (VIII.B1 and VIII.B2), the PWR and BWR feedwater system (VIII.D1 and VIII.D2), the auxiliary feedwater system (VIII.G, PWR only), the BWR reactor water cleanup system (VII.E3), the open or closed cycle cooling water systems (VII.C1 or VII.C2), and the condensate storage facility.

VIII E	STEAM AND POWER CONVERSION SYSTEM Condensate System						
Item	**Link**	**Structure and/or Component**	**Material**	**Environment**	**Aging Effect/ Mechanism**	**Aging Management Program (AMP)**	**Further Evaluation**
VIII.E.S-25	VIII.E-2(S-25)	Heat exchanger components	Stainless steel	Closed-cycle cooling water	Loss of material due to pitting and crevice corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VIII.E.SP-117	VIII.E-3(S-26)	Heat exchanger components	Stainless steel	Raw water	Loss of material due to general, pitting, crevice, galvanic, and microbiologically-influenced corrosion; fouling that leads to corrosion	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VIII.E.S-23	VIII.E-5(S-23)	Heat exchanger components	Steel	Closed-cycle cooling water	Loss of material due to general, pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VIII.E.SP-146	VIII.E-6(S-24)	Heat exchanger components	Steel	Raw water	Loss of material due to general, pitting, crevice, galvanic, and microbiologically-influenced corrosion; fouling that leads to corrosion	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VIII.E.SP-77	VIII.E-7(S-18)	Heat exchanger components	Steel	Treated water	Loss of material due to general, pitting, crevice, and galvanic corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.E.SP-80	VIII.E-4(S-21)	Heat exchanger components and tubes	Stainless steel	Treated water	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.E.SP-57	VIII.E-8(SP-57)	Heat exchanger tubes	Copper alloy	Closed-cycle cooling water	Reduction of heat transfer due to fouling	Chapter XI.M21A, "Closed Treated Water Systems"	No

VIII STEAM AND POWER CONVERSION SYSTEM
E Condensate System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VIII.E.SP-56	VIII.E-9(SP-56)	Heat exchanger tubes	Copper alloy	Raw water	Reduction of heat transfer due to fouling	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VIII.E.SP-100	VIII.E-10(SP-58)	Heat exchanger tubes	Copper alloy	Treated water	Reduction of heat transfer due to fouling	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.E.SP-41	VIII.E-11(SP-41)	Heat exchanger tubes	Stainless steel	Closed-cycle cooling water	Reduction of heat transfer due to fouling	Chapter XI.M21A, "Closed Treated Water Systems"	No
VIII.E.S-28	VIII.E-12(S-28)	Heat exchanger tubes	Stainless steel	Raw water	Reduction of heat transfer due to fouling	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VIII.E.SP-96	VIII.E-13(SP-40)	Heat exchanger tubes	Stainless steel	Treated water	Reduction of heat transfer due to fouling	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.E.SP-64	VIII.E-14(SP-64)	Heat exchanger tubes	Steel	Closed-cycle cooling water	Reduction of heat transfer due to fouling	Chapter XI.M21A, "Closed Treated Water Systems"	No
VIII.E.SP-90	VIII.E-15(SP-24)	Piping, piping components, and piping elements	Aluminum	Treated water	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.E.SP-8	VIII.E-16(SP-8)	Piping, piping components, and piping elements	Copper alloy	Closed-cycle cooling water	Loss of material due to pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VIII.E.SP-92	VIII.E-17(SP-32)	Piping, piping components, and piping elements	Copper alloy	Lubricating oil	Loss of material due to pitting and crevice corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No

VIII STEAM AND POWER CONVERSION SYSTEM
E Condensate System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VIII.E.SP-31	VIII.E-18(SP-31)	Piping, piping components, and piping elements	Copper alloy	Raw water	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VIII.E.SP-29	VIII.E-19(SP-29)	Piping, piping components, and piping elements	Copper alloy (>15% Zn or >8% Al)	Closed-cycle cooling water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VIII.E.SP-30	VIII.E-20(SP-30)	Piping, piping components, and piping elements	Copper alloy (>15% Zn or >8% Al)	Raw water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VIII.E.SP-55	VIII.E-21(SP-55)	Piping, piping components, and piping elements	Copper alloy (>15% Zn or >8% Al)	Treated water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VIII.E.SP-26	VIII.E-22(SP-26)	Piping, piping components, and piping elements	Gray cast iron	Soil	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VIII.E.SP-27	VIII.E-23(SP-27)	Piping, piping components, and piping elements	Gray cast iron	Treated water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VIII.E.SP-39	VIII.E-24(SP-39)	Piping, piping components, and piping elements	Stainless steel	Closed-cycle cooling water	Loss of material due to pitting and crevice corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VIII.E.SP-54	VIII.E-25(SP-54)	Piping, piping components, and piping elements	Stainless steel	Closed-cycle cooling water >60°C (>140°F)	Cracking due to stress corrosion cracking	Chapter XI.M21A, "Closed Treated Water Systems"	No
VIII.E.SP-95	VIII.E-26(SP-38)	Piping, piping components, and piping elements	Stainless steel	Lubricating oil	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No

VIII STEAM AND POWER CONVERSION SYSTEM
E Condensate System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VIII.E.SP-36	VIII.E-27(SP-36)	Piping, piping components, and piping elements	Stainless steel	Raw water	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VIII.E.SP-94	VIII.E-28(SP-37)	Piping, piping components, and piping elements	Stainless steel	Soil or concrete	Loss of material due to pitting and crevice corrosion	Chapter XI.M41, "Buried and Underground Piping and Tanks"	No
VIII.E.SP-87	VIII.E-29(SP-16)	Piping, piping components, and piping elements	Stainless steel	Treated water	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.E.SP-88	VIII.E-30(SP-17)	Piping, piping components, and piping elements	Stainless steel	Treated water >60°C (>140°F)	Cracking due to stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.E.SP-91	VIII.E-32(SP-25)	Piping, piping components, and piping elements	Steel	Lubricating oil	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
VIII.E.SP-73	VIII.E-33(S-09)	Piping, piping components, and piping elements	Steel	Treated water	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.E.S-16	VIII.E-35(S-16)	Piping, piping components, and piping elements	Steel	Treated water	Wall thinning due to flow-accelerated corrosion	Chapter XI.M17, “Flow-Accelerated Corrosion”	No
VIII.E.SP-118		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated

VIII E STEAM AND POWER CONVERSION SYSTEM
Condensate System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VIII.E.SP-127		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Loss of material due to pitting and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated
VIII.E.SP-145	VIII.E-1(S-01)	Piping, piping components, and piping elements; tanks	Steel (with coating or wrapping)	Soil or concrete	Loss of material due to general, pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M41, "Buried and Underground Piping and Tanks"	No
VIII.E.SP-81	VIII.E-36(S-22)	PWR heat exchanger components	Stainless steel	Treated water	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.E.SP-78	VIII.E-37(S-19)	PWR heat exchanger components	Steel	Treated water	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.E.SP-140		Tanks	Aluminum	Air – outdoor (External)	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M29, “Aboveground Metallic Tanks”	No
VIII.E.SP-139		Tanks	Aluminum	Soil or Concrete	Loss of material due to pitting, and crevice corrosion	Chapter XI.M29, “Aboveground Metallic Tanks”	No
VIII.E.SP-138		Tanks	Stainless steel	Air – outdoor (External)	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M29, “Aboveground Metallic Tanks”	No
VIII.E.SP-137		Tanks	Stainless steel	Soil or Concrete	Loss of material due to pitting, and crevice corrosion	Chapter XI.M29, “Aboveground Metallic Tanks”	No

VIII STEAM AND POWER CONVERSION SYSTEM
E Condensate System

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VIII.E.SP-97	VIII.E-38(SP-42)	Tanks	Stainless steel	Treated water >60°C (>140°F)	Cracking due to stress corrosion cracking	Chapter XI.M2, "Water Chemistry," and Chapter XI.M32, "One-Time Inspection"	No
VIII.E.S-31	VIII.E-39(S-31)	Tanks	Steel	Air – outdoor (External)	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M29, "Aboveground Metallic Tanks"	No
VIII.E.SP-115		Tanks	Steel	Soil or Concrete	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M29, "Aboveground Metallic Tanks"	No
VIII.E.SP-75	VIII.E-40(S-13)	Tanks	Steel; stainless steel	Treated water	Loss of material due to general (steel only), pitting, and crevice corrosion	Chapter XI.M2, "Water Chemistry," and Chapter XI.M32, "One-Time Inspection"	No

F. STEAM GENERATOR BLOWDOWN SYSTEM (PWR)

Systems, Structures, and Components

This section addresses the steam generator blowdown system for pressurized water reactors (PWRs), which extends from the steam generator through the blowdown condenser and includes the containment isolation valves and small bore piping less than nominal pipe size (NPS) 2 in. (including instrumentation lines).

Based on Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” the portion of the blowdown system extending from the steam generator up to the isolation valve outside the containment and including the isolation valves is governed by Group B or C Quality Standards. The remaining portions of the steam generator blowdown system consist of components governed by Group D Quality Standards.

Pump and valve internals perform their intended functions with moving parts or with a change in configuration. They are subject to replacement based on qualified life or a specified time period. Pursuant to 10 CFR 54.21(a)(1), therefore, they are not subject to an aging management review.

Aging management programs for the degradation of the external surfaces of components and miscellaneous bolting are included in VIII.H. Common miscellaneous material/environment combinations, where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation, are included in VIII.I.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The systems that interface with the blowdown system include the steam generator (IV.D1 and IV.D2) and the open- or closed-cycle cooling water systems (VII.C1 or VII.C2).

VIII STEAM AND POWER CONVERSION SYSTEM
F Steam Generator Blowdown System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VIII.F.SP-56	VIII.F-6(SP-56)	Heat exchanger components	Copper alloy	Raw water	Reduction of heat transfer due to fouling	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VIII.F.S-25	VIII.F-1(S-25)	Heat exchanger components	Stainless steel	Closed-cycle cooling water	Loss of material due to pitting and crevice corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VIII.F.SP-117	VIII.F-2(S-26)	Heat exchanger components	Stainless steel	Raw water	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion; fouling that leads to corrosion	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VIII.F.SP-85	VIII.F-3(S-39)	Heat exchanger components	Stainless steel	Treated water >60°C (>140°F)	Cracking due to stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.F.S-23	VIII.F-4(S-23)	Heat exchanger components	Steel	Closed-cycle cooling water	Loss of material due to general, pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VIII.F.SP-146	VIII.F-5(S-24)	Heat exchanger components	Steel	Raw water	Loss of material due to general, pitting, crevice, galvanic, and microbiologically-influenced corrosion; fouling that leads to corrosion	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VIII.F.SP-100	VIII.F-7(SP-58)	Heat exchanger tubes	Copper alloy	Treated water	Reduction of heat transfer due to fouling	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No

VIII STEAM AND POWER CONVERSION SYSTEM
F Steam Generator Blowdown System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VIII.F.SP-41	VIII.F-8(SP-41)	Heat exchanger tubes	Stainless steel	Closed-cycle cooling water	Reduction of heat transfer due to fouling	Chapter XI.M21A, "Closed Treated Water Systems"	No
VIII.F.S-28	VIII.F-9(S-28)	Heat exchanger tubes	Stainless steel	Raw water	Reduction of heat transfer due to fouling	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VIII.F.SP-96	VIII.F-10(SP-40)	Heat exchanger tubes	Stainless steel	Treated water	Reduction of heat transfer due to fouling	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.F.SP-64	VIII.F-11(SP-64)	Heat exchanger tubes	Steel	Closed-cycle cooling water	Reduction of heat transfer due to fouling	Chapter XI.M21A, "Closed Treated Water Systems"	No
VIII.F.SP-90	VIII.F-12(SP-24)	Piping, piping components, and piping elements	Aluminum	Treated water	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.F.SP-8	VIII.F-13(SP-8)	Piping, piping components, and piping elements	Copper alloy	Closed-cycle cooling water	Loss of material due to pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VIII.F.SP-31	VIII.F-14(SP-31)	Piping, piping components, and piping elements	Copper alloy	Raw water	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VIII.F.SP-101	VIII.F-15(SP-61)	Piping, piping components, and piping elements	Copper alloy	Treated water	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No

VIII STEAM AND POWER CONVERSION SYSTEM
F Steam Generator Blowdown System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VIII.F.SP-29	VIII.F-16(SP-29)	Piping, piping components, and piping elements	Copper alloy (>15% Zn or >8% Al)	Closed-cycle cooling water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VIII.F.SP-30	VIII.F-17(SP-30)	Piping, piping components, and piping elements	Copper alloy (>15% Zn or >8% Al)	Raw water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VIII.F.SP-55	VIII.F-18(SP-55)	Piping, piping components, and piping elements	Copper alloy (>15% Zn or >8% Al)	Treated water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VIII.F.SP-27	VIII.F-19(SP-27)	Piping, piping components, and piping elements	Gray cast iron	Treated water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VIII.F.SP-39	VIII.F-20(SP-39)	Piping, piping components, and piping elements	Stainless steel	Closed-cycle cooling water	Loss of material due to pitting and crevice corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VIII.F.SP-54	VIII.F-21(SP-54)	Piping, piping components, and piping elements	Stainless steel	Closed-cycle cooling water >60°C (>140°F)	Cracking due to stress corrosion cracking	Chapter XI.M21A, "Closed Treated Water Systems"	No
VIII.F.SP-36	VIII.F-22(SP-36)	Piping, piping components, and piping elements	Stainless steel	Raw water	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VIII.F.SP-87	VIII.F-23(SP-16)	Piping, piping components, and piping elements	Stainless steel	Treated water	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No

VIII STEAM AND POWER CONVERSION SYSTEM
F Steam Generator Blowdown System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VIII.F.SP-88	VIII.F-24(SP-17)	Piping, piping components, and piping elements	Stainless steel	Treated water >60°C (>140°F)	Cracking due to stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.F.SP-74	VIII.F-25(S-10)	Piping, piping components, and piping elements	Steel	Treated water	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.F.S-16	VIII.F-26(S-16)	Piping, piping components, and piping elements	Steel	Treated water	Wall thinning due to flow-accelerated corrosion	Chapter XI.M17, “Flow-Accelerated Corrosion”	No
VIII.F.SP-118		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated
VIII.F.SP-127		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Loss of material due to pitting and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated
VIII.F.SP-81	VIII.F-27(S-22)	PWR heat exchanger components	Stainless steel	Treated water	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.F.SP-78	VIII.F-28(S-19)	PWR heat exchanger components	Steel	Treated water	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No

G. AUXILIARY FEEDWATER SYSTEM (PWR)

Systems, Structures, and Components

This section addresses the auxiliary feedwater (AFW) system for pressurized water reactors (PWRs), which extends from the condensate storage or backup water supply system to the steam generator or to the main feedwater (MFW) line. They consist of AFW piping, AFW pumps, pump turbine oil coolers, and valves, including the containment isolation valves.

Based on Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," portions of the AFW system extending from the secondary side of the steam generator up to the second isolation valve and including the containment isolation valves are governed by Group B Quality Standards. In addition, portions of the AFW system that are required for their safety functions and that either do not operate during any mode of normal reactor operation or cannot be tested adequately are also governed by Group B Quality Standards. The remainder of the structures and components covered in this section are governed by Group C Quality Standards.

Pump and valve internals perform their intended functions with moving parts or with a change in configuration. They are subject to replacement based on qualified life or a specified time period. Pursuant to 10 CFR 54.21(a)(1), therefore, they are not subject to an aging management review.

Aging management programs for the degradation of the external surfaces of components and miscellaneous bolting are included in VIII.H. Common miscellaneous material/environment combinations, where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation, are included in VIII.I.

The system piping includes all pipe sizes, including instrument piping.

System Interfaces

The systems that interface with the AFW system include the steam generator (IV.D1 and IV.D2), the main steam system (VIII.B1), the PWR feedwater system (VIII.D1), the condensate system (VIII.E), and the open- or closed-cycle cooling water systems (VII.C1 or VII.C2).

VIII STEAM AND POWER CONVERSION SYSTEM
G Auxiliary Feedwater System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VIII.G.S-25	VIII.G-2(S-25)	Heat exchanger components	Stainless steel	Closed-cycle cooling water	Loss of material due to pitting and crevice corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VIII.G.SP-79	VIII.G-3(S-20)	Heat exchanger components	Stainless steel	Lubricating oil	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
VIII.G.SP-117	VIII.G-4(S-26)	Heat exchanger components	Stainless steel	Raw water	Loss of material due to general, pitting, crevice, galvanic, and microbiologically-influenced corrosion; fouling that leads to corrosion	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VIII.G.S-23	VIII.G-5(S-23)	Heat exchanger components	Steel	Closed-cycle cooling water	Loss of material due to general, pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VIII.G.SP-76	VIII.G-6(S-17)	Heat exchanger components	Steel	Lubricating oil	Loss of material due to general, pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No

VIII STEAM AND POWER CONVERSION SYSTEM
G Auxiliary Feedwater System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VIII.G.SP-146	VIII.G-7(S-24)	Heat exchanger components	Steel	Raw water	Loss of material due to general, pitting, crevice, galvanic, and microbiologically-influenced corrosion; fouling that leads to corrosion	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VIII.G.SP-113		Heat exchanger components and tubes	Aluminum	Lubricating oil	Reduction of heat transfer due to fouling	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
VIII.G.SP-99	VIII.G-8(SP-53)	Heat exchanger tubes	Copper alloy	Lubricating oil	Reduction of heat transfer due to fouling	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
VIII.G.SP-56	VIII.G-9(SP-56)	Heat exchanger tubes	Copper alloy	Raw water	Reduction of heat transfer due to fouling	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VIII.G.SP-100	VIII.G-10(SP-58)	Heat exchanger tubes	Copper alloy	Treated water	Reduction of heat transfer due to fouling	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.G.SP-41	VIII.G-11(SP-41)	Heat exchanger tubes	Stainless steel	Closed-cycle cooling water	Reduction of heat transfer due to fouling	Chapter XI.M21A, "Closed Treated Water Systems"	No
VIII.G.SP-102	VIII.G-12(SP-62)	Heat exchanger tubes	Stainless steel	Lubricating oil	Reduction of heat transfer due to fouling	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No

VIII STEAM AND POWER CONVERSION SYSTEM
G Auxiliary Feedwater System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VIII.G.S-28	VIII.G-13(S-28)	Heat exchanger tubes	Stainless steel	Raw water	Reduction of heat transfer due to fouling	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VIII.G.SP-64	VIII.G-14(SP-64)	Heat exchanger tubes	Steel	Closed-cycle cooling water	Reduction of heat transfer due to fouling	Chapter XI.M21A, "Closed Treated Water Systems"	No
VIII.G.SP-103	VIII.G-15(SP-63)	Heat exchanger tubes	Steel	Lubricating oil	Reduction of heat transfer due to fouling	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
VIII.G.S-27	VIII.G-16(S-27)	Heat exchanger tubes	Steel	Raw water	Reduction of heat transfer due to fouling	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VIII.G.SP-114		Piping, piping components, and piping elements	Aluminum	Lubricating oil	Loss of material due to pitting and crevice corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
VIII.G.SP-90	VIII.G-17(SP-24)	Piping, piping components, and piping elements	Aluminum	Treated water	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.G.SP-8	VIII.G-18(SP-8)	Piping, piping components, and piping elements	Copper alloy	Closed-cycle cooling water	Loss of material due to pitting, crevice, and galvanic corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No
VIII.G.SP-92	VIII.G-19(SP-32)	Piping, piping components, and piping elements	Copper alloy	Lubricating oil	Loss of material due to pitting and crevice corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No

VIII STEAM AND POWER CONVERSION SYSTEM
G Auxiliary Feedwater System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VIII.G.SP-31	VIII.G-20(SP-31)	Piping, piping components, and piping elements	Copper alloy	Raw water	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VIII.G.SP-29	VIII.G-21(SP-29)	Piping, piping components, and piping elements	Copper alloy (>15% Zn or >8% Al)	Closed-cycle cooling water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VIII.G.SP-30	VIII.G-22(SP-30)	Piping, piping components, and piping elements	Copper alloy (>15% Zn or >8% Al)	Raw water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VIII.G.SP-55	VIII.G-23(SP-55)	Piping, piping components, and piping elements	Copper alloy (>15% Zn or >8% Al)	Treated water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VIII.G.SP-28	VIII.G-24(SP-28)	Piping, piping components, and piping elements	Gray cast iron	Raw water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VIII.G.SP-26	VIII.G-25(SP-26)	Piping, piping components, and piping elements	Gray cast iron	Soil	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VIII.G.SP-27	VIII.G-26(SP-27)	Piping, piping components, and piping elements	Gray cast iron	Treated water	Loss of material due to selective leaching	Chapter XI.M33, “Selective Leaching”	No
VIII.G.SP-39	VIII.G-27(SP-39)	Piping, piping components, and piping elements	Stainless steel	Closed-cycle cooling water	Loss of material due to pitting and crevice corrosion	Chapter XI.M21A, "Closed Treated Water Systems"	No

VIII STEAM AND POWER CONVERSION SYSTEM
G Auxiliary Feedwater System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VIII.G.SP-54	VIII.G-28(SP-54)	Piping, piping components, and piping elements	Stainless steel	Closed-cycle cooling water >60°C (>140°F)	Cracking due to stress corrosion cracking	Chapter XI.M21A, "Closed Treated Water Systems"	No
VIII.G.SP-95	VIII.G-29(SP-38)	Piping, piping components, and piping elements	Stainless steel	Lubricating oil	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, “One-Time Inspection”	No
VIII.G.SP-36	VIII.G-30(SP-36)	Piping, piping components, and piping elements	Stainless steel	Raw water	Loss of material due to pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M20, "Open-Cycle Cooling Water System"	No
VIII.G.SP-94	VIII.G-31(SP-37)	Piping, piping components, and piping elements	Stainless steel	Soil or concrete	Loss of material due to pitting and crevice corrosion	Chapter XI.M41, "Buried and Underground Piping and Tanks"	No
VIII.G.SP-87	VIII.G-32(SP-16)	Piping, piping components, and piping elements	Stainless steel	Treated water	Loss of material due to pitting and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.G.SP-88	VIII.G-33(SP-17)	Piping, piping components, and piping elements	Stainless steel	Treated water >60°C (>140°F)	Cracking due to stress corrosion cracking	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No
VIII.G.SP-60	VIII.G-34(SP-60)	Piping, piping components, and piping elements	Steel	Condensation (Internal)	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No

VIII STEAM AND POWER CONVERSION SYSTEM
G Auxiliary Feedwater System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VIII.G.SP-91	VIII.G-35(SP-25)	Piping, piping components, and piping elements	Steel	Lubricating oil	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M39, "Lubricating Oil Analysis," and Chapter XI.M32, "One-Time Inspection"	No
VIII.G.S-11	VIII.G-37(S-11)	Piping, piping components, and piping elements	Steel	Treated water	Cumulative fatigue damage due to fatigue	Fatigue is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation. See the SRP, Section 4.3 "Metal Fatigue," for acceptable methods for meeting the requirements of 10 CFR 54.21(c)(1).	Yes, TLAA
VIII.G.SP-74	VIII.G-38(S-10)	Piping, piping components, and piping elements	Steel	Treated water	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M2, "Water Chemistry," and Chapter XI.M32, "One-Time Inspection"	No
VIII.G.S-16	VIII.G-39(S-16)	Piping, piping components, and piping elements	Steel	Treated water	Wall thinning due to flow-accelerated corrosion	Chapter XI.M17, "Flow-Accelerated Corrosion"	No
VIII.G.SP-118		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Cracking due to stress corrosion cracking	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated
VIII.G.SP-127		Piping, piping components, and piping elements; tanks	Stainless steel	Air – outdoor	Loss of material due to pitting and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	Yes, environmental conditions need to be evaluated

VIII G STEAM AND POWER CONVERSION SYSTEM
Auxiliary Feedwater System (PWR)

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VIII.G.SP-145	VIII.G-1(S-01)	Piping, piping components, and piping elements; tanks	Steel (with coating or wrapping)	Soil or concrete	Loss of material due to general, pitting, crevice, and microbiologically-influenced corrosion	Chapter XI.M41, "Buried and Underground Piping and Tanks"	No
VIII.G.SP-136	VIII.G-36(S-12)	Steel Piping, piping components, and piping elements exposed to Raw water	Steel	Raw water	Loss of material due to general, pitting, crevice, galvanic, and microbiologically-influenced corrosion; fouling that leads to corrosion	Chapter XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components"	No
VIII.G.S-31	VIII.G-40(S-31)	Tanks	Steel	Air – outdoor (External)	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M29, “Aboveground Metallic Tanks”	No
VIII.G.SP-116		Tanks	Steel	Soil or Concrete	Loss of material due to general, pitting, and crevice corrosion	Chapter XI.M29, “Aboveground Metallic Tanks”	No
VIII.G.SP-75	VIII.G-41(S-13)	Tanks	Steel; stainless steel	Treated water	Loss of material due to general (steel only), pitting, and crevice corrosion	Chapter XI.M2, “Water Chemistry,” and Chapter XI.M32, “One-Time Inspection”	No

H. EXTERNAL SURFACES OF COMPONENTS AND MISCELLANEOUS BOLTING

Systems, Structures, and Components

This section includes the aging management programs for the degradation of external surfaces of all steel structures and components, including closure bolting in the steam and power conversion system in pressurized water reactors (PWRs) and boiling water reactors (BWRs). For the steel components in PWRs, this section addresses only boric acid corrosion of external surfaces as a result of dripping borated water leaking from an adjacent PWR component.

System Interfaces

The structures and components covered in this section belong to the Steam and Power Conversion Systems in PWRs and BWRs (for example, see system interfaces in VIII.A to VIII.G for details).

VIII STEAM AND POWER CONVERSION SYSTEM
H External Surfaces of Components and Miscellaneous Bolting

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VIII.H.SP-149		Bolting	Copper alloy	Any environment	Loss of preload due to thermal effects, gasket creep, and self-loosening	Chapter XI.M18, "Bolting Integrity"	No
VIII.H.SP-150		Bolting	Nickel alloy	Any environment	Loss of preload due to thermal effects, gasket creep, and self-loosening	Chapter XI.M18, "Bolting Integrity"	No
VIII.H.SP-143		Bolting	Stainless steel	Soil or concrete	Loss of material due to pitting and crevice corrosion	Chapter XI.M41, "Buried and Underground Piping and Tanks"	No
VIII.H.SP-144		Bolting	Stainless steel	Soil	Loss of preload	Chapter XI.M18, "Bolting Integrity"	No
VIII.H.S-40	VIII.H-2(S-40)	Bolting	Steel	Air with borated water leakage	Loss of material due to boric acid corrosion	Chapter XI.M10, “Boric Acid Corrosion”	No
VIII.H.SP-141		Bolting	Steel	Soil or concrete	Loss of material due to general, pitting and crevice corrosion	Chapter XI.M41, "Buried and Underground Piping and Tanks"	No
VIII.H.SP-142		Bolting	Steel	Soil	Loss of preload	Chapter XI.M18, "Bolting Integrity"	No
VIII.H.SP-82	VIII.H-1(S-32)	Bolting	Steel; stainless steel	Air – outdoor (External)	Loss of material due to general (steel only), pitting, and crevice corrosion	Chapter XI.M18, "Bolting Integrity"	No

VIII STEAM AND POWER CONVERSION SYSTEM
H External Surfaces of Components and Miscellaneous Bolting

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VIII.H.SP-151		Bolting	Steel; stainless steel	Air – outdoor (External)	Loss of preload due to thermal effects, gasket creep, and self-loosening	Chapter XI.M18, "Bolting Integrity"	No
VIII.H.S-03	VIII.H-3(S-03)	Closure bolting	High-strength steel	Air with steam or water leakage	Cracking due to cyclic loading, stress corrosion cracking	Chapter XI.M18, "Bolting Integrity"	No
VIII.H.S-02	VIII.H-6(S-02)	Closure bolting	Steel	Air with steam or water leakage	Loss of material due to general corrosion	Chapter XI.M18, "Bolting Integrity"	No
VIII.H.SP-84	VIII.H-4(S-34)	Closure bolting	Steel; stainless steel	Air – indoor, uncontrolled (External)	Loss of material due to general (steel only), pitting, and crevice corrosion	Chapter XI.M18, "Bolting Integrity"	No
VIII.H.SP-83	VIII.H-5(S-33)	Closure bolting	Steel; stainless steel	Air – indoor, uncontrolled (External)	Loss of preload due to thermal effects, gasket creep, and self-loosening	Chapter XI.M18, "Bolting Integrity"	No
VIII.H.S-29	VIII.H-7(S-29)	External surfaces	Steel	Air – indoor, uncontrolled (External)	Loss of material due to general corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
VIII.H.S-41	VIII.H-8(S-41)	External surfaces	Steel	Air – outdoor (External)	Loss of material due to general corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
VIII.H.S-30	VIII.H-9(S-30)	External surfaces	Steel	Air with borated water leakage	Loss of material due to boric acid corrosion	Chapter XI.M10, “Boric Acid Corrosion”	No

VIII STEAM AND POWER CONVERSION SYSTEM
H External Surfaces of Components and Miscellaneous Bolting

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VIII.H.S-42	VIII.H-10(S-42)	External surfaces	Steel	Condensation (External)	Loss of material due to general corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
VIII.H.SP-147		Piping, piping components, and piping elements	Aluminum	Air - outdoor	Loss of material due to pitting and crevice corrosion	Chapter XI.M36, "External Surfaces Monitoring of Mechanical Components"	No
VIII.H.SP-161		Underground piping, piping components, and piping elements	Steel; stainless steel; copper alloy; aluminum	Air-indoor uncontrolled or condensation (External)	Loss of material due to general (steel only), pitting and crevice corrosion	Chapter XI.M41, “Buried and Underground Piping and Tanks”	No

I. COMMON MISCELLANEOUS MATERIAL/ENVIRONMENT COMBINATIONS

Systems, Structures, and Components

This section includes the aging management programs for miscellaneous material/environment combinations which may be found throughout the steam and power conversion system's structures and components. For the material/environment combinations in this part, aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation. Therefore, no resulting aging management programs for these structures and components are required.

System Interfaces

The structures and components covered in this section belong to the steam and power conversion system in pressurized water reactors (PWRs) and boiling water reactors (BWRs) (for example, see system interfaces in VIII.A to VIII.G2 for details).

VIII STEAM AND POWER CONVERSION SYSTEM
I Common Miscellaneous Material/Environment Combinations

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VIII.I.SP-33	VIII.I-4(SP-33)	Piping elements	Glass	Air	None	None	No
VIII.I.SP-9	VIII.I-5(SP-9)	Piping elements	Glass	Air – indoor, uncontrolled (External)	None	None	No
VIII.I.SP-108		Piping elements	Glass	Air – outdoor	None	None	No
VIII.I.SP-67		Piping elements	Glass	Air with borated water leakage	None	None	No
VIII.I.SP-70		Piping elements	Glass	Closed-cycle cooling water	None	None	No
VIII.I.SP-68		Piping elements	Glass	Condensation	None	None	No
VIII.I.SP-111		Piping elements	Glass	Condensation (Internal/External)	None	None	No
VIII.I.SP-69		Piping elements	Glass	Gas	None	None	No
VIII.I.SP-10	VIII.I-6(SP-10)	Piping elements	Glass	Lubricating oil	None	None	No

VIII STEAM AND POWER CONVERSION SYSTEM
I Common Miscellaneous Material/Environment Combinations

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VIII.I.SP-34	VIII.I-7(SP-34)	Piping elements	Glass	Raw water	None	None	No
VIII.I.SP-35	VIII.I-8(SP-35)	Piping elements	Glass	Treated water	None	None	No
VIII.I.SP-93		Piping, piping components, and piping elements	Aluminum	Air – indoor, uncontrolled (Internal/External)	None	None	No
VIII.I.SP-23	VIII.I-1(SP-23)	Piping, piping components, and piping elements	Aluminum	Gas	None	None	No
VIII.I.SP-6	VIII.I-2(SP-6)	Piping, piping components, and piping elements	Copper alloy	Air – indoor, uncontrolled (External)	None	None	No
VIII.I.SP-5	VIII.I-3(SP-5)	Piping, piping components, and piping elements	Copper alloy	Gas	None	None	No
VIII.I.SP-104		Piping, piping components, and piping elements	Copper alloy (≤15% Zn and ≤8% Al)	Air with borated water leakage	None	None	No

VIII STEAM AND POWER CONVERSION SYSTEM
I Common Miscellaneous Material/Environment Combinations

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VIII.I.SP-11	VIII.I-9(SP-11)	Piping, piping components, and piping elements	Nickel alloy	Air – indoor, uncontrolled (External)	None	None	No
VIII.I.SP-148		Piping, piping components, and piping elements	Nickel alloy	Air with borated water leakage	None	None	No
VIII.I.SP-152		Piping, piping components, and piping elements	PVC	Air – indoor, uncontrolled	None	None	No
VIII.I.SP-153		Piping, piping components, and piping elements	PVC	Condensation (Internal)	None	None	No
VIII.I.SP-12	VIII.I-10(SP-12)	Piping, piping components, and piping elements	Stainless steel	Air – indoor, uncontrolled (External)	None	None	No
VIII.I.SP-86		Piping, piping components, and piping elements	Stainless steel	Air – indoor, uncontrolled (Internal)	None	None	No
VIII.I.SP-13	VIII.I-11(SP-13)	Piping, piping components, and piping elements	Stainless steel	Concrete	None	None	No

VIII STEAM AND POWER CONVERSION SYSTEM
I Common Miscellaneous Material/Environment Combinations

Item	Link	Structure and/or Component	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Further Evaluation
VIII.I.SP-15	VIII.I-12(SP-15)	Piping, piping components, and piping elements	Stainless steel	Gas	None	None	No
VIII.I.SP-1	VIII.I-13(SP-1)	Piping, piping components, and piping elements	Steel	Air – indoor controlled (External)	None	None	No
VIII.I.SP-154	VIII.I-14(SP-2)	Piping, piping components, and piping elements	Steel	Concrete	None	None, provided: 1) attributes of the concrete are consistent with ACI 318 or ACI 349 (low water-to-cement ratio, low permeability, and adequate air entrainment) as cited in NUREG-1557, and 2) plant OE indicates no degradation of the concrete	No, if conditions are met.
VIII.I.SP-4	VIII.I-15(SP-4)	Piping, piping components, and piping elements	Steel	Gas	None	None	No

CHAPTER IX

SELECTED DEFINITIONS AND USE OF TERMS FOR STRUCTURES, COMPONENTS, MATERIALS, ENVIRONMENTS, AGING EFFECTS, AND AGING MECHANISMS

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

A. Introduction

B. Structures and Components

C. Materials

D. Environments

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F. Significant Aging Mechanisms

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A. Introduction

The format and content of the aging management review (AMR) tables presented here (GALL Report, Rev. 2), have been revised to enhance the report's applicability to future plant license renewal applications. Several types of changes are incorporated in this revision to achieve the objective. One of these changes is to incorporate additional material, environment, aging effect and program (MEAP) combinations established by precedents based on a strong technical justification from earlier license renewal applications (LRAs) and the corresponding NRC safety evaluation reports (SERs).

The NRC has added several new definitions and clarified some of those that were in the GALL Report , Rev.1.

B. Structures and Components

The GALL Report does not address scoping of structures and components for license renewal. Scoping is plant-specific, and the results depend on individual plant design and its current licensing basis. The inclusion of a certain structure or component in the GALL Report does not mean that this particular structure or component is within the scope of license renewal for all plants. Conversely, the omission of a certain structure or component in the GALL Report does not mean that this particular structure or component is omitted from the scope of license renewal for any plant.

IX.B Selected Definitions & Use of Terms for Describing and Standardizing STRUCTURES AND COMPONENTS

Term	Definition as used in this document
Bolting	Bolting can refer to structural bolting, closure bolting, or all other bolting. Within the scope of license renewal, both Class 1 and non-Class 1 systems and components contain bolted closures that are necessary for the pressure boundary of the components being joined or closed. Closure bolting in high-pressure or high-temperature systems is defined as that in which the pressure exceeds 275 psi or 200°F (93°C). Closure bolting is used to join pressure boundaries or where a mechanical seal is required.
Ducting and components	Ducting and components include heating, ventilation, and air-conditioning (HVAC) components. Examples include ductwork, ductwork fittings, access doors, equipment frames and housing, housing supports, including housings for valves, dampers (including louvers, gravity, and fire dampers), and ventilation fans (including exhaust fans, intake fans, and purge fans). In some cases, this includes HVAC closure bolts or HVAC piping.
Encapsulation components/ valve chambers	These are airtight enclosures that function as a secondary containment boundary to completely enclose containment sump lines and isolation valves. Encapsulation components and features (e.g., emergency core cooling system, containment spray system, and containment isolation system, and refueling water storage tank, etc.) can include encapsulation vessels, piping, and valves.
"Existing programs" components	Per EPRI MRP-227 [Ref. 1] guidance on inspection and evaluation, PWR vessel internals (GALL AMP XI.M16A) were assigned to one of the following four groups: Primary, Expansion, Existing Programs, and No Additional Measures. Existing program components are those PWR internals that are susceptible to the effects of at least one of the aging mechanisms identified in MRP-227 and for which generic and plant-specific existing AMP elements are capable of managing those effects.
"Expansion" components	Per EPRI MRP-227 guidance on inspection and evaluation, PWR vessel internals (GALL AMP XI.M16A) were assigned to one of the following four

IX.B Selected Definitions & Use of Terms for Describing and Standardizing STRUCTURES AND COMPONENTS

Term	Definition as used in this document
	groups: Primary, Expansion, Existing Programs, and No Additional Measures. “Expansion” components are those PWR internals that are highly or moderately susceptible to the effects of at least one of the aging mechanisms addressed by MRP-227, but for which functionality assessment has shown a degree of tolerance to those effects. (See MRP-227, Section 3.3)
External surfaces	In the context of structures and components, the term “external surfaces” is used to represent the external surfaces of structures and components, such as tanks, that are not specifically listed elsewhere.
Heat exchanger components	A heat exchanger is a device that transfers heat from one fluid to another without the fluids coming in contact with each other. This includes air handling units and other devices that cool or heat fluids. Heat exchanger components may include, but are not limited to, air handling unit cooling and heating coils, piping/tubing, shell, tubesheets, tubes, valves, and bolting. Although tubes are the primary heat transfer components, heat exchanger internals, including tubesheets and fins, contribute to heat transfer and may be affected by reduction of heat transfer due to fouling [Ref. 2]. The inclusion of components such as tubesheets is dependent on manufacturer specifications.
High voltage insulators	An insulator is an insulating material in a configuration designed to physically support a conductor and separate the conductor electrically from other conductors or objects. The high voltage insulators that are evaluated for license renewal are those used to support and insulate high voltage electrical components in switchyards, switching stations and transmission lines.
Metal enclosed bus	“Metal enclosed bus” (MEB) is the term used in electrical and industry standards (IEEE and ANSI) for electrical buses installed on electrically-insulated supports constructed with all phase conductors enclosed in a metal enclosure.

IX.B Selected Definitions & Use of Terms for Describing and Standardizing STRUCTURES AND COMPONENTS

Term	Definition as used in this document
"No Additional Measures" components	Per EPRI MRP-227 guidance on inspection and evaluation, PWR vessel internals (GALL AMP XI.M16A) were assigned to one of the following four groups: Primary, Expansion, Existing Programs, and No Additional Measures. Additional components were placed in the "No Additional Measures," group as a result of the Failure Mode, Effects, and Criticality Analysis and the functionality assessment. Note: Components with no additional measures are not uniquely identified in GALL tables (see AMR Items IV.B2.RP-265, IV.B2.RP-267, IV.B3.RP-306, IV.B3.RP-307, IV.B4.RP-236, and IV.B4.RP-237. Components with no additional measures are defined in Section 3.3.1 of MRP-227, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines."
Piping, piping components, piping elements, and tanks	This general category includes features of the piping system within the scope of license renewal. Examples include piping, fittings, tubing, flow elements/indicators, demineralizers, nozzles, orifices, flex hoses, pump casings and bowls, safe ends, sight glasses, spray heads, strainers, thermowells, and valve bodies and bonnets. For reactor coolant pressure boundary components in Chapter IV that are subject to cumulative fatigue damage, this category also can include flanges, nozzles and safe ends, penetrations, instrument connections, vessel heads, shells, welds, weld inlays and weld overlays, stub tubes, and miscellaneous Class 1 components (e.g., pressure housings, etc.). As used in AMP XI.M41, buried piping and tanks are in direct contact with soil or concrete (e.g., a wall penetration). Underground piping and tanks are below grade, but are contained within a tunnel or vault such that they are in contact with air and are located where access for inspection is restricted.

IX.B Selected Definitions & Use of Terms for Describing and Standardizing STRUCTURES AND COMPONENTS

Term	Definition as used in this document
Piping elements	The category of “piping elements” is a sub-category of piping, piping components, and piping elements that in GALL Report, Rev. 2 applies only to components made of glass (e.g., sight glasses and level indicators, etc.). In the GALL Report, Chapters V, VII, and VIII, piping elements are thus called out separately.
Pressure housing	The term “pressure housing” only refers to pressure housing for the control rod drive (CRD) head penetration (it is only of concern in Section A2 for PWR reactor vessels).
“Primary” components	Per EPRI MRP-227 guidance on inspection and evaluation, PWR vessel internals (GALL AMP XI.M16A) were assigned to one of the following four groups: Primary, Expansion, Existing Programs, and No Additional Measures. Primary components are those PWR internals that are highly susceptible to the effects of at least one of the aging mechanisms addressed by MRP-227. The Primary group also includes components which have shown a degree of tolerance to a specific aging degradation effect, but for which no highly susceptible component exists or for which no highly susceptible component is accessible.
Reactor coolant pressure boundary components	Reactor coolant pressure boundary components include, but are not limited to, piping, piping components, piping elements, flanges, nozzles, safe ends, pressurizer vessel shell heads and welds, heater sheaths and sleeves, penetrations, and thermal sleeves.
Seals, gaskets, and moisture barriers (caulking, flashing, and other sealants)	This category includes elastomer components used as sealants or gaskets.
Steel elements: liner; liner anchors; integral attachments	This category includes steel liners used in suppression pools or spent fuel pools.
Switchyard bus	Switchyard bus is the uninsulated, unenclosed, rigid electrical conductor or pipe used in switchyards and switching stations to connect two or more elements of an electrical power circuit, such as active disconnect switches and passive transmission conductors.

IX.B Selected Definitions & Use of Terms for Describing and Standardizing STRUCTURES AND COMPONENTS

Term	Definition as used in this document
Tanks	Tanks are large reservoirs used as hold-up volumes for liquids or gases. Tanks may have an internal liquid and/or vapor space and may be partially buried or in close proximity to soils or concrete. Tanks are treated separately from piping due to their potential need for different aging management programs (AMP). One example is GALL AMP XI.M29, "Aboveground Metallic Tanks," for tanks partially buried or in contact with soil or concrete that experience general corrosion as the aging effect at the soil or concrete interface.
Transmission conductors	Transmission conductors are uninsulated, stranded electrical cables used in switchyards, switching stations, and transmission lines to connect two or more elements of an electrical power circuit, such as active disconnect switches, power circuit breakers, and transformers and passive switchyard bus.
Vibration isolation elements	This category includes non-steel supports used for supporting components prone to vibration.

C. Materials

The following table defines many generalized materials used in the preceding GALL AMR tables in Chapters II through VIII of GALL Report, Rev. 2.

IX.C Selected Definitions & Use of Terms for Describing and Standardizing MATERIALS

Term	Definition as used in this document
Boraflex	Boraflex is a material that is composed of 46% silica, 4% polydimethyl siloxane polymer, and 50% boron carbide, by weight. It is a neutron-absorbing material used in spent fuel storage racks. Degradation of Boraflex panels under gamma radiation can lead to a loss of their ability to absorb neutrons in spent fuel storage pools. The aging management program for Boraflex is found in GALL AMP XI.M22, "Boraflex Monitoring."
Boral®, boron steel	Boron steel is steel with a boron content ranging from one to several percent. Boron steel absorbs neutrons and is often used as a control rod to help control the neutron flux. Boral® is a cermet consisting of a core of aluminum and boron carbide powder sandwiched between sheets of aluminum. Boral refers to patented Aluminum-Boron master alloys; these alloys can contain up to 10% boron as AlB_{12} intermetallics.
Cast austenitic stainless steel (CASS)	CASS alloys, such as CF-3, CF-8, CF-3M, and CF-8M, have been widely used in LWRs. These CASS alloys 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.
Concrete and cementitious material	When used generally, this category of concrete applies to concrete in many different configurations (block, cylindrical, etc.) and prestressed or reinforced concrete. Cementitious material can be defined as any material having cementing properties, which contributes to the formation of hydrated calcium silicate compounds. When mixing concrete, the following have cementitious properties: Portland cement, blended hydraulic cement, fly ash, ground granulated blast furnace slag, silica fume, calcined clay, metakaolin, calcined shale, and rice husk ash. This category may include asbestos cement.

IX.C Selected Definitions & Use of Terms for Describing and Standardizing MATERIALS

Term	Definition as used in this document
Copper alloy (≤15% Zn and ≤8% Al)	This category applies to those copper alloys whose critical alloying elements are less than the thresholds that keep the alloy from being susceptible to aging effects. For example, copper, copper nickel, brass, bronze ≤15% zinc (Zn), and aluminum bronze ≤8% aluminum (Al) are resistant to stress corrosion cracking, selective leaching, and pitting and crevice corrosion. They may be identified simply as “copper alloy” when these aging mechanisms are not at issue.
Copper alloy (>15% Zn or >8% Al)	This category applies to those copper alloys whose critical alloying elements are above the thresholds that make them susceptible to aging effects. Copper-zinc alloys >15% zinc are susceptible to stress corrosion cracking (SCC), selective leaching (except for inhibited brass), and pitting and crevice corrosion. Additional copper alloys, such as aluminum bronze > 8% aluminum, also may be susceptible to SCC or leaching. The elements that are most commonly alloyed with copper are zinc (forming brass), tin (forming bronze), nickel, silicon, aluminum (forming aluminum-bronze), cadmium, and beryllium. Additional copper alloys may be susceptible to these aging effects if they fall above the threshold for the critical alloying element. [Ref. 3]
Elastomers	Elastomers are flexible materials such as rubber, EPT, EPDM, PTFE, ETFE, viton, vitril, neoprene, and silicone elastomer. Hardening and loss of strength of elastomers can be induced by elevated temperature (over about 95°F or 35°C), and additional aging factors (e.g., exposure to ozone, oxidation, and radiation, etc.). [Ref. 4]
Galvanized steel	Galvanized steel is steel coated with zinc, usually by immersion or electrodeposition. The zinc coating protects the underlying steel because the corrosion rate of the zinc coating in dry, clean air is very low. In the presence of moisture, galvanized steel is classified under the category “Steel.”
Glass	This category includes any glass material. Glass is a hard, amorphous, brittle, super-cooled liquid made by fusing together one or more of the oxides of silicon, boron, or phosphorous with certain basic oxides (e.g., Na, Mg, Ca, K), and cooling the product rapidly to prevent crystallization or devitrification.
Graphitic tool steel	Graphitic tool steels (such as AISI O6, which is oil-hardened, and, AISI A10, which is air-hardened), have excellent non-seizing properties. The graphite particles provide self-lubricity and hold applied lubricants.

IX.C Selected Definitions & Use of Terms for Describing and Standardizing MATERIALS

Term	Definition as used in this document
Gray cast iron	Gray cast iron is an iron alloy made by adding larger amounts of carbon to molten iron than would be used to make steel. Most steel has less than about 1.2% by weight carbon, while cast irons typically have between 2.5 to 4%. Gray cast iron contains flat graphite flakes that reduce its strength and form cracks, inducing mechanical failures. They also cause the metal to behave in a nearly brittle fashion, rather than experiencing the elastic, ductile behavior of steel. Fractures in this type of metal tend to take place along the flakes, which give the fracture surface a gray color, hence the name of the metal. Gray cast iron is susceptible to selective leaching, resulting in a significant reduction of the material's strength due to the loss of iron from the microstructure, leaving a porous matrix of graphite. In some environments, gray cast iron is categorized with the group "Steel."
Insulation materials (e.g., bakelite, phenolic melamine or ceramic, molded polycarbonate)	Insulation materials in this category are bakelite, phenolic melamine or ceramic, molded polycarbonate, etc. used in electrical fuse holders.
Low-alloy steel, yield strength >150 ksi	Low-alloy steel includes AISI steels 4140, 4142, 4145, 4140H, 4142H, and 4145H (UNS#: G41400, G41420, G41450, H41400, H41420, H41450). Low-alloy steel bolting material, SA 193 Gr. B7, is a ferritic, low-alloy steel for high-temperature service. High-strength low-alloy (Fe-Cr-Ni-Mo) steel bolting materials have a maximum tensile strength of <1172 MPa (<170 ksi). They may be subject to stress corrosion cracking if the actual measured yield strength, S_y, ≥ 150 ksi (1034 MPa). Bolting fabricated from high-strength (actual measured yield strength, S_y, ≥ 150 ksi or 1034 MPa) low-alloy steel, SA 193 Gr. B7, is susceptible to stress corrosion cracking. Examples of high-strength alloy steels that comprise this category include SA540-Gr. B23/24, SA193-Gr. B8, and Grade L43 (AISI4340).
Lubrite®	Lubrite® refers to a patented technology in which the bearing substrate (bronze is commonly used, but in unusual environments can range from stainless steel and nodular-iron to tool-steel) is fastened to lubricant. Lubrite® is often defined as bronze attached to ASTM B22, alloy 905, with G10 lubricant. Even though Lubrite® bearings are characterized as maintenance-free because of the differences in installation,

IX.C Selected Definitions & Use of Terms for Describing and Standardizing MATERIALS

Term	Definition as used in this document
	fineness of the surfaces, and lubricant characteristics, they can experience mechanical wear and fretting. Bearings generally have not shown adverse conditions related to the use of Lubrite®. The unique environment and precise installation tolerances required for installing the bearings require bearing-specific examinations. The vendor's (Lubrite® Technologies) literature shows ten lubricant types used in the bearings, ranging from G1 (General Duty) to AE7 (temperature- and radiation-tested) lubricants. The type of lubricant used depends on the plant-specific requirements. Careful installation and clearing out any obstructions during installation ensures that the required tolerances of the bearings are met and reduces the likelihood of functional problems during challenging loading conditions (such as design basis accident [DBA] or safe shutdown earthquake [SSE]). The associated aging effects could include malfunctioning, distortion, dirt accumulation, and fatigue under vibratory and cyclic thermal loads. The potential aging effects could be managed by incorporating its periodic examination in ASME Section XI, Subsection IWF (AMP XI.S3) or in Structures Monitoring (AMP XI.S6).
Malleable iron	The term "Malleable iron" usually means malleable cast iron, characterized by exhibiting some elongation and reduction in area in a tensile test. Malleable iron is one of the materials in the category of "Porcelain, Malleable iron, aluminum, galvanized steel, cement."
Nickel alloys	Nickel alloys are nickel-chromium-iron (molybdenum) alloys and include the Alloys 600 and 690. Examples of nickel alloys include Alloy 182, 600, and 690, Gr. 688 (X-750), Inconel 182, Inconel 82, NiCrFe, SB-166, -167, and -168, and X-750. [Ref. 5]
Polymer	This category generally includes flexible polymeric materials (such as rubber) and rigid polymers (like PVC). As used in GALL Report, Rev. 2 AMR Items VI.A.LP-33, VI.A.LP-34, and VI.A.LP-35, polymers used in electrical applications include EPR (ethylene-propylene rubber), SR (silicone rubber), EPDM (ethylene propylene diene monomer), and XLPE (crosslinked polyethylene). XLPE is a cross-linked polyethylene thermoplastic resin, such as polyethylene and polyethylene copolymers. EPR and EPDM are ethylene-propylene rubbers in the category of thermosetting elastomers.

IX.C Selected Definitions & Use of Terms for Describing and Standardizing MATERIALS

Term	Definition as used in this document
Porcelain	Hard-quality porcelain is used as an insulator for supporting high-voltage electrical insulators. Porcelain is a hard, fine-grained ceramic that consists of kaolin, quartz, and feldspar fired at high temperatures.
SA508-Cl 2 forgings clad with stainless steel using a high-heat-input welding process	This category consists of quenched and tempered vacuum-treated carbon and alloy steel forgings for pressure vessels.
Stainless steel	Products grouped under the term “stainless steel” include wrought or forged austenitic, ferritic, martensitic, precipitation-hardened (PH), or duplex stainless steel (Cr content >11%). 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. In some cases, when the recommended AMP is the same for PH stainless steel or cast austenitic stainless steel (CASS) as for stainless steel, PH stainless steel or CASS are included as a part of the stainless steel classification. However, CASS is quite susceptible to loss of fracture toughness due to thermal and neutron irradiation embrittlement. Therefore, when this aging effect is being considered, CASS is specifically designated in an AMR line-item. Steel with stainless steel cladding also may 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 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, and Types 304, 304NG, 308, 308L, 309, 309L, 316, 347, 403, and 416. Examples of CASS designations include CF-3, -8, -3M, and -8M. [Ref. 6, 7]
Steel	In some environments, carbon steel, alloy steel, cast iron, gray cast iron, malleable iron, and high-strength low-alloy steel are vulnerable to general, pitting, and crevice corrosion, even though the rates of aging may vary. Consequently, these metal types are generally grouped under the broad term “steel.” Note that this does not include stainless steel, which has its own category. However, gray cast iron also is susceptible to selective leaching, and high-strength low-alloy steel is susceptible to stress corrosion cracking. Therefore, when these aging effects are being considered, these

IX.C Selected Definitions & Use of Terms for Describing and Standardizing MATERIALS

Term	Definition as used in this document
	materials are specifically identified. Galvanized steel (zinc-coated carbon steel) is also included in the category of "steel" when exposed to moisture. Malleable iron is specifically called out in the phrase "Porcelain, Malleable iron, aluminum, galvanized steel, cement," which is used to define the high voltage insulators in GALL Chapter VI. Examples of steel designations included in this category are ASTM A36, ASTM A285, ASTM A759, SA36, SA106-Gr. B, SA155-Gr. KCF70, SA193-Gr. B7, SA194 -Gr. 7, SA302-Gr B, SA320-Gr. L43 (AISI 4340), SA333-Gr. 6, SA336, SA508-64, class 2, SA508-Cl 2 or Cl 3, SA516-Gr. 70, SA533-Gr. B, SA540-Gr. B23/24, and SA582. [Ref. 6, 7]
Superaustenitic stainless steel	Superaustenitic stainless steels have the same structure as the common austenitic alloys, but they have enhanced levels of elements such as chromium, nickel, molybdenum, copper, and nitrogen, which give them superior strength and corrosion resistance. Compared to conventional austenitic stainless steels, superaustenitic materials have a superior resistance to pitting and crevice corrosion in environments containing halides. Several NPPs have installed superaustenitic stainless steel (AL-6XN) buried piping.
Titanium	The category titanium includes unalloyed titanium (ASTM grades 1-4) and various related alloys (ASTM grades 5, 7. 9, and 12). The corrosion resistance of titanium is a result of the formation of a continuous, stable, highly adherent protective oxide layer on the metal surface. Titanium and titanium alloys may be susceptible to crevice corrosion in saltwater environments at elevated temperatures (>160^{o}F). Titanium Grades 5 and 12 are resistant to crevice corrosion in seawater at temperatures as high as 500^{o}F. Stress corrosion cracking of titanium and its alloys is considered applicable in sea water or brackish raw water systems if the titanium alloy contains more than 5% aluminum or more than 0.20% oxygen or any amount of tin. ASTM Grades 1, 2, 7, 11, or 12 are not susceptible to stress corrosion cracking in seawater or brackish raw water [Ref. 8].
Wood	Wood piles or sheeting exposed to flowing or standing water is subject to loss of material or changes in material properties due to weathering, chemical degradation, insect infestation, repeated wetting and drying, or fungal decay.
Zircaloy-4	Zircaloy-4, (Zry-4), is a member in the group of high-zirconium (Zr) alloys. Such zircaloys are used in nuclear

IX.C Selected Definitions & Use of Terms for Describing and Standardizing MATERIALS

Term	Definition as used in this document
	technology, as Zr has very low absorption cross-section of thermal neutrons. In the GALL Report, Zry-4 is referenced in AMR Item IV.B3.RP-357 for incore instrumentation thimble tubes. Zry-4 consists of 98.23 weight % zirconium with 1.45% tin, 0.21% iron, 0.1% chromium, and 0.01% hafnium.

D. Environments

The following table defines many of the standardized environments used in the preceding GALL AMR tables in Chapters II through VIII of the GALL Report, Rev. 2.

The usage of temperature thresholds for describing aging effects are continued as in the GALL Report, Rev. 1.

<u>Temperature threshold of 95°F (35°C) for thermal stresses in elastomers</u>: In general, if the ambient temperature is less than about 95°F (35°C), then thermal aging may be considered not significant for rubber, butyl rubber, neoprene, nitrile rubber, silicone elastomer, fluoroelastomer, EPR, and EPDM [Ref. 3]. Hardening and loss of strength of elastomers can be induced by thermal aging, exposure to ozone, oxidation, and radiation. When applied to the elastomers used in electrical 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.

<u>Temperature threshold of 140°F (60°C) for SCC in stainless steel:</u> Stress corrosion cracking (SCC) occurs very rarely in austenitic stainless steels below 140°F (60°C). Although SCC has been observed in stagnant, oxygenated borated water systems at lower temperatures than this 140°F threshold, all of these instances have identified a significant presence of contaminants (halogens, specifically chlorides) in the failed components. With a harsh enough environment (i.e., significant contamination), SCC can occur in austenitic stainless steel at ambient temperature. However, these conditions are considered event-driven, resulting from a breakdown of chemistry controls [Ref. 8, 9].

<u>Temperature threshold of 482°F (250°C) for thermal embrittlement in CASS:</u> CASS subjected to sustained temperatures below 250°C (482°F) will not result in a reduction of room temperature Charpy impact energy below 50 ft-lb for exposure times of approximately 300,000 hours (for CASS with ferrite content of 40% and approximately 2,500,000 hours for CASS with ferrite content of 14%) [Fig. 2; Ref. 10]. For a maximum exposure time of approximately 420,000 hours (48 EFPY), a screening temperature of 482°F is conservatively chosen because (1) the majority of nuclear grade materials is expected to contain a ferrite content well below 40%, and (2) the 50 ft-lb limit is very conservative when applied to cast austenitic materials. It is typically applied to ferritic materials, e.g., 10 CFR 50 Appendix G. For CASS components in the reactor coolant pressure boundary, this threshold is supported by the GALL AMP XI.M12, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)," with the exception of niobium-containing steels, which require evaluation on a case-by-case basis.

IX.D Selected Definitions & Use of Terms for Describing and Standardizing ENVIRONMENTS

Term	Definition as used in this document
Adverse localized environment	An adverse localized environment is an environment limited to the immediate vicinity of a component that is hostile to the component material, thereby leading to potential aging effects. As used in GALL, the conductor insulation used for electrical cables in instrumentation circuits can be subjected to an adverse localized environment. As represented by a specific GALL AMR Item, an adverse localized environment can be due to any of the following: (1) exposure to significant moisture (LP-35), (2) heat, radiation, or moisture (L-01 or LP-34), or (3) heat, radiation, moisture, or voltage (L-05).
Aggressive environment (steel in concrete)	This environment affects steel embedded in concrete with a pH <5.5 or a chloride concentration >500 ppm or sulfate > 1500 ppm. [Ref. 11]
Air – indoor controlled	This environment is one to which the specified internal or external surface of the component or structure is exposed; a humidity-controlled (i.e., air conditioned) environment. For electrical purposes, control must be sufficient to eliminate the cited aging effects of contamination and oxidation without affecting the resistance.
Air – indoor uncontrolled	Uncontrolled indoor air is associated with systems with temperatures higher than the dew point (i.e., condensation can occur, but only rarely; equipment surfaces are normally dry).
Air – indoor uncontrolled >35°C (>95°F) (Internal/External)	Uncontrolled indoor air >35°C (>95°F) is above a thermal stress threshold for elastomers (i.e., <95°F). It is an environment to which the internal or external surface of the component or structure can be exposed. If the ambient temperature is maintained <95°F, any resultant thermal aging of organic materials can be considered as insignificant over the 60-yr period of extended operation. [Ref. 3] However, elastomers can be subjected to aging effects from other factors, such as exposure to ozone, oxidation, and radiation.
Air – outdoor	The outdoor environment consists of moist, possibly salt-laden atmospheric air, ambient temperatures and humidity, and exposure to weather, including precipitation and wind. The component is exposed to air and local weather conditions, including salt water spray (if present). A component is considered susceptible to a wetted environment when it is submerged, has the potential to collect water, or is subject to external condensation.

IX.D Selected Definitions & Use of Terms for Describing and Standardizing ENVIRONMENTS

Term	Definition as used in this document
Air with borated water leakage	Air and untreated borated water leakage on indoor or outdoor systems with temperatures either above or below the dew point. The water from leakage is considered to be untreated, due to the potential for water contamination at the surface (germane to PWRs).
Air with leaking secondary-side water and/or steam	This environment applies to steel components in the pressure boundary and structural parts of the once-through steam generator that may be exposed to air with leaking secondary-side water and/or steam.
Air with metal temperature up to 288°C (550°F)	This environment is synonymous with the more commonly-used phrase “system temperature up to 288°C (550°F).”
Air with reactor coolant leakage	Air and reactor coolant or steam leakage on high temperature systems (germane to BWRs)
Air with steam or water leakage	Air and untreated steam or water leakage on indoor or outdoor systems with temperatures above or below the dew point.
Air, dry	Air that has been treated to reduce its dew point well below the system operating temperature. Within piping, unless otherwise specified, this encompasses either internal or external.
Air, moist	Air with enough moisture to facilitate the loss of material in steel caused by general, pitting, and crevice corrosion. Moist air in the absence of condensation also is potentially aggressive (e.g., under conditions where hygroscopic surface contaminants are present, etc.).
Any	This could be any indoor or outdoor environment where the aging effects are not dependent on environmental conditions.
Buried and underground	As referenced in AMP XI.M41, “Buried and Underground Piping and Tanks,” buried piping and tanks are those in direct contact with soil or concrete (e.g., a wall penetration). Underground piping and tanks are below grade, but are contained within a tunnel or vault such that they are in contact with air and are located where access for inspection is restricted.

IX.D Selected Definitions & Use of Terms for Describing and Standardizing ENVIRONMENTS

Term	Definition as used in this document
Closed-cycle cooling water	Treated water subject to the closed-cycle cooling water chemistry program is included in this environment. Closed-cycle cooling water >60°C (>140°F) makes the SCC of stainless steel possible. Examples of descriptors that comprise this category can include: • chemically-treated, borated water, and treated component cooling water • demineralized water on one side and closed-cycle cooling water (treated water) on the other side • chemically treated borated water on the tube side and closed-cycle cooling water on the shell side.
Concrete	This environment consists of components embedded in concrete.
Condensation (internal/external)	Condensation on the surfaces of systems at temperatures below the dew point is considered “raw water” due to the potential for internal or external surface contamination. Under certain circumstances, the former terms “moist air” or “warm moist air” are subsumed by the definition of “condensation,” which describes an environment where there is enough moisture for corrosion to occur.
Containment environment (inert)	A drywell environment 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.
Diesel exhaust	This environment consists of gases, fluids, and particulates present in diesel engine exhaust.
Fuel oil	Diesel oil, No. 2 oil, or other liquid hydrocarbons used to fuel diesel engines. Fuel oil used for combustion engines may be contaminated with water, which may promote additional aging effects.

IX.D Selected Definitions & Use of Terms for Describing and Standardizing ENVIRONMENTS

Term	Definition as used in this document
Gas	Internal gas environments include dry air or inert, non-reactive gases. This generic term is used only with "Common Miscellaneous Material/Environment," where aging effects are not expected to degrade the ability of the structure or component to perform its intended function for the period of extended operation. The term "gas" is not meant to comprehensively include all gases in the fire suppression system. The GALL AMP XI.M26, "Fire Protection," is used for the periodic inspection and testing of the halon/carbon dioxide fire suppression system.
Ground water/soil	Ground water is subsurface water that can be detected in wells, tunnels, or drainage galleries, or that flows naturally to the earth's surface via seeps or springs. Soil is a mixture of organic and inorganic materials produced by the weathering of rock and clay minerals or the decomposition of vegetation. Voids containing air and moisture can occupy 30 to 60 percent [Ref.12] of the soil volume. Concrete subjected to a ground water/soil environment can be vulnerable to an increase in porosity and permeability, cracking, loss of material (spalling, scaling)/, or aggressive chemical attack. Other materials with prolonged exposures to ground water or moist soils are subject to the same aging effects as those systems and components exposed to raw water.
Lubricating oil	Lubricating oils are low-to-medium viscosity hydrocarbons that can contain contaminants and/or moisture. This definition also functionally encompasses hydraulic oil (non-water based). These oils are used for bearing, gear, and engine lubrication. The GALL AMP XI.M39, *Lubricating Oil Analysis,* addresses this environment. Piping, piping components, and piping elements, whether copper, stainless steel, or steel, when exposed to lubricating oil with some water, will have limited susceptibility to aging degradation due to general or localized corrosion.
Raw water	Raw water consists of untreated surface or ground water, whether fresh, brackish, or saline in nature. This includes water for use in open-cycle cooling water systems and may include potable water, water that is used for drinking or other personal use. See also "condensation."

IX.D Selected Definitions & Use of Terms for Describing and Standardizing ENVIRONMENTS

Term	Definition as used in this document
Reactor coolant	Reactor coolant is treated water in the reactor coolant system and connected systems at or near full operating temperature, including steam associated with BWRs.
Reactor coolant >250°C (>482°F)	Treated water above the thermal embrittlement threshold for CASS.
Reactor coolant >250°C (>482°F) and neutron flux	Treated water in the reactor coolant system and connected systems above the thermal embrittlement threshold for CASS.
Reactor coolant and high fluence (>1 x 10^{21} n/cm^2 E >0.1 MeV)	Reactor coolant subjected to a high fluence (>1 x 10^{21} n/cm^2 E >0.1 MeV).
Reactor coolant and neutron flux	The reactor core environment that will result in a neutron fluence exceeding 10^{17} n/cm^2 (E >1 MeV) at the end of the license renewal term.
Reactor coolant and secondary feedwater/steam	Water in the reactor coolant system and connected systems at or near full operating temperature and the PWR feedwater or steam at or near full operating temperature, subject to the secondary water chemistry program (GALL AMP XI.M2).
Secondary feedwater	Within the context of the recirculating steam generator, components such as steam generator feedwater impingement plate and support may be subjected to loss of material due to erosion in a secondary feedwater environment. More generally, the environment of concern is a secondary feedwater/steam combination.
Secondary feedwater/steam	PWR feedwater or steam at or near full operating temperature, subject to the secondary water chemistry program (GALL AMP XI.M2).
Sodium pentaborate solution	Treated water that contains a mixture of borax and boric acid.

IX.D Selected Definitions & Use of Terms for Describing and Standardizing ENVIRONMENTS

Term	Definition as used in this document
Soil	Soil is a mixture of inorganic materials produced by the weathering of rock and clay minerals, and organic material produced by the decomposition of vegetation. Voids containing air and moisture occupy 30 to 60 percent [Ref.26] of the soil volume. Properties of soil that can affect degradation kinetics include moisture content, pH, ion exchange capacity, density, and hydraulic conductivity. External environments included in the soil category consist of components at the air/soil interface, buried in the soil, or exposed to ground water in the soil. See also "ground water/soil."
Steam	The steam environment is managed by the BWR water chemistry program or PWR secondary plant water chemistry program. Defining the temperature of the steam is not considered necessary for analysis.
System temperature up to 288°C (550°F)	This environment consists of a metal temperature of BWR components <288°C (550°F).
System temperature up to 340°C (644°F)	This environment consists of a maximum metal temperature <340°C (644°F).
Treated borated water	Borated (PWR) water is a controlled water system The Chemical and Volume Control System (CVCS) maintains the proper water chemistry in the reactor coolant system while adjusting the boron concentration during operation to match long-term reactivity changes in the core.
Treated borated water >250°C (>482°F)	Treated water with boric acid above the 250°C (>482°F) thermal embrittlement threshold for CASS
Treated borated water >60°C (>140°F)	Treated water with boric acid in PWR systems above the 60°C (>140°F) SCC threshold for stainless steel

IX.D Selected Definitions & Use of Terms for Describing and Standardizing ENVIRONMENTS

Term	Definition as used in this document
Treated water	Treated water is water whose chemistry has been altered and is maintained (as evidenced by testing) in a state which differs from naturally-occurring sources so as to meet a desired set of chemical specifications. Treated water generally falls into one of two categories. (1) The first category is based on demineralized water and, with the possible exception of boric acid (for PWRs only), generally contains minimal amounts of any additions. This water is generally characterized by high purity, low conductivity, and very low oxygen content. This category of treated water is generally used as BWR coolant and PWR primary and secondary water. (2) The second category may but need not be based on demineralized water. It contains corrosion inhibitors and also may contain biocides or other additives. This water will generally be comparatively higher in conductivity and oxygen content than the first category of treated water. This category of treated water is generally used in HVAC systems, auxiliary boilers, and diesel engine cooling systems. Closed-cycle cooling water is a subset of this category of treated water
Treated water >60°C (>140°F)	Treated water above the 60°C stress corrosion cracking threshold for stainless steel
Waste water	Radioactive, potentially radioactive, or non-radioactive waters that are collected from equipment and floor drains. Waste waters may contain contaminants, including oil and boric acid, depending on location, as well as originally treated water that is not monitored by a chemistry program.
Water-flowing	Water that is refreshed; thus, it has a greater impact on leaching and can include rainwater, raw water, ground water, or water flowing under a foundation
Water-standing	Water that is stagnant and unrefreshed, thus possibly resulting in increased ionic strength up to saturation

E. Aging Effects

The following table explains the selected usage of many of the standardized aging effects due to associated aging mechanisms used in the preceding GALL AMR tables in Chapters II through VIII of GALL Report, Rev. 2.

IX.E Selected Use of Terms for Describing and Standardizing AGING EFFECTS

Term	Usage in this document
Changes in dimensions	Changes in dimension can result from various phenomena, such as void swelling and, on a macroscopic level, denting
Concrete cracking and spalling	Cracking and exfoliation of concrete as the result of freeze-thaw, aggressive chemical attack, and reaction with aggregates
Corrosion of connector contact surfaces	Corrosion of exposed connector contact surfaces when caused by borated water intrusion
Crack growth	Increase in crack size attributable to cyclic loading
Cracking	This term is synonymous with the phrase “crack initiation and growth” in metallic substrates. Cracking in concrete when caused by restraint shrinkage, creep, settlement, and aggressive environment.
Cracking, loss of bond, and loss of material (spalling, scaling)	Cracking, loss of bond, and loss of material (spalling, scaling) when caused by corrosion of embedded steel in concrete.
Cracks; distortion; increase in component stress level	Within concrete structures, cracks, distortion, and increase in component stress level when caused by settlement. Although settlement can occur in a soil environment, the symptoms can be manifested in either an air-indoor uncontrolled or air-outdoor environment.
Cumulative fatigue damage	Cumulative fatigue damage is due to fatigue, as defined by ASME Boiler and Pressure Vessel Code.
Denting	Denting in steam generators can result from corrosion of carbon steel tube support plates.
Expansion and cracking	Within concrete structures, expansion and cracking can result from reaction with aggregates.
Fatigue	Fatigue in metallic fuse holder clamps can result from ohmic heating, thermal cycling, electrical transients, frequent manipulation, and vibration. [Ref. 13]

IX.E Selected Use of Terms for Describing and Standardizing AGING EFFECTS

Term	Usage in this document
Fretting or lockup	Fretting is accelerated deterioration at the interface between contacting surfaces as the result of corrosion and slight oscillatory movement between the two surfaces. In essence, both fretting and lockup are due to mechanical wear.
Hardening and loss of strength	Hardening (loss of flexibility) and loss of strength (loss of ability to withstand tensile or compressive stress) can result from elastomer degradation of seals and other elastomeric components. Weathered elastomers can experience increased hardness, shrinkage, and loss of strength.
Increase in porosity and permeability, cracking, loss of material (spalling, scaling), loss of strength	Porosity and permeability, cracking, and loss of material (spalling, scaling) in concrete can increase due to aggressive chemical attack. In concrete, the loss of material (spalling, scaling) and cracking can result from the freeze-thaw processes. Loss of strength can result from leaching of calcium hydroxide in the concrete.
Increased resistance of connection	Increased resistance of connection is an aging effect that can be caused by the loosening of bolts resulting from thermal cycling and ohmic heating. [VI.A. LP-25, Ref. 14, 15] In Chapter VI AMR line-items, increased resistance to connection is also said to be caused by the following aging mechanisms: • chemical contamination, corrosion, and oxidation (in an air, indoor controlled environment, increased resistance of connection due to chemical contamination, corrosion and oxidation do not apply) [VI.A. LP-23] • thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination, corrosion, and oxidation [VI.A. LP-30] • fatigue caused by frequent manipulation or vibration [VI.A. LP-31] • corrosion of connector contact surfaces caused by intrusion of borated water [VI.A. LP-36] • oxidation or loss of pre-load [VI.A. LP-39, VI.A. LP-48]

IX.E Selected Use of Terms for Describing and Standardizing AGING EFFECTS

Term	Usage in this document
Ligament cracking	Steel tube support plates can experience ligament cracking due to corrosion. As previously noted in IN 96-09, tube support plate signal anomalies found during eddy-current testing of SG tubes may be indicative of support plate damage or ligament cracking.
Loss of conductor strength	Transmission conductors can experience loss of conductor strength due to corrosion.
Loss of fracture toughness	Loss of fracture toughness can result from various aging mechanisms, including thermal aging embrittlement and neutron irradiation embrittlement
Loss of leak tightness	Steel airlocks can experience loss of leak tightness in the closed position resulting from mechanical wear of locks, hinges, and closure mechanisms
Loss of material	Loss of material may be due to general corrosion, boric acid corrosion, pitting corrosion, galvanic corrosion, crevice corrosion, erosion, fretting, flow-accelerated corrosion, MIC, fouling, selective leaching, wastage, wear, and aggressive chemical attack. In concrete structures, loss of material can also be caused by abrasion or cavitation or corrosion of embedded steel. For high-voltage insulators, loss of material can be attributed to mechanical wear or wind-induced abrasion. Ref. 14]
Loss of material, loss of form	In earthen water-control structures, the loss of material and loss of form can result from erosion, settlement, sedimentation, frost action, waves, currents, surface runoff, and seepage.
Loss of mechanical function	Loss of mechanical function in Class 1 piping and components (such as constant and variable load spring hangers, guides, stops, sliding surfaces, and vibration isolators) fabricated from steel or other materials, such as Lubrite®, can occur through the combined influence of a number of aging mechanisms. Such aging mechanisms can include corrosion, distortion, dirt, overload, fatigue due to vibratory and cyclic thermal loads, or elastomer hardening. Clearances being less than the design requirements can also contribute to loss of mechanical function.

IX.E Selected Use of Terms for Describing and Standardizing AGING EFFECTS

Term	Usage in this document
Loss of preload	Loss of preload can be due to gasket creep, thermal effects (including differential expansion and creep or stress relaxation), and self-loosening (which includes vibration, joint flexing, cyclic shear loads, thermal cycles). [Ref. 15, 16]
Loss of prestress	Loss of prestress in structural steel anchorage components can result from relaxation, shrinkage, creep, or elevated temperatures.
Loss of sealing; leakage through containment	Loss of sealing and leakage through containment in such materials as seals, elastomers, rubber, and other similar materials can result from deterioration of seals, gaskets, and moisture barriers (caulking, flashing, and other sealants). Loss of sealing in elastomeric phase bus enclosure assemblies can result from moisture intrusion.
None	Certain material/environment combinations may not be subject to significant aging mechanisms; thus, there are no relevant aging effects that require management.
Reduction in concrete anchor capacity due to local concrete degradation	Reduction in concrete anchor capacity due to local concrete degradation can result from a service-induced cracking or other concrete aging mechanisms.
Reduction in foundation strength, cracking, differential settlement	Reduction in foundation strength, cracking, and differential settlement can result from erosion of porous concrete subfoundation.
Reduction of heat transfer	Reduction of heat transfer can result from fouling on the heat transfer surface. Although in heat exchangers the tubes are the primary heat transfer component, heat exchanger internals, including tubesheets and fins, contribute to heat transfer and may be affected by the reduction of heat transfer due to fouling. Although GALL Report, Rev. 2 does not include reduction of heat transfer for any heat exchanger surfaces other than tubes, reduction in heat transfer is of concern for other heat exchanger surfaces.

IX.E Selected Use of Terms for Describing and Standardizing AGING EFFECTS

Term	Usage in this document
Reduced insulation resistance	Reduced insulation resistance is an aging effect used exclusively in GALL Report, Rev. 2 for Chapter VI, Electrical Components and is said to result from the following aging mechanisms: • thermal/thermoxidative degradation of organics/thermoplastics, radiation-induced oxidation, moisture/debris intrusion, and ohmic heating [VI.A.LP-26] • presence of salt deposits or surface contamination [VI.A.LP-28] • thermal/thermoxidative degradation of organics, radiolysis, and photolysis (UV sensitive materials only) of organics; radiation-induced oxidation; moisture intrusion [VI.A.LP-33, VI.A.LP-34] • moisture [VI.A.LP-35]
Reduction of neutron-absorbing capacity	Reduction of neutron-absorbing capacity can result from Boraflex degradation.
Reduction of strength and modulus	In concrete, reduction of strength and modulus can be attributed to elevated temperatures (>150°F general; >200°F local).
Reduction or loss of isolation function	Reduction or loss of isolation function in polymeric vibration isolation elements can result from elastomers exposed to radiation hardening, temperature, humidity, sustained vibratory loading.
Wall thinning	Wall thinning is a specific type of loss of material attributed in the AMR line-items to general corrosion or flow-accelerated corrosion.

F. Significant Aging Mechanisms

An aging mechanism is considered to be significant when it may result in aging effects that produce a loss of functionality of a component or structure during the current or license renewal period if allowed to continue without mitigation.

The following table defines many of the standardized aging mechanisms used in the preceding GALL AMR tables in Chapters II through VIII of GALL Report, Rev. 2.

IX.F Selected Definitions & Use of Terms for Describing and Standardizing AGING MECHANISMS

Term	Definition as used in this document
Abrasion	As used in the context of GALL Chpt III, "Structures and Component Supports," as water migrates over a concrete surface, it may transport material that can abrade the concrete. The passage of water also may create a negative pressure at the water/air-to-concrete interface that can result in abrasion and cavitation degradation of the concrete. This may result in pitting or aggregate exposure due to loss of cement paste. [Ref. 17]
Aggressive chemical attack	Concrete, being highly alkaline (pH >12.5), is degraded by strong acids. Chlorides and sulfates of potassium, sodium, and magnesium may attack concrete, depending on their concentrations in soil/ground water that comes into contact with the concrete. Exposed surfaces of Class 1 structures may be subject to sulfur-based acid-rain degradation. The minimum thresholds causing concrete degradation are 500 ppm chlorides and 1500 ppm sulfates. [Ref. 17]
Boraflex degradation	Boraflex degradation may involve gamma radiation-induced shrinkage of Boraflex and the potential to develop tears or gaps in the material. A more significant potential degradation is the gradual release of silica and the depletion of boron carbide from Boraflex, following gamma irradiation and long-term exposure to the wet pool environment. The loss of boron carbide from Boraflex is characterized by slow dissolution of the Boraflex matrix from the surface of the Boraflex and a gradual thinning of the material. The boron carbide loss can result in a significant increase in the reactivity within the storage racks. An additional consideration is the potential for silica transfer through the fuel transfer canal into the reactor core during refueling operations and its effect on the fuel-clad heat transfer capability. [Ref. 18]

IX.F Selected Definitions & Use of Terms for Describing and Standardizing AGING MECHANISMS

Term	Definition as used in this document
Borated Water Intrusion	The influx of borated water.
Boric acid corrosion	Corrosion by boric acid, which can occur where there is borated water leakage in an environment described as air with borated water leakage (see Corrosion).
Cavitation	Formation and instantaneous collapse of innumerable tiny voids or cavities within a liquid subjected to rapid and intense pressure changes. Cavitation caused by severe turbulent flow can potentially lead to cavitation damage.
Chemical contamination	Presence of chemicals that do not occur under normal conditions at concentrations that could result in the degradation of the component.
Cladding breach	This refers to the various aging mechanisms breaking metallic cladding via any applicable process. Unique problems with stainless cladding have been identified for HHSI pumps as discussed in NRC Information Notice 94-63, "Boric Acid Corrosion of Charging Pump Casings Caused by Cladding Cracks." It is only used in AMR line-items in the Engineered Safety Features and Auxiliary System to describe the loss of material in PWR emergency core cooling system pump casing constructed of steel with stainless steel cladding and the PWR chemical and volume control system pump casing constructed of steel with stainless steel cladding.
Cladding degradation	This refers to the degradation of the stainless steel cladding via any applicable degradation process and is a precursor to cladding breach. It is only used to describe the loss of material due to pitting and crevice corrosion (only for steel after lining/cladding degradation) of piping, piping components, and piping elements fabricated from steel, with elastomer lining or stainless steel cladding.
Corrosion	Chemical or electrochemical reaction between a material, usually a metal, and the environment or between two dissimilar metals that produces a deterioration of the material and its properties.

IX.F Selected Definitions & Use of Terms for Describing and Standardizing AGING MECHANISMS

Term	Definition as used in this document
Corrosion of carbon steel tube support plate	Corrosion can occur on the carbon steel tube support plates, which are plate-type components providing tube-to-tube mechanical support for the tubes in the tube bundle of the steam generator (recirculating) system of a PWR. The tubes pass through drilled holes in the plate. The secondary coolant flows through the tube supports via flow holes between the tubes. [Ref. 19, 20]
Corrosion of embedded steel	If the pH of concrete in which steel is embedded is reduced below 11.5 by intrusion of aggressive ions (e.g., chlorides > 500 ppm) in the presence of oxygen, embedded steel may corrode. A reduction in pH may be caused by the leaching of alkaline products through cracks, entry of acidic materials, or carbonation. Chlorides may 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. [Ref. 21]
Creep	Creep, for a metallic material, refers to a time-dependent continuous deformation process under constant stress. It is an elevated temperature process and is not a concern for low-alloy steel below 700°F, for austenitic alloys below 1000°F, or for Ni-based alloys below 1800°F. [Ref.22, 23] Creep, in concrete, is related to the loss of absorbed water from the hydrated cement paste. It is a function of the modulus of elasticity of the aggregate. It may result in loss of prestress in the tendons used in prestressed concrete containment. [Ref. 19]
Crevice corrosion	Crevice corrosion occurs in a wetted or buried environment when a crevice or area of stagnant or low flow exists that allows a corrosive environment to develop in a component. It occurs most frequently in joints and connections, or points of contact between metals and non-metals, such as gasket surfaces, lap joints, and under bolt heads. Carbon steel, cast iron, low alloy steels, stainless steel, copper, and nickel base alloys are all susceptible to crevice corrosion. Steel can be subject to crevice corrosion in some cases after lining/cladding degradation. Localized corrosion of a metal surface at, or immediately adjacent to, an area that is shielded from full exposure to the environment because of the close proximity of the metal to the surface of another dissimilar material.

IX.F Selected Definitions & Use of Terms for Describing and Standardizing AGING MECHANISMS

Term	Definition as used in this document
Cyclic loading	One source of cyclic loading is the periodic application of pressure loads and forces due to thermal movement of piping transmitted through penetrations and structures to which penetrations are connected. The typical result of cyclic loads on metal components is fatigue cracking and failure; however, the cyclic loads also may cause changes in dimensions that result in functional failure.
Deterioration of seals, gaskets, and moisture barriers (caulking, flashing, and other sealants)	Seals, gaskets, and moisture barriers (caulking, flashing, and other sealants) are subject to loss of sealing and leakage due to containment caused by aging degradation of these components.
Distortion	The aging mechanism of distortion (as associated with component supports in GALL Chpt III.B2) can be caused by time-dependent strain or by gradual elastic and plastic deformation of metal that is under constant stress at a value lower than its normal yield strength.
Elastomer degradation	Elastomer materials are substances whose elastic properties are similar to those of natural rubber. The term elastomer is sometimes used to technically distinguish synthetic rubbers and rubber-like plastics from natural rubber. Degradation may include mechanisms such as cracking, crazing, fatigue breakdown, abrasion, chemical attacks, and weathering. [Ref. 24, 25]
Electrical transients	An electrical transient is a stressor caused by a voltage spike that can contribute to aging degradation. Certain types of high-energy electrical transients can contribute to electromechanical forces, ultimately resulting in fatigue or loosening of bolted connections. Transient voltage surges are a major contributor to the early failure of sensitive electrical components
Elevated temperature	Elevated temperature is referenced as an aging mechanism only in the context of LWR containments (GALL Chpt. II). In concrete, reduction of strength and modulus can be attributed to elevated temperatures (>150°F general; >200°F local).
Erosion	Erosion, or the progressive loss of material from a solid surface, is due to mechanical interaction between that surface and a fluid, a multicomponent fluid, or solid particles carried by the fluid.

IX.F Selected Definitions & Use of Terms for Describing and Standardizing AGING MECHANISMS

Term	Definition as used in this document
Erosion settlement	Erosion settlement is the subsidence of a containment structure that may occur due to changes in the site conditions, e.g., erosion or changes in the water table). The amount of settlement depends on the foundation material. [Ref. 21] Another synonymous term is “erosion of the porous concrete subfoundation.”
Erosion, settlement, sedimentation, frost action, waves, currents, surface runoff, seepage	In earthen water-control structures, the loss of material and loss of form can result from erosion, settlement, sedimentation, frost action, waves, currents, surface runoff, and seepage.
Fatigue	Fatigue is a phenomenon leading to fracture under repeated or fluctuating stresses having a maximum value less than the tensile strength of the material. Fatigue fractures are progressive, and grow under the action of the fluctuating stress. Fatigue due to vibratory and cyclic thermal loads is defined as the structural degradation that can occur from repeated stress/strain cycles caused by fluctuating loads (e.g., from vibratory loads) and temperatures, giving rise to thermal loads. After repeated cyclic loading of sufficient magnitude, microstructural damage may accumulate, leading to macroscopic crack initiation at the most vulnerable regions. Subsequent mechanical or thermal cyclic loading may lead to growth of the initiated crack. Vibration may result in component cyclic fatigue, as well as in cutting, wear, and abrasion, if left unabated. Vibration is generally induced by external equipment operation. It may also result from flow resonance or movement of pumps or valves in fluid systems. Crack initiation and growth resistance is governed by factors including stress range, mean stress, loading frequency, surface condition, and the presence of deleterious chemical species. [Ref. 26]
Flow-accelerated corrosion (FAC)	Flow-accelerated corrosion, also termed “erosion-corrosion,” is 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 and -R3 recommendations for an effective FAC program. [Ref. 27]

IX.F Selected Definitions & Use of Terms for Describing and Standardizing AGING MECHANISMS

Term	Definition as used in this document
Fouling	Fouling is an accumulation of deposits on the surface of a component or structure. This term includes accumulation and growth of aquatic organisms on a submerged metal surface or the accumulation of deposits (usually inorganic) on heat exchanger tubing. Biofouling, a subset of fouling, can be caused by either macro-organisms (e.g., barnacles, Asian clams, zebra mussels, and others found in fresh and salt water) or micro-organisms (e.g., algae, etc.). Fouling also can be categorized as particulate fouling from sediment, silt, dust, and corrosion products, or marine biofouling, or macrofouling (e.g., peeled coatings, debris, etc.). Fouling in a raw water system can occur on the piping, valves, and heat exchangers. Fouling can result in a reduction of heat transfer or loss of material.
Freeze-thaw, frost action	Repeated freezing and thawing can cause severe degradation of concrete, characterized by scaling, cracking, and spalling. The cause is water freezing within the pores of the concrete, creating hydraulic pressure. If unrelieved, this pressure will lead to freeze-thaw degradation. If the temperature cannot be controlled, other factors that enhance the resistance of concrete to freeze-thaw degradation are (a) adequate air content (i.e., within ranges specified in ACI 301-84), (b) low permeability, (c) protection until adequate strength has developed, and (d) surface coating applied to frequently wet-dry surfaces. [Ref. 21, 28]
Fretting	Fretting is an aging effect due to accelerated deterioration at the interface between contacting surfaces that experience a slight, differential oscillatory movement as the result of corrosion.
Galvanic corrosion	Galvanic corrosion is accelerated corrosion of a metal because of an electrical contact with a more noble metal or nonmetallic conductor in a corrosive electrolyte. It is also called bimetallic corrosion, contact corrosion, dissimilar metal corrosion, or two-metal corrosion. Galvanic corrosion is an applicable aging mechanism for steel materials coupled to more noble metals in heat exchangers; galvanic corrosion of copper is of concern when coupled with the nobler stainless steel.

IX.F Selected Definitions & Use of Terms for Describing and Standardizing AGING MECHANISMS

Term	Definition as used in this document
General corrosion	General corrosion, also known as uniform corrosion, proceeds at approximately the same rate over a metal surface. Loss of material due to general corrosion is an aging effect requiring management for low-alloy steel, carbon steel, and cast iron in outdoor environments. Some potential for pitting and crevice corrosion may exist even when pitting and crevice corrosion is not explicitly listed in the aging effects/aging mechanism column in GALL Report, Rev. 2 AMR Items and when the descriptor may only be loss of material due to general corrosion. For example, the AMP XI.M36, "External Surfaces Monitoring of Mechanical Components," calls for the inspection of general corrosion of steel through visual inspection of external surfaces for evidence of material loss and leakage. It acts as a de facto screening for pitting and crevice corrosion, since the symptoms of general corrosion will be noticed first. Wastage is thinning of component walls due to general corrosion.
Intergranular attack (IGA)	In austenitic stainless steels, the precipitation of chromium carbides, usually at grain boundaries, on exposure to temperatures of about 550-850°C, leaves the grain boundaries depleted of Cr and, therefore, susceptible to preferential attack (intergranular attack) by a corroding (oxidizing) medium.
Intergranular stress corrosion cracking (IGSCC)	IGSCC is SCC in which the cracking occurs along grain boundaries.
Irradiation-assisted stress corrosion cracking (IASCC)	Failure by intergranular cracking in aqueous environments of stressed materials exposed to ionizing radiation has been termed irradiation-assisted stress corrosion cracking (IASCC). Irradiation by high-energy neutrons can promote SCC by affecting material microchemistry (e.g., radiation-induced segregation of elements such as P, S, Si, and Ni to the grain boundaries), material composition and microstructure (e.g., radiation hardening), as well as water chemistry (e.g., radiolysis of the reactor water to make it more aggressive).

IX.F Selected Definitions & Use of Terms for Describing and Standardizing AGING MECHANISMS

Term	Definition as used in this document
Leaching of calcium hydroxide and carbonation	Water passing through cracks, inadequately prepared construction joints, or areas that are not sufficiently consolidated during placing may dissolve some calcium-containing products (of which calcium hydroxide is the most-readily soluble, depending on the solution pH) in concrete. Once the calcium hydroxide has been leached away, other cementatious constituents become vulnerable to chemical decomposition, finally leaving only the silica and alumina gels behind with little strength. The water's aggressiveness in the leaching of calcium hydroxide depends on its salt content, pH, and temperature. This leaching action is effective only if the water passes through the concrete. [Ref. 21]
Low-temperature crack propagation	Low-temperature crack propagation (LTCP) is IGSCC at low temperatures (~130-170^{o}F).
Mechanical loading	Applied loads of mechanical origins rather than from other sources, such as thermal.
Mechanical wear	See “Wear.”
Microbiologically-influenced corrosion (MIC)	Any of the various forms of corrosion influenced by the presence and activities of such microorganisms as bacteria, fungi, and algae, and/or the products produced in their metabolism. Degradation of material that is accelerated due to conditions under a biofilm or microfouling tubercle, for example, anaerobic bacteria that can set up an electrochemical galvanic reaction or inactivate a passive protective film, or acid-producing bacterial that might produce corrosive metabolites.
Moisture intrusion	Influx of moisture through any viable process.
Neutron irradiation embrittlement	Irradiation by neutrons results in embrittlement of carbon and low-alloy steels. It may produce changes in mechanical properties by increasing tensile and yield strengths with a corresponding decrease in fracture toughness and ductility. The extent of embrittlement depends on neutron fluence, temperature, and trace material chemistry. [Ref. 23]

IX.F Selected Definitions & Use of Terms for Describing and Standardizing AGING MECHANISMS

Term	Definition as used in this document
Ohmic heating	Ohmic heating is induced by current flow through a conductor and can be calculated using first principles of electricity and heat transfer. Ohmic heating is a thermal stressor and can be induced by conductors passing through electrical penetrations, for example. Ohmic heating is especially significant for power circuit penetrations. [Ref. 14]
Outer diameter stress corrosion cracking (ODSCC)	ODSCC is SCC initiating in the outer diameter (secondary side) surface of steam generator tubes. The secondary side is part of the secondary system consisting of the shell side of the steam generator, high- and low-pressure turbines, moisture/separator reheaters, main electrical stages and interconnecting piping. This differs from PWSCC, which describes inner diameter (SG primary side) initiated cracking. [Ref. 20] The primary loop basically consists of the reactor vessel, reactor coolant pumps, pressurizer steam generator tubes, and interconnecting piping.
Overload	Overload is one of the aging mechanisms that can cause loss of mechanical function in Class 1 piping and components, such as constant and variable load spring hangers, guides, stops, sliding surfaces, design clearances, and vibration isolators, fabricated from steel or other materials, such as Lubrite®.
Oxidation	Oxidation involves two types of reactions: (a) an increase in valence resulting from a loss of electrons, or (b) a corrosion reaction in which the corroded metal forms an oxide. [Ref. 24]
Photolysis	Chemical reactions induced or assisted by light
Pitting corrosion	Localized corrosion of a metal surface, confined to a point or small area, which takes the form of cavities called pits
Plastic deformation	Time-dependent strain, or gradual elastic and plastic deformation, of metal that is under constant stress at a value lower than its normal yield strength

IX.F Selected Definitions & Use of Terms for Describing and Standardizing AGING MECHANISMS

Term	Definition as used in this document
Presence of any salt deposits	The surface contamination (and increased electrical conductivity) resulting from the aggressive environment associated with the presence of salt deposits can degrade high voltage insulator quality. Although this aging mechanism may be due to temporary, transient environmental conditions, the net result may be long-lasting and cumulative for plants located in the vicinity of saltwater bodies.
Primary water stress corrosion cracking (PWSCC)	PWSCC is an intergranular cracking mechanism that requires the presence of high applied and/or residual stress, susceptible tubing microstructures (few intergranular carbides), and also high temperatures. This aging mechanism is most likely a factor for nickel alloys in the PWR environment. [Ref. 19]
Radiation hardening, temperature, humidity, sustained vibratory loading	Reduction or loss of isolation function in polymeric vibration isolation elements can result from a combination of radiation hardening, temperature, humidity, and sustained vibratory loading.
Radiation-induced oxidation	Two types of reactions that are affected by radiation are (a) an increase in valence resulting from a loss of electrons, or (b) a corrosion reaction in which the corroded metal forms an oxide. This is a very limited form of oxidation and is referenced in GALL Chpt. VI for MEB insulation. [Ref. 24]
Radiolysis	Radiolysis is a chemical reaction induced or assisted by radiation. Radiolysis and photolysis aging mechanisms can occur in UV-sensitive organic materials.
Reaction with aggregate	The presence of reactive alkalis in concrete can lead to subsequent reactions with aggregates that may be present. These alkalis are introduced mainly by cement, but also may come from admixtures, salt-contamination, seawater penetration, or solutions of deicing salts. These reactions include alkali-silica reactions, cement-aggregate reactions, and aggregate-carbonate reactions. These reactions may lead to expansion and cracking. [Ref. 11, 29]
Restraint shrinkage	Restraint shrinkage can cause cracking in concrete transverse to the longitudinal construction joint.

IX.F Selected Definitions & Use of Terms for Describing and Standardizing AGING MECHANISMS

Term	Definition as used in this document
Selective leaching	Selective leaching is also known as dealloying (e.g., dezincification or graphitic corrosion) and involves selective corrosion of one or more components of a solid solution alloy.
Service-induced cracking or other concrete aging mechanisms	Cracking of concrete under load over time of service (e.g., from shrinkage or creep, or other concrete aging mechanisms) that may include freeze-thaw, leaching, aggressive chemicals, reaction with aggregates, corrosion of embedded steels, elevated temperatures, irradiation, abrasion, and cavitation [Ref. 17]
Settlement	This term is referenced as an aging mechanism in GALL Chpt. II, *Containment Structures*. Settlement of a containment structure may occur due to changes in the site conditions (e.g., water table, etc.). The amount of settlement depends on the foundation material. [Ref. 20]
Stress corrosion cracking (SCC)	SCC is the cracking of a metal produced by the combined action of corrosion and tensile stress (applied or residual), especially at elevated temperature. SCC is highly chemically specific in that certain alloys are likely to undergo SCC only when exposed to a small number of chemical environments. For PWR internal components, in Chapters IV.B2, IV.B3 and IV.B4, SCC includes intergranular stress corrosion cracking, transgranular stress corrosion cracking, primary water stress corrosion cracking, and low temperature crack propagation as aging mechanisms.
Stress relaxation	Many of the bolts in reactor internals are stressed to a cold initial preload. When subject to high operating temperatures, over time these bolts may loosen and the preload may be lost. Radiation can also cause stress relaxation in highly stressed members such as bolts. [Ref. 15] Relaxation in structural steel anchorage components can be an aging mechanism contributing to the aging effect of loss of prestress.
Surface contamination	Contamination of the surfaces by corrosive constituents or fouling.
Sustained vibratory loading	Vibratory loading over time

IX.F Selected Definitions & Use of Terms for Describing and Standardizing AGING MECHANISMS

Term	Definition as used in this document
Thermal aging embrittlement	Also termed “thermal aging” or “thermal embrittlement.” At operating temperatures of 500 to 650°F, cast austenitic stainless steels (CASS) exhibit a spinoidal decomposition of the ferrite phase into ferrite-rich and chromium-rich phases. This may give rise to significant embrittlement (reduction in fracture toughness), depending on the amount, morphology, and distribution of the ferrite phase and the composition of the steel. Thermal aging of materials other than CASS is a time- and temperature-dependent degradation mechanism that decreases material toughness. It includes temper embrittlement and strain aging embrittlement. Ferritic and low-alloy steels are subject to both of these types of embrittlement, but wrought stainless steel is not affected by either of these processes. [Ref. 23]
Thermal effects, gasket creep, and self-loosening	Loss of preload due to gasket creep, thermal effects (including differential expansion and creep or stress relaxation), and self-loosening (which includes vibration, joint flexing, cyclic shear loads, thermal cycles) [Ref. 15, 16]
Thermal and mechanical loading	Loads (stress) due to mechanical or thermal (temperature) sources
Thermal degradation of organic materials	Organic materials, in this case, are polymers. This category includes both short-term thermal degradation and long-term thermal degradation. Thermal energy absorbed by polymers can result in crosslinking and chain scission. Crosslinking will generally result in such aging effects as increased tensile strength and hardening of material, with some loss of flexibility and eventual decrease in elongation-at-break and increased compression set. Scission generally reduces tensile strength. Other reactions that may occur include crystallization and chain depolymerization.

IX.F Selected Definitions & Use of Terms for Describing and Standardizing AGING MECHANISMS

Term	Definition as used in this document
Thermal fatigue	Fatigue is the progressive and localized structural damage that occurs when a material is subjected to cyclic loading. The maximum stress values are less than the ultimate tensile stress limit, and may be below the yield stress limit of the material. Higher temperatures generally decrease fatigue strength. Thermal fatigue can result from phenomena such as thermal loading, thermal cycling, where there is cycling of the thermal loads, and thermal stratification and turbulent penetration. Thermal stratification is a thermo-hydraulic condition with a definitive hot and cold water boundary inducing thermal fatigue of the piping. Turbulent penetration is a thermo-hydraulic condition where hot and cold water mix as a result of turbulent flow conditions, leading to thermal fatigue of the piping. The GALL AMP XI.M32, “One-Time Inspection,” inspects for cracking induced by thermal stratification, and for turbulent penetration via volumetric (RT or UT) techniques.
Thermoxidative degradation of organics/thermoplastics	Degradation of organics/thermoplastics via oxidation reactions (loss of electrons by a constituent of a chemical reaction) and thermal means (see Thermal degradation of organic materials). [Ref. 22]
Transgranular stress corrosion cracking	Transgranular stress corrosion cracking (TGSCC) is stress corrosion cracking in which cracking occurs across the grains
Void swelling	Vacancies created in reactor (metallic) materials as a result of irradiation may accumulate into voids that may, in turn, lead to changes in dimensions (swelling) of the material. Void swelling may occur after an extended incubation period.
Water trees	Water trees occur when the insulating materials are exposed to long-term electrical stress and moisture; these trees eventually result in breakdown of the dielectric and ultimate failure. The growth and propagation of water trees is somewhat unpredictable. Water treeing is a degradation and long-term failure phenomenon.

IX.F Selected Definitions & Use of Terms for Describing and Standardizing AGING MECHANISMS

Term	Definition as used in this document
Wear	Wear is defined as the removal of surface layers due to relative motion between two surfaces or under the influence of hard, abrasive particles. Wear occurs in parts that experience intermittent relative motion, frequent manipulation, or in clamped joints where relative motion is not intended, but may occur due to a loss of the clamping force. [Ref. 23]
Weathering	Weathering is the mechanical or chemical degradation of external surfaces of materials when exposed to an outside environment.
Wind-induced abrasion	(See Abrasion) The fluid carrier of abrading particles is wind rather than water/liquids.

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CHAPTER X

TIME-LIMITED AGING ANALYSES EVALUATION OF AGING MANAGEMENT PROGRAMS UNDER 10 CFR 54.21(C)(1)(iii)

TIME-LIMITED AGING ANALYSES (TLAAs)

X.M1 Fatigue Monitoring
X.S1 Concrete Containment Tendon Prestress
X.E1 Environmental Qualification (EQ) of Electric Components

X.M1 FATIGUE MONITORING

Program Description

Fatigue usage factor is a computed mechanical parameter suitable for gauging fatigue damage in components subjected to fluctuating stresses. Crack initiation is assumed to have started in a structural component when the fatigue usage factor at a point of the component reaches the value of 1, the design limit on fatigue. In order not to exceed the design limit on fatigue usage, the aging management program (AMP) monitors and tracks the number of critical thermal and pressure transients for the selected components. The program also verifies that the severity of the monitored transients are bounded by the design transient definition for which they are classified.

The AMP addresses the effects of the reactor coolant environment on component fatigue life (to determine an environmentally-adjusted cumulative usage factor, or CUF_{en}) by assessing the impact of the reactor coolant environment on a set of sample critical components for the plant. Examples of critical components are identified in NUREG/CR-6260. Environmental effects on fatigue for these critical components may be evaluated using one of the following sets of formulae:

- <u>Carbon and Low Alloy Steels</u>
    - Those provided in NUREG/CR-6583, using the applicable ASME Section III fatigue design curve
    - Those provided in Appendix A of NUREG/CR-6909, using either the applicable ASME Section III fatigue design curve or the fatigue design curve for carbon and low alloy steel provided in NUREG/CR-6909 (Figures A.1 and A.2, respectively, and Table A.1)
    - A staff approved alternative

- <u>Austenitic Stainless Steels</u>
    - Those provided in NUREG/CR-5704, using the applicable ASME Section III fatigue design curve
    - Those provided in NUREG/CR-6909, using the fatigue design curve for austenitic stainless steel provided in NUREG/CR-6909 (Figure A.3 and Table A.2)
    - A staff approved alternative

- <u>Nickel Alloys</u>
    - Those provided in NUREG/CR-6909, using the fatigue design curve for austenitic stainless steel provided in NUREG/CR-6909 (Figure A.3 and Table A.2)
    - A staff approved alternative

Any one option may be used for calculating the CUF_{en} for each material.

Evaluation and Technical Basis

1. ***Scope of Program:*** The scope includes those components that have been identified to have a fatigue TLAA. The program monitors and tracks the number of critical thermal and pressure transients for the selected components. The program ensures the fatigue usage remaining within the allowable limit, thus minimizing fatigue cracking of metal components caused by anticipated cyclic strains in the material.

   For purposes of monitoring and tracking, applicants should include, for a set of sample reactor coolant system components, fatigue usage calculations that consider the effects of the reactor water environment. This sample set should include the locations identified in NUREG/CR-6260 and additional plant-specific component locations in the reactor coolant pressure boundary if they may be more limiting than those considered in NUREG/CR-6260.

2. ***Preventive Actions:*** The program prevents the fatigue TLAAs from becoming invalid by assuring that the fatigue usage resulting from actual operational transients does not exceed the Code design limit of 1.0, including environmental effects where applicable. This could be caused by the numbers of actual plant transients exceeding the numbers used in the fatigue analyses or by the actual transient severity exceeding the bounds of the design transient definitions. However, in either of these cases, if the analysis is revised to account for the increased number or severity of transients such that the CUF value remains below 1.0, the program remains effective.

3. ***Parameters Monitored/Inspected:*** The program monitors all plant design transients that cause cyclic strains, which are significant contributors to the fatigue usage factor. The number of occurrences of the plant transients that cause significant fatigue usage for each component is to be monitored. Alternatively, more detailed monitoring of local pressure and thermal conditions may be performed to allow the actual fatigue usage for the specified critical locations to be calculated.

4. ***Detection of Aging Effects:*** The program provides for updates of the fatigue usage calculations on an as-needed basis if an allowable cycle limit is approached, or in a case where a transient definition has been changed, unanticipated new thermal events are discovered, or the geometry of components have been modified.

5. ***Monitoring and Trending:*** Trending is assessed to ensure that the fatigue usage factor remains below the design limit during the period of extended operation, thus minimizing fatigue cracking of metal components caused by anticipated cyclic strains in the material.

6. ***Acceptance Criteria:*** The acceptance criterion is maintaining the cumulative fatigue usage below the design limit through the period of extended operation, with consideration of the reactor water environmental fatigue effects described in the program description and scope of program.

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 the period of extended operation. Scope expansion includes consideration of other locations with the highest expected cumulative usage factors when considering environmental effects. As discussed in the Appendix for GALL, the staff

finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

8. ***Confirmation 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process and administrative controls.

9. ***Administrative Controls:*** As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the administrative controls.

10. ***Operating Experience:*** The program reviews industry experience relevant to fatigue cracking. Applicable operating experience relevant to fatigue cracking is to be considered in selecting the locations for monitoring. As discussed in NRC Regulatory Issue Summary 2008-30, the use of certain simplified analysis methodology to demonstrate compliance with the ASME Code fatigue acceptance criteria could be nonconservative; therefore, a confirmatory analysis is recommended.

References

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NUREG/CR-6909, *Effects of LWR Coolant Environments on the Fatigue Life of Reactor Materials,* U.S. Nuclear Regulatory Commission, February 2007.

X.S1 CONCRETE CONTAINMENT TENDON PRESTRESS

Program Description

This aging management program provides reasonable assurance of the adequacy of prestressing forces in prestressed concrete containment tendons during the period of extended operation under 10 CFR 54.21(c)(1)(iii). The program consists of an assessment of inspections performed in accordance with the requirements of Subsection IWL of the American Society of Mechanical Engineers (ASME) Code, Section XI, as supplemented by the requirements of 10 CFR 50.55a(b)(2)(viii). The assessment related to the adequacy of the prestressing force establishes (a) acceptance criteria in accordance with U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.35.1 and (b) trend lines based on the guidance provided in NRC Information Notice (IN) 99-10.

As evaluated below, this time-limited aging analysis (TLAA) is an acceptable option to manage containment tendon prestress forces. However, it is recommended that the staff further evaluate an applicant's operating experience related to the containment tendon prestress force. Programs related to the adequacy of prestressing force for containments with grouted tendons are reviewed on a case-by-case basis.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program addresses the assessment of containment tendon prestressing force when an applicant performs the containment prestress force TLAA using 10 CFR 54.21(c)(1)(iii).

2. ***Preventive Actions:*** Maintaining the prestress above the minimum required value (MRV), as described under the acceptance criteria below, ensures that the structural and functional adequacy of the containment are maintained.

3. ***Parameters Monitored:*** The parameters monitored are the containment tendon prestressing forces in accordance with requirements specified in Subsection IWL of Section XI of the ASME Code, as incorporated by reference in 10 CFR 50.55a.

4. ***Detection of Aging Effects:*** The loss of containment tendon prestressing forces is detected by the program.

5. ***Monitoring and Trending:*** The estimated and measured prestressing forces are plotted against time, and the predicted lower limit (PLL), MRV, and trending lines are developed for the period of extended operation. NRC RG 1.35.1 provides guidance for calculating PLL and MRV. The trend line represents the trend of prestressing forces based on the actual measured forces. NRC IN 99-10 provides guidance for constructing the trend line.

6. ***Acceptance Criteria:*** The prestressing force trend lines indicate that existing prestressing forces in the containment tendon would not be below the MRVs prior to the next scheduled inspection, as required by 10 CFR 50.55a(b)(2)(viii)(B). The acceptance criteria normally consists of PLL and the minimum required prestressing force, also called MRV. The goal is to keep the trend line above the PLL because, as a result of any inspection performed in accordance with ASME Section XI, Subsection IWL, if the trend line crosses the PLL, the existing prestress in the containment tendon could go below the MRV soon after the inspection and would not meet the requirements of 10 CFR 50.55a(b)(2)(viii)(B).

7. ***Corrective Actions:*** If acceptance criteria are not met, then either systematic retensioning of tendons or a reanalysis of the containment is warranted to ensure the design adequacy of the containment. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

8. ***Confirmation Process:*** The confirmation process ensures that preventive actions are adequate and that appropriate corrective actions have been completed and are effective. The confirmation process for this program is implemented through the site's quality assurance (QA) program in accordance with the requirements of 10 CFR Part 50, Appendix B.

9. ***Administrative Controls:*** The administrative controls for this program provide for a formal review and approval of corrective actions. The administrative controls for this program are implemented through the site's QA program in accordance with the requirements of 10 CFR Part 50, Appendix B.

10. ***Operating Experience:*** The program incorporates the relevant operating experience that has occurred at the applicant's plant as well as at other plants. The applicable portions of the experience with prestressing systems described in NRC IN 99-10 could be useful. Additional industry operating experience has been documented in NUREG/CR-4652 and in the May/June 1994 *Concrete International* publication by H. Ashar, C. P. Tan, and D. Naus. However, tendon operating experience may be different at plants with prestressed concrete containments. The difference could be due to the prestressing system design (e.g., button-headed, wedge, or swaged anchorages), environment, and type of reactor (i.e., pressurized water reactor and boiling water reactor). Thus, the applicant's plant-specific operating experience should be further evaluated for license renewal.

References

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X.E1 ENVIRONMENTAL QUALIFICATION (EQ) OF ELECTRIC COMPONENTS

Program Description

The Nuclear Regulatory Commission (NRC) has established nuclear station environmental qualification (EQ) requirements in 10 CFR Part 50, Appendix A, Criterion 4, and 10 CFR 50.49. 10 CFR 50.49 specifically requires that an EQ program be established to demonstrate that certain electrical components located in harsh plant environments (that is, those areas of the plant that could be subject to the harsh environmental effects of a loss of coolant accident (LOCA), high energy line breaks, or post-LOCA environment) are qualified to perform their safety function in those harsh environments after the effects of inservice aging. 10 CFR 50.49 requires that the effects of significant aging mechanisms be addressed as part of environmental qualification.

All operating plants shall meet the requirements of 10 CFR 50.49 for certain electrical components important to safety. 10 CFR 50.49 defines the scope of components to be included, requires the preparation and maintenance of a list of in-scope components, and requires the preparation and maintenance of a qualification file that includes component performance specifications, electrical characteristics, and the environmental conditions to which the components could be subjected. 10 CFR 50.49(e)(5) contains provisions for aging that require, in part, consideration of all significant types of aging degradation that can affect component functional capability. 10 CFR 50.49(e)(5) also requires replacement or refurbishment of components not qualified for the current license term prior to the end of designated life, unless additional life is established through ongoing qualification. 10 CFR 50.49(f) establishes four methods of demonstrating qualification for aging and accident conditions. 10 CFR 50.49(k) and (i) permit different qualification criteria to apply based on plant and component vintage. Supplemental EQ regulatory guidance for compliance with these different qualification criteria is provided in the Division of Operating Reactors (DOR) Guidelines; Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors; NUREG-0588, “Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment”; and Regulatory Guide 1.89, Rev. 1, “Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants.” Compliance with 10 CFR 50.49 provides reasonable assurance that the component can perform its intended functions during accident conditions after experiencing the effects of inservice aging.

EQ programs manage component thermal, radiation, and cyclical aging through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods. As required by 10 CFR 50.49, EQ components not qualified for the current license term are refurbished, replaced, or have their qualification extended prior to reaching the aging limits established in the evaluation. Aging evaluations for EQ components that specify a qualification of at least 40 years are considered time-limited aging analyses (TLAAs) for license renewal.

Under 10 CFR 54.21(c)(1)(iii), plant EQ programs, which implement the requirements of 10 CFR 50.49 (as further defined and clarified by the DOR Guidelines, NUREG-0588, and Regulatory Guide 1.89, Rev. 1), are viewed as aging management programs (AMPs) for license renewal. Reanalysis of an aging evaluation to extend the qualification of components under 10 CFR 50.49(e) is performed on a routine basis as part of an EQ program. Important attributes for the reanalysis of an aging evaluation include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria, and corrective actions (if acceptance criteria are not met). These attributes are discussed in the "EQ Component Reanalysis Attributes" section.

This reanalysis program can be applied to EQ components now qualified for the current operating term (i.e., those components now qualified for 40 years or more). As evaluated below, this is an acceptable AMP. Thus, no further evaluation is recommended for license renewal if an applicant elects this option under 10 CFR 54.21(c)(1)(iii) to evaluate the TLAA of EQ of electric equipment. The reanalysis showing the 60-year qualification is established prior to the plant entering the period of extended operation. As defined in 10 CFR 50.49(j), a record of the qualification must be maintained in an auditable form for the entire period of extended operation during which the covered item is installed in the nuclear power plant or is stored for future use; this permits verification that each item of electric equipment important to safety covered by this section (a) is qualified for its application and (b) meets its specified performance requirements when it is subjected to the conditions predicted to be present when it must perform a safety function up to the end of qualified life.

EQ Component Reanalysis Attributes

The reanalysis of an aging evaluation is normally performed to extend the qualification by reducing excess conservatism incorporated in the prior evaluation. Reanalysis of an aging evaluation to extend the qualification of a component is performed on a routine basis pursuant to 10 CFR 50.49(e) as part of an EQ program. While a component life limiting condition may be due to thermal, radiation, or cyclical aging, the vast majority of component aging limits are based on thermal conditions. Conservatism may exist in aging evaluation parameters, such as the assumed ambient temperature of the component, an unrealistically low activation energy, or in the application of a component (de-energized versus energized). The reanalysis of an aging evaluation is documented according to the station's quality assurance program requirements, which requires the verification of assumptions and conclusions. As already noted, important attributes of a reanalysis include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria, and corrective actions (if acceptance criteria are not met). These attributes are discussed below.

Analytical Methods: The analytical models used in the reanalysis of an aging evaluation are the same as those previously applied during the prior evaluation. The Arrhenius methodology is an acceptable thermal model for performing a thermal aging evaluation. The analytical method used for a radiation aging evaluation is to demonstrate qualification for the total integrated dose (that is, normal radiation dose for the projected installed life plus accident radiation dose). For license renewal, one acceptable method of establishing the 60-year normal radiation dose is to multiply the 40-year normal radiation dose by 1.5 (that is, 60 years/40 years). The result is added to the accident radiation dose to obtain the total integrated dose for the component. For cyclical aging, a similar approach may be used. Other models may be justified on a case-by-case basis.

Data Collection and Reduction Methods: Reducing excess conservatism in the component service conditions (for example, temperature, radiation, cycles) used in the prior aging evaluation is the chief method used for a reanalysis. Temperature data used in an aging evaluation is conservative and based on plant design temperatures or on actual plant temperature data. When used, plant temperature data can be obtained in several ways, including monitors used for technical specification compliance, other installed monitors, measurements made by plant operators during rounds, and temperature sensors on large motors (while the motor is not running). A representative number of temperature measurements are conservatively evaluated to establish the temperatures used in an aging evaluation. Plant temperature data may be used in an aging evaluation in different ways, such as (a) directly applying the plant temperature data in the evaluation, or (b) using the plant temperature data to demonstrate conservatism when using plant design temperatures for an evaluation. Any changes to material activation energy values as part of a reanalysis are justified on a plant-specific basis. Similar methods of reducing excess conservatism in the component service conditions used in prior aging evaluations can be used for radiation and cyclical aging.

Underlying Assumptions: EQ component aging evaluations contain sufficient conservatism to account for most environmental changes occurring due to plant modifications and events. When unexpected adverse conditions are identified during operational or maintenance activities that affect the normal operating environment of a qualified component, the affected EQ component is evaluated and appropriate corrective actions are taken, which may include changes to the qualification bases and conclusions.

Acceptance Criteria and Corrective Actions: The reanalysis of an aging evaluation could extend the qualification of the component. If the qualification cannot be extended by reanalysis, the component is refurbished, replaced, or requalified prior to exceeding the period for which the current qualification remains valid. A reanalysis is performed in a timely manner (that is, sufficient time is available to refurbish, replace, or requalify the component if the reanalysis is unsuccessful).

Evaluation and Technical Basis

1. ***Scope of Program:*** EQ programs apply to certain electrical components that are important to safety and could be exposed to harsh environment accident conditions, as defined in 10 CFR 50.49 and Regulatory Guide 1.89, Rev.1.

2. ***Preventive Actions:*** 10 CFR 50.49 does not require actions that prevent aging effects. EQ program actions that could be viewed as preventive actions include (a) establishing the component service condition tolerance and aging limits (for example, qualified life or condition limit) and (b) where applicable, requiring specific installation, inspection, monitoring, or periodic maintenance actions to maintain component aging effects within the bounds of the qualification basis.

3. ***Parameters Monitored/Inspected:*** EQ component qualified life is not based on condition or performance monitoring. However, pursuant to Regulatory Guide 1.89, Rev. 1, such monitoring programs are an acceptable basis to modify a qualified life through reanalysis. Monitoring or inspection of certain environmental conditions or component parameters may be used to ensure that the component is within the bounds of its qualification basis, or as a means to modify the qualified life.

4. ***Detection of Aging Effects:*** 10 CFR 50.49 does not require the detection of aging effects for in-service components. Monitoring or inspection of certain environmental conditions or component parameters may be used to ensure that the component is within the bounds of its qualification basis, or as a means to modify the qualified life.

5. ***Monitoring and Trending:*** 10 CFR 50.49 does not require monitoring and trending of component condition or performance parameters of in-service components to manage the effects of aging. EQ program actions that could be viewed as monitoring include monitoring how long qualified components have been installed. Monitoring or inspection of certain environmental, condition, or component parameters may be used to ensure that a

component is within the bounds of its qualification basis, or as a means to modify the qualification.

6. ***Acceptance Criteria:*** 10 CFR 50.49 acceptance criteria are that an inservice EQ component is maintained within the bounds of its qualification basis, including (a) its established qualified life and (b) continued qualification for the projected accident conditions. 10 CFR 50.49 requires refurbishment, replacement, or requalification prior to exceeding the qualified life of each installed device. When monitoring is used to modify a component qualified life, plant-specific acceptance criteria are established based on applicable 10 CFR 50.49(f) qualification methods.

7. ***Corrective Actions:*** If an EQ component is found to be outside the bounds of its qualification basis, corrective actions are implemented in accordance with the station's corrective action program. When unexpected adverse conditions are identified during operational or maintenance activities that affect the environment of a qualified component, the affected EQ component is evaluated and appropriate corrective actions are taken, which may include changes to the qualification bases and conclusions. When an emerging industry aging issue is identified that affects the qualification of an EQ component, the affected component is evaluated and appropriate corrective actions are taken, which may include changes to the qualification bases and conclusions. Confirmatory actions, as needed, are implemented as part of the station's corrective action program, pursuant to 10 CFR 50, Appendix B. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

8. ***Confirmation Process:*** Confirmatory actions, as needed, are implemented as part of the station's corrective action program, pursuant to 10 CFR 50, Appendix B. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process.

9. ***Administrative Controls:*** EQ programs are implemented through the use of station policy, directives, and procedures. EQ programs continue to comply with 10 CFR 50.49 throughout the renewal period, including development and maintenance of qualification documentation demonstrating reasonable assurance that a component can perform required functions during harsh accident conditions. EQ program documents identify the applicable environmental conditions for the component locations. EQ program qualification files are maintained at the plant site in an auditable form for the duration of the installed life of the component. EQ program documentation is controlled under the station's quality assurance program. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the administrative controls.

10. ***Operating Experience:*** EQ programs include consideration of operating experience to modify qualification bases and conclusions, including qualified life. Compliance with 10 CFR 50.49 provides reasonable assurance that components can perform their intended functions during accident conditions after experiencing the effects of inservice aging.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

10 CFR 50.49, *Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

10 CFR 54.21, *Contents of Application—Technical Information*, Office of the Federal Register, National Archives and Records Administration, 2009.

DOR Guidelines, *Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors,* November 1979.

NRC Regulatory Guide 1.89, Rev. 1, *Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants,* U. S. Nuclear Regulatory Commission, June 1984.

NRC Regulatory Issue Summary 2003-09, *Environmental Qualification of Low-Voltage Instrumentation and Control Cables*, May 2, 2003.

NUREG-0588, *Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment,* U. S. Nuclear Regulatory Commission, July 1981.

CHAPTER XI

AGING MANAGEMENT PROGRAMS (AMPS)

Guidance on Use of Later Editions/Revisions of Various Industry Documents

XI.M1	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD
XI.M2	Water Chemistry
XI.M3	Reactor Head Closure Stud Bolting
XI.M4	BWR Vessel ID Attachment Welds
XI.M5	BWR Feedwater Nozzle
XI.M6	BWR Control Rod Drive Return Line Nozzle
XI.M7	BWR Stress Corrosion Cracking
XI.M8	BWR Penetrations
XI.M9	BWR Vessel Internals
XI.M10	Boric Acid Corrosion
XI.M11B	Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in Reactor Coolant Pressure Boundary Components (PWRs only)
XI.M12	Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)
XI.M16 A	PWR Vessel Internals
XI.M17	Flow-Accelerated Corrosion
XI.M18	Bolting Integrity
XI.M19	Steam Generators
XI.M20	Open-Cycle Cooling Water System
XI.M21A	Closed Treated Water Systems
XI.M22	Boraflex Monitoring
XI.M23	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems
XI.M24	Compressed Air Monitoring
XI.M25	BWR Reactor Water Cleanup System
XI.M26	Fire Protection
XI.M27	Fire Water System
XI.M29	Aboveground Metallic Tanks
XI.M30	Fuel Oil Chemistry
XI.M31	Reactor Vessel Surveillance
XI.M32	One-Time Inspection
XI.M33	Selective Leaching
XI.M35	One-Time Inspection of ASME Code Class 1 Small Bore-Piping
XI.M36	External Surfaces Monitoring of Mechanical Components
XI.M37	Flux Thimble Tube Inspection
XI.M38	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components
XI.M39	Lubricating Oil Analysis
XI.M40	Monitoring of Neutron-Absorbing Materials Other than Boraflex
XI.M41	Buried and Underground Piping and Tanks
XI.S1	ASME Section XI, Subsection IWE
XI.S2	ASME Section XI, Subsection IWL
XI.S3	ASME Section XI, Subsection IWF

XI.S4	10 CFR 50, Appendix J
XI.S5	Masonry Walls
XI.S6	Structures Monitoring
XI.S7	RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants
XI.S8	Protective Coating Monitoring and Maintenance Program
XI.E1	Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements
XI.E2	Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits
XI.E3	Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements
XI.E4	Metal-Enclosed Bus
XI.E5	Fuse Holders
XI.E6	Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

GUIDANCE ON USE OF LATER EDITIONS/REVISIONS OF VARIOUS INDUSTRY DOCUMENTS

To aid applicants in the development of their license renewal applications, the staff has developed a list of aging management programs (AMPs) in the GALL Report that are based entirely or in part on specific editions/revisions of various industry codes (other than the ASME Code), standards, and other industry-generated guidance documents. License renewal applicants may use later editions/revisions of these industry generated documents, subject to the following provisions:

(i) If the later edition/revision has been explicitly reviewed and approved/endorsed by the NRC staff for license renewal via an NRC Regulatory Guide endorsement, a safety evaluation for generic use (such as for a BWRVIP), incorporation into 10 CFR, or a license renewal interim staff guidance.

(ii) If the later edition/revision has been explicitly reviewed and approved on a plant-specific basis by the NRC staff in their safety evaluation report for another applicant's license renewal application (a precedent exists). Applicants may reference this and justify applicability to their facility via the exception process in NEI 95-10.

If either of these methods is used as justification for adopting a later edition/revision than specified in the GALL Report, the applicant shall make available for the staff's review the information pertaining to the NRC endorsement/approval of the later edition/revision.

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

Program Description

Title 10 of 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 of these components is covered in Subsections IWB, IWC, and IWD, respectively, in the 2004 edition.[1] 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. Repair/replacement activities for these components are covered in Subsection IWA of the ASME code.

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. 10 CFR 50.55a imposes additional limitations, modifications, and augmentations of ISI requirements specified in ASME Code, Section XI, and those limitations, modifications, or augmentations described in 10 CFR 50.55a are included as part of this program. 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 Generic Aging Lessons Learned (GALL) Report.

Evaluation and Technical Basis

1. ***Scope of Program:*** The ASME Section XI program provides the requirements for ISI, repair, and replacement of code Class 1, 2, or 3 pressure-retaining components and their integral attachments in light-water cooled nuclear power plants. The components within the scope of the program are specified in ASME Code, Section XI, Subsections IWB-1100, IWC-1100, and IWD-1100 for Class 1, 2, and 3 components, respectively. The components described in Subsections IWB-1220, IWC-1220, and IWD-1220 are exempt from the volumetric and surface examination requirements, but not exempt from visual exam requirements of Subsections IWB-2500, IWC-2500, and IWD-2500.

2. ***Preventive Actions:*** This is a condition monitoring program. It does not implement preventive actions.

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.

   The program uses three types of examination—visual, surface, and volumetric—in accordance with the 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

[1] Refer to the GALL Report, Chapter I, for applicability of other editions of the ASME Code, Section XI.

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.

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 aging effects are discovered and repaired before the loss of intended function of the component. Inspection can reveal cracking, loss of material due to corrosion, leakage of coolant, and indications of degradation due to wear or stress relaxation (such as changes in 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. For boiling water reactors (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 BWR, are included in the approved Boiling Water Reactor Vessel and Internals Project Report (BWRVIP-03).

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-3000 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-2410 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 for Class 1, 2, or, 3 components, respectively.

6. ***Acceptance Criteria:*** Any indication or relevant conditions of degradation 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 by comparing the results with the acceptance standards of IWB-3400 and IWB-3500, or IWC-3400 and IWC-3500, 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. 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.

7. ***Corrective Actions:*** Repair and replacement activities are performed in conformance with IWA-4000. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

8. ***Confirmation Process:*** Site quality assurance 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process and administrative controls.

9. ***Administrative Controls:*** As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the administrative controls.

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).

    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. IGSCC has also occurred in a number of vessel internal components, such as core shrouds, access hole covers, top guides, and core spray spargers (U.S. Nuclear Regulatory Commission [NRC] Bulletin 80-13, NRC Information Notice [IN] 95-17, NRC Generic Letter [GL] 94-03, and NUREG-1544). Cracking due to thermal and mechanical loading has occurred in high-pressure coolant injection piping (NRC IN 89-80) and instrument lines (NRC Licensee Event Report [LER] 50-249/99-003-01). 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) caused by leaky valves has also occurred (NRC LER 50-219/98-014-00).

    PWR Primary System: Although the primary pressure boundary piping of PWRs has generally not been found to be affected by stress corrosion cracking (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), control rod drive 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 been observed in plants in the United States. Cracking due to thermal and mechanical loading has occurred in high-pressure injection and safety injection piping (NRC IN 97-46 and NRC Bulletin 88-08). Through-wall circumferential cracking has been found in reactor pressure vessel head control rod drive penetration nozzles (NRC IN 2001-05). Evidence of reactor coolant leakage, together with crack-like indications, has been found in bottom-mounted instrumentation nozzles (NRC IN 2003-11 and IN 2003-11, Supplement 1). Cracking in pressurizer safety and relief line nozzles and in surge line nozzles has been detected (NRC IN 2004-11), and circumferential cracking in stainless steel pressurizer heater sleeves has also been found (NRC IN 2006-27). Also, primary water stress corrosion cracking (PWSCC) has been observed in steam generator drain bowl welds inspected as part of a licensee's Alloy 600/82/182 program (NRC IN 2005-02).

PWR Secondary System: Steam generator tubes have experienced outside diameter stress corrosion cracking (OGSCC), intergranular attack, 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 Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

10 CFR 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2009.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, The ASME Boiler and Pressure Vessel Code, 2004 edition as approved in 10 CFR 50.55a, The 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.

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 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 Information Notice 2001-05, *Through-Wall Circumferential Cracking of Reactor Pressure Vessel Head Control Rod Drive Mechanism Penetration Nozzles at Oconee Nuclear Station, Unit 3*, U.S. Nuclear Regulatory Commission, April 30, 2001.

NRC Information Notice 2003-11, *Leakage Found on Bottom-Mounted Instrumentation Nozzles*, U.S. Nuclear Regulatory Commission, August 13, 2003.

NRC Information Notice 2003-11, Supplement 1, *Leakage Found on Bottom-Mounted Instrumentation Nozzles*, U.S. Nuclear Regulatory Commission, January 8, 2004.

NRC Information Notice 2004-11, *Cracking in Pressurizer Safety and Relief Nozzles and in Surge Line Nozzles*, U.S. Nuclear Regulatory Commission, May 4, 2004.

NRC Information Notice 2005-02, *Pressure Boundary Leakage Identified on Steam Generator Drain Bowl Welds*, U.S. Nuclear Regulatory Commission, February 4, 2005.

NRC Information Notice 2006-27, *Circumferential Cracking in the Stainless Steel Pressurizer Heater Sleeves of Pressurized Water Reactors*, U.S. Nuclear Regulatory Commission, December 11, 2006.

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.

NUREG-1544, *Status Report: Intergranular Stress Corrosion Cracking of BWR Core Shrouds and Other Internal Components*, U.S. Nuclear Regulatory Commission, March 1, 1996.

XI.M2 WATER CHEMISTRY

Program Description

The main objective of this program is to mitigate loss of material due to corrosion, cracking due to stress corrosion cracking (SCC) and related mechanisms, and reduction of heat transfer due to fouling in components exposed to a treated water environment. The program includes periodic monitoring of the treated water in order to minimize loss of material or cracking.

The water chemistry program for boiling water reactors (BWRs) relies on monitoring and control of reactor water chemistry based on industry guidelines contained in the Boiling Water Reactor Vessel and Internals Project (BWRVIP)-190 (Electric Power Research Institute [EPRI] 1016579). The BWRVIP-190 has three sets of guidelines: one for reactor water, one for condensate and feedwater, and one for control rod drive (CRD) mechanism cooling water. The water chemistry program for PWRs relies on monitoring and control of reactor water chemistry based on industry guidelines contained in EPRI 1014986 (PWR Primary Water Chemistry Guidelines-Revision 6) and EPRI 1016555 (PWR Secondary Water Chemistry Guidelines-Revision 7).

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’s intended function is maintained during the period of extended operation. 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 components in the reactor coolant system, the engineered safety features, the auxiliary systems, and the steam and power conversion system. This program addresses the metallic components subject to aging management review that are exposed to a treated water environment controlled by the water chemistry program.

2. ***Preventive Actions:*** The program includes specifications for chemical species, impurities and additives, 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 cracking caused by SCC. For BWRs, maintaining high water purity reduces susceptibility to SCC, and chemical additive programs such as hydrogen water chemistry, or noble metal chemical application also may be used. For PWRs, additives are used for reactivity control and to control pH and inhibit corrosion.

3. ***Parameters Monitored/Inspected:*** The concentrations of corrosive impurities listed in the EPRI water chemistry guidelines are monitored to mitigate loss of material, cracking, and reduction of heat transfer. Water quality also is maintained in accordance with the guidance. Chemical species and water quality are monitored by in-process methods or through sampling. The chemical 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.

4. ***Detection of Aging Effects:*** This is a mitigation program and does not provide for detection of any aging effects of concern for the components within its scope. The monitoring methods and frequency of water chemistry sampling and testing is performed in accordance with the EPRI water chemistry guidelines and based on plant operating conditions. The main objective of this program is to mitigate loss of material due to corrosion and cracking due to SCC in components exposed to a treated water environment.

5. ***Monitoring and Trending:*** Chemistry parameter data are recorded, evaluated, and trended in accordance with the EPRI water chemistry guidelines.

6. ***Acceptance Criteria:*** Maximum levels for various chemical parameters are maintained within the system-specific limits as indicated by the limits specified in the corresponding EPRI water chemistry guidelines.

7. ***Corrective Actions:*** Any evidence of aging effects or unacceptable water chemistry results are evaluated, the cause identified, and the condition corrected. When measured water chemistry parameters are outside the specified range, corrective actions are taken to bring the parameter back within the acceptable range (or to change the operational mode of the plant) within the time period specified in the EPRI water chemistry guidelines. Whenever corrective actions are taken to address an abnormal chemistry condition, increased sampling or other appropriate actions may be used to verify the effectiveness of these actions. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process.

9. ***Administrative Controls:*** Site quality assurance 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address 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 systems, and reactor water cleanup 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] Bulletin 80-13, NRC Information Notice [IN] 95-17, NRC Generic 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 potential for SCC-type mechanisms might normally occur because of inadvertent introduction of contaminants into the primary coolant system, including contaminants introduced from the free surface of the spent fuel pool (which can be a natural collector of airborne contaminants) or the introduction of oxygen during plant cooldowns (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. 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 (NRC INs 89-33, 94-87, 97-88, 90-10, and 96-11; NRC Bulletin 89-01 and its two supplements). IGSCC-induced circumferential cracking has occurred in PWR pressurizer heater sleeves (NRC IN 2006-27).

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). Extensive buildup of deposits at steam generator tube support holes can result in flow-induced vibrations and tube cracking (NRC IN 2007-37).

Such operating experience has provided feedback to revisions of the EPRI water chemistry guideline documents.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

BWRVIP-190 (EPRI 1016579), *BWR Vessel and Internals Project: BWR Water Chemistry Guidelines-2008 Revision*, Electric Power Research Institute, Palo Alto, CA, October 2008.

EPRI 1016555, *PWR Secondary Water Chemistry Guidelines–Revision 7*, Electric Power Research Institute, Palo Alto, CA, February 2009.

EPRI 1014986, *PWR Primary Water Chemistry Guidelines,* Revision 6, Volumes 1 and 2, Electric Power Research Institute, Palo Alto, CA, December 2007.

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

NRC Bulletin 89-01, *Failure of Westinghouse Steam Generator Tube Mechanical Plugs*, U.S. Nuclear Regulatory Commission, May 15, 1989.

NRC Bulletin 89-01, Supplement 1, *Failure of Westinghouse Steam Generator Tube Mechanical Plugs*, U.S. Nuclear Regulatory Commission, November 14, 1989.

NRC 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.

NRC Information Notice 2006-27, *Circumferential Cracking in the Stainless Steel Pressurizer Heater Sleeves of Pressurized Water Reactors*, December 11, 2006.

NRC Information Notice 2007-37, *Buildup of Deposits in Steam Generators*, November 23, 2007.

NUREG-1544*, Status Report: Intergranular Stress Corrosion Cracking of BWR Core Shrouds and Other Internal Components*, U.S. Nuclear Regulatory Commission, March 1, 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.M3 REACTOR HEAD CLOSURE STUD BOLTING

Program Description

This program includes (a) inservice inspection (ISI) in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Code, Section XI, Subsection IWB (2004 edition,[2] no addenda), Table IWB 2500-1; and (b) preventive measures to mitigate cracking. The program also relies on recommendations to address reactor head stud bolting degradation as delineated in NUREG-1339 and Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.65.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program manages the aging effects of cracking due to stress corrosion cracking (SCC) or intergranular stress corrosion cracking (IGSCC) and loss of material due to wear or corrosion for reactor vessel closure stud bolting (studs, washers, bushings, nuts, and threads in flange) for both boiling water reactors (BWRs) and pressurized water reactors (PWRs).

2. ***Preventive Actions:*** Preventive measures include:

   (a) avoiding the use of metal-plated stud bolting to prevent degradation due to corrosion or hydrogen embrittlement;

   (b) using manganese phosphate or other acceptable surface treatments;

   (c) using stable lubricants. Of particular note, use of molybdenum disulfide (MoS_2) as a lubricant has been shown to be a potential contributor to SCC and should not be used (RG 1.65); and

   (d) using bolting material for closure studs that has an actual measured yield strength less than 1,034 megapascals (MPa) (150 kilo-pounds per square inch) (NUREG-1339).

   Implementation of these mitigation measures can reduce potential for SCC or IGSCC, thus making this program effective.

3. ***Parameters Monitored/Inspected:*** The ASME Section XI ISI program detects and sizes cracks, detects loss of material, and detects coolant leakage by following the examination and inspection requirements specified in Table IWB-2500-1.

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 aging effects are discovered and repaired before the loss of intended function of the component. Inspection can reveal cracking, loss of material due to corrosion or wear, and leakage of coolant.

   The program uses visual, surface, and volumetric examinations in accordance with the general requirements of Subsection IWA-2000. Surface examination uses magnetic particle or liquid penetrant examinations to indicate the presence of surface discontinuities and

[2] Refer to the GALL Report, Chapter I, for applicability of other editions of the ASME Code, Section XI.

flaws. Volumetric examination uses radiographic or ultrasonic examinations to indicate the presence of discontinuities or flaws throughout the volume of material. Visual VT-2 examination detects evidence of leakage from pressure-retaining components, as required during the system pressure test.

Components are examined and tested in accordance with ASME Code, Section XI, Table IWB-2500-1, Examination Category B-G-1, for pressure-retaining bolting greater than 2 inches in diameter. Examination Category B-P for all pressure-retaining components specifies visual VT-2 examination of all pressure-retaining boundary components during the system leakage test. Table IWB-2500-1 specifies the extent and frequency of the inspection and examination methods, and IWB-2400 specifies the schedule of the inspection.

5. ***Monitoring and Trending:*** The Inspection schedule of IWB-2400 and the extent and frequency of IWB-2500-1 provide timely detection of cracks, loss of material, and leakage.

6. ***Acceptance Criteria:*** Any indication or relevant condition of degradation in closure stud bolting is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3500.

7. ***Corrective Actions:*** Repair and replacement are performed in accordance with the requirements of IWA-4000 and the material and inspection guidance of RG 1.65. The maximum yield strength of replacement material should be limited as recommended in NUREG-1339. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

8. ***Confirmation Process:*** Site quality assurance 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process and administrative controls.

9. ***Administrative Controls:*** As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the administrative controls.

10. ***Operating Experience:*** SCC has occurred in BWR pressure vessel head studs (Stoller, 1991). The aging management program has provisions regarding inspection techniques and evaluation, material specifications, corrosion prevention, and other aspects of reactor pressure vessel head stud cracking. Implementation of the program provides reasonable assurance that the effects of cracking due to SCC or IGSCC and loss of material due to wear are adequately managed so that the intended functions of the reactor head closure studs and bolts are maintained consistent with the current licensing basis for the period of extended operation. Degradation of threaded bolting and fasteners in closures for the reactor coolant pressure boundary has occurred from boric acid corrosion, SCC, and fatigue loading (NRC Inspection and Enforcement Bulletin 82-02, NRC Generic Letter 91-17).

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

10 CFR 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2009.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, The ASME Boiler and Pressure Vessel Code, 2004 edition as approved in 10 CFR 50.55a The American Society of Mechanical Engineers, New York, NY.

NRC Regulatory Guide 1.65, *Material and Inspection for Reactor Vessel Closure Studs*, Revision 1, U.S. Nuclear Regulatory Commission, April 2010.

NRC Inspection and Enforcement Bulletin 82-02, *Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants*, June 2, 1982.

NUREG-1339, *Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants*, June 1990.

NRC Generic Letter 91-17, *Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants,* October 17, 1991.

Stoller, S. M., Reactor Head Closure Stud Cracking, Material Toughness Outside FSAR - SCC in Thread Roots, Nuclear Power Experience, BWR-2, III, 58, p. 30, 1991.

XI.M4 BWR VESSEL ID ATTACHMENT WELDS

Program Description

The program includes inspection and flaw evaluation in accordance with the guidelines of a staff-approved boiling water reactor vessel and internals project (BWRVIP-48-A) to provide reasonable assurance of the long-term integrity and safe operation of boiling water reactor (BWR) vessel inside diameter (ID) attachment welds.

The guidelines of BWRVIP-48-A include inspection recommendations and evaluation methodologies for the attachment welds between the vessel wall and vessel ID brackets that attach safety-related components to the vessel (e.g., jet pump riser braces and core spray piping brackets). In some cases, the attachment is a simple weld; in others, it includes a weld build-up pad on the vessel. The BWRVIP-48-A guidelines include information on the geometry of the vessel ID attachments; evaluate susceptible locations and safety consequence of failure; provide recommendations regarding the method, extent, and frequency of inspection; and discuss acceptable methods for evaluating the structural integrity significance of flaws detected during these examinations.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program is focused on managing the effects of cracking due to stress corrosion cracking (SCC), including intergranular stress corrosion cracking (IGSCC). The program is an augmented inservice inspection program that uses the inspection and flaw evaluation criteria in BWRVIP-48-A to detect cracking and monitor the effects of cracking on the intended function of the components. The program provides for repair and/or replacement, as needed, to maintain the ability to perform the intended function. The program is applicable to structural welds for BWR reactor vessel internal integral attachments.

2. ***Preventive Actions:*** The BWR Vessel ID Attachment Welds Program is a condition monitoring program and has no preventive actions. Maintaining high water purity reduces susceptibility to SCC or IGSCC. Reactor coolant water chemistry is monitored and maintained in accordance with the Water Chemistry Program. The program description, evaluation, and technical basis of water chemistry are presented in GALL AMP XI.M2, “Water Chemistry.”

3. ***Parameters Monitored/Inspected:*** The program monitors for cracks induced by SCC and IGSCC on the intended function of BWR vessel ID attachment welds. The program looks for surface discontinuities that may indicate the presence of a crack in the component in accordance with the guidelines of approved BWRVIP-48-A and the requirements of the American Society of Mechanical Engineers (ASME) Code, Section XI, Table IWB 2500-1 (2004 edition[3]).

4. ***Detection of Aging Effects:*** The extent and schedule of the inspection and test techniques prescribed by BWRVIP-48-A guidelines are designed to maintain structural integrity and ensure that aging effects are discovered and repaired before the loss of intended function. Inspection can reveal cracking. Vessel ID attachment welds are inspected in accordance with the requirements of ASME Section XI, Subsection IWB, Examination Category B-N-2.

[3] Refer to the GALL Report, Chapter I, for applicability of other editions of the ASME Code, Section XI.

The ASME Code, Section XI inspection specifies visual VT-1 examination to detect discontinuities and imperfections on the surfaces of components and visual VT-3 examination to determine the general mechanical and structural condition of the component supports. The inspection and evaluation guidelines of BWRVIP-48-A recommend more stringent inspections for certain attachments. The guidelines recommend enhanced visual VT-1 examination of all safety-related attachments and those non-safety-related attachments identified as being susceptible to IGSCC. Visual VT-1 examination is capable of achieving 1/32-inch resolution; the enhanced visual VT-1 examination method is capable of achieving a 1/2 mil (0.0005 inch) wire resolution. The nondestructive examination (NDE) techniques appropriate for inspection of BWR vessel internals, including the uncertainties inherent in delivering and executing NDE techniques in a BWR, are included in BWRVIP-03.

5. ***Monitoring and Trending:*** Inspections scheduled in accordance with ASME Code, Section XI, IWB-2400 and approved BWRVIP-48-A guidelines provide timely detection of cracks. If flaws are detected, the scope of examination is expanded. Any indication detected is evaluated in accordance with ASME Code, Section XI or the staff-approved BWRVIP-48-A guidelines. Applicable and approved BWRVIP-14-A, BWRVIP-59-A, and BWRVIP-60-A documents provide guidelines for evaluation of crack growth in stainless steels, nickel alloys, and low-alloy steels, respectively.

6. ***Acceptance Criteria:*** Acceptance criteria are given in BWRVIP-48-A and ASME Code, Section XI.

7. ***Corrective Actions:*** Repair and replacement procedures are equivalent to those requirements in ASME Code, Section XI. Corrective action is performed in accordance with ASME Code, Section XI, IWA-4000. As discussed in the Appendix for GALL, the staff finds that licensee implementation of the corrective action guidelines in BWRVIP-48-A provides an acceptable level of quality in accordance with 10 CFR Part 50, Appendix B corrective actions.

8. ***Confirmation Process:*** Site quality assurance 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 for GALL, the staff finds that licensee implementation of the guidelines in BWRVIP-48-A provides an acceptable level of quality in accordance with the 10 CFR Part 50, Appendix B confirmation process and administrative controls.

9. ***Administrative Controls:*** As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B acceptable to address the administrative controls.

10. ***Operating Experience:*** Cracking due to SCC, including IGSCC, has occurred in BWR components. The program guidelines are based on an evaluation of available information, including BWR inspection data and information on the elements that cause IGSCC, to determine which attachment welds may be susceptible to cracking. Implementation of this program provides reasonable assurance that cracking will be adequately managed and that the intended functions of the vessel ID attachments will be maintained consistent with the current licensing basis for the period of extended operation.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

10 CFR 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2009.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, The ASME Boiler and Pressure Vessel Code, 2004 edition as approved in 10 CFR 50.55a, The American Society of Mechanical Engineers, New York, NY.

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

BWRVIP-14-A (EPRI 1016569), *Evaluation of Crack Growth in BWR Stainless Steel RPV Internals*, September 2008.

BWRVIP-48-A (EPRI 1009948), *BWR Vessel and Internals Project, Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines*, November 2004.

BWRVIP-59-A (EPRI 1014874), *Evaluation of Crack Growth in BWR Nickel-Base Austenitic Alloys in RPV Internals*, May 2007.

BWRVIP-60-A (EPRI 1008871), *BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Low Alloy Steel RPV Internals*, June 2003.

BWRVIP-62 (EPRI 108705), *BWR Vessel and Internals Project, Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection*, March 7, 2000.

BWRVIP-190 (EPRI 1016579), *BWR Vessel and Internals Project: BWR Water Chemistry Guidelines—2008 Revision*, October 2008.

XI.M5 BWR FEEDWATER NOZZLE

Program Description

This program includes enhanced inservice inspection (ISI) in accordance with (a) the requirements of the American Society of Mechanical Engineers (ASME) Code, Section XI, Subsection IWB, Table IWB 2500-1 (2004 edition[4]); (b) the recommendation of General Electric (GE) NE-523-A71-0594, Rev. 1, *Alternate BWR Feedwater Nozzle Inspection Requirements*; and (c) NUREG-0619 recommendations for system modifications to mitigate cracking. The program specifies periodic ultrasonic inspection of critical regions of the boiling water reactor (BWR) feedwater nozzle.

Systems modifications to mitigate cracking may have been made, such as removal of stainless steel cladding and installation of improved spargers. Mitigation also is accomplished by changes to plant-operating procedures, such as improved feedwater control to decrease the magnitude and frequency of temperature fluctuations. These modifications are design and operating changes and were instituted for many BWRs during their initial 40-year operating period.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program includes enhanced ISI to monitor the effects of cracking due to cyclic loading and its impact on the intended function of BWR feedwater nozzles.

2. ***Preventive Actions:*** This program is a condition monitoring program and has no preventive actions.

3. ***Parameters Monitored/Inspected:*** The aging management program (AMP) monitors for cracking due to cyclic loading and its impact on the intended function of the BWR feedwater nozzle by detection and sizing of cracks by ISI in accordance with ASME Code, Section XI, Subsection IWB; the recommendation of GE NE-523-A71-0594, Rev. 1; and NUREG-0619 recommendations.

4. ***Detection of Aging Effects:*** The extent and schedule of the inspection prescribed by the program are designed to ensure that aging effects are discovered and repaired before the loss of intended function of the component. Inspection can reveal cracking.

   GE NE-523-A71-0594, Rev. 1 specifies ultrasonic testing (UT) of specific regions of the blend radius and bore. The UT examination techniques and personnel qualifications are in accordance with the guidelines of GE NE-523-A71-0594, Rev. 1. Based on the inspection method and techniques and plant-specific fracture mechanics assessments, the inspection schedule is in accordance with Table 6-1 of GE NE-523-A71-0594, Rev. 1. Leakage monitoring may be used to modify the inspection interval.

5. ***Monitoring and Trending:*** Inspections scheduled in accordance with GE NE-523-A71-0594, Rev. 1 provide timely detection of cracks.

6. ***Acceptance Criteria:*** Any cracking is evaluated in accordance with ASME Code, Section XI, IWB-3100 by comparing inspection results with the acceptance standards of ASME Code, Section XI, IWB-3400 and IWB-3500.

[4] Refer to the GALL Report, Chapter I, for applicability of other editions of the ASME Code, Section XI.

7. ***Corrective Actions:*** Repair and replacement are in conformance with ASME Code, Section XI, Subsection IWA-4000. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B acceptable to address the corrective actions.

8. ***Confirmation Process:*** Site quality assurance 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B acceptable to address the confirmation process and administrative controls.

9. ***Administrative Controls:*** As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B acceptable to address the administrative controls.

10. ***Operating Experience:*** Cracking has occurred in several BWR plants (NUREG-0619, U.S. Nuclear Regulatory Commission [NRC] Generic Letter 81-11). This AMP has been implemented for nearly 30 years and has been found to be effective in managing the effects of cracking on the intended function of feedwater nozzles.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, The ASME Boiler and Pressure Vessel Code, 2004 edition as approved in 10 CFR 50.55a, The American Society of Mechanical Engineers, New York, NY.

GE-NE-523-A71-0594, Rev. 1, *Alternate BWR Feedwater Nozzle Inspection Requirements*, BWR Owner’s Group, August 1999.

NRC Generic Letter 81-11, *BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking (NUREG-0619),* U.S. Nuclear Regulatory Commission, February 29, 1981.

NUREG-0619, *BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking*, U.S. Nuclear Regulatory Commission, November 1980.

XI.M6 BWR CONTROL ROD DRIVE RETURN LINE NOZZLE

Program Description

This program is a condition monitoring program for boiling water reactor (BWR) control rod drive return line (CRDRL) nozzles that is based on the staff's recommended position in NUREG-0619 for thermal fatigue. This program is also intended to address stress corrosion cracking (SCC) discussed in NRC IN 2004-08. The augmented inspections performed in accordance with the recommendations in NUREG-0619 supplement those in-service inspections that are required for these nozzles in accordance with the American Society of Mechanical Engineers (ASME) Code, Section XI, Table IWB-2500-1, as mandated through reference in 10 CFR 50.55a. Thus, this program includes (a) mandatory in-service inspection (ISI) in accordance with the ASME Code, Section XI, Table IWB 2500-1 (2004 edition[5]), and (b) augmented ISI examinations in accordance with applicant's commitments to U.S. Nuclear Regulatory Commission (NRC) Generic Letter (GL) 80-095 to implement the recommendations in NUREG-0619.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program manages the effects of cracking on the intended pressure boundary function of CRDRL nozzles. The scope of this program is applicable to BWRs whose reactor vessel (RV) design includes a welded CRDRL nozzle design. The scope of the program includes CRDRL nozzles and their nozzle-to-RV welds, which are ASME Code Class 1 components. The scope of the program also includes a CRDRL nozzle cap (including any CRDRL nozzle-to-cap welds) if, to mitigate cracking, an applicant has cut the piping to the CRDRL nozzle, and capped the CRDRL nozzle.

2. ***Preventive Actions:*** Activities for preventing or mitigating cracking in CRDRL nozzles are consistent with a BWR facility's past preventive or mitigation actions/activities in its current licensing basis as stated in the applicant's docketed response to NRC GL 80-095 and made to address the recommendations in NUREG-0619. Maintaining high water purity reduces susceptibility to SCC. Reactor coolant water chemistry is monitored and maintained in accordance with the Water Chemistry Program. The program description, evaluation, and technical basis of water chemistry are addressed through implementation of GALL AMP XI.M2, "Water Chemistry."

3. ***Parameters Monitored/Inspected:*** The aging management program (AMP) manages the effects of cracking on the intended function of the RV, the CRDRL nozzle, and for capped nozzles, the nozzle caps, and cap-to-nozzle welds. For liquid penetrant test (PT) examinations that are implemented in accordance with this AMP, the AMP monitors for linear indications that may be indicative of surface breaking cracks. For the volumetric ultrasonic test (UT) examinations that are performed in accordance with this AMP, the AMP monitors and evaluates signals that may indicate the presence of a planar flaw (crack).

4. ***Detection of Aging Effects:*** The extent and schedule of inspection, as delineated in NUREG-0619, assures detection of cracks before the loss of intended function of the CRDRL nozzles. Inspection and test recommendations include PT of CRDRL nozzle bend radius and bore regions and the RV wall area beneath the nozzle, control rod drive system performance testing, and for capped nozzles, the nozzle caps and cap-to-nozzle welds. The

[5] Refer to the GALL Report, Chapter I, for applicability of other editions of the ASME Code, Section XI.

inspection is to include base metal to a distance of one-pipe-wall thickness or 0.5 inches, whichever is greater, on both sides of the weld.

5. ***Monitoring and Trending:*** The inspection schedule of NUREG-0619 provides timely detection of cracks. Indications of cracking are evaluated and trended in accordance with the ASME Code, Section XI, IWB-3100, against applicable acceptance standard criteria that are specified in the ASME Code, Section XI, IWB-3400 or IWB-3500.

6. ***Acceptance Criteria:*** Any cracking is evaluated in accordance with ASME Code, Section XI, IWB-3100 by comparing inspection results with the acceptance standards of ASME Code, Section XI, IWB-3400 and ASME Code, Section XI, IWB-3500.

7. ***Corrective Actions*** Corrective action is performed in conformance with ASME Code, Section XI, IWA-4000. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B acceptable to address the corrective actions.

8. ***Confirmation Process:*** Site quality assurance 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B acceptable to address the confirmation process and administrative controls.

9. ***Administrative Controls:*** As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B acceptable to address the administrative controls.

10. ***Operating Experience:*** Cracking of CRDRL nozzle-to-vessel and nozzle-to-cap welds has occurred in several BWR plants (NUREG-0619 and Information Notice 2004-08). The present AMP has been implemented for nearly 30 years and has been found to be effective in managing the effects of cracking on the intended function of CRDRL nozzles.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

10 CFR 50.55a, *Codes and Standards,* Office of the Federal Register, National Archives and Records Administration, 2009.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, The ASME Boiler and Pressure Vessel Code, 2004 edition as approved in 10 CFR 50.55a, The American Society of Mechanical Engineers, New York, NY.

Letter from D. G. Eisenhut, U.S. Nuclear Regulatory Commission, to R. Gridley, General Electric Company, *forwarding NRC Generic Technical Activity A-10*, January 28, 1980.

NRC Generic Letter 80-095, (Untitled), November 13, 1980.[6]

[6] This GL forwarded NUREG-0619 to members of the U.S nuclear power industry and requested that licensees owning BWR model reactors provide confirmation of their intent to implement the recommendations of NUREG-0619, as applied to the design of their BWRs.

NRC Generic Letter 81-11, (Untitled), February 29, 1981.[7]

NRC Information Notice 2004-08, *Reactor Coolant Pressure Boundary Leakage Attributable To Propagation of Cracking In Reactor Vessel Nozzle Welds*, U.S. Nuclear Regulatory Commission, April 22, 2004.

NUREG-0619, *BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking*, U.S. Nuclear Regulatory Commission, November 1980.

[7] This GL was issued primarily to provide additional clarification on the contents of the confirmatory response that was requested in NRC GL 80-095.

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-based alloy components is delineated in NUREG-0313, Rev. 2, and Nuclear Regulatory Commission (NRC) Generic Letter (GL) 88-01 and its Supplement 1. The material includes base metal and welds. The comprehensive program outlined in NUREG-0313, Rev 2 and NRC GL 88-01 describes improvements that, in combination, will reduce the susceptibility to IGSCC. The elements to cause IGSCC consist of a susceptible (sensitized) material, a significant tensile stress, and an aggressive environment. Sensitization of nonstabilized austenitic stainless steels containing greater than 0.035 weight percent carbon involves precipitation of chromium carbides at the grain boundaries during certain fabrication or welding processes. The formation of carbides creates a 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 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-A) report allows for modifications to the inspection extent and schedule described 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 in-service inspection to monitor IGSCC and its effects on the intended function of BWR piping components within the scope of license renewal. The program is applicable to all BWR piping and piping welds made of austenitic SS and nickel alloy that are 4 inches or larger in nominal diameter containing 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, Rev. 2 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 BWR Stress Corrosion Cracking Program is primarily a condition monitoring program. Maintaining high water purity reduces susceptibility to SCC or IGSCC. Reactor coolant water chemistry is monitored and maintained in accordance with the Water Chemistry Program. The program description, evaluation and technical basis of water chemistry are addressed through implementation of GALL AMP XI.M2, “Water Chemistry.” In addition, NUREG-0313, Rev. 2 and GL 88-01 delineate the guidance for selection of resistant materials and processes that provide resistance to IGSCC such as solution heat treatment and stress improvement processes.

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-A 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-A are designed to maintain structural integrity and ensure that aging effects are discovered and repaired before the loss of intended function of the component. Modifications to the extent and schedule of inspection in NRC GL 88-01 are allowed in accordance with the inspection guidance in approved BWRVIP-75-A. The program uses volumetric examinations to detect IGSCC. Inspection can reveal cracking 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).

5. ***Monitoring and Trending:*** The extent and schedule for inspection, in accordance with the recommendations of NRC GL 88-01 or approved BWRVIP-75-A guidelines, provide timely detection of cracks and leakage of coolant. Indications of cracking are evaluated and trended in accordance with the American Society of Mechanical Engineers (ASME) Code, Section XI, IWA-3000.

   Applicable and approved BWRVIP-14-A, BWRVIP-59-A, BWRVIP-60-A, and BWRVIP-62 reports 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.

6. ***Acceptance Criteria:*** Any cracking is evaluated in accordance with ASME Code, Section XI, IWA-3000 by comparing inspection results with the acceptance standards of ASME Code, Section XI, IWB-3000, IWC-3000 and IWD-3000 for Class 1, 2 and 3 components, respectively.

7. ***Corrective Actions***: The guidance for weld overlay repair and stress improvement or replacement is provided in NRC GL 88-01. Corrective action is performed in accordance with IWA-4000. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B acceptable to address the corrective actions.

8. ***Confirmation Process:*** Site quality assurance 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B acceptable to address the confirmation process and administrative controls.

9. ***Administrative Controls:*** As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B acceptable to address the administrative controls.

10. ***Operating Experience:*** Intergranular SCC has occurred in small- and large-diameter BWR piping made of austenitic SS and nickel-base alloys. Cracking has occurred in recirculation, core spray, residual heat removal, CRD return line penetrations, and reactor water cleanup

system piping welds (NRC GL 88-01 and NRC Information Notices [INs] 82-39, 84-41, and 04-08). The comprehensive program outlined in NRC GL 88-01, NUREG-0313, Rev. 2, and in the staff-approved BWRVIP-75-A report addresses mitigating measures for SCC or IGSCC (e.g., susceptible material, significant tensile stress, and an aggressive environment). The GL 88-01 program, with or without the modifications allowed by the staff-approved BWRVIP-75-A report, has been effective in managing IGSCC in BWR reactor coolant pressure-retaining components and will adequately manage IGSCC degradation.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

10 CFR 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2009.

ASME Code Case N-504-1, *Alternative Rules for Repair of Class 1, 2, and 3 Austenitic Stainless Steel Piping*, Section XI, Division 1, 1995 edition, ASME Boiler and Pressure Vessel Code – Code Cases – Nuclear Components, 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, 2004 edition as approved in 10 CFR 50.55a, The American Society of Mechanical Engineers, New York, NY.

BWRVIP-14-A (EPRI 1016569), *BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Stainless Steel RPV Internals,* Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation, September 2008.

BWRVIP-59-A, (EPRI 1014874), *BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Nickel-Base Austenitic Alloys in RPV Internals*, Final Report by the Office of Nuclear Reactor Regulation, May 2007.

BWRVIP-60-A (EPRI 108871), *BWR Vessel and Internals Project, Evaluation of Stress Corrosion Crack Growth in Low Alloy Steel Vessel Materials in the BWR Environment,* Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation, June 2003.

BWRVIP-61 (EPRI 112076), *BWR Vessel and Internals Induction Heating Stress Improvement Effectiveness on Crack Growth in Operating Reactors,* Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation, January 29, 1999.

BWRVIP-62 (EPRI 108705), *BWR Vessel and Internals Project, Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection,* Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation, March 7, 2000.

BWRVIP-75-A (EPRI 1012621), *BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (NUREG-0313),* Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation, October 2005.

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 04-08, *Reactor Coolant Pressure Boundary Leakage Attributable to Propagation of Cracking in Reactor Vessel Nozzle Welds,* U.S. Nuclear Regulatory Commission, April 22, 2004.

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.M8 BWR PENETRATIONS

Program Description

The program for boiling water reactor (BWR) vessel instrumentation penetrations, control rod drive (CRD) housing and incore-monitoring housing (ICMH) penetrations and standby liquid control (SLC) nozzles/Core ΔP nozzles includes inspection and flaw evaluation in conformance with the guidelines of staff-approved boiling water reactor vessel and internals project (BWRVIP) Topical Reports BWRVIP-49-A, BWRVIP-47-A and BWRVIP-27-A. The inspection and evaluation guidelines of BWRVIP-49-A, BWRVIP-47-A and BWRVIP-27-A contain generic guidelines intended to present appropriate inspection recommendations to assure safety function integrity. The guidelines of BWRVIP-49-A provide information on the type of instrument penetration, evaluate their susceptibility and consequences of failure, and define the inspection strategy to assure safe operation. The guidelines of BWRVIP-47-A provide information on components located in the lower plenum region of BWRs, evaluate their susceptibility and consequences of failure, and define the inspection strategy to assure safe operation. The guidelines of BWRVIP-27-A are applicable to plants in which the SLC system injects sodium pentaborate into the bottom head region of the vessel (in most plants, as a pipe within a pipe of the core plate ΔP monitoring system). The BWRVIP-27-A guidelines address the region where the ΔP and SLC nozzle or housing penetrates the vessel bottom head and include the safe ends welded to the nozzle or housing. Guidelines for repair design criteria are provided in BWRVIP-57-A for instrumentation penetrations and BWRVIP-53-A for SLC line.

Although this is a condition monitoring program, control of water chemistry helps prevent stress corrosion cracking (SCC) and intergranular stress corrosion cracking (IGSCC). The water chemistry program for BWRs relies on monitoring and control of reactor water chemistry based on industry guidelines, such BWRVIP-190 (Electric Power Research Institute [EPRI] 1016579) or later revisions. BWRVIP-190 has three sets of guidelines: one for primary water, one for condensate and feedwater, and one for control rod drive (CRD) mechanism cooling water. Adequate aging management activities for these components provide reasonable assurance that the long-term integrity and safe operation of BWR vessel instrumentation nozzles, CRD housing and incore-monitoring housing (ICMH) penetrations and SLC nozzles/Core ΔP nozzles.

Evaluation and Technical Basis

1. ***Scope of Program:*** The scope of this program is applicable to BWR instrumentation penetrations, CRD housing and incore-monitoring housing (ICMH) penetrations and BWR SLC nozzles/Core ΔP nozzles. The program manages cracking due to cyclic loading or SCC and IGSCC using inspection and flaw evaluation in accordance with the guidelines of staff-approved BWRVIP-49-A, BWRVIP-47-A and BWRVIP-27-A.

2. ***Preventive Actions:*** This program is a condition monitoring program and has no preventive actions. However, maintaining high water purity reduces susceptibility to SCC or IGSCC. The program description, evaluation and technical basis of water chemistry are presented in GALL AMP XI.M2, "Water Chemistry."

3. ***Parameters Monitored/Inspected:*** The program manages the effects of cracking due to SCC/IGSCC on the intended function of the BWR instrumentation nozzles, CRD housing and incore-monitoring housing (ICMH) penetrations, and BWR SLC nozzles/Core ΔP nozzles. The program accomplishes this by inspection for cracks in accordance with the

guidelines of approved BWRVIP-49-A, BWRVIP-47-A or BWRVIP-27-A and the requirements of the ASME Code, Section XI, Table IWB 2500-1 (2004 edition[8]).

4. ***Detection of Aging Effects:*** The evaluation guidelines of BWRVIP-49-A, BWRVIP-47-A and BWRVIP-27-A provide that the existing inspection requirements in ASME Code, Section XI, Table IWB-2500-1, are sufficient to monitor for indications of cracking in BWR instrumentation nozzles, CRD housing and incore-monitoring housing (ICMH) penetrations and BWR SLC nozzles/Core ΔP nozzles, and should continue to be followed for the period of extended operation. The extent and schedule of the inspection and test techniques prescribed by the ASME Code, Section XI program are designed to maintain structural integrity and ensure that aging effects are discovered and repaired before the loss of intended function of the component.

   Instrument penetrations, CRD housing and incore-monitoring housing (ICMH) penetrations and SLC system nozzles or housings are inspected in accordance with the requirements in the ASME Code, Section XI. These examination categories include volumetric examination methods (ultrasonic testing or radiography testing), surface examination methods (liquid penetrant testing or magnetic particle testing), and VT-2 visual examination methods.

5. ***Monitoring and Trending:*** Inspections scheduled in accordance with ASME Code, Section XI, IWB-2400 and approved BWRVIP-49-A, BWRVIP-47-A, or BWRVIP-27-A provides timely detection of cracks. The scope of examination and reinspection is expanded beyond the baseline inspection if flaws are detected. Any indication detected is evaluated in accordance with ASME Code, Section XI or other acceptable flaw evaluation criteria, such as the staff-approved BWRVIP-49-A, BWRVIP-47-A, or BWRVIP-27-A guidelines. Applicable and approved BWRVIP-14-A, BWRVIP-59-A, and BWRVIP-60-A documents provide additional guidelines for the evaluation of crack growth in stainless steels (SSs), nickel alloys, and low-alloy steels, respectively.

6. ***Acceptance Criteria:*** Acceptance criteria are given in BWRVIP-49-A for instrumentation nozzles, BWRVIP-47-A for CRD housing and incore-monitoring housing (ICMH) penetrations, and BWRVIP-27A for BWR SLC nozzles/Core ΔP nozzles.

7. ***Corrective Actions:*** Repair and replacement procedures in staff-approved BWRVIP-57-A and BWRVIP-53-A are equivalent to those required in ASME Code, Section XI. Guidelines for repair design criteria are provided in BWRVIP-57-A for instrumentation penetrations and BWRVIP-53-A for SLC line. As discussed in the Appendix for GALL, the staff finds that licensee implementation of the guidelines in BWRVIP-49-A, BWRVIP-47-A, and BWRVIP-27-A provides an acceptable level of quality in accordance with 10 CFR Part 50, Appendix B corrective actions. However, any repair in accordance with ASME Code is acceptable.

8. ***Confirmation Process:*** Site quality assurance 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 for GALL, the staff finds that licensee implementation of the guidelines in BWRVIP-49-A, BWRVIP-47-A, and BWRVIP-27A, as modified, provides an acceptable level of quality for inspection and flaw evaluation of the safety-related components addressed in accordance with the 10 CFR Part 50, Appendix B confirmation process and administrative controls.

[8] Refer to the GALL Report, Chapter I, for applicability of other editions of the ASME Code, Section XI.

9. ***Administrative Controls:*** As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B acceptable to address the administrative controls.

10. ***Operating Experience:*** Cracking due to SCC or IGSCC has occurred in BWR components made of austenitic SSs and nickel alloys. The program guidelines are based on an evaluation of available information, including BWR inspection data and information about the elements that cause IGSCC, to determine which locations may be susceptible to cracking. Implementation of the program provides reasonable assurance that cracking will be adequately managed so the intended functions of the instrument penetrations and SLC system nozzles or housings will be maintained consistent with the current licensing basis for the period of extended operation.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

10 CFR 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2009.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, The ASME Boiler and Pressure Vessel Code, 2004 edition as approved in 10 CFR 50.55a, The American Society of Mechanical Engineers, New York, NY.

BWRVIP-14-A (EPRI 1016569), *BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Stainless Steel RPV Internals,* Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation, September 2008.

BWRVIP-27-A (EPRI 1007279), *BWR Vessel and Internals Project, BWR Standby Liquid Control System/Core Plate ΔP Inspection and Flaw Evaluation Guidelines,* Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation, August 2003.

BWRVIP-47-A (EPRI 1009947), *BWR Vessel and Internals Project, BWR Lower Plenum Inspection and Flaw Evaluation Guidelines,* Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation, November 2004.

BWRVIP-49-A (EPRI 1006602), BWR Vessel and Internals Project, Instrument Penetration Inspection and Flaw Evaluation Guidelines, Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation.

BWRVIP-53-A (EPRI 1012120), BWR Vessel and Internals Project, Standby Liquid Control Line Repair Design Criteria Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation, September 2005.

BWRVIP-57-A (EPRI 1012111), BWR Vessel and Internals Project, Instrument Penetration Repair Design Criteria, Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation, September 2005.

BWRVIP-59-A (EPRI 1014874), BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Nickel-Base Austenitic Alloys in RPV Internals, Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation, May 2007.

BWRVIP-60-A (EPRI 1008871), BWR Vessel and Internals Project, Evaluation of Stress Corrosion Crack Growth in Low Alloy Steel Vessel Materials in the BWR Environment, Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation, June 2003.

BWRVIP-190 (EPRI 1016579), BWR Vessel and Internals Project, BWR Water Chemistry Guidelines-2008 Revision, Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation, October 2008.

XI.M9 BWR VESSEL INTERNALS

Program Description

The program includes inspection and flaw evaluations in conformance with the guidelines of applicable and staff-approved boiling water reactor vessel and internals project (BWRVIP) documents to provide reasonable assurance of the long-term integrity and safe operation of boiling water reactor (BWR) vessel internal components.

The BWRVIP documents provide generic guidelines intended to present the applicable inspection recommendations to assure safety function integrity of the subject safety-related reactor pressure vessel internal components. The guidelines provide information on component description and function; evaluate susceptible locations and safety consequences of failure; provide recommendations for methods, extent, and frequency of inspection; discuss acceptable methods for evaluating the structural integrity significance of flaws detected during these examinations; and recommend repair and replacement procedures.

In addition, this program provides screening criteria to determine the susceptibility of cast austenitic stainless steels (CASS) components to thermal aging on the basis of casting method, molybdenum content, and percent ferrite, in accordance with the criteria set forth in the May 19, 2000 letter from Christopher Grimes, Nuclear Regulatory Commission (NRC), to Mr. Douglas Walters, Nuclear Energy Institute (NEI). The susceptibility to thermal aging embrittlement of CASS components is determined in terms of casting method, molybdenum content, and ferrite content. For low-molybdenum content steels (SA-351 Grades CF3, CF3A, CF8, CF8A, or other steels with ≤0.5 wt.% molybdenum), only static-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement. Static-cast low-molybdenum steels with >20% ferrite and all centrifugal-cast low-molybdenum steels are not susceptible. For high-molybdenum content steels (SA-351 Grades CF3M, CF3MA, CF8M or other steels with 2.0 to 3.0 wt.% molybdenum), static-cast steels with >14% ferrite and centrifugal-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement. Static-cast high-molybdenum steels with ≤14% ferrite and centrifugal-cast high-molybdenum steels with ≤20% ferrite are not susceptible. In the susceptibility screening method, ferrite content is calculated by using the Hull's equivalent factors (described in NUREG/CR-4513, Rev. 1) or a staff approved method for calculating delta ferrite in CASS materials.

The screening criteria are applicable to all cast stainless steel primary pressure boundary and reactor vessel internal components with service conditions above 250°C (482°F). The screening criteria for susceptibility to thermal aging embrittlement are not applicable to niobium-containing steels; such steels require evaluation on a case-by-case basis. For "potentially susceptible" components, the program considers loss of fracture toughness due to neutron embrittlement or thermal aging embrittlement.

This AMP addresses aging degradation of X-750 alloy-, and precipitation-hardened (PH) martensitic stainless steel (e.g., 15-5 and 17-4 PH steel) materials and martensitic stainless steel (e.g., 403, 410, 431 steel) that are used in BWR vessel internal components. When exposed to a BWR reactor temperature of 550°F, these materials can experience neutron embrittlement and a decrease in fracture toughness. PH-martensitic stainless steels and martensitic stainless steels are also susceptible to thermal embrittlement. Effects of thermal and neutron embrittlement can cause failure of these materials in vessel internal components. In addition, X-750 alloy in a BWR environment is susceptible to intergranular stress corrosion cracking (IGSCC).

Evaluation and Technical Basis

1. ***Scope of Program:*** The program is focused on managing the effects of cracking due to stress corrosion cracking (SCC), IGSCC, or irradiation-assisted stress corrosion cracking (IASCC), cracking due to fatigue and loss of material due to wear. This program also includes loss of toughness due to neutron and thermal embrittlement. The program applies to wrought and cast reactor vessel internal components. The program contains in-service inspection (ISI) to monitor the effects of cracking on the intended function of the components, uses NRC-approved BWRVIP reports as the basis for inspection, evaluation, repair and/or replacement, as needed, and evaluates the susceptibility of CASS, X-750 alloy, precipitation-hardened (PH) martensitic stainless steel (e.g., 15-5 and 17-4 PH steel), and martensitic stainless steel (e.g., 403, 410, 431 steel) components to neutron and/or thermal embrittlement.

   The scope of the program includes the following BWR reactor vessel (RV) and RV internal components as subject to the following NRC-approved applicable BWRVIP guidelines:

   Core shroud: BWRVIP-76-A provides guidelines for inspection and evaluation; BWRVIP-02-A, Rev. 2, provides guidelines for repair design criteria.

   Core plate: BWRVIP-25 provides guidelines for inspection and evaluation; BWRVIP-50-A provides guidelines for repair design criteria.

   Core spray: BWRVIP-18-A provides guidelines for inspection and evaluation; BWRVIP-16-A and 19A provides guidelines for replacement and repair design criteria, respectively.

   Shroud support: BWRVIP-38 provides guidelines for inspection and evaluation; BWRVIP-52-A provides guidelines for repair design criteria.

   Jet pump assembly: BWRVIP-41 provides guidelines for inspection and evaluation; BWRVIP-51-A provides guidelines for repair design criteria.

   Low-pressure coolant injection (LPCI) coupling: BWRVIP-42-A provides guidelines for inspection and evaluation; BWRVIP-56-A provides guidelines for repair design criteria.

   Top guide: BWRVIP-26-A and BWRVIP-183 provide guidelines for inspection and evaluation; BWRVIP-50-A provides guidelines for repair design criteria. Inspect five percent (5%) of the top guide locations using enhanced visual inspection technique, EVT-1 within six years after entering the period of extended operation. An additional 5% of the top guide locations will be inspected within twelve years after entering the period of extended operation.

   Reinspection Criteria:

   BWR/2-5 - Inspect 10% of the grid beam cells containing control rod drives/blades every twelve years with at least 5% to be performed within six years.

   BWR/6 - Inspect the rim areas containing the weld and heat affected zone (HAZ) from the top surface of the top guide and two cells in the same plane/axis as the weld every six years.

The top guide inspection locations are those that have high neutron fluences exceeding the IASCC threshold. The extent of the examination and its frequency will be based on a ten percent sample of the total population, which includes all grid beam and beam-to-beam crevice slots.

Control rod drive (CRD) housing: BWRVIP-47-A provides guidelines for inspection and evaluation; BWRVIP-58-A provides guidelines for repair design criteria.

Lower plenum components: BWRVIP-47-A provides guidelines for inspection and evaluation; BWRVIP-57-A provides guidelines for repair design criteria for instrument penetrations.

Steam Dryer: BWRVIP-139 provides guidelines for inspection and evaluation for the steam dryer components.

Although BWRVIP repair design criteria provide criteria for repairs, aging management strategies for repairs are provided by the repair designer, not the BWRVIP.

2. ***Preventive Actions:*** The BWR Vessel Internals Program is a condition monitoring program and has no preventive actions. Maintaining high water purity reduces susceptibility to SCC or IGSCC. Reactor coolant water chemistry is monitored and maintained in accordance with the Water Chemistry Program. The program description, evaluation and technical basis of water chemistry are presented in GALL AMP XI.M2, "Water Chemistry." In addition, for core shroud repairs or other IGSCC repairs, the program maintains operating tensile stresses below a threshold limit that precludes IGSCC of X-750 material.

3. ***Parameters Monitored/Inspected:*** The program monitors the effects of cracking on the intended function of the component by detection and sizing of cracks by inspection in accordance with the guidelines of applicable and approved BWRVIP documents and the requirements of the American Society of Mechanical Engineers (ASME) Code, Section XI, Table IWB 2500-1 (2004 edition[9]).

   Loss of fracture toughness due to neutron embrittlement in CASS materials can occur with a neutron fluence greater than $1x10^{17}$ n/cm^2 (E>1 MeV). Loss fracture toughness of CASS material due to thermal embrittlement is dependent on the material's casting method, molybdenum content, and ferrite content. The program does not directly monitor for loss of fracture toughness that is induced by thermal aging or neutron irradiation embrittlement. The impact of loss of fracture toughness on component integrity is indirectly managed by using visual or volumetric examination techniques to monitor for cracking in the components.

   Neutron embrittlement of X-750 alloys, PH-martensitic stainless steels, and martensitic stainless steels cannot be identified by typical in-service inspection activities. However, by performing visual or other inspections, applicants can identify cracks that could lead to failure of a potentially embrittled component prior to component failure. Applicants can thus indirectly manage the effects of embrittlement in the PH steels, martensitic stainless steels, and X-750 components by identifying aging degradation (i.e., cracks), implementing early corrective actions, and monitoring and trending age-related degradation.

[9] Refer to the GALL Report, Chapter I, for applicability of other editions of the ASME Code, Section XI.

4. ***Detection of Aging Effects:*** The extent and schedule of the inspection and test techniques prescribed by the applicable and NRC-approved BWRVIP guidelines are designed to maintain structural integrity and ensure that aging effects will be discovered and repaired before the loss of intended function of BWR vessel internals. Inspection can reveal cracking. Vessel internal components are inspected in accordance with the requirements of ASME Section XI, Subsection IWB, Examination Category B-N-2. The ASME Section XI inspection specifies visual VT-1 examination to detect discontinuities and imperfections, such as cracks, corrosion, wear, or erosion, on the surfaces of components. This inspection also specifies visual VT-3 examination to determine the general mechanical and structural condition of the component supports by (a) verifying parameters, such as clearances, settings, and physical displacements, and (b) detecting discontinuities and imperfections, such as loss of integrity at bolted or welded connections, loose or missing parts, debris, corrosion, wear, or erosion. BWRVIP program requirements provide for inspection of BWR reactor internals to manage loss of material and cracking using appropriate examination techniques such as visual examinations (e.g., EVT-1, VT-1) and volumetric examinations (e.g., UT).

   The applicable and NRC-approved BWRVIP guidelines recommend more stringent inspections, such as EVT-1 examinations or ultrasonic methods of volumetric inspection, for certain selected components and locations. The nondestructive examination (NDE) techniques appropriate for inspection of BWR vessel internals, including the uncertainties inherent in delivering and executing NDE techniques in a BWR, are included in BWRVIP-03.

   Thermal and/or neutron embrittlement in susceptible CASS, PH-martensitic steels, martensitic stainless steels, and X-750 components are indirectly managed by performing periodic visual inspections capable of detecting cracks in the component. The 10-year ISI program during the renewal period may include a supplemental inspection covering portions of the susceptible components determined to be limiting from the standpoint of thermal aging susceptibility, neutron fluence, and cracking susceptibility (i.e., applied stress, operating temperature, and environmental conditions). The inspection technique is capable of detecting the critical flaw size with adequate margin. The critical flaw size is determined based on the service loading condition and service-degraded material properties. One example of a supplemental examination is VT-1 examination of ASME Code, Section XI, IWA-2210. The initial inspection is performed either prior to or within 5 years after entering the period of extended operation. If cracking is detected after the initial inspection, the frequency of re-inspection should be justified by the applicant based on fracture toughness properties appropriate for the condition of the component. The sample size is 100% of the accessible component population, excluding components that may be in compression during normal operations.

5. ***Monitoring and Trending:*** Inspections are scheduled in accordance with the applicable and approved BWRVIP guidelines provide timely detection of cracks. Each BWRVIP guideline recommends baseline inspections that are used as part of data collection towards trending. The BWRVIP guidelines provide recommendations for expanding the sample scope and re-inspecting the components if flaws are detected. Any indication detected is evaluated in accordance with ASME Code, Section XI or the applicable BWRVIP guidelines. BWRVIP-14-A, BWRVIP-59-A, BWRVIP-60-A, BWRVIP-80NP-A and BWRVIP-99-A documents provide additional guidelines for evaluation of crack growth in stainless steels (SSs), nickel alloys, and low-alloy steels, respectively.

Inspections scheduled in accordance with ASME Code, Section XI, IWB-2400 and reliable examination methods provide timely detection of cracks. The fracture toughness of PH-martensitic steels, martensitic stainless steels, and X-750 alloys susceptible to thermal and/or neutron embrittlement need to be assessed on a case-by-case basis.

6. ***Acceptance Criteria:*** Acceptance criteria are given in the applicable BWRVIP documents or ASME Code, Section XI. Flaws detected in CASS components are evaluated in accordance with the applicable procedures of ASME Code, Section XI, IWB-3500. Flaw tolerance evaluation for components with ferrite content up to 25% is performed according to the principles associated with ASME Code, Section XI, IWB-3640 procedures for SAWs, disregarding the ASME Code restriction of 20% ferrite. Extensive research data indicate that the lower-bound fracture toughness of thermally aged CASS materials with up to 25% ferrite is similar to that for SAWs with up to 20% ferrite (Lee et al., 1997). Flaw evaluation for CASS components with >25% ferrite is performed on a case-by-case basis by using fracture toughness data provided by the applicant. A fracture toughness value of 255 kJ/m^2 (1,450 $in.\text{-}lb/in.^2$) at a crack depth of 2.5 mm (0.1 in.) is used to differentiate between CASS materials that are susceptible to thermal aging embrittlement and those that are not. Extensive research data indicate that for non-susceptible CASS materials, the saturated lower-bound fracture toughness is greater than 255 kJ/m^2 (NUREG/CR-4513, Rev. 1).

   Acceptance criteria for the assessment of PH-martensitic steels, martensitic stainless steels, and X-750 alloys susceptible to thermal aging and/or neutron embrittlement are assessed on a case-by-case basis.

7. ***Corrective Actions:*** Repair and replacement procedures are equivalent to those requirements in ASME Code Section XI. Repair and replacement is performed in conformance with the applicable and NRC-approved BWRVIP guidelines listed above. For top guides where cracking is observed, sample size and inspection frequencies are increased. As discussed in the Appendix for GALL, the staff finds that licensee implementation of the corrective action guidelines in the staff-approved BWRVIP reports will provide an acceptable level of quality accordance with 10 CFR Part 50, Appendix B.

8. ***Confirmation Process:*** Site quality assurance 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 for GALL, the staff finds that licensee implementation of the guidelines in the staff-approved BWRVIP reports will provide an acceptable level of quality for inspection and flaw evaluation of the safety-related components addressed in accordance with the 10 CFR Part 50, Appendix B, confirmation process and administrative controls.

9. ***Administrative Controls:*** As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B acceptable to address the administrative controls.

10. ***Operating Experience:*** There is documentation of cracking in both the circumferential and axial core shroud welds, and in shroud supports. Extensive cracking of circumferential core shroud welds has been documented in NRC Generic Letter 94-03 and extensive cracking in vertical core shroud welds has been documented in NRC Information Notice 97-17. It has affected shrouds fabricated from Type 304 and Type 304L SS, which is generally considered to be more resistant to SCC. Weld regions are most susceptible to SCC, although it is not clear whether this is due to sensitization and/or impurities associated with

the welds or the high residual stresses in the weld regions. This experience is reviewed in NRC GL 94-03 and NUREG-1544; some experiences with visual inspections are discussed in NRC IN 94-42.

Both circumferential (NRC IN 88-03) and radial cracking (NRC IN 92-57) have been observed in the shroud support access hole covers that are made from Alloy 600. Instances of cracking in core spray spargers have been reviewed in NRC Bulletin 80-13, and cracking in core spray pipe has been reviewed in BWRVIP-18.

Cracking of the core plate has not been reported, but the creviced regions beneath the plate are difficult to inspect. BWRVIP-06R1-A and BWRVIP-25 address the safety significance and inspection requirements for the core plate assembly. Only inspection of core plate bolts (for plants without retaining wedges) or inspection of the retaining wedges is required. NRC IN 95-17 discusses cracking in top guides of United States and overseas BWRs. Related experience in other components is reviewed in NRC GL 94-03 and NUREG-1544. Cracking has also been observed in the top guide of a Swedish BWR.

Instances of cracking have occurred in the jet pump assembly (NRC Bulletin 80-07), hold-down beam (NRC IN 93-101), and jet pump riser pipe elbows (NRC IN 97-02).

Cracking of dry tubes has been observed at 14 or more BWRs. The cracking is intergranular and has been observed in dry tubes without apparent sensitization, suggesting that IASCC may also play a role in the cracking.

Two CRDM lead screw male couplings were fractured in a pressurized-water reactor (PWR), designed by Babcock and Wilcox (B&W), at Oconee Nuclear Station (ONS), Unit 3. The fracture was due to thermal embrittlement of 17-4 PH material (NRC IN 2007-02). While this occurred at a PWR, it also needs to be considered for BWRs.

IGSCC in the X-750 materials of a tie rod coupling and jet pump hold-down beam was observed in a domestic plant.

The program guidelines outlined in applicable and approved BWRVIP documents are based on an evaluation of available information, including BWR inspection data and information on the elements that cause SCC, IGSCC, or IASCC, to determine which components may be susceptible to cracking. Implementation of the program provides reasonable assurance that cracking will be adequately managed so the intended functions of the vessel internal components will be maintained consistent with the current licensing basis (CLB) for the period of extended operation.

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XI.M10 BORIC ACID CORROSION

Program Description

The program relies in part on implementation of recommendations in Nuclear Regulatory Commission (NRC) Generic Letter (GL) 88-05 to monitor the condition of the reactor coolant pressure boundary for borated water leakage. Periodic visual inspection of adjacent structures, components, and supports for evidence of leakage and corrosion is an element of the NRC GL 88-05 monitoring program. Potential improvements to boric acid corrosion programs have been identified because of recent operating experience with cracking of certain nickel alloy pressure boundary components (NRC Regulatory Issue Summary 2003-013).

Borated water leakage from piping and components that are outside the scope of the program established in response to NRC GL 88-05 may affect structures and components that are subject to aging management review (AMR). Therefore, the scope of the monitoring and inspections of this program includes all components that contain borated water and that are in proximity to structures and components that are subject to AMR. The scope of the evaluations, assessments, and corrective actions include all observed leakage sources and the affected structures and components.

Borated water leakage may be discovered through activities other than those established specifically to detect such leakage. Therefore, the program includes provisions for triggering evaluations and assessments when leakage is discovered by other activities. The effects of boric acid corrosion on reactor coolant pressure boundary materials in the vicinity of nickel alloy components are managed by GALL AMP XI.M11B, "Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-induced Corrosion in Reactor Coolant Pressure Boundary Components."

Evaluation and Technical Basis

1. ***Scope of Program:*** The program covers any structures or components on which boric acid corrosion may occur (e.g., steel, copper alloy >15% zinc, and aluminum) and electrical components onto which borated reactor water may leak. The program includes provisions in response to the recommendations of NRC GL 88-05. NRC GL 88-05 provides a program consisting of systematic measures to ensure that corrosion caused by leaking borated coolant does not lead to degradation of the leakage source or adjacent structures and components, and provides assurance that the reactor coolant pressure boundary will have an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture. Such a program provides for (a) determination of the principal location of leakage, (b) examinations and procedures for locating small leaks, and (c) engineering evaluations and corrective actions to ensure that boric acid corrosion does not lead to degradation of the leakage source or adjacent structures or components, which could cause the loss of intended function of the structures or components.

2. ***Preventive Actions:*** This program is a condition monitoring program; thus, there are no preventive actions. However, minimizing reactor coolant leakage by frequent monitoring of the locations where potential leakage could occur and timely repair if leakage is detected prevents or mitigates boric acid corrosion.

3. ***Parameters Monitored/Inspected:*** The aging management program monitors the aging effects of loss of material due to boric acid corrosion on the intended function of an affected

structure and component by detection of borated water leakage. Borated water leakage results in deposits of white boric acid crystals and the presence of moisture that can be observed by visual examination. Boric acid deposits, borated water leakage, or the presence of moisture that could lead to the identification of loss of material can be monitored through visual examination.

4. ***Detection of Aging Effects:*** Degradation of the component due to boric acid corrosion cannot occur without leakage of borated water. Conditions leading to boric acid corrosion, such as crystal buildup and evidence of moisture, are readily detectable by visual inspection, though removal of insulation may be required in some cases. However, for leakage examinations of components with external insulation surfaces and joints under insulation or not visible for direct visual examination, the surrounding area (including the floor, equipment surfaces, and other areas where leakage may be channeled) is examined for evidence of component leakage. Discoloration, staining, boric acid residue, and other evidence of leakage on insulation surfaces and the surrounding area are given particular consideration as evidence of component leakage. If evidence of leakage is found, removal of insulation to determine the exact source may be required. The program delineated in NRC GL 88-05 includes guidelines for locating small leaks, conducting examinations, and performing engineering evaluations. In addition, the program includes appropriate interfaces with other site programs and activities, such that borated water leakage that is encountered by means other than the monitoring and trending established by this program is evaluated and corrected. Thus, the use of the NRC GL 88-05 program assures detection of leakage before the loss of the intended function of the affected components.

5. ***Monitoring and Trending:*** The program provides monitoring and trending activities as delineated in NRC GL 88-05, timely evaluation of evidence of borated water leakage identified by other means, and timely detection of leakage by observing boric acid crystals during normal plant walkdowns and maintenance.

6. ***Acceptance Criteria*:** Any detected borated water leakage, white or discolored crystal buildup, or rust-colored deposits are evaluated to confirm or restore the intended functions of affected structures and components consistent with the design basis prior to continued service.

7. ***Corrective Actions*:** The NRC finds that the requirements of 10 CFR Part 50, Appendix B, with additional consideration of the guidance in NRC GL 88-05, are acceptable to implement the corrective actions related to this program. Borated water leakage and areas of resulting boric acid corrosion are evaluated and corrected in accordance with the applicable provisions of NRC GL 88-05 and the corrective action program. Any detected boric acid crystal buildup or deposits should be cleaned. NRC GL 88-05 recommends that corrective actions to prevent recurrences of degradation caused by borated water leakage be included in the program implementation. These corrective actions include any modifications to be introduced in the present design or operating procedures of the plant that (a) reduce the probability of primary coolant leaks at locations where they may cause corrosion damage and (b) entail the use of suitable corrosion resistant materials or the application of protective coatings or claddings.

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 for GALL, the

staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process and administrative controls.

9. ***Administrative Controls:*** The administrative controls for this program provide for a formal review and approval of corrective actions. The administrative controls for this program are implemented through the site's QA program in accordance with the requirements of 10 CFR Part 50, Appendix B.

10. ***Operating Experience:*** Boric acid corrosion has been observed in nuclear power plants (NRC Information Notice [IN] 86-108 [and supplements 1 through 3] and NRC IN 2003-02) and has resulted in significant impairment of component-intended functions in areas that are difficult to access/observe (NRC Bulletin 2002-01).

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XI.M11B CRACKING OF NICKEL-ALLOY COMPONENTS AND LOSS OF MATERIAL DUE TO BORIC ACID-INDUCED CORROSION IN REACTOR COOLANT PRESSURE BOUNDARY COMPONENTS (PWRs ONLY)

Program Description

This program replaces AMPs XI.M11, “Nickel-Alloy Nozzles and Penetrations” and XI.M11A, “Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of Pressurized Water Reactors.” It addresses the issue of cracking of nickel-alloy components and loss of material due to boric acid-induced corrosion in susceptible, safety-related components in the vicinity of nickel-alloy reactor coolant pressure boundary components. A final rule (September 2008) updating 10 CFR 50.55a requires the following American Society of Mechanical Engineer (ASME) Boiler and Pressure Vessel (B&PV) Code Cases: (a) N-722, “Additional Examinations for PWR Pressure Retaining Welds in Class 1 Components Fabricated with Alloy 600/82/182 Materials, Section XI, Division 1” to establish long-term inspection requirements for the pressurized water reactor (PWR) vessel, steam generator, pressurizer components and piping if they contain the primary water stress corrosion cracking (PWSCC) susceptible materials designated alloys 600/82/182; and (b) N-729-1, “Alternative Examination Requirements for PWR Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds, Section XI, Division 1” to establish new requirements for the long-term inspection of reactor pressure vessel upper heads.

In addition, dissimilar metal welds need additional examinations to provide reasonable assurance of structural integrity. The U.S. Nuclear Regulatory Commission (NRC) issued Regulatory Information Summary (RIS) 2008-25, “Regulatory Approach for Primary Water Stress Corrosion Cracking (PWSCC) of Dissimilar Metal Butt Welds in Pressurized Water Reactor Primary Coolant System Piping” (October 2008) which stated the regulatory approach for addressing PWSCC of dissimilar metal butt welds. The RIS documents the NRC’s approach to ensuring the integrity of primary coolant system piping containing dissimilar metal butt welds in PWRs and, in conjunction with the mandated inspections of ASME Code Case N-722, ensures that augmented in-service inspections (ISI) of all nickel-based alloy components and welds in the reactor coolant system (RCS) continue to perform their intended functions.

As stated in this RIS, the NRC has found that MRP-139, “Primary System Piping Butt Weld Inspection and Evaluation Guideline” (2005), and MRP interim guidance letters provide adequate protection of public health and safety for addressing PWSCC in dissimilar metal butt welds pending the incorporation of ASME Code Case N-770, containing comprehensive inspection requirements, into 10 CFR 50.55a. It is the intention of the NRC to replace MRP-139 by incorporating the requirements of ASME Code Case N-770 into 10 CFR 50.55a.

The impacts of boric acid leakage from non-nickel alloy reactor coolant pressure boundary components are addressed in AMP XI.M10, “Boric Acid Corrosion.” The Water Chemistry program for PWRs relies on monitoring and control of reactor water chemistry based on industry guidelines as described in AMP XI.M2, “Water Chemistry.”

Evaluation and Technical Basis

1. ***Scope of Program:*** The program is focused on managing the effects of cracking due to PWSCC of all susceptible nickel alloy-based components of the reactor coolant pressure boundary (including nickel-alloy welds). The program also manages the loss of material due to boric acid corrosion in susceptible components in the vicinity of nickel-alloy components.

These components could include, but are not limited to, the reactor vessel components (reactor pressure vessel upper head), steam generator components (nozzle-to-pipe connections, instrument connections, and drain tube penetrations), pressurizer components (nozzle-to-pipe connections, instrument connections, and heater penetrations), and reactor coolant system piping (instrument connections and full penetration welds).

2. ***Preventive Actions:*** This program is a condition monitoring program and does not include preventive or mitigative measures. However, maintaining high water purity reduces susceptibility to PWSCC. Reactor coolant water chemistry is monitored and maintained in accordance with the Water Chemistry program. The program description and the evaluation and technical basis of monitoring and maintaining reactor water chemistry are presented in GALL AMP XI.M2, "Water Chemistry."

   At the discretion of the applicant, preventive actions to mitigate PWSCC may be addressed by various measures (e.g., weld overlays, replacement of components with more PWSCC-resistant materials, etc.).

3. ***Parameters Monitored/Inspected:*** This is a condition monitoring program that monitors cracking/PWSCC for nickel-alloy components and loss of material by boric acid corrosion for potentially affected steel component. Reactor coolant pressure boundary cracking and leakage are monitored by the applicant's in-service inspection program in accordance with 10 CFR 50.55a and industry guidelines (e.g., MRP-139). Boric acid deposits, borated water leakage, or the presence of moisture that could lead to the identification of cracking or loss of material can be monitored through visual examination.

4. ***Detection of Aging Effects:*** The program detects the effect of aging by various methods, including non-destructive examination techniques. Reactor coolant pressure boundary leakage can be monitored through the use of radiation air monitoring and other general area radiation monitoring, and technical specifications for reactor coolant pressure boundary leakage. The specific types of non-destructive examinations are dependent on the component's susceptibility to PWSCC and its accessibility to inspection. Inspection methods, schedules, and frequencies for the susceptible components are implemented in accordance with 10 CFR 50.55a and industry guidelines (e.g., MRP-139).

5. ***Monitoring and Trending:*** Reactor coolant pressure boundary leakage is calculated and trended on a routine basis in accordance with technical specification to detect changes in the leakage rates. Flaw evaluation through 10 CFR 50.55a is a means to monitor cracking.

6. ***Acceptance Criteria:*** Acceptance criteria for all indications of cracking and loss of material due to boric acid-induced corrosion are defined in 10 CFR 50.55a and industry guidelines (e.g., MRP-139).

7. ***Corrective Actions:*** Relevant flaw indications of susceptible components within the scope of this program found to be unacceptable for further services are corrected through implementation of appropriate repair or replacement as dictated by 10 CFR 50.55a and industry guidelines (e.g., MRP-139). In addition, detection of leakage or evidence of cracking in susceptible components within the scope of this program require scope expansion of current inspection and increased inspection frequencies of some components, as required by 10 CFR 50.55a and industry guidelines (e.g., MRP-139).

Repair and replacement procedures and activities must either comply with ASME Section XI, as incorporated in 10 CFR 50.55a or conform to applicable ASME Code Cases that have been endorsed in 10 CFR 50.55a by referencing the latest version of NRC Regulatory Guide 1.147.

As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

8. ***Confirmation Process:*** Site quality assurance procedures and review and approval processes are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address confirmation process.

9. ***Administrative Controls:*** As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the administrative controls.

10. ***Operating Experience:*** This new program addresses reviews of related operating experience, including plant-specific information, generic industry findings, and international data. Within the current regulatory requirements, as necessary, the applicant maintains a record of operating experience through the required update of the facility's inservice inspection program in accordance with 10 CFR 50.55a. Additionally, the applicant follows mandated industry guidelines developed to address operating experience in accordance with NEI-03-08, "Guideline for the Management of Materials Issues."

Cracking of Alloy 600 has occurred in domestic and foreign PWRs (NRC Information Notice [IN] 90-10). Furthermore, ingress of demineralizer resins also has occurred in operating plants (NRC IN 96-11). The Water Chemistry program, AMP XI.M2, manages the effects of such excursions through monitoring and control of primary water chemistry. NRC GL 97-01 is effective in managing the effect of PWSCC. PWSCC also is occurring in the vessel head penetration (VHP) nozzle of U.S. PWRs as described in NRC Bulletins 2001-01, 2002-01 and 2002-02.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

10 CFR Part 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2009.

ASME Code Case N-722, *Additional Examinations for PWR Pressure Retaining Welds in Class 1 Components Fabricated with Alloy 600/82/182 Materials*, July 5, 2005.

ASME Code Case N-729-1, *Alternative Examination Requirements for PWR Reactor Vessel Upper Heads with Nozzles Having Pressure-Retaining Partial-Penetration Welds*, March 28, 2006.

ASME Code Case N-770, *Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds Fabricated with UNS N06082 or UNS*

W86182 Weld Filler Material With or Without Application of Listed Mitigation Activities, January 26, 2009.

MRP-139, Revision 1, *Primary System Piping Butt Weld Inspection and Evaluation Guideline*, Materials Reliability Program, December 16, 2008.

NEI-03-08, *Guideline for the Management of Materials Issues,* Nuclear Energy Institute, May 2003.

NRC Bulletin 2001-01, *Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles*, U.S. Nuclear Regulatory Commission, August 3, 2001.

NRC Bulletin 2002-01, *Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity*, U.S. Nuclear Regulatory Commission, March 18, 2002.

NRC Bulletin 2002-02, *Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs*, U.S. Nuclear Regulatory Commission, August 9, 2002.

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 90-10, *Primary Water Stress Corrosion Cracking (PWSCC) of Inconel 600*, U.S. Nuclear Regulatory Commission, February 23, 1990.

NRC Information Notice 96-11, *Ingress of Demineralizer Resins Increases Potential for Stress Corrosion Cracking of Control Rod Drive Mechanism Penetrations*, U.S. Nuclear Regulatory Commission, February 14, 1996.

NRC Inspection Manual, Inspection Procedure 71111.08, *Inservice Inspection Activities*, March 23, 2009.

NRC Inspection Manual, Temporary Instruction 2515/172, *Reactor Coolant System Dissimilar Metal Butt Welds*, February 21, 2008.

NRC Regulatory Guide 1.147, Revision 15, *Inservice Inspection Code Case Acceptability*, ASME Section XI, Division 1, U.S. Nuclear Regulatory Commission, January 2004.

NRC Regulatory Information Summary 2008-25, *Regulatory Approach for Primary Water Stress Corrosion Cracking of Dissimilar Metal Butt Welds in Pressurized Water Reactor Primary Coolant System Piping*, U.S. Nuclear Regulatory Commission, October 22, 2008.

NUREG-1823, *U.S. Plant Experience with Alloy 600 Cracking and Boric Acid Corrosion of Light-Water Reactor Pressure Vessel Materials*, U.S. Nuclear Regulatory Commission, April 2005.

XI.M12 THERMAL AGING EMBRITTLEMENT OF CAST AUSTENITIC STAINLESS STEEL (CASS)

Program Description

The reactor coolant system components are inspected in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI. This inspection is augmented to detect the effects of loss of fracture toughness due to thermal aging embrittlement of cast austenitic stainless steel (CASS) piping components except for pump casings and valve bodies. This aging management program (AMP) includes determination of the susceptibility of CASS components to thermal aging embrittlement based on casting method, molybdenum (Mo) content, and percent ferrite. For "potentially susceptible" components, as defined below, aging management is accomplished through either (a) qualified visual inspections, such as enhanced visual examination (EVT-1); (b) a qualified ultrasonic testing (UT) methodology; or (c) a component-specific flaw tolerance evaluation in accordance with the ASME Code, Section XI, 2004 edition.[10] Additional inspection or evaluations to demonstrate that the material has adequate fracture toughness are not required for components that are not susceptible to thermal aging embrittlement.

For pump casings and valve bodies, based on the results of the assessment documented in the letter dated May 19, 2000, from Christopher Grimes, Nuclear Regulatory Commission (NRC), to Douglas Walters, Nuclear Energy Institute (NEI) (May 19, 2000 NRC letter), screening for susceptibility to thermal aging embrittlement is not required. The existing ASME Code, Section XI inspection requirements, including the alternative requirements of ASME Code Case N-481 for pump casings, are adequate for all pump casings and valve bodies.

Aging management of CASS reactor internal components of pressurized water reactors (PWRs) are discussed in AMP XI.M16A and of CASS reactor internal components of boiling water reactors (BWRs) in AMP XI.M9.

Evaluation and Technical Basis

1. ***Scope of Program:*** This program manages loss of fracture toughness in potentially susceptible ASME Code Class 1 piping components made from CASS. The program includes screening criteria to determine which CASS components are potentially susceptible to thermal aging embrittlement and require augmented inspection. The screening criteria are applicable to all primary pressure boundary components constructed from cast austenitic stainless steel with service conditions above 250°C (482°F). The screening criteria for susceptibility to thermal aging embrittlement are not applicable to niobium-containing steels; such steels require evaluation on a case-by-case basis.

   Based on the criteria set forth in the May 19, 2000, NRC letter, the susceptibility to thermal aging embrittlement of CASS materials is determined in terms of casting method, molybdenum content, and ferrite content. For low-molybdenum content steels (SA-351 Grades CF3, CF3A, CF8, CF8A or other steels with ≤ 0.5 weight percent [wt.%] Mo), only static-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement. Static-cast low-molybdenum steels with ≤20% ferrite and all centrifugal-cast low-molybdenum steels are not susceptible. For high-molybdenum content steels (SA-351 Grades CF3M, CF3MA, and CF8M or other steels with 2.0 to 3.0 wt.% Mo), static-cast

[10] Refer to the GALL Report, Chapter I, for applicability of other editions of ASME Code, Section XI.

steels with >14% ferrite and centrifugal-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement. Static-cast high-molybdenum steels with ≤14% ferrite and centrifugal-cast high-molybdenum steels with ≤20% ferrite are not susceptible. In the susceptibility screening method, ferrite content is calculated by using the Hull's equivalent factors (described in NUREG/CR-4513, Rev. 1) or a staff-approved method for calculating delta ferrite in CASS materials. A fracture toughness value of 255 kilojoules per square meter (kJ/m^2) (1,450 inches-pounds per square inch) at a crack depth of 2.5 millimeters (0.1 inch) is used to differentiate between CASS materials that are not susceptible and those that are potentially susceptible to thermal aging embrittlement. Extensive research data indicate that for CASS materials not susceptible to thermal aging embrittlement, the saturated lower-bound fracture toughness is greater than 255 kJ/m^2 (NUREG/CR-4513, Rev. 1).

For pump casings and valve bodies, screening for susceptibility to thermal aging embrittlement is not needed (and thus there are no aging management review line items). For all pump casings and valve bodies greater than a nominal pipe size (NPS) of 4 inches, the existing ASME Code, Section XI inspection requirements, including the alternative requirements of ASME Code Case N-481 for pump casings, are adequate. ASME Code, Section XI, Subsection IWB requires only surface examination of valve bodies less than a NPS of 4 inches. For these valve bodies less than a NPS of 4 inches, the adequacy of inservice inspection (ISI) according to ASME Code, Section XI has been demonstrated by an NRC-performed bounding integrity analysis (May 19, 2000 letter).

2. ***Preventive Actions:*** This program is a condition monitoring program and does not mitigate thermal aging embrittlement.

3. ***Parameters Monitored/Inspected:*** The program monitors the effects of loss of fracture toughness on the intended function of the component by identifying the CASS materials that are susceptible to thermal aging embrittlement.

   The program does not directly monitor for loss of fracture toughness that is induced by thermal aging; instead, the impact of loss of fracture toughness on component integrity is indirectly managed by using visual or volumetric examination techniques to monitor for cracking in the components.

4. ***Detection of Aging Effects:*** For pump casings, valve bodies, and other "not susceptible" CASS piping components, no additional inspection or evaluations are needed to demonstrate that the material has adequate fracture toughness.

   For "potentially susceptible" piping components, the AMP provides for qualified inspections of the base metal, such as enhanced visual examination (EVT-1) or a qualified UT methodology, with the scope of the inspection covering the portions determined to be limiting from the standpoint of applied stress, operating time, and environmental considerations. Examination methods that meet the criteria of the ASME Code, Section XI, Appendix VIII are acceptable. Alternatively, a plant-specific or component-specific flaw tolerance evaluation, using specific geometry, stress information, material properties, and ASME Code, Section XI can be used to demonstrate that the thermally-embrittled material has adequate toughness. Current UT methodology cannot detect and size cracks; thus, EVT-1 is used until qualified UT methodology for CASS can be established. A description of EVT-1 is found in Boiling Water Reactor Vessel and Internals Project (BWRVIP)-03 (Revision 6) and Materials Reliability Program (MRP)-228 for PWRs.

5. ***Monitoring and Trending:*** Inspection schedules in accordance with ASME Code, Section XI, IWB-2400 or IWC-2400, reliable examination methods, and qualified inspection personnel provide timely and reliable detection of cracks. If flaws are detected, the period of acceptability is determined from analysis of the flaw, depending on the crack growth rate and mechanism.

6. ***Acceptance Criteria:*** Flaws detected in CASS components are evaluated in accordance with the applicable procedures of ASME Code, Section XI, IWB-3500 or ASME Code, Section XI, IWC-3500. Flaw tolerance evaluation for components with ferrite content up to 25% is performed according to the principles associated with ASME Code, Section XI, IWB-3640 procedures for SAWs, disregarding the ASME Code restriction of 20% ferrite. Extensive research data indicates that the lower-bound fracture toughness of thermally aged CASS materials with up to 25% ferrite is similar to that for SAWs with up to 20% ferrite (Lee et al., 1997). Flaw tolerance evaluation for piping with >25% ferrite is performed on a case-by-case basis by using the applicant’s fracture toughness data.

7. ***Corrective Actions:*** Repair and replacement are performed in accordance with ASME Code, Section XI, IWA-4000. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B acceptable to address the corrective actions.

8. ***Confirmation Process:*** Site quality assurance 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B acceptable to address the confirmation process and administrative controls.

9. ***Administrative Controls:*** The administrative controls for this program provide for a formal review and approval of corrective actions. The administrative controls for this program are implemented through the site's QA program in accordance with the requirements of 10 CFR Part 50, Appendix B.

10. ***Operating Experience:*** The AMP was developed by using research data obtained on both laboratory-aged and service-aged materials. Based on this information, the effects of thermal aging embrittlement on the intended function of CASS components will be effectively managed.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

10 CFR Part 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2009.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, The ASME Boiler and Pressure Vessel Code, 2004 edition as approved in 10 CFR 50.55a, The American Society of Mechanical Engineers, New York, NY.

ASME Code Case N-481, *Alternative Examination Requirements for Cast Austenitic Pump Casings*, Section XI, Division 1.

BWRVIP-03, Rev. 6, *BWR Vessel and Internals Project: Reactor Pressure Vessel and Internals Examination Guidelines* (EPRI TR-105696).

Lee, S., Kuo, P. T., Wichman, K., and Chopra, O., *Flaw Evaluation of Thermally-Aged Cast Stainless Steel in Light-Water Reactor Applications*, Int. J. Pres. Vessel and Piping, pp 37-44, 1997.

Letter from Christopher I. Grimes, U.S. Nuclear Regulatory Commission, License Renewal and Standardization Branch, to Douglas J. Walters, Nuclear Energy Institute, License Renewal Issue No. 98-0030, *Thermal Aging Embrittlement of Cast Stainless Steel Components*, May 19, 2000. (ADAMS Accession No. ML003717179)

Letter from Mark J. Maxin, to Rick Libra (BWRVIP Chairman), Safety Evaluation for Electric Power Research Institute (EPRI) Boiling Water Reactor Vessel and Internals project (BWRVIP) Report TR-105696-R6 (BWRVIP-03), Revision 6, BWR Vessel and Internals Examination Guidelines (TAC No MC2293)," June 30, 2008 (ADAMS Accession No ML081500814)

MRP-228, *Materials Reliability Program: Inspection Standard for PWR Internals,* 2009.

NUREG/CR-4513, Rev. 1, *Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems*, U.S. Nuclear Regulatory Commission, August 1994.

XI.M16A PWR VESSEL INTERNALS

Program Description

This program relies on implementation of the Electric Power Research Institute (EPRI) Report No. 1016596 (MRP-227) and EPRI Report No. 1016609 (MRP-228) to manage the aging effects on the reactor vessel internal (RVI) components.

This program is used to manage the effects of age-related degradation mechanisms that are applicable in general to the PWR RVI components at the facility. These aging effects include (a) various forms of cracking, including stress corrosion cracking (SCC), which also encompasses primary water stress corrosion cracking (PWSCC), irradiation-assisted stress corrosion cracking (IASCC), or cracking due to fatigue/cyclical loading; (b) loss of material induced by wear; (c) loss of fracture toughness due to either thermal aging or neutron irradiation embrittlement; (d) changes in dimension due to void swelling; and (e) loss of preload due to thermal and irradiation-enhanced stress relaxation or creep.

The program applies the guidance in MRP-227 for inspecting, evaluating, and, if applicable, dispositioning non-conforming RVI components at the facility. The program conforms to the definition of a sampling-based condition monitoring program, as defined by the Branch Technical Position RSLB-1, with periodic examinations and other inspections of highly-affected internals locations. These examinations provide reasonable assurance that the effects of age-related degradation mechanisms will be managed during the period of extended operation. The program includes expanding periodic examinations and other inspections if the extent of the degradation effects exceeds the expected levels.

The MRP-227 guidance for selecting RVI components for inclusion in the inspection sample is based on a four-step ranking process. Through this process, the reactor internals for all three PWR designs were assigned to one of the following four groups: Primary, Expansion, Existing Programs, and No Additional Measures components. Definitions of each group are provided in GALL Chapter IX.B.

The result of this four-step sample selection process is a set of Primary Internals Component locations for each of the three plant designs that are expected to show the leading indications of the degradation effects, with another set of Expansion Internals Component locations that are specified to expand the sample should the indications be more severe than anticipated. The degradation effects in a third set of internals locations are deemed to be adequately managed by Existing Programs, such as ASME Code, Section XI,[11] Examination Category B-N-3 examinations of core support structures. A fourth set of internals locations are deemed to require no additional measures. As a result, the program typically identifies 5 to 15% of the RVI locations as Primary Component locations for inspections, with another 7 to 10% of the RVI locations to be inspected as Expansion Components, as warranted by the evaluation of the inspection results. Another 5 to 15% of the internals locations are covered by Existing Programs, with the remainder requiring no additional measures. This process thus uses appropriate component functionality criteria, age-related degradation susceptibility criteria, and failure consequence criteria to identify the components that will be inspected under the program in a manner that conforms to the sampling criteria for sampling-based condition monitoring programs in Section A.1.2.3.4 of NRC Branch Position RLSB-1. Consequently, the sample

[11] Refer to the GALL Report, Chapter I, for applicability of various editions of the ASME Code, Section XI.

selection process is adequate to assure that the intended function(s) of the PWR reactor internal components are maintained during the period of extended operation.

The program's use of visual examination methods in MRP-227 for detection of relevant conditions (and the absence of relevant conditions as a visual examination acceptance criterion) is consistent with the ASME Code, Section XI rules for visual examination. However, the program's adoption of the MRP-227 guidance for visual examinations goes beyond the ASME Code, Section XI visual examination criteria because additional guidance is incorporated into MRP-227 to clarify how the particular visual examination methods will be used to detect relevant conditions and describes in more detail how the visual techniques relate to the specific RVI components and how to detect their applicable age-related degradation effects.

The technical basis for detecting relevant conditions using volumetric ultrasonic testing (UT) inspection techniques can be found in MRP-228, where the review of existing bolting UT examination technical justifications has demonstrated the indication detection capability of at least two vendors, and where vendor technical justification is a requirement prior to any additional bolting examinations. Specifically, the capability of program's UT volumetric methods to detect loss of integrity of PWR internals bolts, pins, and fasteners, such as baffle-former bolting in B&W and Westinghouse units, has been well demonstrated by operating experience. In addition, the program's adoption of the MRP-227 guidance and process incorporates the UT criteria in MRP-228, which calls for the technical justifications that are needed for volumetric examination method demonstrations, required by the ASME Code, Section V.

The program also includes future industry operating experience as incorporated in periodic revisions to MRP-227. The program thus provides reasonable assurance for the long-term integrity and safe operation of reactor internals in all commercial operating U.S. PWR nuclear power plants.

Age-related degradation in the reactor internals is managed through an integrated program. Specific features of the integrated program are listed in the following ten program elements. Degradation due to changes in material properties (e.g., loss of fracture toughness) was considered in the determination of inspection recommendations and is managed by the requirement to use appropriately degraded properties in the evaluation of identified defects. The integrated program is implemented by the applicant through an inspection plan that is submitted to the NRC for review and approval with the application for license renewal.

Evaluation and Technical Basis

1. ***Scope of Program:*** The scope of the program includes all RVI components at the [*as an administrative action item for the AMP, the applicant to fill in the name of the applicant's nuclear facility, including applicable units*], which [*is/are*] built to a [*applicant to fill in Westinghouse, CE, or B&W, as applicable*] NSSS design. The scope of the program applies the methodology and guidance in the most recently NRC-endorsed version of MRP-227, which provides augmented inspection and flaw evaluation methodology for assuring the functional integrity of safety-related internals in commercial operating U.S. PWR nuclear power plants designed by B&W, CE, and Westinghouse. The scope of components considered for inspection under MRP-227 guidance includes core support structures (typically denoted as Examination Category B-N-3 by the ASME Code, Section XI), those RVI components that serve an intended license renewal safety function pursuant to criteria in 10 CFR 54.4(a)(1), and other RVI components whose failure could prevent satisfactory accomplishment of any of the functions identified in 10 CFR 54.4(a)(1)(i), (ii), or (iii). The

scope of the program does not include consumable items, such as fuel assemblies, reactivity control assemblies, and nuclear instrumentation, because these components are not typically within the scope of the components that are required to be subject to an aging management review (AMR), as defined by the criteria set in 10 CFR 54.21(a)(1). The scope of the program also does not include welded attachments to the internal surface of the reactor vessel because these components are considered to be ASME Code Class 1 appurtenances to the reactor vessel and are adequately managed in accordance with an applicant's AMP that corresponds to GALL AMP XI.M1, "ASME Code, Section XI Inservice Inspection, Subsections IWB, IWC, and IWD."

The scope of the program includes the response bases to applicable license renewal applicant action items (LRAAIs) on the MRP-227 methodology, and any additional programs, actions, or activities that are discussed in these LRAAI responses and credited for aging management of the applicant's RVI components. The LRAAIs are identified in the staff's safety evaluation on MRP-227 and include applicable action items on meeting those assumptions that formed the basis of the MRP's augmented inspection and flaw evaluation methodology (as discussed in Section 2.4 of MRP-227), and NSSS vendor-specific or plant-specific LRAAIs as well. The responses to the LRAAIs on MRP-227 are provided in Appendix C of the LRA.

The guidance in MRP-227 specifies applicability limitations to base-loaded plants and the fuel loading management assumptions upon which the functionality analyses were based. These limitations and assumptions require a determination of applicability by the applicant for each reactor and are covered in Section 2.4 of MRP-227.

2. ***Preventive Actions:*** The guidance in MRP-227 relies on PWR water chemistry control to prevent or mitigate aging effects that can be induced by corrosive aging mechanisms (e.g., loss of material induced by general, pitting corrosion, crevice corrosion, or stress corrosion cracking or any of its forms [SCC, PWSCC, or IASCC]). Reactor coolant water chemistry is monitored and maintained in accordance with the Water Chemistry Program. The program description, evaluation, and technical basis of water chemistry are presented in GALL AMP XI.M2, "Water Chemistry."

3. ***Parameters Monitored/Inspected:*** The program manages the following age-related degradation effects and mechanisms that are applicable in general to the RVI components at the facility: (a) cracking induced by SCC, PWSCC, IASCC, or fatigue/cyclical loading; (b) loss of material induced by wear; (c) loss of fracture toughness induced by either thermal aging or neutron irradiation embrittlement; (d) changes in dimension due to void swelling and irradiation growth, distortion, or deflection; and (e) loss of preload caused by thermal and irradiation-enhanced stress relaxation or creep. For the management of cracking, the program monitors for evidence of surface breaking linear discontinuities if a visual inspection technique is used as the non-destruction examination (NDE) method, or for relevant flaw presentation signals if a volumetric UT method is used as the NDE method. For the management of loss of material, the program monitors for gross or abnormal surface conditions that may be indicative of loss of material occurring in the components. For the management of loss of preload, the program monitors for gross surface conditions that may be indicative of loosening in applicable bolted, fastened, keyed, or pinned connections. The program does not directly monitor for loss of fracture toughness that is induced by thermal aging or neutron irradiation embrittlement, or by void swelling and irradiation growth; instead, the impact of loss of fracture toughness on component integrity is indirectly managed by using visual or volumetric examination techniques to monitor for cracking in the

components and by applying applicable reduced fracture toughness properties in the flaw evaluations if cracking is detected in the components and is extensive enough to warrant a supplemental flaw growth or flaw tolerance evaluation under the MRP-227 guidance or ASME Code, Section XI requirements. The program uses physical measurements to monitor for any dimensional changes due to void swelling, irradiation growth, distortion, or deflection.

Specifically, the program implements the parameters monitored/inspected criteria for [*as an administrative action item for the AMP, applicant is to select one of the following to finish the sentence, as applicable to its NSSS vendor for its internals: "for B&W designed Primary Components in Table 4-1 of MRP-227"; "for CE designed Primary Components in Table 4-2 of MRP-227"; and "for Westinghouse designed Primary Components in Table 4-3 of MRP-227"*]. Additionally, the program implements the parameters monitored/inspected criteria for [*as an administrative action item for the AMP, applicant is to select one of the following to finish the sentence, as applicable to its NSSS vendor for its internals: "for B&W designed Expansion Components in Table 4-4 of MRP-227"; "for CE designed Expansion Components in Table 4-5 of MRP-227"; and "for Westinghouse designed Expansion Components in Table 4-6 of MRP-227"*]. The parameters monitored/inspected for Existing Program Components follow the bases for referenced Existing Programs, such as the requirements for ASME Code Class RVI components in ASME Code, Section XI, Table IWB-2500-1, Examination Categories B-N-3, as implemented through the applicant's ASME Code, Section XI program, or the recommended program for inspecting Westinghouse-designed flux thimble tubes in GALL AMP XI.M37, "Flux Thimble Tube Inspection." No inspections, except for those specified in ASME Code, Section XI, are required for components that are identified as requiring "No Additional Measures," in accordance with the analyses reported in MRP-227.

4. ***Detection of Aging Effects:*** The detection of aging effects is covered in two places: (a) the guidance in Section 4 of MRP-227 provides an introductory discussion and justification of the examination methods selected for detecting the aging effects of interest; and (b) standards for examination methods, procedures, and personnel are provided in a companion document, MRP-228. In all cases, well-established methods were selected. These methods include volumetric UT examination methods for detecting flaws in bolting, physical measurements for detecting changes in dimension, and various visual (VT-3, VT-1, and EVT-1) examinations for detecting effects ranging from general conditions to detection and sizing of surface-breaking discontinuities. Surface examinations may also be used as an alternative to visual examinations for detection and sizing of surface-breaking discontinuities.

   Cracking caused by SCC, IASCC, and fatigue is monitored/inspected by either VT-1 or EVT-1 examination (for internals other than bolting) or by volumetric UT examination (bolting). The VT-3 visual methods may be applied for the detection of cracking only when the flaw tolerance of the component or affected assembly, as evaluated for reduced fracture toughness properties, is known and has been shown to be tolerant of easily detected large flaws, even under reduced fracture toughness conditions. In addition, VT-3 examinations are used to monitor/inspect for loss of material induced by wear and for general aging conditions, such as gross distortion caused by void swelling and irradiation growth or by gross effects of loss of preload caused by thermal and irradiation-enhanced stress relaxation and creep.

   In addition, the program adopts the recommended guidance in MRP-227 for defining the Expansion criteria that need to be applied to inspections of Primary Components and

Existing Requirement Components and for expanding the examinations to include additional Expansion Components. As a result, inspections performed on the RVI components are performed consistent with the inspection frequency and sampling bases for Primary Components, Existing Requirement Components, and Expansion Components in MRP-227, which have been demonstrated to be in conformance with the inspection criteria, sampling basis criteria, and sample Expansion criteria in Section A.1.2.3.4 of NRC Branch Position RLSB-1.

Specifically, the program implements the parameters monitored/inspected criteria and bases for inspecting the relevant parameter conditions for [*as an administrative action item for the AMP, applicant is to select one of the following to finish the sentence, as applicable to its NSSS vendor for its internals: "B&W designed Primary Components in Table 4-1 of MRP-227"; "CE designed Primary Components in Table 4-2 of MRP-227;" or "Westinghouse designed Primary Components in Table 4-3 of MRP-227"*] and for [*as an administrative action item for the AMP, applicant is to select one of the following to finish the sentence, as applicable to its NSSS vendor for its internals: "for B&W designed Expansion Components in Table 4-4 of MRP-227;" "for CE designed expansion components in Table 4-5 of MRP-227;" and "for Westinghouse designed Expansion Components in Table 4-6 of MRP-227"*].

The program is supplemented by the following plant-specific Primary Component and Expansion Component inspections for the program (as applicable): [*As a relevant license renewal applicant action item, the applicant is to list (using criteria in MRP-227) each additional RVI component that needs to be inspected as an additional plant-specific Primary Component for the applicant's program and each additional RVI component that needs to be inspected as an additional plant-specific Expansion Component for the applicant's program. For each plant specific component added as an additional primary or Expansion Component, the list should include the applicable aging effects that will be monitored for, the inspection method or methods used for monitoring, and the sample size and frequencies for the examinations*].

In addition, in some cases (as defined in MRP-227), physical measurements are used as supplemental techniques to manage for the gross effects of wear, loss of preload due to stress relaxation, or for changes in dimension due to void swelling, deflection or distortion. The physical measurements methods applied in accordance with this program include [*Applicant to input physical measure methods identified by the MRP in response to NRC RAI No. 11 in the NRC's Request for Additional Information to Mr. Christen B. Larson, EPRI MRP on Topical Report MRP-227 dated November 12, 2009*].

5. ***Monitoring and Trending:*** The methods for monitoring, recording, evaluating, and trending the data that result from the program's inspections are given in Section 6 of MRP-227 and its subsections. The evaluation methods include recommendations for flaw depth sizing and for crack growth determinations as well for performing applicable limit load, linear elastic and elastic-plastic fracture analyses of relevant flaw indications. The examinations and re-examinations required by the MRP-227 guidance, together with the requirements specified in MRP-228 for inspection methodologies, inspection procedures, and inspection personnel, provide timely detection, reporting, and corrective actions with respect to the effects of the age-related degradation mechanisms within the scope of the program. The extent of the examinations, beginning with the sample of susceptible PWR internals component locations identified as Primary Component locations, with the potential for inclusion of Expansion Component locations if the effects are greater than anticipated, plus the continuation of the Existing Programs activities, such as the ASME Code, Section XI, Examination Category B-

N-3 examinations for core support structures, provides a high degree of confidence in the total program.

6. ***Acceptance Criteria:*** Section 5 of MRP-227 provides specific examination acceptance criteria for the Primary and Expansion Component examinations. For components addressed by examinations referenced to ASME Code, Section XI, the IWB-3500 acceptance criteria apply. For other components covered by Existing Programs, the examination acceptance criteria are described within the Existing Program reference document.

   The guidance in MRP-227 contains three types of examination acceptance criteria:

   - For visual examination (and surface examination as an alternative to visual examination), the examination acceptance criterion is the absence of any of the specific, descriptive relevant conditions; in addition, there are requirements to record and disposition surface breaking indications that are detected and sized for length by VT-1/EVT-1 examinations;

   - For volumetric examination, the examination acceptance criterion is the capability for reliable detection of indications in bolting, as demonstrated in the examination Technical Justification; in addition, there are requirements for system-level assessment of bolted or pinned assemblies with unacceptable volumetric (UT) examination indications that exceed specified limits; and

   - For physical measurements, the examination acceptance criterion for the acceptable tolerance in the measured differential height from the top of the plenum rib pads to the vessel seating surface in B&W plants are given in Table 5-1 of MRP-227. The acceptance criterion for physical measurements performed on the height limits of the Westinghouse-designed hold-down springs are [*The incorporation of this sentence is a license renewal applicant action item for Westinghouse PWR applicants only – insert the applicable sentence incorporating the specified physical measurement criteria only if the applicant's facility is based on a Westinghouse NSSS design: the Westinghouse applicant is to incorporate the applicable language and then specify the fit up limits on the hold down springs, as established on a plant-specific basis for the design of the hold-down springs at the applicant's Westinghouse-designed facility*].

7. ***Corrective Actions:*** Corrective actions following the detection of unacceptable conditions are fundamentally provided for in each plant's corrective action program. Any detected conditions that do not satisfy the examination acceptance criteria are required to be dispositioned through the plant corrective action program, which may require repair, replacement, or analytical evaluation for continued service until the next inspection. The disposition will ensure that design basis functions of the reactor internals components will continue to be fulfilled for all licensing basis loads and events. Examples of methodologies that can be used to analytically disposition unacceptable conditions are found in the ASME Code, Section XI or in Section 6 of MRP-227. Section 6 of MRP-227 describes the options that are available for disposition of detected conditions that exceed the examination acceptance criteria of Section 5 of the report. These include engineering evaluation methods, as well as supplementary examinations to further characterize the detected condition, or the alternative of component repair and replacement procedures. The latter are subject to the requirements of the ASME Code, Section XI. The implementation of the guidance in MRP-227, plus the implementation of any ASME Code requirements, provides

an acceptable level of aging management of safety-related components addressed in accordance with the corrective actions of 10 CFR Part 50, Appendix B or its equivalent, as applicable.

Other alternative corrective action bases may be used to disposition relevant conditions if they have been previously approved or endorsed by the NRC. Examples of previously NRC-endorsed alternative corrective actions bases include those corrective actions bases for Westinghouse-design RVI components that are defined in Tables 4-1, 4-2, 4-3, 4-4, 4-5, 4-6, 4-7 and 4-8 of Westinghouse Report No. WCAP-14577-Rev. 1-A, or for B&W-designed RVI components in B&W Report No. BAW-2248. Westinghouse Report No. WCAP-14577-Rev. 1-A was endorsed for use in an NRC SE to the Westinghouse Owners Group, dated February 10, 2001. B&W Report No. BAW-2248 was endorsed for use in an SE to Framatome Technologies on behalf of the B&W Owners Group, dated December 9, 1999. Alternative corrective action bases not approved or endorsed by the NRC will be submitted for NRC approval prior to their implementation.

8. ***Confirmation Process:*** Site quality assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B, or their equivalent, as applicable. It is expected that the implementation of the guidance in MRP-227 will provide an acceptable level of quality for inspection, flaw evaluation, and other elements of aging management of the PWR internals that are addressed in accordance with the 10 CFR Part 50, Appendix B, or their equivalent (as applicable), confirmation process, and administrative controls.

9. ***Administrative Controls:*** The administrative controls for such programs, including their implementing procedures and review and approval processes, are under existing site 10 CFR 50 Appendix B Quality Assurance Programs, or their equivalent, as applicable. Such a program is thus expected to be established with a sufficient level of documentation and administrative controls to ensure effective long-term implementation.

10. ***Operating Experience:*** Relatively few incidents of PWR internals aging degradation have been reported in operating U.S. commercial PWR plants. A summary of observations to date is provided in Appendix A of MRP-227-A. The applicant is expected to review subsequent operating experience for impact on its program or to participate in industry initiatives that perform this function.

    The application of the MRP-227 guidance will establish a considerable amount of operating experience over the next few years. Section 7 of MRP-227 describes the reporting requirements for these applications, and the plan for evaluating the accumulated additional operating experience.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Pla*nts, Office of the Federal Register, National Archives and Records Administration, 2009.

10 CFR Part 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2009.

ASME Boiler & Pressure Vessel Code, Section V, *Nondestructive Examination*, 2004 Edition, American Society of Mechanical Engineers, New York, NY.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, The ASME Boiler and Pressure Vessel Code, 2004 edition as approved in 10 CFR 50.55a, The American Society of Mechanical Engineers, New York, NY.

B&W Report No. BAW-2248, *Demonstration of the Management of Aging Effects for the Reactor Vessel Internals*, Framatome Technologies (now AREVA Technologies), Lynchburg VA, July 1997. (NRC Microfiche Accession Number A0076, Microfiche Pages 001 - 108).

EPRI 1014986, *PWR Primary Water Chemistry Guidelines*, Volume 1, Revision 6, Electric Power Research Institute, Palo Alto, CA, December 2007. (Non-publicly available ADAMS Accession Number ML081140278). The non-proprietary version of the report may accessed by members of the public at ADAMS Accession Number ML081230449

EPRI 1016596, *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines* (MRP-227-Rev. 0), Electric Power Research Institute, Palo Alto, CA: 2008.

EPRI 1016609, *Materials Reliability Program: Inspection Standard for PWR Internals* (MRP-228), Electric Power Research Institute, Palo Alto, CA, July 2009. (Non-publicly available ADAMS Accession Number ML092120574). The non-proprietary version of the report may accessed by members of the public at ADAMS Accession Number ML092750569.

NRC RAI No. 11 in the *NRC's Request for Additional Information* to the Mr. Christen B. Larson, EPRI MRP on Topical Report MRP-227 dated November 12, 2009.

NRC Safety Evaluation from C. I. Grimes [NRC] to R. A, Newton [Chairman, Westinghouse Owners Group], *Acceptance for Referencing of Generic License Renewal Program Topical Report Entitled "License Renewal Evaluation: Aging Management for Reactor Internals,"* WCAP-14577, Revision 1, February 10, 2001. (ADAMS Accession Number ML010430375).

NRC Safety Evaluation from C. I. Grimes [NRC] to W. R. Gray [Framatome Technologies], *Acceptance for Referencing of Generic License Renewal Program Topical Report Entitled "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals,"* February 10, 2001. (ADAMS Accession Number ML993490288).

NUREG-1800, Revision 2, *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants*, Appendix A.1, "Aging Management Review - Generic (Branch Technical Position RLSB-1)," U.S. Nuclear Regulatory Commission, Washington, DC, 2010.

Westinghouse Non-Proprietary Class 3 Report No. WCAP-14577-Rev. 1-A, *License Renewal Evaluation: Aging Management for Reactor Internals*, Westinghouse Electric Company, Pittsburgh, PA [March 2001]. Report was submitted to the NRC Document Control Desk in a letter dated April 9, 2001. (ADAMS Accession Number ML011080790).

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 or R3 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. NSAC-202L-R2 or R3 provides general guidelines for the FAC program. To provide reasonable assurance 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 or R3 to satisfy the criteria specified in 10 CFR Part 50, Appendix B, for development of procedures and control of special processes.

Evaluation and Technical Basis

1. ***Scope of Program:*** The FAC program, described by the EPRI guidelines in NSAC-202L-R2 or R3, 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 89-08.

2. ***Preventive Actions:*** The FAC program is an analysis, inspection, and verification program; no preventive action has been recommended in this program. 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 monitors the effects of loss of material due to wall thinning 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-R2 or R3 consists of identification of susceptible locations, as indicated by operating conditions or special considerations. Ultrasonic or radiographic testing is used to detect wall thinning. A representative sample of components is selected based on the most susceptible locations for wall thickness measurements at a frequency in accordance with NSAC 202L guidelines to ensure that degradation is identified and mitigated before the component integrity is challenged. The extent and schedule of the inspections ensure 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. The analysis is bounding because in general the predicted wear rates and component thicknesses are conservative when compared to actual field measurements. It is recognized that CHECWORKS is not

always conservative in predicting component thickness; therefore, when measurements show the predictions to be non-conservative, the model must be re-calibrated using the latest field data. CHECWORKS was developed and benchmarked by comparing CHECWORKS predictions against actual measured component thickness measurements 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. Inspection results are evaluated to determine if additional inspections are needed to ensure that the extent of wall thinning is adequately determined, that intended function will not be lost, and that corrective actions are adequately identified. Previous wear rate predictions due to FAC may change after a power uprate is implemented. Wear rates are updated in CHECWORKS according to power uprate conditions. Subsequent field measurements are used to calibrate or benchmark the predicted wear rates.

6. ***Acceptance Criteria:*** Inspection results are input for 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 wall thickness before the next scheduled outage, corrective action should be considered.

7. ***Corrective Actions:*** Prior to service, components for which the acceptance criteria are not satisfied are reevaluated, repaired, or replaced. Long-term corrective actions could include adjusting operating parameters or selecting materials resistant to FAC. When susceptible components are replaced with resistant materials, such as high Cr material, the downstream components should be monitored closely to mitigate any increased wear. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process.

9. ***Administrative Controls:*** Site 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the administrative controls.

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 Notice [IN] 81-28, IN 92-35, IN 95-11, IN 2006-08) and in two-phase piping in extraction steam lines (NRC IN 89-53, IN 97-84) and moisture separation reheater and feedwater heater drains (NRC IN 89-53, IN 91-18, IN 93-21, IN 97-84). Observed wall thinning may be due to mechanisms other than FAC, which require alternate materials to resolve the issue (Licensee Event Report 50-237/2007-003-00). Operating experience shows that the present program, when properly implemented, is effective in managing FAC in high-energy carbon steel piping and components.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

10 CFR Part 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2009.

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 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.

NRC Information Notice 99-19, *Rupture of the Shell Side of a Feedwater Heater at the Point Beach Nuclear Plant,* U.S. Nuclear Regulatory Commission, June 23, 1999.

NSAC-202L-R2, *Recommendations for an Effective Flow Accelerated Corrosion Program*, Electric Power Research Institute, Nuclear Safety Analysis Center, Palo Alto, CA, April 8, 1999.

NSAC-202L-R3, *Recommendations for an Effective Flow Accelerated Corrosion Program*, (1011838), Electric Power Research Institute, Nuclear Safety Analysis Center, Palo Alto, CA, May 2006.

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

NRC Information Notice 2006-08, *Secondary Piping Rupture at the Mihama Power Station in Japan*, U.S. Nuclear Regulatory Commission, March 16, 2006.

NRC Licensee Event Report 50‐237/2007‐003‐00, *Unit 2 High Pressure Coolant Injection System Declared Inoperable*, U.S. Nuclear Regulatory Commission, September 24, 2007.

NRC Licensee Event Report 1999-003-01, *Manual Reactor Trip due to Heater Drain System Pipe Rupture Caused by Flow Accelerated Corrosion,* U.S. Nuclear Regulatory Commission, May 1, 2000.

XI.M18 BOLTING INTEGRITY

Program Description

The program manages aging of closure bolting for pressure retaining components. The program relies on recommendations for a comprehensive bolting integrity program, as delineated in NUREG-1339, and industry recommendations, as delineated in the following documents:

- NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants."

- Electric Power Research Institute (EPRI) NP-5769, "Degradation and Failure of Bolting in Nuclear Power Plants" (with the exceptions noted in NUREG-1339 for safety-related bolting).

- EPRI TR-104213, "Bolted Joint Maintenance and Application Guide."

The program generally includes periodic inspection of closure bolting for indication of loss of preload, cracking, and loss of material due to corrosion, rust, etc. The program also includes preventive measures to preclude or minimize loss of preload and cracking.

Aging management program (AMP) XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," includes inspection of safety-related and non-safety-related closure bolting and supplements this bolting integrity program. AMPs XI.S1, "ASME Section XI, Subsection IWE"; XI.S3, "ASME Section XI, Subsection IWF"; XI.S6, "Structures Monitoring"; XI.S7, "RG 1.127, "Inspection of Water-Control Structures Associated with Nuclear Power Plants"; and XI.M23, "Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems," manage inspection of safety-related and non-safety related structural bolting.

Evaluation and Technical Basis

1. ***Scope of Program:*** This program manages aging of closure bolting for pressure retaining components within the scope of license renewal, including both safety-related and non-safety-related bolting. This program does not manage aging of reactor head closure stud bolting (AMP XI.M3) or structural bolting (AMPs XI.S1, XI.S3, XI.S6, XI.S7, and XI.M23).

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. NUREG-1339 takes exception to certain items in EPRI NP-5769 and recommends additional measures with regard to them. Of particular note, use of molybdenum disulfide (MoS_2) as a lubricant has been shown to be a potential contributor to stress corrosion cracking (SCC) and should not be used. Preventive measures also include using bolting material that has an actual measured yield strength limited to less than 1,034 megapascals (MPa) (150 kilo-pounds per square inch [ksi]). Bolting replacement activities include proper torquing of the bolts and checking for uniformity of the gasket compression after assembly. Maintenance practices require the application of an appropriate preload based on guidance in EPRI documents, manufacturer recommendations, or engineering evaluation.

3. ***Parameters Monitored/Inspected:*** This program monitors the effects of aging on the intended function of bolting. Specifically, bolting for safety-related pressure retaining components is inspected for leakage, loss of material, cracking, and loss of preload/loss of prestress. Bolting for other pressure retaining components is inspected for signs of leakage. High strength closure bolting (actual yield strength greater than or equal to 1,034 MPa [150 ksi]), if used, should be monitored for cracking.

4. ***Detection of Aging Effects:*** The American Society of Mechanical Engineers (ASME) Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program implements inspection of Class 1, Class 2, and Class 3 pressure retaining bolting in accordance with requirements of ASME Code Section XI,[12] Tables IWB-2500-1, IWC-2500-1, and IWD-2500-1. These include volumetric and visual (VT-1) examinations, as appropriate. In addition, for both ASME Code class bolting and non-ASME Code class bolting, periodic system walkdowns and inspections (at least once per refueling cycle) ensure detection of leakage at bolted joints before the leakage becomes excessive. Bolting inspections should include consideration of the guidance applicable for pressure boundary bolting in NUREG-1339 and in EPRI NP-5769 and EPRI TR-104213.

   Degradation of pressure boundary closure bolting due to crack initiation, loss of preload, or loss of material may result in leakage from the mating surfaces or joint connections of pressure boundary components. Periodic inspection of pressure boundary components for signs of leakage ensures that age-related degradation of closure bolting is detected and corrected before component leakage becomes excessive. Accordingly, pressure retaining bolted connections should be inspected at least once per refueling cycle. The inspections may be performed as part of ASME Code Section XI leakage tests or as part of other periodic inspection activities, such as system walkdowns or an external surfaces monitoring program.

   High strength closure bolting (actual yield strength greater than or equal to 1,034 MPa (150 ksi) may be subject to stress corrosion cracking. For high strength closure bolts (regardless of code classification), volumetric examination in accordance to that of ASME Code Section XI, Table IWB-2500-1, Examination Category B-G-1, should be performed.

5. ***Monitoring and Trending:*** The inspection schedules of ASME Section XI components are effective and ensure timely detection of applicable aging effects. If a bolting connection for pressure retaining components not covered by ASME Section XI is reported to be leaking, it may be inspected daily or in accordance with the corrective action process. If the leak rate is increasing, more frequent inspections may be warranted.

6. ***Acceptance Criteria:*** Any indications of aging effects in ASME pressure retaining bolting are evaluated in accordance with Section XI of the ASME Code. For other pressure retaining bolting, indications of aging should be dispositioned in accordance with the corrective action process.

7. ***Corrective Actions:*** Replacement of ASME pressure retaining bolting is performed in accordance with appropriate requirements of Section XI of the ASME Code, as subject to the additional guidelines and recommendations of EPRI NP-5769. Replacement of other pressure retaining bolting (i.e., non-ASME code class bolting) is performed in accordance with the guidelines and recommendations of EPRI TR-104213. As discussed in the

[12] Refer to the GALL Report, Chapter I, for applicability of other editions of the ASME Code, Section XI.

Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

8. ***Confirmation Process:*** Site quality assurance 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process and administrative controls.

9. ***Administrative Controls:*** As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the administrative controls.

10. ***Operating Experience:*** Degradation of threaded bolting and fasteners in closures for the reactor coolant pressure boundary has occurred from boric acid corrosion, SCC, and fatigue loading (U.S. Nuclear Regulatory Commission [NRC] IE Bulletin 82-02, NRC Generic Letter 91-17). SCC has occurred in high strength bolts used for nuclear steam supply system component supports (EPRI NP-5769). The bolting integrity program developed and implemented in accordance with the applicant's docketed responses to NRC communications on bolting events have provided an effective means of ensuring bolting reliability. These programs are documented in EPRI NP-5769 and TR-104213 and represent industry consensus.

    Degradation related failures have occurred in downcomer Tee-quencher bolting in boiling water reactors (BWRs) designed with drywells (ADAMS Accession Number ML050730347). Leakage from bolted connections has been observed in reactor building closed cooling systems of BWRs (LER 50-341/2005-001).

    The applicant is to evaluate applicable operating experience to support the conclusion that the effects of aging are adequately managed.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

10 CFR 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2009.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, The ASME Boiler and Pressure Vessel Code, 2004 edition as approved in 10 CFR 50.55a, The 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, April 1988.

EPRI TR-104213, *Bolted Joint Maintenance & Application Guide,* Electric Power Research Institute, 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.

NRC Morning Report, *Failure of Safety/Relief Valve Tee-Quencher Support Bolts*, March 14, 2005. (ADAMS Accession Number ML050730347)

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

XI.M19 STEAM GENERATORS

Program Description

The Steam Generator program is applicable to managing the aging of steam generator tubes, plugs, sleeves, and secondary side components that are contained within the steam generator (i.e., secondary side internals).

The establishment of a steam generator program for ensuring steam generator tube integrity is required by plant technical specifications. The steam generator tube integrity portion of the technical specifications at each PWR contains the same fundamental requirements as outlined in the standard technical specifications of NUREG-1430, Volume 1, Rev. 3, for Babcock & Wilcox pressurized water reactors (PWRs); NUREG-1431, Volume 1, Rev. 3, for Westinghouse PWRs; and NUREG-1432, Volume 1, Rev. 3, for Combustion Engineering PWRs. The requirements pertaining to steam generators in these three versions of the standard technical specifications are essentially identical. The technical specifications require tube integrity to be maintained and specify performance criteria, condition monitoring requirements, inspection scope and frequency, acceptance criteria for the plugging or repair of flawed tubes, acceptable tube repair methods, and leakage monitoring requirements.

The nondestructive examination techniques used to inspect tubes, plugs, sleeves, and secondary side internals are intended to identify components (e.g., tubes, plugs) with degradation that may need to be removed from service or repaired.

The Steam Generator program at PWRs is modeled after Nuclear Energy Institute (NEI) 97-06, Revision 2, “Steam Generator Program Guidelines.” This program references a number of industry guidelines (e.g., the EPRI PWR Steam Generator Examination Guidelines, PWR Primary-to-Secondary Leak Guidelines, PWR Primary Water Chemistry Guidelines, PWR Secondary Water Chemistry Guidelines, Steam Generator Integrity Assessment Guidelines, Steam Generator In Situ Pressure Test Guidelines) and incorporates a balance of prevention, mitigation, inspection, evaluation, repair, and leakage monitoring measures. The NEI 97-06 document (a) includes performance criteria that are intended to provide assurance that tube integrity is being maintained consistent with the plant’s licensing basis and (b) provides guidance for monitoring and maintaining the tubes to provide assurance that the performance criteria are met at all times between scheduled inspections of the tubes. Steam generator tube integrity can be affected by degradation of steam generator plugs, sleeves, and secondary side internals. Therefore, all of these components are addressed by this aging management program (AMP). The NEI 97-06 program has been effective at managing the aging effects associated with steam generator tubes, plugs, sleeves, and secondary side internals.

Evaluation and Technical Basis

1. ***Scope of Program:*** This program addresses degradation associated with steam generator tubes, plugs, sleeves, and secondary side components that are contained within the steam generator (i.e., secondary side internals). It does not cover degradation associated with the steam generator shell, channelhead, nozzles, or welds associated with these components.

2. ***Preventive Actions:*** This program includes preventive and mitigative actions for addressing degradation. Preventive and mitigative measures that are part of the Steam Generator program include foreign material exclusion programs, and other primary and secondary side maintenance activities. The program includes foreign material exclusion as a means to

inhibit wear degradation and secondary side maintenance activities, such as sludge lancing, for removing deposits that may contribute to degradation. Guidance on foreign material exclusion is provided in NEI 97-06. Guidance on maintenance of secondary side integrity is provided in the EPRI Steam Generator Integrity Assessment Guidelines. Primary side preventive maintenance activities include replacing plugs made with corrosion susceptible materials with more corrosion resistant materials and preventively plugging tubes susceptible to degradation.

Extensive deposit buildup in the steam generators could affect tube integrity. The EPRI Steam Generator Integrity Assessment Guidelines, which are referenced in NEI 97-06, provide guidance on maintenance on the secondary side of the steam generator, including secondary side cleaning. Secondary side water chemistry plays an important role in controlling the introduction of impurities into the steam generator and potentially limiting their deposition on the tubes. Maintaining high water purity reduces susceptibility to SCC or IGSCC. Water chemistry is monitored and maintained in accordance with the Water Chemistry program. The program description and evaluation and technical basis of monitoring and maintaining water chemistry are addressed in GALL AMP XI.M2, “Water Chemistry.”

3. ***Parameters Monitored/Inspected:*** There are currently three types of steam generator tubing used in the United States: mill annealed Alloy 600, thermally treated Alloy 600, and thermally treated Alloy 690. Mill annealed Alloy 600 steam generator tubes have experienced degradation due to corrosion (e.g., primary water stress corrosion cracking, outside diameter stress corrosion cracking, intergranular attack, pitting, and wastage) and mechanically induced phenomena (e.g., denting, wear, impingement damage, and fatigue). Thermally treated Alloy 600 steam generator tubes have experienced degradation due to corrosion (primarily cracking) and mechanically induced phenomena (primarily wear). Thermally treated Alloy 690 tubes have only experienced tube degradation due to mechanically induced phenomena (primarily wear). Degradation of tube plugs, sleeves, and secondary side internals have also been observed, depending, in part, on the material of construction of the specific component.

   The program includes an assessment of the forms of degradation to which a component is susceptible and implementation of inspection techniques capable of detecting those forms of degradation. The parameter monitored is specific to the component and the acceptance criteria for the inspection. For example, the severity of tube degradation may be evaluated in terms of the depth of degradation or measured voltage, dependent on whether a depth-based or voltage-based tube repair criteria (acceptance criteria) is being implemented for that specific degradation mechanism. Other parameters monitored include signals of excessive deposit buildup (e.g., steam generator water level oscillations), which may result in fatigue failure of tubes or corrosion of the tubes; water chemistry parameters, which may indicate unacceptable levels of impurities; primary-to-secondary leakage, which may indicate excessive tube, plug, or sleeve degradation; and the presence of loose parts or foreign objects on the primary and secondary side of the steam generator, which may result in tube damage.

   Water chemistry parameters are also monitored as discussed in AMP XI.M2. The EPRI PWR Steam Generator Primary-to-Secondary Leakage Guidelines (EPRI 1008219) provides guidance on monitoring primary-to-secondary leakage. The EPRI Steam Generator Integrity Assessment Guidelines (EPRI 1012987) provide guidance on secondary side activities.

In summary, the NEI 97-06 program provides guidance on parameters to be monitored or inspected.

4. ***Detection of Aging Effects:*** The technical specifications require that a Steam Generator program be established and implemented to ensure that steam generator tube integrity is maintained. This requirement ensures that components that could compromise tube integrity are properly evaluated or monitored (e.g., degradation of a secondary side component that could result in a loss of tube integrity is managed by this program). The inspection requirements in the technical specifications are intended to detect degradation (i.e., aging effects), if they should occur.

   The technical specifications are performance-based, and the actual scope of the inspection and the expansion of sample inspections are justified based on the results of the inspections. The goal is to perform inspections at a frequency sufficient to provide reasonable assurance of steam generator tube integrity for the period of time between inspections.

   The general condition of some components (e.g., plugs and secondary side components) may be monitored visually, and, subsequently, more detailed inspections may be performed if degradation is detected.

   NEI 97-06 provides additional guidance on inspection programs to detect degradation of tubes, sleeves, plugs, and secondary side internals. The frequencies of the inspections are based on technical assessments. Guidance on performing these technical assessments is contained in NEI 97-06 and the associated industry guidelines.

   The inspections and monitoring are performed by qualified personnel using qualified techniques in accordance with approved licensee procedures. The EPRI PWR Steam Generator Examination Guidelines (EPRI 1013706) contains guidance on the qualification of steam generator tube inspection techniques.

   The primary-to-secondary leakage monitoring program provides a potential indicator of a loss of steam generator tube integrity. NEI 97-06 and the associated EPRI guidelines provide information pertaining to an effective leakage monitoring program.

5. ***Monitoring and Trending:*** Condition monitoring assessments are performed to determine whether the structural- and accident-induced leakage performance criteria were satisfied during the prior operating interval. Operational assessments are performed to verify that structural and leakage integrity will be maintained for the planned operating interval before the next inspection. If tube integrity cannot be maintained for the planned operating interval before the next inspection, corrective actions are taken in accordance with the plant’s corrective action program. Comparisons of the results of the condition monitoring assessment to the predictions of the previous operational assessment are performed to evaluate the adequacy of the previous operational assessment methodology. If the operational assessment was not conservative in terms of the number and/or severity of the condition, corrective actions are taken in accordance with the plant’s corrective action program.

   The technical specifications require condition monitoring and operational assessments to be performed (although the technical specifications do not explicitly require operational assessments, these assessments are necessary to ensure that the tube integrity will be maintained until the next inspection). Condition monitoring and operational assessments are done in accordance with the technical specification requirements and guidance in NEI 97-06 and the EPRI Steam Generator Integrity Assessment Guidelines.

The goal of the inspection program for all components covered by this AMP is to ensure that the components continue to function consistent with the design and licensing basis of the facility (including regulatory safety margins).

Assessments of the degradation of steam generator secondary side internals are performed in accordance with the guidance in the EPRI Steam Generator Integrity Assessment Guidelines to ensure the component continues to function consistent with the design and licensing basis and to ensure technical specification requirements are satisfied.

6. ***Acceptance Criteria:*** Assessment of tube and sleeve integrity and plugging or repair criteria of flawed and sleeved tubes is in accordance with plant technical specifications. The criteria for plugging or repairing steam generator tubes and sleeves are based on U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.121 and are incorporated into plant technical specifications. Guidance on assessing the acceptability of flaws is also provided in NEI 97-06 and the associated EPRI guidelines, including the EPRI Steam Generator In-Situ Pressure Test Guidelines and EPRI Steam Generator Integrity Assessment Guidelines.

   Degraded plugs, degraded secondary side internals, and leaving a loose part or a foreign object in the steam generator are evaluated for continued acceptability on a case-by-case basis. NEI 97-06 and the associated EPRI guidelines provide guidance on the performance of these evaluations. The intent of these evaluations is to ensure that the components affected by parts or objects have adequate integrity consistent with the design and licensing basis of the facility.

   Guidance on the acceptability of primary-to-secondary leakage and water chemistry parameters also are discussed in NEI 97-06 and the associated EPRI guidelines.

7. ***Corrective Actions:*** For degradation of steam generator tubes and sleeves (if applicable), the technical specifications provide requirements on the actions to be taken when the acceptance criteria are not met. For degradation of other components, the appropriate corrective action is evaluated per NEI 97-06 and the associated EPRI guidelines, the American Society of Mechanical Engineers (ASME) Code Section XI (2004 Edition),[13] 10 CFR 50.65, and 10 CFR Part 50, Appendix B, as appropriate. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable for ensuring effective corrective actions.

8. ***Confirmation Process:*** Site quality assurance (QA) procedures, review and approval processes, and site quality assurance 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process and administrative controls.

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[13] Refer to the GALL Report, Chapter 1, for applicability of other editions of the ASME Code.

In addition, the adequacy of the preventive measures in the Steam Generator program is confirmed through periodic inspections.

9. ***Administrative Controls:*** As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the administrative controls.

10. ***Operating Experience:*** Several generic communications have been issued by the NRC related to the steam generator programs implemented at plants. The reference section lists many of these generic communications. In addition, NEI 97-06 provides guidance to the industry for routinely sharing pertinent steam generator operating experience and for incorporating lessons learned from plant operation into guidelines referenced in NEI 97-06. The latter includes providing interim guidance to the industry, when needed.

    The NEI 97-06 program has been effective at managing the aging effects associated with steam generator tubes, plugs, sleeves, and secondary side components that are contained within the steam generator (i.e., secondary side internals), such that the steam generators can perform their intended safety function.

References

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NRC Information Notice 88-99, *Detection and Monitoring of Sudden and/or Rapidly Increasing Primary-to-Secondary Leakage*, U.S. Nuclear Regulatory Commission, December 20, 1988.

NRC Information Notice 89-65, *Potential for Stress Corrosion Cracking in Steam Generator Tube Plugs Supplied by Babcock and Wilcox*, U.S. Nuclear Regulatory Commission, September 8, 1989.

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

NRC Information Notice 91-43, *Recent Incidents Involving Rapid Increases in Primary-to-Secondary Leak Rate*, U.S. Nuclear Regulatory Commission, July 5, 1991.

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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 94-88, *Inservice Inspection Deficiencies Result in Severely Degraded Steam Generator Tubes*, U.S. Nuclear Regulatory Commission, December 23, 1994.

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XI.M20 OPEN-CYCLE COOLING WATER SYSTEM

Program Description

The program relies on implementation of the recommendations of the Nuclear Regulatory Commission (NRC) Generic Letter (GL) 89-13 to ensure that the effects of aging on the open-cycle cooling water (OCCW) (or service water) system will be managed for the period of extended operation. NRC GL 89-13 defines the OCCW system as a system or systems that transfer heat from safety-related structures, systems, and components (SSCs) to the ultimate heat sink (UHS). The guidelines of NRC GL 89-13 for managing an OCCW include (a) surveillance and control of biofouling (see Chapter IX of NUREG-1801); (b) a test program to verify heat transfer capabilities; (c) routine inspection and a maintenance program to ensure that corrosion, erosion, protective coating failure, sediment deposition (silting), and biofouling cannot degrade the performance of safety-related systems serviced by OCCW; (d) a system walkdown inspection to ensure compliance with the licensing basis; and (e) a review of maintenance, operating, and training practices and procedures.

In accordance with guidance of NRC GL 89-13, the OCCW aging management program manages aging effects of components in raw water systems, such as the service water or river water, by using a combination of preventive, condition, and performance monitoring activities. These include (a) surveillance and control techniques to manage aging effects caused by biofouling, corrosion, erosion, protective coating failures, and silting in the OCCW system or structures and components serviced by the OCCW system; (b) inspection of critical components for signs of corrosion, erosion, and biofouling; and (c) testing of the heat transfer capability of heat exchangers that remove heat from components important to safety.

For buried OCCW piping, the aging effects on the external surfaces are managed by XI.M41, but the internal surfaces are managed by this program. The aging management of closed-cycle cooling water (CCCW) systems is described in XI.M21A, “Closed Treated Water Systems,” and is not included as part of this program. The OCCW System program applies to components constructed of various materials, including steel, stainless steel, aluminum, copper alloys, titanium, polymeric materials, and concrete. Piping may be lined with internal coatings or unlined.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program addresses the aging effects of material loss and fouling due to micro- or macro-organisms and various corrosion mechanisms generally found in OCCW systems and OCCW steel piping components with or without protective coating as described in the applicant’s response to NRC GL 89-13. OCCW systems, as defined by NRC GL 89-13, include the service water system and any other cooling system exposed to raw water that transfers heat from safety-related SSCs to the UHS. The OCCW System program applies to components constructed of various materials, including steel, stainless steel, aluminum, copper alloys, titanium, polymeric materials, and concrete. Piping may be lined with internal coatings or unlined.

2. ***Preventive Actions:*** Preventive actions begin with the use of appropriate material for construction. Steel piping system components are typically lined or coated to protect the underlying metal surfaces from exposure to corrosive cooling water environments. Implementation of NRC GL 89-13 includes control or preventive measures, such as chemical treatment whenever the potential for biological fouling exists or flushing of

infrequently used systems. Treatment with chemicals mitigates microbiologically-influenced corrosion (MIC) and buildup of macroscopic biological fouling debris from biota, such as blue mussels, oysters, or clams. Periodic flushing of the system removes accumulations of biofouling agents, corrosion products, and debris or silt.

3. ***Parameters Monitored/Inspected:*** This program manages the aging effects, such as loss of heat transfer capability, loss of material, and corrosion effects. Adverse effects on system or component performance are caused by accumulations of biofouling agents, corrosion products, and silt. Cleanliness and material integrity of piping, components, heat exchangers, elastomers, and their internal linings or coatings (when applicable) that are part of the OCCW system or that are cooled by the OCCW system are periodically inspected, monitored, or tested to ensure their heat transfer capabilities. The program ensures (a) removal of accumulations of biofouling agents, corrosion products, and silt and (b) detection of defective protective coatings and corroded OCCW system piping and components that could adversely affect performance of their intended safety functions.

4. ***Detection of Aging Effects:*** Inspection scope, methods (e.g., visual or nondestructive examination), and testing frequencies are in accordance with the applicant's docketed response to NRC GL 89-13. Inspections for biofouling, damaged coatings, and degraded material condition are conducted. Visual inspections are typically performed to determine whether corrosion, erosion, or biofouling are occurring in the system. Examinations of polymeric materials should be consistent with the examinations described in AMP XI.M38. Nondestructive testing, such as ultrasonic testing and eddy current testing, are effective methods to measure surface conditions or the extent of wall thinning associated with the service water system piping and components.

5. ***Monitoring and Trending:*** Heat transfer testing results are documented in plant test procedures and are trended in accordance with the applicant's docketed response to NRC GL 89-13. If corrosion buildup or fouling is noted, the system also is evaluated for their impact on the heat transfer capability of the system. Evidence of corrosion in these systems also is evaluated for its potential impact on the integrity of the piping. For relevant indications, inspections or nondestructive testing is used to determine the extent of biofouling, the condition of the surface coating, the magnitude of localized pitting, and the amount of MIC, if applicable.

6. ***Acceptance Criteria:*** The acceptance criteria are in accordance with the applicant's docketed response to NRC GL 89-13. Corrosion, erosion, and biofouling can cause significant loss of material in components. Inspected components should exhibit adequate design margin regarding design dimensions (e.g., minimum required wall thickness). As applicable, coatings or linings should be intact to protect the underlying metal. Heat removal capability is within allowable values for the system and components tested, in accordance with NRC GL 89-13.

7. ***Corrective Actions:*** Evaluations are performed for test or inspection results that do not satisfy established acceptance criteria, and a problem or condition report is initiated to document the concern in accordance with plant administrative procedures. The corrective actions program ensures that the conditions adverse to quality are promptly corrected. If the deficiency is assessed to be significantly adverse to quality, the cause of the condition is determined, and an action plan is developed to preclude repetition. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

8. ***Confirmation Process:*** Site quality assurance 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process controls.

9. ***Administrative Controls:*** As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the administrative controls.

10. ***Operating Experience:*** Significant MIC (NRC Information Notice [IN] 85-30, IN 07-06), failure of protective coatings (NRC IN 85-24), and fouling (NRC IN 81-21, IN 86-96, IN 07-04, IN 07-28) have been observed in a number of heat exchangers. The guidance of NRC GL 89-13 has been implemented for more than 20 years and has been effective in managing aging effects due to biofouling, corrosion, erosion, protective coating failures, and silting in structures and components serviced by OCCW systems.

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XI.M21A CLOSED TREATED WATER SYSTEMS

Program Description

Nuclear power plants contain many closed, treated water systems. These systems undergo water treatment to control water chemistry and prevent corrosion (i.e., treated water systems). These systems are also recirculating systems in which the rate of recirculation is much higher than the rate of addition of makeup water (i.e., closed systems). The program includes (a) water treatment, including the use of corrosion inhibitors, to modify the chemical composition of the water such that the function of the equipment is maintained and such that the effects of corrosion are minimized; (b) chemical testing of the water to ensure that the water treatment program maintains the water chemistry within acceptable guidelines; and (c) inspections to determine the presence or extent of corrosion and/or cracking. Depending on the industry standard selected for use in association with this aging management program (AMP) and/or plant operating experience, this program also may include corrosion monitoring (e.g., corrosion coupon testing) and microbiological testing.

Evaluation and Technical Basis

1. ***Scope of Program:*** This program manages the aging effects of reduction of heat transfer due to fouling, or the loss of material from and cracking due to corrosion and/or stress corrosion cracking of the internal surfaces of piping, piping components, and piping elements fabricated from any material and exposed to treated water. Not included are those piping systems that are managed by another AMP. Examples of systems managed by this AMP include closed-cycle cooling water systems (as defined by U.S. Nuclear Regulatory Commission [NRC] Generic Letter [GL] 89-13[14]); closed portions of heating, ventilation, and air conditioning systems; diesel generator cooling water; and auxiliary boiler systems. Examples of systems not addressed by this AMP include boiling water reactor (BWR) coolant, pressurized water reactor (PWR) primary and secondary water, and PWR/BWR condensate systems. Aging in these systems is managed by the water chemistry AMP (XI.M2) and American Society of Mechanical Engineers (ASME) Section XI, Inservice Inspection, Subsections IWB, IWC, and IWD AMP (XI.M1). Treated fire water systems, if present, are also not included in this AMP. The water used in systems covered by this AMP may, but need not, be demineralized. The water used in systems covered by this AMP receives chemical treatment, including corrosion inhibitors. Untreated water systems are addressed using other AMPs, such as Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (XI.M38).

2. ***Preventive Actions:*** This program mitigates aging effects of loss of material and cracking that are due to corrosion and stress corrosion cracking through water treatment. The water treatment program includes corrosion inhibitors and is designed to maintain the function of associated equipment and minimize the corrosivity of the water.

3. ***Parameters Monitored/Inspected:*** This program monitors water chemistry (preventive monitoring) and the visual appearance of surfaces exposed to the water (condition monitoring). Depending on the industry standard selected for use in association with this

[14] NRC GL 89-13 defines a service water system as “the system or systems that transfer heat from safety-related structures, systems, or components to the ultimate heat sink.” NRC GL 89-13 further defines a closed-cycle system as a part of the service water system that is not subject to significant sources of contamination, one in which water chemistry is controlled and in which heat is not directly rejected to an ultimate heat sink.

AMP and/or plant operating experience, this program may also include corrosion monitoring (e.g., corrosion coupon testing) and microbiological testing. These parameters (such as the concentration of iron, copper, silica, oxygen; and hardness, alkalinity, specific conductivity, and pH) are monitored because maintenance of optimal water chemistry prevents loss of material and cracking due to corrosion and stress corrosion cracking. In addition, the visual appearance of surfaces provides evidence of the existence of loss of material or cracking. The specific water chemistry parameters monitored and the acceptable range of values for these parameters are in accordance with industry standard guidance documents produced by the Electric Power Research Institute (EPRI), the American Society of Heating Refrigeration and Air-Conditioning Engineers, the Cooling Technology Institute, the American Boiler Manufacturer's Association, ASTM standards, water chemistry guidelines recommended by the equipment manufacturer, Nalco Water Handbook, or the ASME. For closed-cycle cooling water systems as defined in NRC GL 89-13, EPRI 1007820 is used. For other systems, the applicant selects an appropriate industry standard document. In all cases, the selected industry standard guidance document is used in its entirety for the water chemistry control or guidance.

4. ***Detection of Aging Effects:*** In this program, aging effects are detected through water testing and periodic inspections. Water testing ensures that the water treatment program is effective in maintaining acceptable water chemistry. Water testing is conducted in accordance with the selected industry standard. The frequency of water testing is in accordance with the selected industry standard, but in no case should the testing interval be greater than quarterly unless justified with an additional analysis. Because the control of water chemistry may not be fully effective in mitigating the aging effects, visual inspections are conducted. Inspections are conducted whenever the system boundary is opened. Additionally, a representative sample of piping and components is selected based on likelihood of corrosion or cracking and inspected at an interval not to exceed once in 10 years. When required by the ASME Code, inspections are conducted in accordance with the applicable code requirements. In the absence of Code inspection requirements, inspections are conducted in accordance with the selected industry standard. In the event that the selected industry standard does not contain inspection requirements, plant-specific inspection and personnel qualification procedures that are capable of detecting corrosion or cracking may be used. If visual examination identifies adverse conditions, additional examinations, including ultrasonic testing, are conducted. Plant operating experience and/or the industry standard program selected for use in association with this AMP may recommend corrosion testing and/or microbiological testing. If warranted, these tests are conducted in accordance with the industry standard selected or other industry standards appropriate for the conduct of corrosion or microbiological testing.

5. ***Monitoring and Trending:*** Water chemistry data are evaluated against the standards contained in the selected industry standard documents. These data are trended with time, so corrective actions are taken, based on trends in water chemistry, prior to loss of intended function. Inspection results also are trended with time so that the progression of any corrosion or cracking can be evaluated and predicted.

6. ***Acceptance Criteria:*** Water chemistry concentrations are maintained within the limits specified in the selected industry standard documents. System components should meet system design requirements, such as minimum wall thickness.

7. ***Corrective Actions:*** Water chemistry concentrations that are not in accordance with the selected industry standard document should be returned to an "in specification" condition in

accordance with the referenced guidelines. Some industry standard documents have time guidelines which govern how rapidly "out of specification" conditions should be corrected. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address corrective actions.

8. ***Confirmation Process:*** Site quality assurance 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process and administrative controls.

9. ***Administrative Controls:*** As discussed in the Generic Aging Lessons Learned (GALL) Report, the staff finds the requirements 10 CFR Part 50, Appendix B, acceptable to address the administrative controls.

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 LER 50-280/91-019-00) has been observed in operating plants. Accordingly, operating experience demonstrates the need for this program.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

EPRI 1007820, *Closed Cooling Water Chemistry Guideline*, Electric Power Research Institute, Palo Alto, CA, April 2004.

Flynn, Daniel. *The Nalco Water Handbook*, Nalco Company, 2009.

NRC Generic Letter 89-13, *Service Water System Problems Affecting Safety-Related Components*, U.S. Nuclear Regulatory Commission, July 18, 1989.

NRC Generic Letter 89-13, Supplement 1, *Service Water System Problems Affecting Safety-Related Components*, U.S. Nuclear Regulatory Commission, April 4, 1990.

NRC Licensee Event Report 50-280/91-019-00, *Loss of Containment Integrity due to Crack in Component Cooling Water Piping*, October 26, 1991.

NRC Licensee Event Report 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.M22 BORAFLEX MONITORING

Program Description

For Boraflex panels in spent fuel storage racks, gamma irradiation and long-term exposure to the wet fuel pool environment causes shrinkage resulting in gap formation, gradual degradation of the polymer matrix, and the release of silica to the spent fuel storage pool water. This results in the loss of boron carbide in the neutron absorber sheets. A monitoring program for the Boraflex panels in the spent fuel storage racks is implemented to assure that no unexpected degradation of the Boraflex material compromises the criticality analysis in support of the design of spent fuel storage racks. This aging management program (AMP) relies on periodic inspection, testing, monitoring, and analysis of the criticality design to assure that the required 5% subcriticality margin is maintained. Therefore, this AMP includes: (a) completing sampling and analysis for silica levels in the spent fuel pool water on a regular basis, such as monthly, quarterly, or annually (depending on Boraflex panel condition), and trending the results by using the EPRI RACKLIFE predictive code or its equivalent; and (b) performing neutron attenuation testing or blackness testing to determine gap formation in Boraflex panels or measuring boron areal density by techniques such as the BADGER device.

Evaluation and Technical Basis

1. ***Scope of Program***: This program manages the effect of reduction in neutron-absorbing capacity due to degradation in sheets of neutron-absorbing material made of Boraflex affixed to spent fuel racks.

2. ***Preventive Actions:*** This program is a performance monitoring program and does not include preventive actions.

3. ***Parameters Monitored/Inspected:*** The parameters monitored include physical conditions of the Boraflex panels, such as gap formation and decreased boron areal density, and the concentration of the silica in the spent fuel pool. These are conditions directly related to degradation of the Boraflex material. When Boraflex is subjected to gamma radiation and long-term exposure to the spent fuel pool environment, the silicon polymer matrix becomes degraded and silica filler and boron carbide are released into the spent fuel pool water. As indicated in the Nuclear Regulatory Commission (NRC) Information Notice (IN) 95-38 and NRC Generic Letter (GL) 96-04, the loss of boron carbide (washout) from Boraflex is characterized by slow dissolution of silica from the surface of the Boraflex and a gradual thinning of the material. Because Boraflex contains about 25% silica, 25% polydimethyl siloxane polymer, and 50% boron carbide, sampling and analysis of the presence of silica in the spent fuel pool provide an indication of depletion of boron carbide from Boraflex; however, the degree to which Boraflex has degraded is ascertained through measurement of the boron areal density.

4. ***Detection of Aging Effects:*** Aging effects on Boraflex panels are detected by monitoring silica levels in the spent fuel storage pool on a regular basis, such as monthly, quarterly, or annually (depending on Boraflex panel condition); by performing blackness testing to measure gap formation or measuring boron areal density on a frequency determined by the material condition of the Boraflex panels, with a minimum frequency of once every 5 years; and by applying predictive methods to the measured results. The amount of boron carbide present in the Boraflex panels is determined through direct measurement of boron areal density by blackness testing or by periodic verification of boron loss through areal density

measurement techniques, such as the BADGER device. Frequent Boraflex testing is sufficient to ensure that Boraflex panel degradation does not compromise criticality analysis for the spent fuel pool storage racks. Additionally, changes in the level of silica present in the spent fuel pool water provide an indication of changes in the rate of degradation of Boraflex panels.

5. ***Monitoring and Trending:*** The periodic inspection measurements and analysis are compared to values of previous measurements and analysis providing a continuing level of data for trend analysis. Sampling and analysis for silica levels in the spent fuel pool water is performed on a regular basis, such as monthly, quarterly, or annually (depending on Boraflex panel condition), and results are trended using the EPRI RACKLIFE predictive code or its equivalent. The frequency to perform blackness testing will be determined by the material condition of the Boraflex panels, with a maximum of 5 years.

6. ***Acceptance Criteria:*** The 5% subcriticality margin of the spent fuel racks is maintained for the period of extended operation.

7. ***Corrective Actions:*** Corrective actions are initiated if the test results find that the 5% subcriticality margin cannot be maintained because of the current or projected future degradation. Corrective actions consist of providing additional neutron-absorbing capacity by Boral® or boron steel inserts or other options which are available to maintain a subcriticality margin of 5%. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

8. ***Confirmation Process:*** Site quality assurance procedures, site 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process and administrative controls.

9. ***Administrative Controls:*** As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the administrative controls.

10. ***Operating Experience:*** NRC IN 87-43 addresses the problems of development of tears and gaps (average 1-2 inches, with the largest 4 inches) in Boraflex sheets due to gamma radiation-induced shrinkage of the material. NRC IN 93-70, NRC IN 95-38, and NRC GL 96-04 address several cases of significant degradation of Boraflex test coupons due to accelerated dissolution of Boraflex caused by pool water flow through panel enclosures and high accumulated gamma dose. Two spent fuel rack cells with about 12 years of service have only 40% of the Boraflex remaining. In such cases, the Boraflex may be replaced by boron steel inserts or by a completely new rack system using Boral®. Experience with boron steel is limited; however, the application of Boral® for use in the spent fuel storage racks predates the manufacturing and use of Boraflex. The experience with Boraflex panels indicates that coupon surveillance programs are not reliable. Therefore, during the period of extended operation, the measurement of boron areal density correlated, through a predictive code, with silica levels in the pool water, is verified. These monitoring programs provide assurance that degradation of Boraflex sheets is monitored so that appropriate actions can be taken in a timely manner if significant loss of neutron-absorbing capability is occurring. These monitoring programs provide reasonable assurance that the Boraflex sheets maintain their integrity and are effective in performing their intended function.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

BNL-NUREG-25582, *Corrosion Considerations in the Use of Boral in Spent Fuel Storage Pool Racks*, January 1979.

EPRI NP-6159, *An Assessment of Boraflex Performance in Spent-Nuclear-Fuel Storage Racks*, Electric Power Research Institute, Palo Alto, CA, December 14, 1988.

EPRI 1003413, *Guidance and Recommended Procedure for Maintaining and Using RACKLIFE Version 1.10*, Electric Power Research Institute, Palo Alto, CA, April 2002.

EPRI TR-101986, *Boraflex Test Results and Evaluation*, Electric Power Research Institute, Palo Alto, CA, March 1, 1993.

EPRI TR-103300, *Guidelines for Boraflex Use in Spent-Fuel Storage Racks*, Electric Power Research Institute, Palo Alto, CA, December 1, 1993.

NRC Generic Letter 96-04, *Boraflex Degradation in Spent Fuel Pool Storage Racks*, U.S. Nuclear Regulatory Commission, June 26, 1996.

NRC Information Notice 87-43, *Gaps in Neutron Absorbing Material in High Density Spent Fuel Storage Racks,* U.S. Nuclear Regulatory Commission, September 8, 1987.

NRC Information Notice 93-70, *Degradation of Boraflex Neutron Absorber Coupons*, U.S. Nuclear Regulatory Commission, September 10, 1993.

NRC Information Notice 95-38, *Degradation of Boraflex Neutron Absorber in Spent Fuel Storage Racks*, U.S. Nuclear Regulatory Commission, September 8, 1995.

NRC Regulatory Guide 1.26, Rev. 3, *Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants (for Comment),* U.S. Nuclear Regulatory Commission, February 1976.

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. Because only a few cranes operate over safety-related equipment, normally fewer than 10 cranes fall within the scope of 10 CFR 54.4.

Many of the systems and components of these cranes perform an intended function with moving parts or with a change in configuration or are 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. 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. The aging management activities specified in this program utilize the guidance provided in American Society of Mechanical Engineers (ASME) Safety Standard B30.2, "Overhead and Gantry Cranes (Top Running Bridge, Single or Multiple Girder, Top Running Trolley Hoist)."

Evaluation and Technical Basis

1. ***Scope of Program:*** The program manages (a) the effects of loss of material due to general corrosion on the bridge rails, bridge, and trolley structural components for those cranes that are within the scope of 10 CFR 54.4 and (b) the effects of wear on the rails in the rail system. The program also manages the effects of loss of preload of bolted connections.

2. ***Preventive Actions:*** This program is a condition monitoring program. No preventive actions are identified.

3. ***Parameters Monitored/Inspected:*** Surface condition is monitored by visual inspection to ensure that loss of material is not occurring due to corrosion or wear. Bolted connections are monitored for loose bolts, missing or loose nuts, and other conditions indicative of loss of preload.

4. ***Detection of Aging Effect:*** Crane rails and structural components are visually inspected at a frequency in accordance ASME B30.2, "Overhead and Gantry Cranes (Top Running Bridge, Single or Multiple Girder, Top Running Trolley Hoist)," or other appropriate standard in the ASME B30 series. For systems that are infrequently in service, such as containment polar cranes, periodic inspections are performed once every refueling cycle just prior to use. Bolted connections are visually inspected for loose bolts or missing nuts at the same frequency as crane rails and structural components.

5. ***Monitoring and Trending:*** Visual inspection activities are performed by personnel qualified in accordance with controlled procedures and processes. Deficiencies are documented using applicant-approved processes and procedures, such that results can be trended; however, the program does not include formal trending.

6. ***Acceptance Criteria:*** Any visual indication of loss of material due to corrosion or wear and any visual sign of loss of bolting pre-load is evaluated according to ASME B30.2 or other applicable industry standard in the ASME B30 series.

7. ***Corrective Actions:*** Repairs are performed as specified in ASME B30.2 or other appropriate standard in the ASME B30 series. 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions, confirmation process, and administrative controls.

8. ***Confirmation Process:*** Site 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process and administrative controls.

9. ***Administrative Controls:*** As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the administrative controls.

10. ***Operating Experience:*** There has been no history of corrosion-related degradation that threatened the ability of a crane to perform its intended function. Likewise, because cranes have not been operated beyond their design lifetime, there have been no significant fatigue-related structural failures. Operating experience indicates that loss of bolt preload has occurred, but not to the extent that it has threatened the ability of a crane structure to perform its intended function.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

10 CFR 54.4, *Scope*, Office of the Federal Register, National Archives and Records Administration, 2009.

ASME Safety Standard B30.2, *Overhead and Gantry Cranes (Top Running Bridge, Single or Multiple Girder, Top Running Trolley Hoist),* American Society of Mechanical Engineers, 2005.

NRC Generic Letter 80-113, *Control of Heavy Loads*, U.S. Nuclear Regulatory Commission, December 22, 1980.

NRC Generic Letter 81-07, *Control of Heavy Loads*, U.S. Nuclear Regulatory Commission, February 3, 1981.

NRC Regulatory Guide 1.160, Rev. 2, *Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*, U.S. Nuclear Regulatory Commission, March 1997.

NUREG-0612, *Control of Heavy Loads at Nuclear Power Plants*, U.S. Nuclear Regulatory Commission, 1980.

XI.M24 COMPRESSED AIR MONITORING

Program Description

The purpose of the compressed air monitoring program is to provide reasonable assurance of the integrity of the compressed air system. The program consists of monitoring moisture content, corrosion, and performance of the compressed air system. This includes (a) preventive monitoring of water (moisture) and other potential contaminants to keep within the specified limits; and (b) inspection of components for indications of loss of material due to corrosion.

The compressed air monitoring aging management program (AMP) is based on results of the plant owner's response to Nuclear Regulatory Commission (NRC) Generic Letter (GL) 88-14 (as applicable to license renewal) and reported in previous NRC Information Notices (IN) 81-38; IN 87-28; IN 87-28, Supplement 1; and by the Institute of Nuclear Power Operations Significant Operating Experience Report (INPO SOER) 88-01. NRC GL 88-14, issued after several years of study of problems and failures of instrument air systems, recommends that each holder of an operating license perform an extensive design and operations review and verification of its instrument air system. NRC GL 88-14 also recommends that the licensees describe their program for maintaining proper instrument air quality. This AMP does not include all aspects of NRC GL 88-14 because many of the issues in the GL are not relevant to license renewal.

This AMP does not change the applicant's docketed response to NRC GL 88-14 for the rest of its operations. The program utilizes the aging management aspects of the applicant's response to NRC GL 88-14 for license renewal with regard to preventative measures, inspections of components, and testing to ensure that the compressed air system will be able to perform its intended function for the period of extended operation. The AMP also incorporates the air quality provisions provided in the guidance of the Electric Power Research Institute (EPRI) NP-7079. EPRI NP-7079 was issued in 1990 to assist utilities in identifying and correcting system problems in the instrument air system and to enable them to maintain required industry safety standards. The American Society of Mechanical Engineers (ASME) operations and maintenance standards and guides (ASME OM-S/G-1998, Part 17) provides additional guidance for maintenance of the instrument air system by offering recommended test methods, test intervals, parameters to be measured and evaluated, acceptance criteria, corrective actions, and records requirements.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program manages the aging effects of loss of material due to corrosion in compressed air systems.

2. ***Preventive Actions:*** For the purposes of aging management, moisture and other corrosive contaminants in the system's air are maintained below specified limits to ensure that the system and components maintain their intended functions. These limits are prepared from consideration of manufacturer's recommendations for individual components and guidelines based on ASME OM-S/G-1998, Part 17; American National Standards Institute (ANSI)/ISA-S7.0.01-1996; EPRI NP-7079; and EPRI TR-108147.

3. ***Parameters Monitored/Inspected:*** Maintaining moisture and other corrosive contaminants below acceptable limits mitigates loss of material due to corrosion. Periodic air samples are taken and analyzed for moisture and other corrosives. Periodic and opportunistic

inspections of accessible internal surfaces are performed for signs of corrosion and abnormal corrosion products that might indicate a loss of material within the system.

4. ***Detection of Aging Effects:*** Moisture and other corrosives increase the potential for loss of material due to corrosion. The program periodically samples and tests the air quality in the compressed system for moisture in accordance with industry standards, such as ANSI/ISA-S7.0.01-1996. Typically, compressed systems have in-line dew point instrumentation that either checks continuously using an automatic alarm system or is checked at least daily to ensure that moisture content is within specifications. Additionally, periodic visual inspections of critical component internal surfaces (compressors, dryers, after-coolers, and filters) are performed for signs of loss of material due to corrosion. ASME O/M-S/G-1998, Part 17 provides guidance for inspection frequency and inspection methods of these components.

5. ***Monitoring and Trending:*** Daily readings of system dew point are recorded and trended. Air quality analysis results are reviewed to determine if alert levels or limits have been reached or exceeded. This review also checks for unusual trends. ASME O/M-S/G-1998, Part 17, provides guidance for monitoring and trending data. Visual inspection results are compared to previous results to ascertain if adverse long-term trends exist. The effects of corrosion are monitored by visual inspection. Test data are analyzed and compared to data from previous tests to provide for the timely detection of aging effects on passive components.

6. ***Acceptance Criteria:*** Acceptance criteria for air quality moisture limits are established based on accepted industry standards, such as ANSI/ISA-S7.0.01-1996. Internal surfaces should not show signs of corrosion (general, pitting, and crevice) that could indicate the potential loss of function of the component. Manufacturers' certifications can be used to demonstrate that the bottled air meets acceptable quality standards.

7. ***Corrective Actions:*** Corrective actions are taken if any parameters are out of acceptable ranges, such as moisture content in the system air. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

8. ***Confirmation Process:*** The 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process and administrative controls.

9. ***Administrative Controls:*** Site 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address administrative controls.

10. ***Operating Experience:*** Potentially significant safety-related problems pertaining to air systems have been documented in NRC IN 81-38; IN 87-28; IN 87-28, Supplement 1; and License Event Report 50-237/94-005-3. Some of the systems that have been significantly degraded or that have failed due to the problems in the air system include the decay heat removal, auxiliary feedwater, main steam isolation, containment isolation, and fuel pool seal systems. In 2008, one plant incurred an unplanned reactor trip from a failure of a mechanical joint in the instrument air system (NRC IN 2008-06). Nevertheless, as a result of

NRC GL 88-14 and in consideration of INPO SOER 88-01, EPRI NP-7079, and EPRI TR-108147, performance of air systems has improved significantly.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

ANSI/ISA-S7.0.01-1996, *Quality Standard for Instrument Air*, American National Standards Institute (ANSI), 1996.

ASME OM-S/G-1998, Part 17, *Performance Testing of Instrument Air Systems Information Notice Light-Water Reactor Power Plants*, 1ISA-S7.0.1-1996, “Quality Standard for Instrument Air,” American Society of Mechanical Engineers, New York, NY, 1998.

EPRI NP-7079, *Instrument Air System: A Guide for Power Plant Maintenance Personnel*, Electric Power Research Institute, Palo Alto, CA, December 1990.

EPRI/NMAC TR-108147, *Compressor and Instrument Air System Maintenance Guide: Revision to NP-7079*, Electric Power Research Institute, Nuclear Maintenance Application Center, Palo Alto, CA, March 1998.

INPO Significant Operating Experience Report 88-01, *Instrument Air System Failures*, Institute of Nuclear Power Operations, May 18, 1988.

NRC Generic Letter 88-14, *Instrument Air Supply Problems Affecting Safety-Related Components*, U.S. Nuclear Regulatory Commission, August 8, 1988.

NRC Information Notice 81-38, *Potentially Significant Components Failures Resulting from Contamination of Air-Operated Systems*, U.S. Nuclear Regulatory Commission, December 17, 1981.

NRC Information Notice 87-28, *Air Systems Problems at U.S. Light Water Reactors*, U.S. Nuclear Regulatory Commission, June 22, 1987.

NRC Information Notice 87-28, Supplement 1, *Air Systems Problems at U.S. Light Water Reactors*, U.S. Nuclear Regulatory Commission, December 28, 1987.

NRC Information Notice 2008-06, *Instrument Air System Failure Resulting In Manual Reactor Trip*, U.S. Nuclear Regulatory Commission, April 10, 2008.

NRC Licensee Event Report 50-237/94-005-3, *Manual Reactor Scram due to Loss of Instrument Air Resulting from Air Receiver Pipe Failure Caused by Improper Installation of Threaded Pipe during Initial Construction*, U.S. Nuclear Regulatory Commission, April 23, 1997.

XI.M25 BWR REACTOR WATER CLEANUP SYSTEM

Program Description

This program provides inspection to manage the aging effects of cracking due to stress corrosion cracking (SCC) or intergranular stress corrosion cracking (IGSCC) on the intended function of austenitic stainless steel (SS) piping outboard of the second primary containment isolation valves in the reactor water cleanup (RWCU) system. Based on the Nuclear Regulatory Commission (NRC) criteria related to inspection guidelines for RWCU piping welds outboard of the second isolation valve, the program includes the measures delineated in NUREG-0313, Rev. 2, and in NRC Generic Letter (GL) 88-01 and its Supplement 1. The aging management review (AMR) Item in the GALL Report that credits this program also credits AMP XI.M2, “Water Chemistry,” to provide mitigation of the aging effects. Reactor coolant water chemistry is monitored and maintained in accordance with the Water Chemistry program.

NRC GL 88-01 applies to all boiling water reactor (BWR) piping made of austenitic SS that is 4 inches or larger in nominal diameter and contains reactor coolant at a temperature above 93.3^{o} C (200^{o} F) during power operation regardless of code classification. NRC GL 88-01 requests, in part, that affected licensees implement an ISI program conforming to staff positions for austenitic SS piping covered under the scope of the letter. In response to NRC GL 88-01, affected licensees undertook ISI in accordance with the scope and schedules described in the letter and included affected portions of RWCU piping outboard of the second isolation valves in their ISI programs.

The NRC issued GL 88-01, Supplement 1, to provide acceptable alternatives to staff positions delineated in NRC GL 88-01. In NRC GL 88-01, Supplement 1, the staff noted, in part, that the position stated in NRC GL 88-01 on inspection sample size of RWCU system welds outboard of the second isolation valves had created an unnecessary hardship for affected licensees because of the very high radiation levels associated with this portion of RWCU piping. The staff also noted that affected licensees had requested that they be exempted from NRC GL 88-01 with regard to inspection of this piping of the RWCU system. Although NRC GL 88-01, Supplement 1, does not provide explicit generic guidance with regard to staff criteria for reduction or elimination of RWCU weld inspections, it does suggest that the staff would be receptive to modifications to a licensee’s original docketed NRC GL 88-01 response for RWCU weld inspections, provided all issues of reactor safety were adequately addressed. The staff has subsequently allowed individual licensees to modify their docketed responses to GL-88-01 to reduce or eliminate their ISI of RWCU welds in the piping outboard of the second isolation valves. This AMP is based on the staff-approved screening criteria for the inspection.

Evaluation and Technical Basis

1. ***Scope of Program:*** This program provides ISI to manage the aging effects of cracking due to SCC or IGSCC in austenitic SS piping outboard of the second containment isolation valves in the RWCU system.

   The components included in this program are the welds in piping that have a nominal diameter of 4 inches or larger and that contain reactor coolant at a temperature above 93°C (200°F) during power operation, regardless of code classification.

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

SCC or IGSCC. These elements are a susceptible (sensitized) material, a significant tensile stress, and an aggressive environment. The program delineated in NUREG-0313 and NRC GL 88-01 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). Special processes are used for existing as well as new and replacement components. These processes include solution heat treatment, heat sink welding, induction heating, and mechanical stress improvement.

3. ***Parameters Monitored/Inspected:*** The aging management program (AMP) monitors SCC or IGSCC of austenitic SS piping by detecting and sizing cracks in accordance with the requirements of American Society of Mechanical Engineers (ASME) Code, Section XI; the guidelines of NUREG-0313, NRC GL 88-01, and NRC GL 88-01, Supplement 1; and the NRC screening criteria as described in Element 4 for the RWCU piping outboard of the second isolation valves.

4. ***Detection of Aging Effects:*** The extent, method, and schedule of the inspection and test techniques delineated in the NRC inspection criteria for RWCU piping and NRC GL 88-01 are designed to maintain structural integrity and to detect aging effects before the loss of intended function of austenitic SS piping and fittings. Guidelines for the inspection schedule, methods, personnel, sample expansion, and leak detection guidelines are based on the guidelines of NRC GL 88-01 and GL 88-01, Supplement 1, and subsequent licensing correspondence. Consistent with the NRC guidelines and with licensees' completion of all actions requested in NRC GL 89-10, no inspection of the outboard piping is required for (a) piping systems that are made of IGSCC-resistant piping materials or (b) piping with no IGSCC detected inboard of the second isolation valves (ongoing GL 88-01 inspection) and outboard of the second isolation valves (after inspecting a minimum of 10% of susceptible piping welds). For piping that includes a non-resistant base or weld material in the scope of the program or piping that has experienced IGSCC, either inboard or outboard of the second isolation valves, an inspection of at least 2% of the welds or two welds, whichever is greater, is performed on the portions of the RWCU system outboard of the second isolation valves every refueling outage.

5. ***Monitoring and Trending:*** The extent and schedule for inspection in accordance with the recommendations of NRC GL 88-01 provide timely detection of cracks and leakage of coolant. Based on inspection results, NRC GL 88-01 provides guidelines for additional samples of welds to be inspected when one or more cracked welds are found in a weld category.

6. ***Acceptance Criteria:*** NRC GL 88-01 recommends that any indication detected be evaluated in accordance with the requirements of ASME Code, Section XI, Subsection IWB-3640.[15]

7. ***Corrective Actions*:** The guidance for weld overlay repair, stress improvement, or replacement is provided in NRC GL 88-01. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

[15] Refer to the GALL Report, Chapter I, for applicability of other editions of the ASME Code, Section XI.

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

9. ***Administrative Controls:*** As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the administrative controls.

10. ***Operating Experience:*** The IGSCC has occurred in small- and large-diameter boiling water reactor (BWR) piping made of austenitic stainless steels. The comprehensive program outlined in NRC GL 88-01 and NUREG-0313 addresses improvements in all elements that cause SCC or IGSCC (e.g., susceptible material, significant tensile stress, and an aggressive environment) and is effective in managing IGSCC in austenitic SS piping in the RWCU system.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, The ASME Boiler and Pressure Vessel Code, 2004 edition as approved in 10 CFR 50.55a The American Society of Mechanical Engineers, New York, NY.

Letter from Joseph W. Shea, U.S. Nuclear Regulatory Commission, to George A. Hunger, Jr., PECO Energy Company, *Reactor Water Cleanup (RWCU) System Weld Inspections at Peach Bottom Atomic Power Station*, *Units 2 and 3 (TAC Nos. M92442 and M92443)*, September 15, 1995. (ADAMS Accession Number ML090930466)

Letter from Robert M. Pulsifer, U.S. Nuclear Regulatory Commission, to Michael A Balduzzi, Vermont Yankee Nuclear Power Corporation, *Review of Request to Discontinue Intergranular Stress Corrosion Cracking Inspection of RWCU Piping Welds Outboard of the Second Containment Isolation Valves (TAC No. MB0468),* March 27, 2001. (ADAMS Accession Number ML010780094)

NRC Generic Letter 88-01, *NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping,* U.S. Nuclear Regulatory Commission, January 25, 1988.

NRC Generic Letter 88-01, Supplement 1, *NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping,* U.S. Nuclear Regulatory Commission, February 4, 1992.

NRC Generic Letter 89-10, *Safety-related Motor Operated Valve Testing and Surveillance,* U.S. Nuclear Regulatory Commission, June 28, 1989; through Supplement 7, January 24, 1996.

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.M26 FIRE PROTECTION

Program Description

For operating plants, the Fire Protection aging management program (AMP) includes a fire barrier 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 AMP also includes periodic inspection and testing of the halon/carbon dioxide (CO_2) fire suppression system.

Evaluation and Technical Basis

1. ***Scope of Program:*** This program manages the effects of loss of material and cracking, increased hardness, shrinkage and loss of strength on the intended function of the penetration seals; fire barrier walls, ceilings, and floors; other fire resistance materials (e.g., flamastic, 3M fire wrapping, spray-on fire proofing material, intumescent coating, etc.) that serve a fire barrier function; 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 halon/CO_2 fire suppression system.

2. ***Preventive Actions:*** This is a condition monitoring program. However, 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 not less than 10% of each type of penetration seal is performed during walkdowns. These inspections examine any sign of degradation, such as cracking, seal separation from walls and components, separation of layers of material, rupture and puncture of seals that are directly caused by increased hardness, and shrinkage of seal material due to loss of material. Visual inspection of the fire barrier walls, ceilings, and floors and other fire barrier materials detects any sign of degradation, such as cracking, spalling, and loss of material caused by freeze-thaw, chemical attack, and reaction with aggregates that could affect their intended fire protection function. Fire-rated doors are visually inspected to detect any degradation of door surfaces.

   The periodic visual inspection and function test are performed to examine for signs of corrosion that may lead to the loss of material of the halon/CO_2 fire suppression system.

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 by fire protection qualified personnel of not less than 10% of each type of seal in walkdowns is performed at a frequency in accordance with an NRC-approved fire protection program (e.g., Technical Requirements Manual, Appendix R program, etc.) or at least once every refueling outage. If any sign of degradation is detected within that sample, the scope of the inspection is expanded to include additional seals. Visual inspection by fire protection qualified personnel of the fire barrier walls, ceilings, floors, doors, and other fire barrier materials performed in walkdowns at a frequency in accordance with an NRC-approved fire protection program ensure timely detection of concrete cracking, spalling, and loss of material. Visual inspection by fire protection qualified personnel detects any sign of

degradation of the fire doors, 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.

Visual inspections of the halon/CO_2 fire suppression system are performed to detect any sign of corrosion. The periodic functional test is performed at least once every 6 months or on a schedule in accordance with an NRC-approved fire protection program. Inspections are performed to detect degradation of the halon/CO_2 fire suppression system before the loss of the component intended function.

5. ***Monitoring and Trending:*** The results of inspections of the aging effects of cracking, spalling, and loss of material on fire barrier penetration seals, fire barriers, and fire doors are used to trend future actions.

   The performance of the halon/CO_2 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 signs of degradation that could result in the loss of the fire protection capability due to loss of material. The acceptance criteria include (a) no visual indications (outside those allowed by approved penetration seal configurations) of cracking, separation of seals from walls and components, separation of layers of material, or ruptures or punctures of seals; (b) no significant indications of concrete cracking, spalling, and loss of material of fire barrier walls, ceilings, and floors and in other fire barrier materials; (c) no visual indications of missing parts, holes, and wear; and (d) no deficiencies in the functional tests of fire doors. Also, no indications of excessive loss of material due to corrosion in the halon/CO_2 fire suppression system is acceptable.

7. ***Corrective Actions:*** For fire protection structures and components identified that are subject to an AMR 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 corrective action program is documented in the final safety analysis report supplement in accordance with 10 CFR 54.21(d). As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions, confirmation process, and administrative controls.

8. ***Confirmation Process:*** As discussed in the Appendix for GALL, 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the administrative controls.

10. ***Operating Experience:*** Silicone foam fire barrier penetration seals have experienced splits, shrinkage, voids, lack of fill, and other failure modes (U.S. Nuclear Regulatory Commission [NRC] Information Notice [IN] 88-56, IN 94-28, and IN 97-70). Degradation of electrical raceway fire barrier such as small holes, cracking, and unfilled seals are found on routine walkdown (NRC IN 91-47 and NRC Generic Letter 92-08). Fire doors have experienced wear of the hinges and handles.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

NRC Generic Letter 92-08, *Thermo-Lag 330-1 Fire Barrier*, U.S. Nuclear Regulatory Commission, December 17, 1992.

NRC Information Notice 88-56, *Potential Problems with Silicone Foam Fire Barrier Penetration Seals*, U.S. Nuclear Regulatory Commission, August 14, 1988.

NRC Information Notice 91-47, *Failure of Thermo-Lag Fire Barrier Material to Pass Fire Endurance Test*, U.S. Nuclear Regulatory Commission, August 6, 1991.

NRC Information Notice 94-28, *Potential Problems with Fire-Barrier Penetration Seals*, U.S. Nuclear Regulatory Commission, April 5, 1994.

NRC Information Notice 97-70, *Potential Problems with Fire Barrier Penetration Seals*, U.S. Nuclear Regulatory Commission, September 19, 1997.

XI.M27 FIRE WATER SYSTEM

Program Description

This aging management program (AMP) applies to water-based fire protection systems that consist of sprinklers, nozzles, fittings, valves, fire pump casings, hydrants, hose stations, standpipes, water storage tanks, and aboveground, buried, 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 tested by using the guidance of NFPA 25, “Inspection, Testing and Maintenance of Water-Based Fire Protection Systems” (1998 Edition), Section 2-3.1.1, or NFPA 25 (2002 Edition), Section 5.3.1.1.1. These NFPA sections state “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 water-based fire protection system piping is subjected to required flow testing in accordance with guidance in NFPA 25 to verify design pressure or evaluated for wall thickness (e.g., non-intrusive volumetric testing or plant maintenance visual inspections) to ensure that 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 determined by engineering evaluation of the fire protection piping to ensure that degradation is detected before the loss of intended function. The purpose of the full flow testing and wall thickness evaluations is to ensure that corrosion, microbiologically influenced corrosion (MIC), or biofouling is managed such that the system function is maintained.

Chapter XI.M41 describes the aging management program for buried and underground water-based fire protection system piping and tanks.

Evaluation and Technical Basis

1. ***Scope of Program:*** The AMP focuses on managing loss of material due to corrosion, MIC, or biofouling of steel components in fire protection systems exposed to water. Fire hose stations and standpipes are considered as piping in the AMP. Fire hoses and gaskets can be excluded from the scope of license renewal if the standards that are relied upon to prescribe replacement of the hose and gaskets are identified in the scoping methodology description.

2. ***Preventive Actions:*** To ensure that no significant corrosion, MIC, or biofouling has occurred in water-based fire protection systems, periodic flushing and system performance testing are conducted in accordance with NFPA 25.

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, or wall thickness evaluations may be performed to ensure that the system maintains its intended function. Testing of sprinklers ensures that degradation is detected in timely manner.

4. ***Detection of Aging Effects:*** The water-based fire protection system testing is performed to ensure 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 are capable of evaluating (a) wall thickness to ensure against catastrophic failure and (b) the inner diameter of the piping as it applies to the design flow 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 is 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 that 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 to detect degradation before a loss of intended function can occur. Sprinkler heads are tested before the end of the 50-year sprinkler head service life and at 10-year intervals thereafter during the period of extended 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 visual inspection is evaluated.

6. ***Acceptance Criteria:*** The acceptance criteria are (a) the water-based fire protection system is able to maintain required pressure, (b) no unacceptable signs of degradation are observed during non-intrusive or visual inspection of components, (c) minimum design pipe wall thickness is maintained, and (d) no biofouling exists in the sprinkler systems that could cause corrosion in the sprinklers.

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 (AMR) for license renewal, the applicant's 10 CFR Part 50, Appendix B, program is used for corrective actions for aging management during the period of extended operation. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

8. ***Confirmation Process:*** For fire water systems and components identified within scope that are subject to an AMR for license renewal, the applicant's 10 CFR Part 50, Appendix B, program is used for confirmation process for aging management during the period of extended operation. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process.

9. ***Administrative Controls:*** For the water-based fire water systems and components identified within scope that are subject to an AMR for license renewal, the applicant's 10 CFR Part 50, Appendix B, program is used for administrative controls for aging management during the period of extended operation. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the administrative controls.

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

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

NFPA 25, *Inspection, Testing and Maintenance of Water-Based Fire Protection Systems*, 1998 Edition, National Fire Protection Association.

NFPA 25, *Inspection, Testing and Maintenance of Water-Based Fire Protection Systems*, 2002 Edition, National Fire Protection Association.

XI.M29 ABOVEGROUND METALLIC TANKS

Program Description

The Aboveground Metallic Tanks aging management program (AMP) manages the effects of loss of material on the outer surfaces of above ground tanks constructed on concrete or soil. If the tank exterior is fully visible, the program for inspection of external surfaces may be used instead (XI.M36). This program credits the standard industry practice of coating or painting the external of steel tanks as a preventive measure to mitigate corrosion. The program relies on periodic inspections to monitor degradation of the protective paint or coating. However, for storage tanks supported on earthen or concrete foundations, corrosion may occur at inaccessible locations, such as the tank bottom. Accordingly, verification of the effectiveness of the program is performed to ensure that significant degradation in inaccessible locations is not occurring and that the component intended function is maintained during the period of extended operation. For reasons set forth below, an acceptable verification program consists of thickness measurement of the tank bottom surface.

Evaluation and Technical Basis

1. ***Scope of Program**:* The program consists of periodic inspections of metallic tanks (with or without coatings) to manage the effects of corrosion on the intended function of these tanks. Inspections cover the entire outer surface of the tank. Because lower portions of the tank are on concrete or soil, this program includes the bottom of the tank as well. If the tank exterior is fully visible, the program for inspection of external surfaces may be used instead (AMP XI.M36).

2. ***Preventive Actions:*** In accordance with industry practice, tanks may be coated with protective paint or coating to mitigate corrosion by protecting the external surface of the tank from environmental exposure. Sealant or caulking may be applied at the external interface between the tank and concrete or earthen foundation to mitigate corrosion of the bottom surface of the tank by minimizing the amount of water and moisture penetrating the interface, which would lead to corrosion of the bottom surface.

3. ***Parameters Monitored/Inspected:*** The AMP utilizes periodic plant inspections to monitor degradation of coatings, sealants, and caulking because it is a condition directly related to the potential loss of materials. Additionally, thickness measurements of the bottoms of the tanks are made periodically for the tanks monitored by this program as an additional measure to ensure that loss of material is not occurring at locations that are inaccessible for inspection.

4. ***Detection of Aging Effects:*** Degradation of an exterior metallic surface can occur in the presence of moisture; therefore, an inspection of the coating is performed to ensure that the surface is protected from moisture. Conducting periodic visual inspections at each outage to confirm that the paint, coating, sealant, and caulking are intact is an effective method to manage the effects of corrosion on the external surface of the component. Potential corrosion of tank bottoms is determined by taking ultrasonic testing (UT) thickness measurements of the tank bottoms whenever the tank is drained and at least once within 5 years of entering the period of extended operation. Measurements are taken to ensure that significant degradation is not occurring and that the component intended function is maintained during the period of extended operation.

5. ***Monitoring and Trending:*** The effects of corrosion of the aboveground external surface are detectable by visual techniques. Based on operating experience, plant inspections during each outage provide for timely detection of aging effects. The effects of corrosion of the inaccessible external surface are detectable by UT thickness measurement of the tank bottom and are monitored and trended if significant material loss is detected where multiple measurements are available.

6. ***Acceptance Criteria:*** Any degradation of paints or coatings (cracking, flaking, or peeling) is reported and requires further evaluation. Drying, cracking, or missing sealant and caulking are unacceptable and need to be evaluated using the corrective action program. The evaluation will determine the need to repair the sealant and caulking. UT thickness measurements of the tank bottom are evaluated against the design thickness and corrosion allowance.

7. ***Corrective Actions:*** The site corrective actions program, quality assurance 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions, confirmation process, and administrative controls. Flaws in the caulking or sealant are repaired.

8. ***Confirmation Process:*** As discussed in the Appendix for GALL, 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the administrative controls.

10. ***Operating Experience:*** Coating degradation, such as flaking and peeling, has occurred in safety-related systems and structures (U.S. Nuclear Regulatory Commission [NRC] Generic Letter 98-04). Corrosion damage near the concrete-metal interface and sand-metal interface has been reported in metal containments (NRC Information Notice [IN] 89-79; IN 89-79, Supplement 1; IN 86-99; and IN 86-99, Supplement 1).

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

NRC Generic Letter 98-04, *Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment*, U.S. Nuclear Regulatory Commission, July 14, 1998.

NRC Information Notice 86-99, *Degradation of Steel Containments*, U.S. Nuclear Regulatory Commission, December 8, 1986.

NRC Information Notice 86-99, Supplement 1, *Degradation of Steel Containments*, U.S. Nuclear Regulatory Commission, February 14, 1991.

NRC Information Notice 89-79, *Degraded Coatings and Corrosion of Steel Containment Vessel*, U.S. Nuclear Regulatory Commission, December 1, 1989.

NRC Information Notice 89-79, Supplement 1, *Degraded Coatings and Corrosion of Steel Containment Vessel*, U.S. Nuclear Regulatory Commission, June 29, 1990.

XI.M30 FUEL OIL CHEMISTRY

Program Description

The program includes (a) surveillance and maintenance procedures to mitigate corrosion and (b) measures to verify the effectiveness of the mitigative actions and confirm the insignificance of an aging effect. Fuel oil quality is maintained by monitoring and controlling fuel oil contamination in accordance with the plant's technical specifications. Guidelines of the American Society for Testing Materials (ASTM) Standards, such as ASTM D 0975-04, D 1796-97, D 2276-00, D 2709-96, D 6217-98, and D 4057-95, also may be used. Exposure to fuel oil contaminants, such as water and microbiological organisms, is minimized by periodic draining or cleaning of tanks and by verifying the quality of new oil before its introduction into the storage tanks. However, corrosion may occur at locations in which contaminants may accumulate, such as tank bottoms. Accordingly, the effectiveness of the program is verified to ensure that significant degradation is not occurring and that the component's intended function is maintained during the period of extended operation. Thickness measurement of tank bottom surfaces is an acceptable verification program.

The fuel oil chemistry program is generally effective in removing impurities from intermediate and high flow areas. This report identifies those circumstances in which the fuel oil chemistry program is to be augmented to manage the effects of aging for license renewal. For example, the fuel oil chemistry program may not be effective in low flow or stagnant flow areas. Accordingly, in certain cases as identified in this report, verification of the effectiveness of the chemistry program is undertaken to ensure that significant degradation is not occurring and that the component's intended function is maintained during the period of extended operation. As discussed in this 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:*** Components within the scope of the program are the diesel fuel oil storage tanks, piping, and other metal components subject to aging management review that are exposed to an environment of diesel fuel oil. The program is focused on managing loss of material due to general, pitting, crevice, and microbiologically-influenced corrosion (MIC) and fouling that leads to corrosion of the diesel fuel tank internal surfaces.

2. ***Preventive Actions:*** The program reduces the potential for (a) exposure of the storage tanks' internal surface to fuel oil contaminated with water and microbiological organisms, reducing the potential for age-related degradation in other components exposed to diesel fuel oil; and (b) transport of corrosion products, sludge, or particulates to components serviced by the fuel oil storage tanks. Biocides or corrosion inhibitors may be added as a preventive measure or are added if periodic testing indicates biological activity or evidence of corrosion. Periodic cleaning of a tank allows removal of sediments, and periodic draining of water collected at the bottom of a tank minimizes the amount of water and the length of contact time. Accordingly, these measures are effective in mitigating corrosion inside diesel fuel oil tanks. Coatings, if used, prevent or mitigate corrosion by protecting the internal surfaces of the tank from contact with water and microbiological organisms.

3. ***Parameters Monitored/Inspected:*** The program is focused on managing loss of material due to general, pitting, crevice, and MIC, and fouling that leads to corrosion of the diesel fuel tank internal surfaces. The aging management program monitors fuel oil quality through

receipt testing and periodic sampling of stored fuel oil. Parameters monitored include water and sediment content, total particulate concentration, and the levels of microbiological organisms in the fuel oil. Water and microbiological organisms in the fuel oil storage tank increase the potential for corrosion. Sediment and total particulate content may be indicative of water intrusion or corrosion.

4. ***Detection of Aging Effects:*** Loss of material due to corrosion of the diesel fuel oil tank or other components exposed to diesel fuel oil cannot occur without exposure of the tank's internal surfaces to contaminants in the fuel oil, such as water and microbiological organisms. Periodic multilevel sampling provides assurance that fuel oil contaminants are below unacceptable levels. If tank design features do not allow for multilevel sampling, a sampling methodology that includes a representative sample from the lowest point in the tank may be used.

   At least once during the 10-year period prior to the period of extended operation, each diesel fuel tank is drained and cleaned, the internal surfaces are visually inspected (if physically possible) and volumetrically-inspected if evidence of degradation is observed during visual inspection, or if visual inspection is not possible. During the period of extended operation, at least once every 10 years, each diesel fuel tank is drained and cleaned, the internal surfaces are visually inspected (if physically possible), and, if evidence of degradation is observed during inspections, or if visual inspection is not possible, these diesel fuel tanks are volumetrically inspected.

   Prior to the period of extended operation, a one-time inspection (i.e., AMP XI.M32) of selected components exposed to diesel fuel oil is performed to verify the effectiveness of the Fuel Oil Chemistry program.

5. ***Monitoring and Trending:*** Water, biological activity, and particulate contamination concentrations are monitored and trended in accordance with the plant's technical specifications or at least quarterly.

6. ***Acceptance Criteria:*** Acceptance criteria for fuel oil quality parameters are as invoked or referenced in a plant's technical specifications. Additional acceptance criteria may be implemented using guidance from industry standards and equipment manufacturer or fuel oil supplier recommendations. ASTM D 0975-04 or other appropriate standards may be used to develop fuel oil quality acceptance criteria. Suspended water concentrations are in accordance with the applicable fuel oil quality specifications. Corrective actions are taken if microbiological activity is detected.

7. ***Corrective Actions:*** Specific corrective actions are implemented in accordance with the plant quality assurance (QA) program. For example, corrective actions are taken to prevent recurrence when the specified limits for fuel oil standards are exceeded or when water is drained during periodic surveillance. If accumulated water is found in a fuel oil storage tank, it is immediately removed. In addition, when the presence of biological activity is confirmed, a biocide is added to fuel oil. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

8. ***Confirmation Process:*** Site 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 for GALL, the staff finds the

requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process and administrative controls.

9. ***Administrative Controls:*** The administrative controls for this program provide for a formal review and approval of corrective actions. The administrative controls for this program are implemented through the site's QA program in accordance with the requirements of 10 CFR Part 50, Appendix B.

10. ***Operating Experience:*** The operating experience at some plants has included identification of water in the fuel, particulate contamination, and biological fouling. In addition, when a diesel fuel oil storage tank at one plant was cleaned and visually inspected, the inside of the tank was found to have unacceptable pitting corrosion (>50% of the wall thickness), which was repaired in accordance with American Petroleum Institute (API) 653 standard by welding patch plates over the affected area.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

API 653, *Tank Inspection, Repair, Alteration, and Reconstruction*, American Petroleum Institute, April 23, 2009.

ASTM D 0975-04, *Standard Specification for Diesel Fuel Oils*, American Society for Testing Materials, West Conshohocken, PA, 2004.

ASTM D 1796-97,*Standard Test Method for Water and Sediment in Fuel Oils by the Centrifuge Method*, American Society for Testing Materials, West Conshohocken, PA, 1997.

ASTM D 2276-00,*Standard Test Method for Particulate Contaminant in Aviation Fuel by Line Sampling,* American Society for Testing Materials, West Conshohocken, PA, 2000.

ASTM D 2709-96,*Standard Test Method for Water and Sediment in Middle Distillate Fuels by Centrifuge*, American Society for Testing Materials, West Conshohocken, PA, 1996.

ASTM D 4057-95, *Standard Practice for Manual Sampling of Petroleum and Petroleum Products*, American Society for Testing Materials, West Conshohocken, PA, 2000.

ASTM D 6217-98, *Standard Test Method for Particulate Contamination in Middle Distillate Fuels by Laboratory Filtration*, American Society for Testing Materials, West Conshohocken, PA, 1998.

NRC Regulatory Guide 1.137, Rev. 1, *Fuel-Oil Systems for Standby Diesel Generators*, U.S. Nuclear Regulatory Commission, October 1979.Safety Evaluation Report Related to the License Renewal of Three Mile Island Nuclear Unit 1, Section 3.0.3.2.12, *Fuel Oil Chemistry – Operating Experience*, June 2009.

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 International Standard Practice E 185-82 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 (2009), Paragraph III.C. Additional surveillance capsules may also be needed for the period of extended operation for this alternative.

The objective of the reactor vessel material surveillance program is to provide sufficient material data and dosimetry to (a) monitor irradiation embrittlement at the end of the period of extended operation and (b) 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.

The program is a condition monitoring program that measures the increase in Charpy V-notch 30 foot-pound (ft-lb) transition temperature and the drop in the upper shelf energy as a function of neutron fluence and irradiation temperature. The data from this surveillance program are used to monitor neutron irradiation embrittlement and are used in the time-limited aging analyses that are described in Section 4.2 of the Standard Review Plan for License Renewal. All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of the 1982 edition of ASTM E 185 (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 Nuclear Regulatory Commission (NRC) prior to implementation. Untested capsules placed in storage must be maintained for possible future insertion.

Evaluation and Technical Basis

The Reactor Vessel Surveillance program is 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.

1. ***Scope of Program:*** The program includes all reactor vessel beltline materials as defined by 10 CFR 50, Appendix G, Section II.F. Materials originally monitored within the scope of the licensee's existing 10 CFR Part 50, Appendix H, materials surveillance program will continue to serve as the basis for the reactor vessel surveillance aging management program unless safety considerations for the term of the renewed license would require the monitoring of additional or alternative materials.

2. ***Preventive Actions:*** The program is a surveillance program; no preventive actions are identified.

3. ***Parameters Monitored/Inspected:*** The program monitors reduction of fracture toughness of reactor vessel beltline materials due to neutron irradiation embrittlement and monitors reactor vessel long term operating conditions (cold leg operating temperature and neutron fluence) that could affect neutron irradiation embrittlement of the reactor vessel. The program uses two parameters to monitor the effects of neutron irradiation: (a) the increase in the Charpy V-notch 30 ft-lb transition temperature and (b) the drop in the Charpy V-notch upper shelf energy. The program uses neutron dosimeters to benchmark neutron fluence calculations. Low melting point elements or eutectic alloys may be used as a check on peak specimen irradiation temperature. Preferably, irradiation temperature will be monitored from cold leg operating temperatures. The Charpy V-notch specimens, neutron dosimeters, and temperature monitors are placed in capsules that are located within the reactor vessel; the capsules are withdrawn periodically to monitor the reduction in fracture toughness due to neutron irradiation.

4. ***Detection of Aging Effects:*** Reactor vessel beltline materials will be monitored by a surveillance program in which surveillance capsules are withdrawn from the reactor vessel and tested in accordance with ASTM E 185-82. This ASTM standard describes the methods used to monitor irradiation embrittlement (described in Element 3, above), selection of materials, and the withdrawal schedule for capsules. 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 III.C. Additional surveillance capsules may also be needed for the period of extended operation for this alternative.

   The plant-specific or integrated surveillance program shall have at least one capsule with a projected neutron fluence equal to or exceeding the 60-year peak reactor vessel wall neutron fluence prior to the end of the period of extended operation. The program withdraws one capsule at an outage in which the capsule receives a neutron fluence of between one and two times the peak reactor vessel wall neutron fluence at the end of the period of extended operation and tests the capsule in accordance with the requirements of ASTM E 185-82.

   It is recommended that the program retain additional capsules within the reactor vessel to support additional testing if, for example, the data from the required surveillance capsule turn out to be invalid or in preparation for operation beyond 60 years. If the projected neutron fluence for these additional capsules is expected to be excessive if left in the reactor vessel, the program may propose to withdraw and place one or more untested capsules in storage for future reinsertion and/or testing.

   If a plant has ample capsules remaining for future use, all pulled and tested samples or capsules placed in storage with reactor vessel neutron fluence less than 50% of the projected neutron fluence at the end of the period of extended operation may be discarded. Pulled and tested samples, unless discarded before August 31, 2000, and capsules with a neutron fluence greater than 50% of the projected reactor vessel neutron fluence at the end of the period of extended operation are placed in storage (these specimens and capsules are saved for future reconstitution and reinsertion use) unless the applicant has gained NRC approval to discard the pulled and tested samples or capsules.

If an applicant does not have ample capsules remaining for future use, all pulled and tested capsules, unless discarded before August 31, 2000, are placed in storage. (These specimens are saved for future reconstitution use, in case the surveillance program is reestablished.)

Plant-specific and fleet operating experience should be considered in determining the withdrawal schedule for all capsules; the withdrawal schedule shall be submitted as part of a license renewal application for NRC review and approval in accordance with 10 CFR Part 50, Appendix H.

If all surveillance capsules have been removed, a licensee may seek membership in an integrated surveillance program unless the integrated surveillance program does not have surveillance material representative of its limiting beltline materials or the program can propose one of the following:

(a) An Active Surveillance Program with Reinstituted Specimens

This program consists of (1) capsules from a surveillance program described above, (2) reconstitution of specimen from tested capsules, (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.

(b) An Alternative Neutron Monitoring Program

Programs without in-vessel capsules use alternative dosimetry to monitor neutron fluence during the period of extended operation.

If all surveillance capsules have been removed, operating restrictions are established to ensure that the plant is operated under conditions to which the surveillance capsules were exposed. 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 or more years is reviewed and, if deemed appropriate, modifications are made to the Reactor Vessel Surveillance program. Any changes to the Reactor Vessel Surveillance program must be submitted for NRC review and approval in accordance with 10 CFR Part 50, Appendix H.

5. ***Monitoring and Trending:*** The program provides reactor vessel material fracture toughness data for the time limited aging analyses (TLAAs) on neutron irradiation embrittlement (e.g., upper-shelf energy, pressurized thermal shock and pressure-temperature limits evaluations, etc.) for 60 years. The program is designed to periodically remove and test capsules for monitoring and trending purposes. Refer to the Standard Review Plan for License Renewal, Section 4.2, for the NRC acceptance criteria and review procedures for reviewing TLAAs for neutron irradiation embrittlement.

The TLAAs are projected in accordance with NRC Regulatory Guide (RG) 1.99, Rev. 2, “Radiation Embrittlement of Reactor Vessel Materials,” and the pressurized thermal shock rules (10 CFR 50.61 or 10 CFR 50.61a). When using NRC RG 1.99, Rev. 2, or equivalent provisions in 10 CFR 50.61, a licensee has a choice of the following:

(a) Neutron Embrittlement Using Chemistry Tables and Upper Shelf Energy Figures

An applicant may use the tables and figures 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 NRC RG 1.99, Rev. 2.

(b) Neutron Embrittlement Using Surveillance Data

When two or more credible surveillance data sets 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 and extended operating term. A plant-specific program or an integrated surveillance program during the period of extended operation provides for the collection of additional data.

A program that determines embrittlement by using NRC RG 1.99, Rev. 2, tables and figures (item [a]) uses the applicable limitations in Regulatory Position 1.3 of NRC RG 1.99, Rev. 2. The limits are based on material properties, temperature, material chemistry, and neutron fluence.

The program that determines embrittlement by using surveillance data (item [b]) 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 NRC RG 1.99, Rev. 2.

The reactor vessel monitoring program provides that if future plant operations exceed these limitations or bounds, 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 is evaluated and the NRC is notified.

6. ***Acceptance Criteria:*** The data are used for reactor vessel embrittlement projections to comply with 10 CFR Part 50, Appendix G, requirements and 10 CFR 50.61 or 10 CFR 50.61a limits through the period of extended operation.

7. ***Corrective Actions:*** There are no acceptance criteria that apply to the surveillance data, but the results of surveillance capsule testing will be incorporated into site operating limitations. The data will be used for reactor vessel embrittlement projections to comply with 10 CFR Part 50, Appendix G, requirements and 10 CFR 50.61 or 10 CFR 50.61a limits through the period of extended operation.

If a capsule is not withdrawn as scheduled, the NRC is notified and a revised withdrawal schedule is submitted to the NRC.

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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

8. ***Confirmation Process:*** Site 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process, and administrative controls.

9. ***Administrative Controls:*** The administrative controls for this program provide for a formal review and approval of corrective actions. The administrative controls for this program are implemented through the site's QA program in accordance with the requirements of 10 CFR Part 50, Appendix B.

10. ***Operating Experience:*** The existing reactor vessel material surveillance program provides sufficient material data and dosimetry to (a) monitor irradiation embrittlement at the end of the period of extended operation and (b) determine the need for operating restrictions on the inlet temperature, neutron fluence, and neutron flux.

References

10 CFR Part 50, Appendix G, *Fracture Toughness Requirements,* Office of the Federal Register, National Archives and Records Administration, 2009.

10 CFR Part 50, Appendix H, *Reactor Vessel Material Surveillance Program Requirements*, Office of the Federal Register, National Archives and Records Administration, 2009.

10 CFR 50.61, *Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events*, Office of the Federal Register, National Archives and Records Administration, January 4, 2010.

10 CFR 50.61a, *Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events*, Office of the Federal Register, National Archives and Records Administration, January 4, 2010.

ASTM E 185-82, *Standard Practice for Conducting Surveillance Tests of Light-Water Cooled Nuclear Power Reactor Vessels,* American Society for Testing Materials, Philadelphia, PA. (Versions of ASTM E 185 to be used for the various aspects of the reactor vessel surveillance program are as specified in 10 CFR Part 50, Appendix H.)

NRC Regulatory Guide 1.99, Rev. 2, *Radiation Embrittlement of Reactor Vessel Materials*, U.S. Nuclear Regulatory Commission, May 1988.

XI.M32 ONE-TIME INSPECTION

Program Description

A one-time inspection of selected components is 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 under the XI.M2, "Water Chemistry," program 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 verify that unacceptable degradation is not occurring. It also may trigger additional actions that ensure the intended functions of affected components are maintained during the period of extended operation.

The program verifies the effectiveness of an AMP and confirms the insignificance of an aging effect. Situations in which additional confirmation is appropriate include (a) an aging effect is not expected to occur, but the data are insufficient to rule it out with reasonable confidence; or (b) an aging effect is expected to progress very slowly in the specified environment, but the local environment may be more adverse than generally expected. For these cases, confirmation demonstrates that either the aging effect is not occurring or that the aging effect is occurring very slowly and does not affect the component's or structure's intended function during the period of extended operation based on prior operating experience data.

This program does not address Class 1 piping less than nominal pipe size (NPS) 4. That piping is addressed in AMP XI.M35, "One Time Inspection of ASME Code Class 1 Small Bore-Piping."

The elements of the program include (a) determination of the sample size of components to be inspected 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 potential for the aging effect to occur; (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.

An acceptable (one-time inspection) program to verify system-wide effectiveness of an AMP may consist of a one-time inspection of selected components and susceptible locations in the selected system. Verification may include a review of routine maintenance, repair, or inspection records to confirm that selected components have been inspected for aging degradation and that significant aging degradation has not occurred. A one-time inspection program is acceptable to verify the effectiveness of AMP XI.M2, "Water Chemistry"; AMP XI.M30, "Fuel Oil Chemistry"; and AMP XI.M39, "Lubricating Oil Analysis," programs or where the environment in the period of extended operation is expected to be equivalent to that in the prior 40 years and for which no aging effects have been observed. However, one-time inspection for environments that do not fall in the above category, or of any other action or program created to verify the effectiveness of an AMP and confirm the absence of an aging effect, is to be reviewed by the staff on a plant-specific basis.

This program cannot be used for structures or components with known age-related degradation mechanisms or when the environment in the period of extended operation is not expected to be equivalent to that in the prior 40 years. Periodic inspections should be proposed in these cases.

Evaluation and Technical Basis

1. ***Scope of Program:*** The scope of this program includes systems and components that are subject to aging management using the GALL AMPs XI.M2, "Water Chemistry"; XI.M30, "Fuel Oil Chemistry"; and XI.M39, "Lubricating Oil Analysis," and for which no aging effects have been observed or for which the aging effect is occurring very slowly and does not affect the component's or structure's intended function during the period of extended operation based on prior operating experience data. The scope of this program also may include other components and materials where the environment in the period of extended operation is expected to be equivalent to that in the prior 40 years and for which no aging effects have been observed.

   The program cannot be used for structures or components subjected to known age-related degradation mechanisms or when the environment in the period of extended operation is not expected to be equivalent to that in the prior 40 years. Periodic inspections should be proposed in these cases.

2. ***Preventive Actions:*** One-time inspection is a condition monitoring program. It does not include methods to mitigate or prevent age-related degradation.

3. ***Parameters Monitored/Inspected:*** The program monitors parameters directly related to the age-related degradation of a component. Examples of parameters monitored and the related aging effect are provided in the table in Element 4, below. Inspection is performed using a variety of nondestructive examination (NDE) methods, including visual, volumetric, and surface techniques.

4. ***Detection of Aging Effects:*** Elements of the program include (a) determination of the sample size of components to be inspected 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 potential for the aging effect to occur; and (c) determination of the examination technique, including acceptance criteria that would be effective in managing the aging effect for which the component is examined. Where practical, the inspection includes a representative sample of the system population and focuses on the bounding or lead components most susceptible to aging due to time in service, and severity of operating conditions. For components managed by the AMP XI.M2, Water Chemistry"; AMP XI.M30, "Fuel Oil Chemistry"; and AMP XI.M39, "Lubricating Oil Analysis," programs, a representative sample size is 20% of the population (defined as components having the same material, environment, and aging effect combination) or a maximum of 25 components. Otherwise, a technical justification of the methodology and sample size used for selecting components for one-time inspection should be included as part of the program's documentation.

   The program relies on established NDE techniques, including visual, ultrasonic, and surface techniques. Inspections are performed by personnel qualified in accordance with site procedures and programs to perform the type of examination specified. For code components, examinations should follow procedures consistent with the American Society of Mechanical Engineers (ASME) Code[16] and 10 CFR Part 50, Appendix B. For non-code components, examinations should follow site procedures that include requirements for items such as lighting, presence of protective coatings, and cleaning processes that ensure an adequate examination. In addition, a description of Enhanced Visual Examination (EVT-1) is found in Boiling Water Reactor Vessel and Internals Project (BWRVIP)-03 and Materials Reliability Program (MRP)-228.

The inspection and test techniques shall have a demonstrated history of effectiveness in detecting the aging effect of concern. Typically, the one-time inspections shall be performed as indicated in the following table.

Examples of Parameters Monitored or Inspected and Aging Effect for Specific Structure or Component[17]			
Aging Effect	**Aging Mechanism**	**Parameter(s) Monitored**	**Inspection Method[18]**
Loss of Material	Crevice Corrosion	Surface Condition, Wall Thickness	Visual (VT-1 or equivalent) and/or Volumetric (ultrasonic testing [UT])
Loss of Material	Galvanic Corrosion	Surface Condition, Wall Thickness	Visual (VT-3 or equivalent) and/or Volumetric (UT)
Loss of Material	General Corrosion	Surface Condition, Wall Thickness	Visual (VT-3 or equivalent) and/or Volumetric (UT)
Loss of Material	MIC	Surface Condition, Wall Thickness	Visual (VT-3 or equivalent) and/or Volumetric (UT)
Loss of Material	Pitting Corrosion	Surface Condition, Wall Thickness	Visual (VT-1 or equivalent) and/or Volumetric (UT)
Loss of Material	Erosion	Surface Condition, Wall Thickness	Visual (VT-3 or equivalent) and/or Volumetric (UT)
Reduction of Heat Transfer	Fouling	Tube Fouling	Visual (VT-3 or equivalent)
Cracking	SCC or Cyclic Loading	Surface Condition, Cracks	Enhanced Visual (EVT-1 or equivalent) or Surface Examination (magnetic particle, liquid penetrant) or Volumetric (radiographic testing or UT)

With respect to inspection timing, the sample 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 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

[16] Refer to the GALL Report, Chapter I, for application of other editions of the ASME Code, Section XI.
[17] 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 this table, a technical justification should be provided as an exception to this AMP. This exception should list the AMR line item component, examination technique, acceptance criteria, evaluation standard, and a description of the justification.
[18] Visual inspection may be used only when the inspection methodology examines the surface potentially experiencing the aging effect.

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 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 operated for at least 30 years before inspections under this program begin, sufficient time will have elapsed for any aging effects to be manifested.

5. ***Monitoring and Trending:*** This is a one-time inspection program. Monitoring and trending are not applicable.

6. ***Acceptance Criteria:*** Any indication or relevant conditions of degradation detected are evaluated. Acceptance criteria may be based on applicable ASME or other appropriate standards, design basis information, or vendor-specified requirements and recommendations. For example, ultrasonic thickness measurements are compared to predetermined limits.

7. ***Corrective Actions:*** Unacceptable inspection findings are evaluated in accordance with the site's corrective action process to determine appropriate corrective actions and the need for subsequent (including periodic) inspections under another AMP. 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions, confirmation process, and administrative controls.

8. ***Confirmation Process:*** Confirmation processes to ensure that preventive actions are adequate and that appropriate corrective actions have been completed and are effective are implemented through the site QA program in accordance with the requirements of 10 CFR Part 50, Appendix B.

9. ***Administrative Controls:*** Administrative controls to provide a formal review and approval for corrective actions are implemented through the site QA program in accordance with the requirements of 10 CFR Part 50, Appendix B.

10. ***Operating Experience:*** The elements that comprise inspections associated with this program (the scope of the inspections and inspection techniques) are consistent with industry practice. An applicant's operating experience with detection of aging effects should be adequate to demonstrate that the program is capable of detecting the presence or noting the absence of aging effects in the components, materials, and environments where one-time inspection is used to confirm system-wide effectiveness of another preventive or mitigative AMP.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

10 CFR 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2009.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, The ASME Boiler and Pressure Vessel Code, 2004 edition as approved in 10 CFR 50.55a, The American Society of Mechanical Engineers, New York, NY.

BWRVIP-03 (EPRI 105696- R6), *BWR Vessel and Internals Project: Reactor Pressure Vessel and Internals Examination Guidelines*, January 6, 2004, Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation, June 2008.

MRP-228, *Materials Reliability Program: Inspection Standard for PWR Internals*, 2009.

XI.M33 SELECTIVE LEACHING

Program Description

This program demonstrates the absence of selective leaching. The program for selective leaching of materials ensures the integrity of the components made of gray cast iron and copper alloys (except for inhibited brass) that contain greater than 15 percent zinc (> 15% Zn) or greater than 8 percent aluminum (>8% Al in the case of aluminum-bronze) exposed to a raw water, closed cooling water, treated water, or ground water environment that may lead to selective leaching of one of the metal components where there has not been previous experience of selective leaching. The AMP includes a one-time visual inspection of selected components that may be susceptible to selective leaching, coupled with either hardness measurements (where feasible, based on form and configuration) or mechanical examination techniques. These techniques can determine whether loss of materials due to selective leaching is occurring and whether selective leaching will affect the ability of the components to perform their intended function for the period of extended operation.

The selective leaching process involves the preferential removal of one of the alloying elements from the material, which leads to the enrichment of the remaining alloying elements. Dezincification (loss of zinc from brass) and graphitization (removal of iron from cast iron) are examples of such a process. Susceptible materials, high temperatures, stagnant-flow conditions, and a corrosive environment, such as acidic solutions for brasses with high zinc content and dissolved oxygen, are conducive to selective leaching.

Although the program does not provide guidance on preventive action, it is noted that monitoring of water chemistry to control pH and concentration of corrosive contaminants and treatment to minimize dissolved oxygen in water are effective in reducing selective leaching. Water chemistry is managed by the Water Chemistry program (AMP XI.M2).

Evaluation and Technical Basis

1. ***Scope of Program:*** This program demonstrates the absence of selective leaching. For materials and environments where selective leaching is currently occurring or for materials in environments where the component has been repaired with the same material, a plant-specific program is required. Components include piping, valve bodies and bonnets, pump casings, and heat exchanger components that are susceptible to selective leaching. The materials of construction for these components may include gray cast iron and uninhibited brass containing greater than 15% zinc. These components may be exposed to raw water, treated water, closed cooling water, ground water, water contaminated fuel oil, or water-contaminated lube oil.

2. ***Preventive Actions:*** This program is a condition monitoring program and it contains no preventive actions.

3. ***Parameters Monitored/Inspected:*** This program monitors selective leaching through the monitoring of surface hardness and visual appearance (color, porosity, abnormal surface conditions).

4. ***Detection of Aging Effects:*** The visual inspection and hardness measurement or other mechanical examination techniques, such as destructive testing (when the opportunity arises), chipping, or scraping, is a one-time inspection conducted within the last 5 years

prior to entering the period of extended operation. Because selective leaching is a slow acting corrosion process, this measurement is performed just prior to the period of extended operation. Follow-up of unacceptable inspection findings includes an evaluation using the corrective action program and a possible expansion of the inspection sample size and location.

Where practical, the inspection includes a representative sample of the system population and focuses on the bounding or lead components most susceptible to aging due to time in service, severity of operating conditions, and lowest design margin. Twenty percent of the population with a maximum sample of 25 constitutes a representative sample size. Otherwise, a technical justification of the methodology and sample size used for selecting components for one-time inspection should be included as part of the program's documentation. Each group of components with different material/environment combinations is considered a separate population.

Selective leaching generally does not cause changes in dimensions and is difficult to detect by visual inspection. However, in certain brasses, it causes plug-type dezincification, which can be detected by visual inspection. One acceptable procedure is to visually inspect the susceptible components closely and conduct Brinell hardness testing (where feasible, based on form and configuration or other industry-accepted mechanical inspection techniques) on the inside surfaces of the selected set of components to determine if selective leaching has occurred. If selective leaching is apparent, an engineering evaluation is initiated to determine acceptability of the affected components for further service.

5. ***Monitoring and Trending:*** This is a one-time inspection to determine if selective leaching is an issue. Monitoring and trending is not required.

6. ***Acceptance Criteria:*** The acceptance criteria are no visible evidence of selective leaching or no more than a 20 percent decrease in hardness. For copper alloys with greater than 15 percent zinc, the criteria is no noticeable change in color from the normal yellow color to the reddish copper color.

7. ***Corrective Actions:*** Engineering evaluations are performed for test or inspection results that do not satisfy established acceptance criteria. The corrective actions program ensures that conditions adverse to quality are promptly corrected. If the deficiency is assessed to be significantly adverse to quality, the cause of the condition is determined and an action plan is developed to preclude repetition. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions. Unacceptable inspection findings result in additional inspection(s) being performed, which may be on a periodic basis, or in component repair or replacement.

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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process and administrative controls.

9. ***Administrative Controls:*** The administrative controls for this program provide for a formal review and approval of corrective actions. The administrative controls for this program are implemented through the site's QA program in accordance with the requirements of 10 CFR Part 50, Appendix B.

10. ***Operating Experience:*** The elements that comprise these inspections (e.g., the scope of the inspections and inspection techniques) are consistent with industry practice and staff expectations. Selective leaching has been detected in components constructed from cast iron, brass, bronze, and aluminum bronze. Components affected have included valve bodies, pump casings, piping, and cast iron fire protection piping buried in soil.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

EPRI TR-107514, *Age Related Degradation Inspection Method and Demonstration*, Electric Power Research Institute, April 1998.

Fontana, M. G., *Corrosion Engineering*, McGraw Hill, p 86-90, 1986.

NUREG-1705, *Safety Evaluation Report Related to the License Renewal of Calvert Cliffs Nuclear Power Plant, Units 1 and 2*, U.S. Nuclear Regulatory Commission, December 1999.

NUREG-1723, *Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station, Units 1, 2, and 3*, U.S. Nuclear Regulatory Commission, March 2000.

NUREG-1930, *Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Units 2 and 3*, U.S. Nuclear Regulatory Commission, November 2009.

Schweitzer, P. A., *Encyclopedia of Corrosion Technology 2nd Ed*, Marcel Dekker, p 201-202. March 17, 2004.

XI.M35 ONE-TIME INSPECTION OF ASME CODE CLASS 1 SMALL-BORE PIPING

Program Description

This program augments the requirements in American Society of Mechanical Engineers (ASME) Code, Section XI, 2004 edition[19], and is applicable to small-bore ASME Code Class 1 piping and systems less than 4 inches nominal pipe size (less than NPS 4) and greater than or equal to NPS 1. The program includes pipes, fittings, branch connections, and all full and partial penetration (socket) welds.

According to Table IWB-2500-1, Examination Category B-J, Item No. B9.21 and B9.40 of the current ASME Code, an external surface examination of small-bore Class 1 piping should be included for piping less than NPS 4. Other ASME Code provisions exempt from examination piping NPS 1 and smaller. This program is augmented to include piping from NPS 1 to less than NPS 4. Also, Examination Category B-P requires system leakage of all Class 1 piping. However, the staff believes that for a one-time inspection to detect cracking resulting from thermal and mechanical loading or intergranular stress corrosion of full-penetration welds, the inspection should be a volumetric examination. For a one-time inspection to detect cracking in socket welds, the inspection should be either a volumetric or opportunistic destructive examination. (Opportunistic destructive examination is performed when a weld is removed from service for other considerations, such as plant modifications. A sampling basis is used if more than 1 weld is removed.) These examinations provide additional assurance that either aging of small-bore ASME Code Class 1 piping is not occurring or the aging is insignificant, such that a plant-specific aging management program (AMP) is not warranted.

This program is applicable to systems that have not experienced cracking of ASME Code Class 1 small-bore piping. This program can also be used for systems that experienced cracking but have implemented design changes to effectively mitigate cracking. (Measure of effectiveness includes (1) the one-time inspection sampling is statistically significant;(2) samples will be selected as described in Element 5, Monitoring and Trending below; and (3) no repeated failures over an extended period of time.) For systems that have experienced cracking and operating experience indicates that design changes have not been implemented to effectively mitigate cracking, periodic inspection is proposed, as managed by a plant-specific AMP. Should evidence of cracking be revealed by a one-time inspection, periodic inspection is implemented using a plant-specific AMP.

If small bore piping in a particular plant system has experienced cracking, small bore piping in all plant systems are evaluated to determine whether the cause for the cracking affects other systems (corrective action program).

Evaluation and Technical Basis

1. ***Scope of Program:*** This program is a one-time inspection of a sample of ASME Code Class 1 piping less than NPS 4 and greater than or equal to NPS 1. This program includes measures to verify that degradation is not occurring, thereby either confirming that there is no need to manage age-related degradation or validating the effectiveness of any existing AMP for the period of extended operation. The one-time inspection program for ASME Code Class 1 small-bore piping includes locations that are susceptible to cracking.

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[19] Refer to the GALL Report, Chapter I, for applicability of other editions of the ASME Code, Section XI.

2. ***Preventive Actions:*** This program is a condition monitoring activity independent of methods to mitigate or prevent degradation.

3. ***Parameters Monitored/Inspected:*** This inspection detects cracking in ASME Code Class 1 small-bore piping.

4. ***Detection of Aging Effects:*** This one-time inspection is designed to provide assurance that aging of ASME Code Class 1 small-bore piping is not occurring, or that the effects of aging are not significant. This inspection does not apply to those plants that have experienced cracking due to stress corrosion, cyclical (including thermal, mechanical, and vibration fatigue) loading, or thermal stratification and thermal turbulence (MRP 146 and MRP 146S). For a one-time inspection to detect cracking in socket welds, the inspection should be either a volumetric or opportunistic destructive examination. (Opportunistic destructive examination is performed when a weld is removed from service for other considerations, such as plant modifications. A sampling basis is used if more than one weld is removed.) For a one-time inspection to detect cracking resulting from thermal and mechanical loading or intergranular stress corrosion of full penetration welds, the inspection should be a volumetric examination. Volumetric examination is performed using demonstrated techniques that are capable of detecting the aging effects in the examination volume of interest. This inspection should be performed at a sufficient number of locations to ensure an adequate sample. This number, or sample size, is based on susceptibility, inspectability, dose considerations, operating experience, and limiting locations of the total population of ASME Code Class 1 small-bore piping locations.

   If an applicant has never experienced a failure in its ASME Code Class 1 piping (a through-wall crack detected in the subject component by evidence of leakage, or through nondestructive or destructive examination) and has extensive operating history (more than 30 years of operation at time of submitting the application), the inspection sample size should be at least 3% of the weld population or a maximum of 10 welds of each weld type for each operating unit. If the applicant has successfully mitigated any failures in its ASME Class 1 piping, the inspection should include 10% of the weld population or a maximum of 25 welds of each weld type (e.g., full penetration or socket weld) for each operating unit using a methodology to select the most susceptible and risk-significant welds. For socket welds, opportunistic destructive examination can be performed in lieu of volumetric examination. Because more information can be obtained from a destructive examination than from nondestructive examination, the applicant may take credit for each weld destructively examined equivalent to having volumetrically examined two welds.

   The one time inspection should be completed within the six year period prior to the period of extended operation.

5. ***Monitoring and Trending:*** This is a one-time inspection to determine whether cracking in ASME Code Class 1 small-bore piping resulting from stress corrosion, cyclical (including thermal, mechanical, and vibration fatigue) loading, or thermal stratification and thermal turbulence (MRP 146 and MRP 146S) is an issue. Evaluation of the inspection results may indicate the need for additional or periodic examinations (i.e., a plant-specific AMP for Class 1 small-bore piping using volumetric inspection methods consistent with ASME Code, Section XI, Subsection IWB).

6. ***Acceptance Criteria:*** If flaws or indications exceed the acceptance criteria of ASME Code, Section XI, Paragraph IWB-3400, they are evaluated in accordance with ASME Code,

Section XI, Paragraph IWB-3131; additional examinations are performed in accordance with ASME Code, Section XI, Paragraph IWB-2430. Evaluation of flaws identified during a volumetric examination of socket welds should be in accordance with IWB-3600.

7. ***Corrective Actions:*** The site corrective action program, quality assurance 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions, confirmation process, and administrative controls. Should evidence of cracking be revealed by a one-time inspection, periodic inspection is implemented, as managed by a plant-specific AMP.

8. ***Confirmation Process:*** As discussed in the Appendix for GALL, 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the administrative controls.

10. ***Operating Experience*:** This inspection uses volumetric inspection techniques with demonstrated capability and a proven industry record to detect cracking in piping weld and base material.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

10 CFR 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2009.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, The ASME Boiler and Pressure Vessel Code, 2004 edition as approved in 10 CFR 50.55a, The American Society of Mechanical Engineers, New York, NY.

EPRI 1011955, *Materials Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines (MRP-146*), June 8, 2005.

EPRI 1018330, *Materials Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines – Supplemental Guidance (MRP-146S)*, December 31, 2008.

NRC Information Notice 97-46, *Unisolable Crack in High-Pressure Injection Piping*, U.S. Nuclear Regulatory Commission, July 9, 1997.

XI.M36 EXTERNAL SURFACES MONITORING OF MECHANICAL COMPONENTS

Program Description

The External Surfaces Monitoring of Mechanical Components program is based on system inspections and walkdowns. This program consists of periodic visual inspections of metallic and polymeric components, such as piping, piping components, ducting, polymeric components, and other components within the scope of license renewal and subject to aging management review (AMR) in order to manage aging effects. The program manages aging effects through visual inspection of external surfaces for evidence of loss of material, cracking, and change in material properties. When appropriate for the component and material, manipulation may be used to augment visual inspection to confirm the absence of elastomer hardening and loss of strength. Loss of material due to boric acid corrosion is managed by the Boric Acid Corrosion program (AMP XI.M10).

Evaluation and Technical Basis

1. ***Scope of Program:*** This program visually inspects the external surface of in-scope mechanical components and monitors external surfaces of metallic components in systems within the scope of license renewal and subject to AMR for loss of material and leakage. Cracking of stainless steel components exposed to an air environment containing halides may also be managed. This program also visually inspects and monitors the external surfaces of polymeric components in mechanical systems within the scope of license renewal and subject to AMR for changes in material properties (such as hardening and loss of strength), cracking, and loss of material due to wear. This program manages the effects of aging of polymer materials in all environments to which these materials are exposed.

   The program may also be credited with managing loss of material from internal surfaces of metallic components and with loss of material, cracking, and change in material properties from the internal surfaces of polymers, for situations in which material and environment combinations are the same for internal and external surfaces such that external surface condition is representative of internal surface condition. When credited, the program should describe the component internal environment and the credited similar external component environment inspected.

2. ***Preventive Actions:*** The External Surfaces Monitoring of Mechanical Components program is a condition monitoring program that does not include preventive actions.

3. ***Parameters Monitored/Inspected:*** The External Surfaces Monitoring of Mechanical Components program utilizes periodic plant system inspections and walkdowns to monitor for material degradation and leakage. This program inspects components such as piping, piping components, ducting, polymeric components, and other components. For metallic components, coatings deterioration is an indicator of possible underlying degradation. The aging effects for flexible polymeric components may be monitored through a combination of visual inspection and manual or physical manipulation of the material. “Manual or physical manipulation of the material” means touching, pressing on, flexing, bending, or otherwise manually interacting with the material. The purpose of the manual manipulation is to reveal changes in material properties, such as hardness, and to make the visual examination process more effective in identifying aging effects such as cracking.

   Examples of inspection parameters for metallic components include:

- corrosion and material wastage (loss of material)
- leakage from or onto external surfaces (loss of material)
- worn, flaking, or oxide-coated surfaces (loss of material)
- corrosion stains on thermal insulation (loss of material)
- protective coating degradation (cracking, flaking, and blistering)
- leakage for detection of cracks on the external surfaces of stainless steel components exposed to an air environment containing halides

Examples of inspection parameters for polymers include:

- surface cracking, crazing, scuffing, and dimensional change (e.g., “ballooning” and “necking”)
- discoloration
- exposure of internal reinforcement for reinforced elastomers
- hardening as evidenced by a loss of suppleness during manipulation where the component and material are appropriate to manipulation

4. ***Detection of Aging Effects:*** This program manages aging effects of loss of material, cracking, and change in material properties using visual inspection. For coated surfaces, confirmation of the integrity of the paint or coating is an effective method for managing the effects of corrosion on the metallic surface.

   When required by the ASME Code, inspections are conducted in accordance with the applicable code requirements. In the absence of applicable code requirements, plant-specific visual inspections are performed of metallic and polymeric component surfaces using plant-specific procedures implemented by inspectors qualified through plant-specific programs. The inspections are capable of detecting age-related degradation and are performed at a frequency not to exceed one refueling cycle. This frequency accommodates inspections of components that may be in locations that are normally only accessible during outages or access is physically restricted (underground). Surfaces that are not readily visible during plant operations and refueling outages are inspected when they are made accessible and at such intervals that would ensure the components’ intended functions are maintained. The inspections of underground components shall be conducted during each 10-year period beginning 10 years prior to entering the period of extended operation. These normally underground components should be clearly identified in the program scope and inspection intervals provided. Surfaces that are insulated may be inspected when the external surface is exposed (i.e., during maintenance) at such intervals that would ensure that the components’ intended functions are maintained. The intervals of inspections may be adjusted, as necessary, based on plant-specific inspection results and industry operating experience.

   Visual inspection will identify indirect indicators of flexible polymer hardening and loss of strength and include the presence of surface cracking, crazing, discoloration, and, for elastomers with internal reinforcement, the exposure of reinforcing fibers, mesh, or underlying metal. Visual inspection should be 100% of accessible components. Visual inspection will identify direct indicators of loss of material due to wear to include dimensional change, scuffing, and for flexible polymeric materials with internal reinforcement, the exposure of reinforcing fibers, mesh, or underlying metal. Manual or physical manipulation can be used to augment visual inspection to confirm the absence of hardening and loss of strength for flexible polymeric materials (e.g., HVAC flexible connectors) where appropriate. The sample size for manipulation should be at least 10 percent of available surface area. Hardening and loss of strength and loss of material due to wear for flexible polymeric materials are expected to be detectable prior to any loss of intended function.

This program is credited with managing the following aging effects.

- loss of material and cracking for external surfaces
- loss of material for internal surfaces exposed to the same environment as the external surface
- cracking and change in material properties (hardening and loss of strength) of flexible polymers

5. ***Monitoring and Trending:*** Visual inspection and manual or physical manipulation activities are performed and associated personnel are qualified in accordance with site controlled procedures and processes. The External Surfaces Monitoring of Mechanical Components program uses standardized monitoring and trending activities to track degradation. Deficiencies are documented using approved processes and procedures, such that results can be trended. However, the program does not include formal trending. Inspections are performed at frequencies identified in Element 4, Detection of Aging Effects.

6. ***Acceptance Criteria:*** For each component/aging effect combination, the acceptance criteria are defined to ensure that the need for corrective actions will be identified before loss of intended functions. For metallic surfaces, any indications of relevant degradation detected are evaluated. For stainless steel surfaces, a clean, shiny surface is expected. The appearance of discoloration may indicate the loss of material on the stainless steel surface. For aluminum and copper alloys exposed to marine or industrial environments, any indications of relevant degradation that could impact their intended function are evaluated. For flexible polymers, a uniform surface texture and uniform color with no unanticipated dimensional change is expected. Any abnormal surface condition may be an indication of an aging effect for metals and for polymers. For flexible materials, changes in physical properties (e.g., the hardness, flexibility, physical dimensions, and color of the material are unchanged from when the material was new) should be evaluated for continued service in the corrective action program. Cracks should be absent within the material. For rigid polymers, surface changes affecting performance, such as erosion, cracking, crazing, checking, and chalking, are subject to further investigation. Acceptance criteria include design standards, procedural requirements, current licensing basis, industry codes or standards, and engineering evaluation.

7. ***Corrective Actions:*** Site quality assurance 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 for GALL, the staff finds the

requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions, confirmation process, and administrative controls.

8. ***Confirmation Process***: As discussed in the Appendix for GALL, 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the administrative controls.

10. ***Operating Experience:*** External surface inspections through system inspections and walkdowns have been in effect at many utilities since the mid 1990s in support of the Maintenance Rule (10 CFR 50.65) and have proven effective in maintaining the material condition of plant systems. The elements that comprise these inspections (e.g., the scope of the inspections and inspection techniques) are consistent with industry practice.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

EPRI Technical Report 1007933, *Aging Assessment Field Guide*, December 2003.

EPRI Technical Report 1009743, *Aging Identification and Assessment Checklist*, August 27, 2004.

INPO Good Practice TS-413, *Use of System Engineers*, INPO 85-033, May 18, 1988.

XI.M37 FLUX THIMBLE TUBE INSPECTION

Program Description

The Flux Thimble Tube Inspection is a condition monitoring program used to inspect for thinning of the flux thimble tube wall, which provides a path for the incore neutron flux monitoring system detectors and forms part of the reactor coolant system (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. A nondestructive examination methodology, such as eddy current testing (ECT) or other applicant-justified and U.S. Nuclear Regulatory Commission (NRC)-accepted inspection method, is used to monitor for wear of the flux thimble tubes. This program implements the recommendations of NRC IE Bulletin 88-09, as described below.

Evaluation and Technical Basis

1. ***Scope of Program:*** The flux thimble tube inspection encompasses all of the flux thimble tubes that form part of the RCS pressure boundary. The instrument guide tubes are not in the scope of this program. Within scope are the licensee responses to IE Bulletin 88-09, as accepted by the staff in its closure letters on the bulletin, and any amendments to the licensee responses as approved by the staff.

2. ***Preventive Actions:*** The program consists of inspection and evaluation and provides no guidance on preventive actions.

3. ***Parameters Monitored/Inspected:*** Flux thimble tube wall thickness is 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 is used to detect loss of material during the period of extended operation. Justification for methods other than ECT should be provided unless use of the alternative method has been previously accepted by the NRC.

   Examination frequency is based upon actual plant-specific wear data and wear predictions that have been technically justified as providing conservative estimates of flux thimble tube wear. The interval between inspections is established such that no flux thimble tube is predicted to incur wear that exceeds the established acceptance criteria before the next inspection. The examination frequency may be adjusted based on plant-specific wear projections. Rebaselining of the examination frequency should be justified using plant-specific wear-rate data unless prior plant-specific NRC acceptance for the re-baselining is received outside the license renewal process. If design changes are made to use more wear-resistant thimble tube materials (e.g., chrome-plated stainless steel), sufficient inspections are conducted at an adequate inspection frequency, as described above, for the new materials.

5. ***Monitoring and Trending:*** Flux thimble tube wall thickness measurements are trended and wear rates are calculated based on plant-specific data. Wall thickness is projected using plant-specific data and a methodology that includes sufficient conservatism to ensure that wall thickness acceptance criteria continue to be met during plant operation between scheduled inspections.

6. ***Acceptance Criteria:*** Appropriate acceptance criteria, such as percent through-wall wear, are established, and inspection results are evaluated and compared with the acceptance criteria. The acceptance criteria are 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 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. Acceptance criteria different from those previously documented in the applicant's response to IE Bulletin 88-09 and amendments thereto, as accepted by the NRC, should be justified.

7. ***Corrective Actions:*** Flux thimble tubes with wall thickness that do not meet the established acceptance criteria are 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 are subject to wear due to restriction or other defects, and that cannot be shown by analysis to be satisfactory for continued service are removed from service to ensure the integrity of the reactor coolant system pressure boundary.

   The site corrective actions program, quality assurance 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions, confirmation process, and administrative controls.

8. ***Confirmation Process:*** As discussed in the Appendix for GALL, 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B acceptable to address the administrative controls.

10. ***Operating Experience:*** In IE Bulletin 88-09 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.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

NRC IE Bulletin 88-09, *Thimble Tube Thinning in Westinghouse Reactors*, July 26, 1988.

NRC Information Notice No. 87-44, *Thimble Tube Thinning in Westinghouse Reactors*, September 16, 1987.

NRC Information Notice No. 87-44, Supplement 1, *Thimble Tube Thinning in Westinghouse Reactors*, March 28, 1988.

XI.M38 INSPECTION OF INTERNAL SURFACES IN MISCELLANEOUS PIPING AND DUCTING COMPONENTS

Program Description

The program consists of inspections of the internal surfaces of metallic piping, piping components, ducting, polymeric components, and other components that are exposed to air-indoor uncontrolled, air outdoor, condensation, and any water system other than open-cycle cooling water system (XI.M20), closed treated water system (XI.M21A), and fire water system (XI.M27). These internal inspections are performed during the periodic system and component surveillances or during the performance of maintenance activities when the surfaces are made accessible for visual inspection. The program includes visual inspections to ensure that existing environmental conditions are not causing material degradation that could result in a loss of component intended functions. For certain materials, such as polymers, physical manipulation or pressurization (e.g., hydrotesting) to detect hardening or loss of strength should be used to augment the visual examinations conducted under this program. If visual inspection of internal surfaces is not possible, then the applicant needs to provide a plant-specific program.

This program is not intended for use on piping and ducts where repetitive failures have occurred from loss of material that resulted in loss of intended function. If operating experience indicates that there have been repetitive failures caused by loss of material, a plant-specific program will be required. Following a failure, this program may be used if the failed material is replaced by one that is more corrosion-resistant in the environment of interest.

Evaluation and Technical Basis

1. ***Scope of Program:*** For metallic components, the program calls for the visual inspection of the internal surface of in-scope components that are not included in other aging management programs for loss of material. For metallic components with polymeric liners or for polymeric and elastomeric components, the program includes visual inspections of the internal polymer surfaces when coupled with additional augmented techniques, such as manipulation or pressurization. This program also includes metallic piping with or without polymeric linings, piping elements, ducting, and components in an internal environment. The program also calls for visual inspection and monitors the internal surfaces of polymeric and elastomeric components in mechanical systems for hardening and loss of strength, cracking, and for loss of material due to wear. The program manages the effects of aging of polymer materials in all environments to which these materials are exposed. Inspections are performed when the internal surfaces are accessible during the performance of periodic surveillances or during maintenance activities or scheduled outages. This program is not intended for piping and ducts where failures have occurred from loss of material from corrosion.

2. ***Preventive Actions:*** This program is a condition monitoring program to detect signs of degradation and does not provide guidance for prevention.

3. ***Parameters Monitored/Inspected:*** Parameters monitored or inspected include visible evidence of loss of material in metallic components.

   This program manages loss of material and possible changes in material properties. This program monitors for evidence of surface discontinuities. For changes in material properties, the visual examinations are supplemented, so changes in the properties are readily observable.

Examples of inspection parameters for metallic components include the following:

- corrosion and material parameters wastage (loss of material)
- leakage from or onto internal surfaces (loss of material)
- worn, flaking, or oxide-coated surfaces (loss of material)

Examples of inspection parameters for polymers are as follows:

- surface cracking, crazing, scuffing, and dimensional change (e.g., “ballooning” and “necking”)
- discoloration
- exposure of internal reinforcement for reinforced elastomers
- hardening as evidenced by a loss of suppleness during manipulation where the component and material are appropriate to manipulation

4. ***Detection of Aging Effects:*** Visual and mechanical inspections conducted under this program are opportunistic in nature; they are conducted whenever piping or ducting are opened for any reason. Visual inspections should include all accessible surfaces. Unless otherwise required (e.g., by the ASME code) all inspections should be carried out using plant-specific procedures by inspectors qualified through plant specific programs. The inspection procedures utilized must be capable of detecting the aging effect(s) under consideration. These inspections provide for the detection of aging effects prior to the loss of component function. Visual inspection of flexible polymeric components is performed whenever the component surface is accessible. Visual inspection can provide indirect indicators of the presence of surface cracking, crazing, and discoloration. For elastomers with internal reinforcement, visual inspection can detect the exposure of reinforcing fibers, mesh, or underlying metal. Visual and tactile inspections are performed when the internal surfaces become accessible during the performance of periodic surveillances or during maintenance activities or scheduled outages. Visual inspection provides direct indicators of loss of material due to wear, including dimensional change, scuffing, and the exposure of reinforcing fibers, mesh, or underlying metal for flexible polymeric materials with internal reinforcement.

   Manual or physical manipulation of flexible polymeric components is used to augment visual inspection, where appropriate, to assess loss of material or strength. The sample size for manipulation is at least 10 percent of available surface area, including visually identified suspect areas. For flexible polymeric materials, hardening, loss of strength, or loss of material due to wear is expected to be detectable prior to any loss of intended function.

5. ***Monitoring and Trending:*** The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program uses standardized monitoring and trending activities to track degradation. Deficiencies are documented using approved processes and procedures such

that results can be trended. However, the program does not include formal trending. Inspections are performed at frequencies identified in Element 4, Detection of Aging Effects.

6. ***Acceptance Criteria:*** For each component/aging effect combination, the acceptance criteria are defined to ensure that the need for corrective actions is identified before loss of intended functions. For metallic surfaces, any indications of relevant degradation detected are evaluated. For stainless steel surfaces, a clean, shiny surface is expected. Discoloration may indicate the loss of material on the stainless steel surface. Any abnormal surface condition may be an indication of an aging effect for metals.

   For flexible polymers, a uniform surface texture and uniform color with no unanticipated dimensional change is expected. Any abnormal surface condition may be an indication of an aging effect for metals and for polymers. For flexible materials to be considered acceptable, the inspection results should indicate that the flexible polymer material is in "as new" condition (e.g., the hardness, flexibility, physical dimensions, and color of the material are unchanged from when the material was new). Cracks should be absent within the material. For rigid polymers, surface changes affecting performance, such as erosion, cracking, crazing, checking, and chalks, are subject to further investigation.

   Acceptance criteria include design standards, procedural requirements, current licensing basis, industry codes or standards, and engineering evaluation.

7. ***Corrective Actions:*** The site corrective actions program, quality assurance 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions, confirmation process, and administrative controls.

8. ***Confirmation Process:*** As discussed in the GALL Report, the staff finds the requirements 10 CFR Part 50, Appendix B, acceptable to address the confirmation process.

9. ***Administrative Controls:*** As discussed in the GALL Report, the staff finds the requirements 10 CFR Part 50, Appendix B, acceptable to address the administrative controls.

10. ***Operating Experience:*** Inspections of internal surfaces during the performance of periodic surveillance and maintenance activities have been in effect at many utilities in support of plant component reliability programs. These activities have proven effective in maintaining the material condition of plant systems, structures, and components.

    The elements that comprise these inspections (e.g., the scope of the inspections and inspection techniques) are consistent with industry practice and staff expectations. However, because the inspection frequency is plant-specific and depends on the plant operating experience, the applicant's plant-specific operating experience or applicable generic operating experience is further evaluated for the period of extended operation. The applicant evaluates recent operating experience and provides objective evidence to support the conclusion that the effects of aging are adequately managed.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

EPRI Technical Report 1007933, *Aging Assessment Field Guide*, December 2003.

EPRI Technical Report 1009743, *Aging Identification and Assessment Checklist*, August 27, 2004.

INPO Good Practice TS-413, *Use of System Engineers*, INPO 85-033, May 18, 1988.

XI.M39 LUBRICATING OIL ANALYSIS

Program Description

The purpose of the Lubricating Oil Analysis program is to ensure that the oil environment in the mechanical systems is maintained to the required quality to prevent or mitigate age-related degradation of components within the scope of this program. This program maintains oil systems contaminants (primarily water and particulates) within acceptable limits, thereby preserving an environment that is not conducive to loss of material or reduction of heat transfer. Lubricating oil testing activities include sampling and analysis of lubricating oil for detrimental contaminants. The presence of water or particulates may also be indicative of inleakage and corrosion product buildup.

Although primarily a sampling program, the lubricating oil analysis program is generally effective in monitoring and controlling impurities. This report identifies when the program is to be augmented to manage the effects of aging for license renewal. Accordingly, in certain cases identified in this report, verification of the effectiveness of the program is undertaken to ensure that significant degradation is not occurring and that the component's intended function is maintained during the period of extended operation. 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 manages the aging effects of loss of material due to corrosion or reduction of heat transfer due to fouling. Components within the scope of the program include piping, piping components, and piping elements; heat exchanger tubes; reactor coolant pump elements; and any other plant components subject to aging management review that are exposed to an environment of lubricating oil (including non-water-based hydraulic oils).

2. ***Preventive Actions:*** The Lubricating Oil Analysis program maintains oil system contaminants (primarily water and particulates) within acceptable limits.

3. ***Parameters Monitored/Inspected:*** This program performs a check for water and a particle count to detect evidence of contamination by moisture or excessive corrosion.

4. ***Detection of Aging Effects:*** Moisture or corrosion products increase the potential for, or may be indicative of, loss of material due to corrosion and reduction of heat transfer due to fouling. The program performs periodic sampling and testing of lubricating oil for moisture and corrosion particles in accordance with industry standards. The program recommends sampling and testing of the old oil following periodic oil changes or on a schedule consistent with equipment manufacturer's recommendations or industry standards (e.g., American Society for Testing of Materials [ASTM] D 6224-02). Plant-specific operating experience also may be used to augment manufacturer's recommendations or industry standards in determining the schedule for periodic sampling and testing when justified by prior sampling results.

   In certain cases, as identified by the AMR Items in this report, inspection of selected components is to be undertaken to verify the effectiveness of the program and to ensure that significant degradation is not occurring and that the component intended function is maintained during the period of extended operation.

5. ***Monitoring and Trending:*** Oil analysis results are reviewed to determine if alert levels or limits have been reached or exceeded. This review also checks for unusual trends.

6. ***Acceptance Criteria:*** Water and particle concentration should not exceed limits based on equipment manufacturer's recommendations or industry standards. Phase-separated water in any amount is not acceptable.

7. ***Corrective Actions:*** Pursuant to 10 CFR Part 50, Appendix B, specific corrective actions are implemented in accordance with the plant quality assurance (QA) program. For example, if a limit is reached or exceeded, actions to address the condition are taken. These may include increased monitoring, corrective maintenance, further laboratory analysis, and engineering evaluation of the system. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

8. ***Confirmation Process:*** Site 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process and administrative controls.

9. ***Administrative Controls:*** The administrative controls for this program provide for a formal review and approval of corrective actions. The administrative controls for this program are implemented through the site's QA program in accordance with the requirements of 10 CFR Part 50, Appendix B.

10. ***Operating Experience:*** The operating experience at some plants has identified (a) water in the lubricating oil and (b) particulate contamination. However, no instances of component failures attributed to lubricating oil contamination have been identified.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

ASTM D 6224-02, *Standard Practice for In-Service Monitoring of Lubricating Oil for Auxiliary Power Plant Equipment*, American Society of Testing Materials, West Conshohocken, PA, 2002.

XI.M40 MONITORING OF NEUTRON-ABSORBING MATERIALS OTHER THAN BORAFLEX

Program Description

A monitoring program is implemented to assure that degradation of the neutron-absorbing material used in spent fuel pools that could compromise the criticality analysis will be detected. The applicable aging management program (AMP) relies on periodic inspection, testing, monitoring, and analysis of the criticality design to assure that the required 5% sub-criticality margin is maintained during the period of license renewal.

Evaluation and Technical Basis

1. ***Scope of Program***: The AMP manages the effects of aging on neutron-absorbing components/materials used in spent fuel racks.

2. ***Preventive Actions***: This AMP is a condition monitoring program, and therefore, there are no preventative actions.

3. ***Parameters Monitored/Inspected***: For these materials, gamma irradiation and/or long-term exposure to the wet pool environment may cause loss of material and changes in dimension (such as gap formation, formation of blisters, pits and bulges) that could result in loss of neutron-absorbing capability of the material. The parameters monitored include the physical condition of the neutron-absorbing materials, such as in-situ gap formation, geometric changes in the material (formation of blisters, pits, and bulges) as observed from coupons or in situ, and decreased boron areal density, etc. The parameters monitored are directly related to determination of the loss of material or loss of neutron absorption capability of the material(s).

4. ***Detection of Aging Effects***: The loss of material and the degradation of the neutron-absorbing material capacity are determined through coupon and/or direct in-situ testing. Such testing should include periodic verification of boron loss through areal density measurement of coupons or through direct in-situ techniques, which may include measurement of boron areal density, geometric changes in the material (blistering, pitting, and bulging), and detection of gaps through blackness testing. The frequency of the inspection and testing depends on the condition of the neutron-absorbing material and is determined and justified with plant-specific operating experience by the licensee, not to exceed 10 years.

5. ***Monitoring and Trending***: The measurements from periodic inspections and analysis are compared to baseline information or prior measurements and analysis for trend analysis. The approach for relating the measurements to the performance of the spent fuel neutron absorber materials is specified by the applicant, considering differences in exposure conditions, vented/non-vented test samples, and spent fuel racks, etc.

6. ***Acceptance Criteria***: Although the goal is to ensure maintenance of the 5% sub-criticality margin for the spent fuel pool, the specific acceptance criteria for the measurements and analyses are specified by the applicant.

7. ***Corrective Actions***: Corrective actions are initiated if the results from measurements and analysis indicate that the 5% sub-criticality margin cannot be maintained because of

current or projected future degradation of the neutron-absorbing material. Corrective actions may consist of providing additional neutron-absorbing capacity with an alternate material, or applying other options, which are available to maintain the sub-criticality margin. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

8. ***Confirmation Process*:** Site quality assurance (QA) procedures, site 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B acceptable to address the confirmation process and administrative controls.

9. ***Administrative Controls*:** As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address administrative controls.

10. ***Operating Experience*:** Applicants for license renewal reference plant-specific operating experience and industry experience to provide reasonable assurance that the program is able to detect degradation of the neutron absorbing material in the applicant's spent fuel pool. Some of the industry operating experience that should be included is listed below:

    1. Loss of material from the neutron absorbing material has been seen at many plants, including loss of aluminum, which was detected by monitoring the aluminum concentration in the spent fuel pool. One instance of this was documented in the Vogtle LRA Water Chemistry Program B.3.28.

    2. Blistering has also been noted at many plants. Examples include blistering at Seabrook and Beaver Valley.

    3. The significant loss of neutron-absorbing capacity of the plate-type carborundum material has been reported at Palisades.

The applicant should describe how the monitoring program described above is capable of detecting the aforementioned degradation mechanisms.

References

Interim Staff Guidance LR-ISG-2009-01, *Aging Management of Spent Fuel Pool Neutron-Absorbing Materials Other Than Boraflex*, 2010.

Letter from Christopher J. Schwarz, Entergy Nuclear Operations, Inc., Palisades Nuclear Plant, to the U.S. Nuclear Regulatory Commission, *Commitments to Address Degraded Spent Fuel Pool Storage Rack Neutron Absorber*, August 27, 2008, (ADAMS Accession No. ML082410132).

Letter from Kevin L. Ostrowski, FirstEnergy Nuclear Operating Company, to the U.S. Nuclear Regulatory Commission, *Supplemental Information for the Review of the Beaver Valley Power Station, Units 1 and 2, License Renewal Application (TAC Nos. MD6593 and MD6594) and License Renewal Application Amendment No. 34*, January 19, 2009, (ADAMS Accession No. ML090220216).

Letter from Mark E. Warner, FPL Energy Seabrook Station, to the U.S. Nuclear Regulatory Commission, *Seabrook Station Boral Spent Fuel Pool Test Coupons Report Pursuant to 10 CFR Part 21.21*, October 6, 2003 (ADAMS Accession No. ML032880525).

License Renewal Application Vogtle Electric Generating Plant Units 1 and 2, Southern Nuclear Operating Company, Inc., June 30, 2007 (ADAMS Accession No. ML071840360).

NRC Information Notice 2009-26, *Degradation of Neutron-Absorbing Materials in the Spent Fuel Pool*, U.S. Nuclear Regulatory Commission, October 28, 2009.

XI.M41 BURIED AND UNDERGROUND PIPING AND TANKS

Program Description

This is a comprehensive program designed to manage the aging of the external surfaces of buried and underground piping and tanks and to augment other programs that manage the aging of internal surfaces of buried and underground piping and tanks. It addresses piping and tanks composed of any material, including metallic, polymeric, cementitious, and concrete materials. This program manages aging through preventive, mitigative, and inspection activities. It manages all applicable aging effects such as loss of material, cracking, and changes in material properties.

Depending on the material, preventive and mitigative techniques may include the material itself, external coatings for external corrosion control, the application of cathodic protection, and the quality of backfill utilized. Also, depending on the material, inspection activities may include electrochemical verification of the effectiveness of cathodic protection, non-destructive evaluation of pipe or tank wall thicknesses, hydrotesting of the pipe, and visual inspections of the pipe or tank from the exterior as permitted by opportunistic or directed excavations.

Management of aging of the internal surfaces of buried and underground piping and tanks is accomplished through the use of other aging management programs (e.g., Open Cycle Cooling Water System (AMP XI.M20), Closed Treated Water System (AMP XI.M21A), Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (AMP XI.M38), Fuel Oil Chemistry (AMP XI.M30), Fire Water System (AMP XI.M27), or Water Chemistry (AMP XI.M2)). However, in some cases, this external surface program may be used in conjunction with the internal surface aging management programs to manage the aging of the internal surfaces of buried and underground piping and tanks. This program does not address selective leaching. The Selective Leaching of Materials (AMP XI.M33) is applied in addition to this program for applicable materials and environments.

The terms “buried” and “underground” are fully defined in Chapter IX of the GALL Report. Briefly, buried piping and tanks are in direct contact with soil or concrete (e.g., a wall penetration). Underground piping and tanks are below grade but are contained within a tunnel or vault such that they are in contact with air and are located where access for inspection is restricted.

Evaluation and Technical Basis

1. ***Scope of Program:*** This program is used to manage the effects of aging for buried and underground piping and tanks constructed of any material including metallic, polymeric, cementitious, and concrete materials. The program addresses aging effects such as loss of material, cracking, and changes in material properties. Typical systems in which buried and underground piping and tanks may be found include service water piping and components, condensate storage transfer lines, fuel oil and lubricating oil lines, fire protection piping and piping components (fire hydrants), and storage tanks. Loss of material due to corrosion of piping system bolting within the scope of this program is managed using this program. Other aging effects associated with piping system bolting are managed through the use of the Bolting Integrity Program (AMP XI.M18).

2. ***Preventive Actions:*** Preventive actions utilized by this program vary with the material of the tank or pipe and the environment (air, soil, or concrete) to which it is exposed. These actions are outlined below:

   a. Preventive Actions – **Buried Piping and Tanks**

      i. Preventive actions for buried piping and tanks are conducted in accordance with Table 2a and its accompanying footnotes.

Table 2a. Preventive Actions for **Buried** Piping and Tanks

Material[1]	**Coating[2]**	**Cathodic Protection[4]**	**Backfill Quality**
Titanium			
Super Austenitic Stainless[8]			
Stainless Steel	X[3]		X[5, 7]
Steel	X	X	X[5]
Copper	X	X	X[5]
Aluminum	X	X	X[5]
Cementitious or Concrete	X[3]		X[5, 7]
Polymer			X[6]

1. Materials classifications are meant to be broadly interpreted (e.g., all alloys of titanium that are commonly used for buried piping are to be included in the titanium category). Material categories are generally aligned with P numbers as found in the ASME Code, Section IX. Steel is defined in Chapter IX of this report. Polymer includes polymeric materials as well as composite materials such as fiberglass.
2. When provided, coatings are in accordance with Table 1 of NACE SP0169-2007 or Section 3.4 of NACE RP0285-2002.
3. Coatings are provided based on environmental conditions (e.g., stainless steel in chloride containing environments). If coatings are not provided, a justification is provided in the LRA.
4. Cathodic protection is in accordance with NACE SP0169-2007 or NACE RP0285-2002. The system should be operated so that the cathodic protection criteria and other considerations described in the standards are met at every location in the system. The duration of deviations from these criteria should not exceed 90 days. The system monitoring interval discussed in section 10.3 of NACE SP0169-2007 may not be extended beyond one year. The equipment used to implement cathodic protection need not be qualified in accordance with 10 CFR 50 Appendix B.
5. Backfill is consistent with SP0169-2007 section 5.2.3. The staff considers backfill that is located within 6 inches of the pipe that meets ASTM D 448-08 size number 67 to meet the objectives of SP0169-2007. For materials other than aluminum, the staff also considers the use of controlled low strength materials (flowable backfill) to meet the objectives of SP0169-2007. Backfill quality may be demonstrated by plant records or by examining the backfill while conducting the inspections conducted in program element 4 of this AMP. Backfill not meeting this standard, in either the initial or subsequent inspections, is acceptable if the inspections conducted in program element 4 of this AMP do not reveal evidence of mechanical damage to pipe coatings due to the backfill.
6. Backfill is consistent with SP0169-2007 section 5.2.3. The staff considers backfill that is located within 6 inches of the pipe that meets ASTM D 448-08 size number 10 to meet the objectives of SP0169-2007. The staff also considers the use of controlled low strength materials (flowable backfill) to meet the objectives of SP0169-2007. Backfill quality may be demonstrated by plant records or by examining the backfill while conducting the inspections conducted in program element 4 of this AMP. Backfill not meeting this standard, in either the initial or subsequent inspections, is acceptable if the inspections conducted in program element 4 of this AMP do not reveal evidence of mechanical damage to pipe coatings due to the backfill.
7. Backfill limits apply only if piping is coated.
8. Super austenitic stainless steel (e.g., Al6XN or 254 SMO).

      ii. Fire mains are installed in accordance with National Fire Protection Association (NFPA) Standard 24. Preventive actions for fire mains beyond those in NFPA 24 need not be provided if the system undergoes either a periodic flow test in accordance with NFPA 25 or the activity of the jockey pump (or equivalent equipment or parameter) is monitored as described in program element 4 of this AMP.

iii. When referenced, NACE SP0169-2007 is to be used in its entirety excepting Section 3, Determination of Need for External Corrosion Control. Use of Section 3 of the standard constitutes an exception to this AMP. Exceptions to the AMP related to the need for external corrosion control should include an analysis of issues such as those described in National Cooperative Highway Research Program (NCHRP) Report 408, "Corrosion of Steel Piling in Non Marine Applications and American Association of State Highway and Transportation Officials (AASHTO) Standard R 27."

b. Preventive Actions – **Underground Piping and Tanks**

i. Preventive actions for underground piping and tanks are conducted in accordance with Table 2b and its accompanying footnotes.

Table 2b. Preventive Actions for **Underground** Piping and Tanks

Material[1]	**Coating Provided**[2]
Titanium	
Super Austenitic Stainless[3]	
Stainless Steel	
Steel	X
Copper	X
Aluminum	
Cementitious or Concrete	
Polymer	

1. Materials classifications are meant to be broadly interpreted (e.g., all alloys of titanium that are commonly used for buried piping are to be included in the titanium category). Material categories are generally aligned with P numbers as found in the ASME Code, Section IX. Steel is defined in chapter IX of this report. Polymer includes polymeric materials as well as composite materials such as fiberglass.
2. When provided, coatings are in accordance with Table 1 of NACE SP0169-2007 or Section 3.4 of NACE RP0285-2002. A broader range of coatings may be used if justification is provided in the LRA.
3. Super austenitic stainless steel (e.g., Al6XN or 254 SMO).

3. ***Parameters Monitored/Inspected:*** The aging effects addressed by this AMP are changes in material properties of polymeric materials, loss of material due to all forms of corrosion and, potentially, cracking due to stress corrosion cracking. Changes in material properties are monitored by manual examinations. Loss of material is monitored by visual appearance of the exterior of the piping or tank and wall thickness of the piping or tank. Wall thickness is determined by a non-destructive examination technique such as ultrasonic testing (UT). Two additional parameters, the pipe-to-soil potential and the cathodic protection current, are monitored for steel, copper, and aluminum piping and tanks in contact with soil to determine the effectiveness of cathodic protection systems and, thereby, the effectiveness of corrosion mitigation.

4. ***Detection of Aging Effects:*** Methods and frequencies used for the detection of aging effects vary with the material and environment of the buried and underground piping and tanks. These methods and frequencies are outlined below.

a. Opportunistic Inspections

i. All buried and underground piping and tanks, regardless of their material of construction, are inspected by visual means whenever they become accessible for any reason. The information in paragraph f of this program element is applied in the event deterioration of piping or tanks is observed.

b. Directed Inspections – **Buried Pipe**

i. Directed inspections for buried piping are conducted in accordance with Table 4a and its accompanying footnotes. Modifications to this table may be appropriate if exceptions to program Element 2, Preventive Actions, are taken or in response to plant specific operating experience.

ii. Unless otherwise indicated, directed inspections as indicated in Table 4a will be conducted during each 10-year period beginning 10 years prior to the entry into the period of extended operation.

iii. Inspection locations are selected based on risk (based on susceptibility to degradation and consequences of failure). Characteristics such as coating type, coating condition, cathodic protection efficacy, backfill characteristics, soil resistivity, pipe contents, and pipe function are considered. Piping systems that are backfilled using controlled low strength material generally experience lower corrosion rates and may be more difficult to excavate than piping systems backfilled using compacted aggregate fill. As a result, piping systems that are backfilled using compacted aggregate should generally be given a higher inspection priority than comparable systems that are completely backfilled using controlled low strength material. For many piping systems, External Corrosion Direct Assessment (ECDA) as described in NACE Standard Practice SP0502-2010 has been demonstrated to be an effective method for use in the identification of pipe locations that merit further inspection.

iv. Visual inspections are supplemented with surface and/or volumetric non-destructive testing (NDT) if significant indications are observed.

v. Opportunistic examinations of non leaking pipes may be credited toward these direct examinations if the location selection criteria in item iii, above, are met.

vi. At multi-unit sites, individual inspections of shared piping may be credited for only one unit.

vii. Visual inspections for polymeric materials are augmented with manual examinations to detect hardening, softening, or other changes in material properties.

viii. The use of guided wave ultrasonic or other advanced inspection techniques is encouraged for the purpose of determining those piping locations that should be inspected but may not be substituted for the inspections listed in the table.

ix. For the purpose of this program element, fire mains will be considered to be code class/safety-related piping and inspected as such unless they are subjected to either a flow test as described in section 7.3 of NFPA 25 at a frequency of at least one test in each 1-year period or the activity of the jockey pump (or equivalent equipment or parameter) is monitored on an interval not to exceed 1 month. At a minimum, a flow test is conducted by the end of the next refueling outage or as directed by current licensing basis, whichever is shorter, when unexplained changes in jockey pump activity (or equivalent equipment or parameter) are observed.

x. Inspection as indicated in either (A) or (B) below may be performed in lieu of the inspections contained in Table 4a for either code class/safety significant or hazmat piping or both:

A. At least 25% of the code class/safety-related or hazmat piping or both constructed from the material under consideration is hydrostatically tested in accordance with 49 CFR 195 subpart E on an interval not to exceed 5 years.

B. At least 25% of the code class/safety-related or hazmat piping or both constructed from the material under consideration is internally inspected by a method capable of precisely determining pipe wall thickness. The inspection method must be capable of detecting both general and pitting corrosion and must be qualified by the applicant and approved by the staff. As of the effective date of this document, guided wave ultrasonic examinations do not meet this paragraph. Internal inspections are to be conducted at an interval not to exceed 5 years. Consideration should be given to NACE SP0169-2007 sections 6.1.2 and 6.3.3.

Table 4a. Inspections of **Buried Pipe**

Material[1]	**Preventive Actions**[2]	**Inspections**[3]	
		Code Class Safety-related[4]	**Hazmat**[5]
Titanium			
Super Austenitic Stainless[7]			
Stainless Steel		1[6]	1[6]
HDPE[8]	A	1[6]	1[6]
	B	2	1%
Other Polymer[9]	A	1[6]	1[6]
	B	2	1%
Cementitious or Concrete		1[6]	1[6]
Steel	C	1[6]	1[6]
	D	1	2%
	E	4[10]	5%[10]
	F	8	10%
Copper	C	1[6]	1[6]
	D	1	1%
	E	1[10]	2%[10]
	F	2	5%
Aluminum	C	1[6]	1[6]
	D	1	2%
	E	1	5%

Table 4a. Inspections of **Buried Pipe**			
Material[1]	**Preventive Actions**[2]	**Inspections**[3]	
		Code Class Safety-related[4]	**Hazmat**[5]
	F	2	10%

1. Materials classifications are meant to be broadly interpreted (e.g., all alloys of titanium that are commonly used for buried piping are to be included in the titanium category). Material categories are generally aligned with P numbers as found in the ASME Code, Section IX. Steel is defined in chapter IX of this report. Polymer includes polymeric materials as well as composite materials such as fiberglass.
2. Preventive actions are categorized as follows:
   A. Backfill is in accordance with Table 2a of this AMP.
   B. Backfill is not in accordance with Table 2a of this AMP.
   C. External corrosion control is provided in accordance with NACE SP0169-2007. Each cathodic protection system (a) was installed at least 5 years prior to the period of extended operation and was operational for 90% of the time during that 5-year period or (b) was operational for 90% of the time since the last inspection conducted under this program.
   D. External corrosion control is provided in accordance with NACE SP0169-2007. Each cathodic protection system (a) was installed less than 5 years prior to the period of extended operation or was operational for less than 90% of the time during that 5-year period or (b) was operational for less than 90% of the time since the last inspection conducted under this program.
   E. Coatings and backfill are in accordance with Table 2a of this AMP, but cathodic protection is not provided or is not consistent with criteria C or D. This category is provided for use during the 10 years prior to the period of extended operation by applicants who are not able to install cathodic protection in accordance with program element 2 prior to entry into the period of extended operation. Following entry into the period of extended operation, consistency with program element 2 or an approved alternative is expected.
   F. Preventive actions provided do not meet criteria C, D, or E. This category is provided for use during the 10 years prior to the period of extended operation by applicants who are not able to install cathodic protection in accordance with program element 2 prior to entry into the period of extended operation. Following entry into the period of extended operation, consistency with program element 2 or an approved alternative is expected.
3. Inspections are listed as either a discrete number of visual examinations (excavations) or as a percentage of the linear length of piping under consideration. The following guidance related to the extent of inspections is provided:
   A. Each inspection will examine either the entire length of a run of pipe or a minimum of 10 feet.
   B. If the number of inspections times the minimum inspection length (10 feet) exceeds 10% of the length of the piping under consideration, only 10% need be inspected.
   C. If the total length of in-scope pipe constructed of a given material times the percentage to be inspected is less than 10 feet, either 10 feet or the total length of pipe present, whichever is less, will be inspected.
4. Code Class and safety-related pipe that also meets the definition of hazmat pipe will be inspected as hazmat pipe.
5. Hazmat pipe is pipe that, during normal operation, contains material that, if released, could be detrimental to the environment. This includes chemical substances such as diesel fuel and radioisotopes. To be considered hazmat, the concentration of radioisotopes within the pipe during normal operation must exceed established standards such as the EPA drinking water standard. In the absence of such standards, the concentration of the radioisotope must exceed the greater of background or reliable level of detection. For tritium, the EPA drinking water standard (20,000 pCi/L) is used. (This approach for defining hazmat is consistent with that used in classifying fluid services in ASME B31.3 appendix M.)
6. Only one inspection is conducted even if both Code Class/safety-related and hazmat pipe are present. No inspections are necessary if all the piping constructed from the material under consideration is fully backfilled using controlled low strength material.
7. Super austenitic stainless steel (e.g., Al6XN or 254 SMO).
8. High Density Polyethylene (HDPE) pipe includes only HDPE pipe approved for use by the NRC for buried applications.
9. Other polymer piping includes some HDPE pipe and all other polymeric materials including composite materials such as fiberglass.
10. Inspections may be reduced to one-half the level indicated in the table when performing the indicated inspections necessitates excavation of piping that has been fully backfilled using controlled low strength material. In conducting these inspections, the backfill may be excavated and the pipe examined, or the soil around the backfill may be excavated and the controlled low strength material backfill examined. The corrosion rate of piping that is fully encased within controlled low strength material backfill that shows no signs of degradation, particularly cracking, is expected to be minimal.

c. Directed Inspections – **Underground Pipe**

i. Directed inspections for underground piping are conducted in accordance with Table 4b and its accompanying footnotes.

ii. Unless otherwise indicated, directed inspections as indicated in Table 4b will be conducted during each 10-year period beginning 10 years prior to the entry into the period of extended operation.

iii. Inspection locations are selected based on risk (based on susceptibility to degradation and consequences of failure). Characteristics such as coating type, coating condition, exact external environment, pipe contents, pipe function, and flow characteristics within the pipe, are considered.

iv. Underground pipes are inspected visually to detect external corrosion and by a volumetric technique such as UT to detect internal corrosion.

v. Opportunistic examinations may be credited toward these direct examinations if the location selection criteria in item iii, above, are met.

vi. At multi-unit sites, individual inspections of shared piping may be credited for only one unit.

vii. When access permits, visual inspections for polymeric materials are augmented with manual examinations to detect hardening, softening, or other changes in material properties.

viii. The use of guided wave ultrasonic or other advanced inspection techniques is encouraged for the purpose of determining those piping locations that should be inspected but may not be substituted for the inspections listed in the table.

ix. For the purpose of this program element, fire mains will be considered to be code class/safety-related piping and inspected as such unless they are subjected to either a flow test as described in section 7.3 of NFPA 25 at an frequency of at least one test in each 1-year period or the activity of the jockey pump (or equivalent equipment or parameter) is monitored on an interval not to exceed 1 month. At a minimum, a flow test is conducted by the end of the next refueling outage or as directed by current licensing basis, whichever is shorter, when unexplained changes in jockey pump activity (or equivalent equipment or parameter) are observed.

Table 4b. Inspections of **Underground Pipe**		
Material[1]	**Inspections[2]**	
	Code Class Safety-related[3]	**Hazmat[4]**
Titanium		
Super Austenitic Stainless[6]		
Stainless Steel	1[5]	1[5]
HDPE[7]	1[5]	1[5]
Other Polymer[8]	1[5]	1[5]
Cementitious or Concrete	1[5]	1[5]
Steel	2	2%
Copper	1	1%
Aluminum	1	1%

1. Materials classifications are meant to be broadly interpreted (e.g., all alloys of titanium that are commonly used for buried piping are to be included in the titanium category). Material categories are generally aligned with P numbers as found in the ASME Code, Section IX. Steel is as defined in chapter IX of this report. Polymer includes polymeric materials as well as composite materials such as fiberglass.
2. Inspections are listed as either a discrete number of visual examinations or as a percentage of the linear length of piping under consideration. The following guidance related to the extent of inspections is provided:
   A. Each inspection will examine either the entire length of a run of pipe or a minimum of 10 feet.
   B. If the number of inspections times the minimum inspection length (10 feet) exceeds 10% of the length of the piping under consideration, only 10% need be inspected.
   C. If the total length of in scope pipe constructed of a given material times the percentage to be inspected is less than 10 feet, either 10 feet or the total length of pipe present, whichever is less, will be inspected.
3. Code Class and safety-related pipe that also meets the definition of hazmat pipe will be inspected as hazmat pipe.
4. Hazmat pipe is pipe that, during normal operation, contains material that, if released, could be detrimental to the environment. This includes chemical substances such as diesel fuel and radioisotopes. To be considered hazmat, concentration of radioisotope within the pipe during normal operation must exceed established standards such as the EPA drinking water standard. In the absence of such standards, the concentration of the radioisotope must exceed the greater of background or reliable level of detection. For tritium, the EPA drinking water standard (20,000 pCi/L) is used. (This approach for defining hazmat is consistent with that used in classifying fluid services in ASME B31.3 appendix M.)
5. Only one inspection is conducted even if both Code Class/safety-related and hazmat pipe are present.
6. Super austenitic stainless steel (e.g., Al6XN or 254 SMO).
7. HDPE pipe includes only HDPE pipe approved for use by the NRC for buried applications.
8. Other polymer piping includes some HDPE pipe and all other polymeric materials including composite materials such as fiberglass.

x. Inspection as indicated in (A), and (B) below may be performed in lieu of the inspections contained in Table 4a for either code class/safety significant or hazmat piping or both:

   A. At least 25% of the code class/safety-related or hazmat piping or both constructed from the material under consideration is hydrostatically tested in accordance with 49 CFR 195 subpart E on an interval not to exceed 5 years.

   B. At least 25% of the code class/safety-related or hazmat piping or both constructed from the material under consideration is internally inspected by a method capable of precisely determining pipe wall thickness. The inspection method must be capable of detecting both general and pitting corrosion and must be qualified by the applicant and approved by the staff. As of the effective date of this document, guided wave ultrasonic examinations do not meet this paragraph. Internal inspections are to be conducted at an interval not to exceed 5 years. Consideration should be given to SP0169-2007 sections 6.1.2 and 6.3.3.

d. Directed Inspections – **Buried Tanks**

i. Directed inspections for buried tanks are conducted in accordance with Table 4c and its accompanying footnotes. Modifications to this table may be appropriate if exceptions to program Element 2, preventive actions, are taken or in response to plant specific operating experience.

ii. Directed inspections as indicated in Table 4c will be conducted during each 10-year period beginning 10 years prior to the entry into the period of extended operation.

iii. Each buried tank is examined if it is Code Class/safety-related or contains hazardous materials as defined in footnote 5 to Table 4a and it is constructed from a material for which an examination is indicated in Table 4c.

iv. Examinations may be conducted from the external surface of the tank using visual techniques or from the internal surface of the tank using volumetric techniques. If the tank is inspected from the external surface, a minimum 25% coverage is required. This area must include at least some of both the top and bottom of the tank. If the tank is inspected internally by UT, at least one measurement is required per square foot of tank surface. UT measurements are distributed uniformly over the surface of the tank. If the tank is inspected internally by another volumetric technique, at least 90% of the surface of the tank must be inspected. Double wall tanks may be examined by monitoring the annular space for leakage.

v. Visual inspections for polymeric materials are augmented with manual examinations to detect hardening, softening, or other changes in material properties.

vi. Opportunistic examinations may be credited toward these direct examinations.

Table 4c. Inspections of **Buried Tanks**

Material[1]	**Preventive Actions**[2]	**Inspections**
Titanium		
Super Austenitic Stainless[3]		
Stainless Steel		
HDPE[4]	A	
	B	X
Other Polymer[5]	A	
	B	X
Cementitious or Concrete		X
Steel	C	
	D	
	E	X

Table 4c. Inspections of **Buried Tanks**		
Material[1]	**Preventive Actions[2]**	**Inspections**
Copper	C	
	D	
	E	X
Aluminum	C	
	D	
	E	X

1. Materials classifications are meant to be broadly interpreted (e.g., all alloys of titanium that are commonly used for buried piping are to be included in the titanium category). Material categories are generally aligned with P numbers as found in the ASME Code, Section IX. Steel is defined in chapter IX of this report. Polymer includes polymeric materials as well as composite materials such as fiberglass.
2. Preventive actions are categorized as follows:
   A. Backfill is in accordance with Table 2a of this AMP.
   B. Backfill is not in accordance with Table 2a of this AMP.
   C. External corrosion control is provided in accordance with NACE RP0285-2002. Each cathodic protection system (a) was installed at least 5 years prior to the period of extended operation and was operational for 90% of the time during that 5-year period or (b) was operational for 90% of the time since the last inspection conducted under this program.
   D. External corrosion control is provided in accordance with NACE RP0285-2002. Each cathodic protection system (a) was installed less than 5 years prior to the period of extended operation or was operational for less than 90% of the time during that 5-year period or (b) was operational for less than 90% of the time since the last inspection conducted under this program.
   E. Cathodic protection is not provided. This category is provided for use during the 10 years prior to the period of extended operation by applicants who are not able to install cathodic protection in accordance with program element 2 prior to entry into the period of extended operation. Following entry into the period of extended operation, consistency with program element 2 or an approved alternative is expected.
3. Super austenitic stainless steel (e.g., Al6XN or 254 SMO).
4. HDPE includes only HDPE material approved for use by the NRC for buried applications.
5. Other polymer includes some HDPE material and all other polymeric materials including composite materials such as fiberglass.

e. Directed Inspections – **Underground Tanks**

   i. Directed inspections for underground tanks are conducted in accordance with Table 4d and its accompanying footnotes.

   ii. Directed inspections as indicated in Table 4d will be conducted during each 10-year period beginning 10 years prior to the entry into the period of extended operation.

Table 4d. Inspections of **Underground Tanks**	
Material[1]	**Inspections**
Titanium	
Super Austenitic Stainless[2]	
Stainless Steel	
HDPE[3]	
Other Polymer[4]	
Cementitious or concrete	
Steel	X

Table 4d. Inspections of **Underground Tanks**	
Material[1]	**Inspections**
Copper	
Aluminum	

1. Materials classifications are meant to be broadly interpreted (e.g., all alloys of titanium that are commonly used for buried piping are to be included in the titanium category). Material categories are generally aligned with P numbers as found in the ASME Code, Section IX. Steel is as defined in chapter IX of this report. Polymer includes polymeric materials as well as composite materials such as fiberglass.
2. Super austenitic stainless steel (e.g., Al6XN or 254 SMO).
3. HDPE includes only HDPE material approved for use by the NRC for buried applications.
4. Other polymer includes some HDPE material and all other polymeric materials including composite materials such as fiberglass.

iii. Each underground tank that is Code Class/safety-related or contains hazardous materials as defined in footnote 5 to Table 4a and is constructed from a material for which an examination is indicated in Table 4d is examined.

iv. Examinations may be conducted from the external surface of the tank using visual techniques or from the internal surface of the tank using volumetric techniques. If the tank is inspected from the external surface, a minimum 25% coverage is required. This area must include at least some of both the top and bottom of the tank. If the tank is inspected internally by UT, at least one measurement is required per square foot of tank surface. If the tank is inspected internally by another volumetric technique, at least 90% of the surface of the tank must be inspected. Double wall tanks may be examined by monitoring the annular space for leakage.

v. Tanks that cannot be examined using volumetric examination techniques are examined visually from the outside.

vi. When access permits, visual inspections for polymeric materials are augmented with manual examinations to detect hardening, softening, or other changes in material properties.

vii. Opportunistic examinations may be credited toward these direct examinations.

f. Adverse indications

i. Adverse indications observed during monitoring of cathodic protection systems or during inspections are entered into the plant corrective action program. Adverse indications that are the result of inspections will result in an expansion of sample size as described in item iv, below. Adverse indications that are the result of monitoring of a cathodic protection system may warrant increased monitoring of the cathodic protection system and/or additional inspections. Examples of adverse indications resulting from inspections include leaks, material thickness less than minimum, the presence of coarse backfill with accompanying coating degradation within 6 inches of a coated pipe or tank (see Table 2a Footnotes 5 and 6), and general or local degradation of coatings so as to expose the base material.

ii Adverse indications that fail to meet the acceptance criteria described in program element 6 of this AMP will result in the repair or replacement of the affected component.

iii. An analysis may be conducted to determine the potential extent of the degradation observed. Expansion of sample size may be limited by the extent of piping or tanks subject to the observed degradation mechanism.

iv. If adverse indications are detected, inspection sample sizes within the affected piping categories are doubled. If adverse indications are found in the expanded sample, the inspection sample size is again doubled. This doubling of the inspection sample size continues as necessary.

5. ***Monitoring and Trending:*** For piping and tanks protected by cathodic protection systems, potential difference and current measurements are trended to identify changes in the effectiveness of the systems and/or coatings. If aging of fire mains is managed through monitoring jockey pump activity (or similar parameter), jockey pump activity (or similar parameter) is trended to identify changes in pump activity that may be the result of increased leakage from buried fire main piping.

6. ***Acceptance Criteria:*** The principal acceptance criteria associated with the inspections contained with this AMP follow:

   a. Criteria for soil-to-pipe potential are listed in NACE RP0285-2002 and SP0169-2007.

   b. For coated piping or tanks, there should be either no evidence of coating degradation or the type and extent of coating degradation should be insignificant as evaluated by an individual possessing a NACE operator qualification or otherwise meeting the qualifications to evaluate coatings as contained in 49 CFR 192 and 195.

   c. If coated or uncoated metallic piping or tanks show evidence of corrosion, the remaining wall thickness in the affected area is determined to ensure that the minimum wall thickness is maintained. This may include different values for large area minimum wall thickness, and local area wall thickness.

   d. Cracking or blistering of nonmetallic piping is evaluated.

   e. Cementitious or concrete piping may exhibit minor cracking and spalling provided there is no evidence of leakage or exposed rebar or reinforcing "hoop" bands.

   f. Backfill is in accordance with specifications described in program element 2 of this AMP.

   g. Flow test results for fire mains are in accordance with NFPA 25 section 7.3.

   h. For hydrostatic tests, the condition "without leakage" as required by 49 CFR 195.302 may be met by demonstrating that the test pressure, as adjusted for temperature, does not vary during the test.

   i. Changes in jockey pump activity (or similar parameter) that cannot be attributed to causes other than leakage from buried piping are not occurring.

7. ***Corrective Actions:*** The 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. The staff finds the

requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions, confirmation process, and administrative controls.

8. ***Confirmation Process:*** The confirmation process ensures that preventive actions are adequate to manage the aging effects and that appropriate corrective actions have been completed and are effective. The confirmation process for this program is implemented through the site's QA program in accordance with the requirements of 10 CFR Part 50, Appendix B.

9. ***Administrative Controls:*** The administrative controls for this program provide for a formal review and approval of corrective actions. The administrative controls for this program are implemented through the site's QA program in accordance with the requirements of 10 CFR Part 50, Appendix B.

10. ***Operating Experience:*** Operating experience shows that buried and underground piping and tanks are subject to corrosion. Corrosion of buried oil, gas, and hazardous materials pipelines have been adequately managed through a combination of inspections and mitigative techniques, such as those prescribed in NACE SP0169-2007 and NACE RP0285-2002. Given the differences in piping and tank configurations between transmission pipelines and those in nuclear facilities, it is necessary for applicants to evaluate both plant-specific and nuclear industry operating experience and to modify its aging management program accordingly. The following industry experience may be of significance to an applicant’s program:

    a. In February 2005, a leak was detected in a 4-inch condensate storage supply line. The cause of the leak was microbiologically influenced corrosion or under deposit corrosion. The leak was repaired in accordance with the American Society of Mechanical Engineers (ASME) Section XI, “Repair/Replacement Plan.”

    b. In September 2005, a service water leak was discovered in a buried service water header. The header had been in service for 38 years. The cause of the leak was either failure of the external coating or damage caused by improper backfill. The service water header was relocated above ground.

    c. In October 2007, degradation of essential service water piping was reported. The riser pipe leak was caused by a loss of pipe wall thickness due to external corrosion induced by the wet environment surrounding the unprotected carbon steel pipe. The corrosion processes that caused this leak affected all eight similar locations on the essential service water riser pipes within vault enclosures and had occurred over many years.

    d. In February 2009, a leak was discovered on the return line to the condensate storage tank. The cause of the leak was coating degradation probably due to the installation specification not containing restrictions on the type of backfill allowing rocks in the backfill. The leaking piping was also located close to water table.

    e. In April 2009, a leak was discovered in an aluminum pipe where it went through a concrete wall. The piping was for the condensate transfer system. The failure was caused by vibration of the pipe within its steel support system. This vibration led to coating failure and eventual galvanic corrosion between the aluminum pipe and the steel supports.

f. In June 2009, an active leak was discovered in buried piping associated with the condensate storage tank. The leak was discovered because elevated levels of tritium were detected. The cause of the through-wall leaks was determined to be the degradation of the protective moisture barrier wrap that allowed moisture to come in contact with the piping resulting in external corrosion.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

49 CFR 195 subpart E, *Transportation of Hazardous Liquids by Pipeline, Pressure Testing*. Office of the Federal Register, National Archives and Records Administration, 2009.

AASHTO R 27, *Standard Practice for Assessment of Corrosion of Steel Piling for Non Marine Applications*, American Association of State Highway and Transportation Officials, Washington DC, 2006.

ASME Boiler and Pressure Vessel Code, Section IX, *Welding and Brazing*, American Society of Mechanical Engineers, 2004.

ASME Standard B31.3, *Process Piping*, Appendix M, American Society of Mechanical Engineers, 2002.

ASTM Standard D 448-08, Standard *Classification for Sizes of Aggregate for Road and Bridge Construction*, 2008.

J. A. Beavers and C. L. Durr, *Corrosion of Steel Piping in Non Marine Applications*, NCHRP Report 408, Transportation Research Board, National Research Council, Washington DC, 1998.

NACE Recommended Practice RP0285-2002, *Standard Recommended Practice Corrosion Control of Underground Storage Tank Systems by Cathodic Protection,* revised April 2002.

NACE Recommended Practice RP0502-2010, *Pipeline External Corrosion Direct Assessment Methodology*, 2010.

NACE Standard Practice SP0169-2007, *Control of External Corrosion on Underground or Submerged Metallic Piping Systems*, 2007.

NFPA Standard 24, *Standard for the Installation of Private Fire Service Mains and Their Appurtenances*, 2010 edition.

NFPA Standard 25, *Standard for the Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems*, 2008 edition.

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 and moisture barriers, and pressure-retaining bolting. This evaluation covers the 2004 edition,[20] 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 to detect aging effects. IWE specifies acceptance criteria, corrective actions, and expansion of the inspection scope when degradation exceeding the acceptance criteria is found.

Subsection IWE requires examination of coatings that are intended to prevent corrosion. AMP XI.S8 is a protective coating monitoring and maintenance program that is recommended to ensure Emergency Core Cooling System (ECCS) operability, whether or not the AMP XI.S8 is credited in AMP XI.S1.

The program attributes are augmented to incorporate aging management activities, recommended in the Final Interim Staff Guidance LR-ISG-2006-01, needed to address the potential loss of material due to corrosion in the inaccessible areas of the boiling water reactor (BWR) Mark I steel containment.

The attributes also are augmented to require surface examination for detection of cracking described in NRC Information Notice (IN) 92-20 and to address recommendations delineated in NUREG-1339 and industry recommendations delineated in the Electric Power Research Institute (EPRI) NP-5769, NP-5067, and TR-104213 for structural bolting. The program is also augmented to require surface examination of dissimilar metal welds of vent line bellows in accordance with examination Category E-F, as specified in the 1992 Edition of the ASME Code, Section XI, Subsection IWE. If surface examination is not possible, appropriate 10 CFR Part 50 Appendix J test may be conducted for pressure boundary components.

Evaluation and Technical Basis

1. ***Scope of Program:*** The scope of this program addresses the components of steel containments and steel liners of concrete containments specified in Subsection IWE-1000 as augmented by LR-ISG-2006-01. 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 moisture barriers, containment pressure-retaining bolting, and metal containment surface areas, including welds and base metal. The concrete

[20] Refer to the GALL Report, Chapter I, for applicability of other editions of the ASME Code, Section XI.

portions of containments are inspected in accordance with Subsection IWL. Subsection IWE requires examination of coatings that are intended to prevent corrosion. XI.S8 is a protective coating monitoring and maintenance program that is recommended to ensure ECCS operability, whether or not the AMP XI.S8 is credited in AMP XI.S1.

Subsection IWE exempts the following from examination:

(a) Components that are outside the boundaries of the containment, as defined in the plant-specific design specification;

(b) Embedded or inaccessible portions of containment components that met the requirements of the original construction code of record;

(c) Components that become embedded or inaccessible as a result of containment structure (i.e., steel containments [Class MC] and steel liners of concrete containments [Class CC]) repair or replacement, provided the requirements of IWE-1232 and IWE-5220 are met; and

(d) 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:*** The ASME Code Section XI, Subsection IWE, is a condition monitoring program. The program is augmented to include preventive actions that ensure that moisture levels associated with an accelerated corrosion rate do not exist in the exterior portion of the BWR Mark I steel containment drywell shell. The actions consist of ensuring that the sand pocket area drains and/or the refueling seal drains are clear. The program is also augmented to require that the selection of bolting material installation torque or tension and the use of lubricants and sealants are in accordance with the guidelines of EPRI NP-5769, EPRI TR-104213, and the additional recommendations of NUREG-1339 to prevent or mitigate degradation and failure of structural bolting. If the structural bolting consists of ASTM A325, ASTM F1852, and/or ASTM A490 bolts, the preventive actions for storage, lubricants, and stress corrosion cracking potential discussed in Section 2 of RCSC (Research Council for Structural Connections) publication “Specification for Structural Joints Using ASTM A325 or A490 Bolts,” need to be considered.

3. ***Parameters Monitored or Inspected:*** Table IWE-2500-1 references the applicable sections in IWE-2300 and IWE-3500 that identify the parameters examined or monitored. Non-coated surfaces are examined for evidence of cracking, discoloration, wear, pitting, excessive corrosion, arc strikes, gouges, surface discontinuities, dents, and other signs of surface irregularities. Painted or coated surfaces are examined for evidence of flaking, blistering, peeling, discoloration, and other signs of distress. Stainless steel penetration sleeves, dissimilar metal welds, bellows, and steel components that are subject to cyclic loading but have no current licensing basis fatigue analysis are monitored for cracking. The moisture barriers are examined for wear, damage, erosion, tear, surface cracks, or other defects that permit intrusion of moisture in the inaccessible areas of the pressure retaining surfaces of

the metal containment shell or liner. Pressure-retaining bolting is examined for loosening and material conditions that cause the bolted connection to affect either containment leak-tightness or structural integrity.

As recommended in LR-ISG-2006-01, license renewal applicants with BWR Mark I steel containments should monitor the sand pocket area drains and/or the refueling seal drains for water leakage. The licensees should ensure the drains are clear to prevent moisture levels associated with accelerated corrosion rates in the exterior portion of the drywell shell.

4. ***Detection of Aging Effects:*** The examination methods, frequency, and scope of examination specified in 10 CFR 50.55a and Subsection IWE ensure that aging effects are detected before they compromise the design-basis requirements. IWE-2500-1 and the requirements of 10 CFR 50.55a provide information regarding the examination categories, parts examined, and examination methods to be used to detect aging.

   As indicated in IWE-2400, inservice examinations 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.

   After 40 years of operation, any future examinations are 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 was 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 as specified in Table IWE-2500-1 and the requirements of 10 CFR 50.55a. The acceptability of inaccessible areas of the BWR Mark I steel containment drywell is evaluated when conditions exist in the adjacent accessible areas that could indicate the presence of moisture or could result in degradation to such inaccessible areas. 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.

   The requirements of ASME Section XI, Subsection IWE and 10 CFR 50.55a are augmented to require surface examination, in addition to visual examination, to detect cracking in stainless steel penetration sleeves, dissimilar metal welds, bellows, and steel components that are subject to cyclic loading but have no current licensing basis fatigue analysis. Where feasible, Appendix J tests (AMP XI.S4) may be performed in lieu of the surface examination.

5. ***Monitoring and Trending:*** With the exception of inaccessible areas, all surfaces are monitored by virtue of the examination requirements on a scheduled basis.

   IWE-2420 specifies that:

(a) The sequence of component examinations established during the first inspection interval shall be repeated.

(b) When examination results require evaluation of flaws or areas of degradation in accordance with IWE-3000, and the component is acceptable for continued service, the areas containing such flaws or areas of degradation shall be reexamined during the next inspection period listed in the schedule of the inspection program of IWE-2411 or IWE-2412, in accordance with Table IWE-2500-1, Examination Category E-C.

(c) When the reexaminations required by IWE-2420(b) reveal that the flaws or areas of degradation remain essentially unchanged for the next inspection period, these areas no longer require augmented examination in accordance with Table IWE-2500-1 and the regular inspection schedule is continued.

Applicants for license renewal for plants with BWR Mark I containment should augment IWE monitoring and trending requirements to address inaccessible areas of the drywell. The applicant should consider the following recommended actions based on plant-specific operating experience.

(a) Develop a corrosion rate that can be inferred from past ultrasonic testing (UT) examinations or establish a corrosion rate using representative samples in similar operating conditions, materials, and environments. If degradation has occurred, provide a technical basis using the developed or established corrosion rate to demonstrate that the drywell shell will have sufficient wall thickness to perform its intended function through the period of extended operation.

(b) Demonstrate that UT measurements performed in response to U.S. Nuclear Regulatory Commission (NRC) Generic Letter (GL) 87-05 did not show degradation inconsistent with the developed or established corrosion rate.

6. ***Acceptance Criteria:*** IWE-3000 provides acceptance standards for components of steel containments and liners of concrete containments. IWE-3410 refers to criteria to evaluate the acceptability of the containment components for service following the preservice examination and each inservice examination. 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 locally exceeding 10% of the nominal containment wall thickness or material loss that is projected to locally exceed 10% of the nominal containment wall thickness before the next examination are documented. Such areas are corrected by repair or replacement in accordance with IWE-3122 or accepted by engineering evaluation. Cracking of stainless steel penetration sleeves, dissimilar metal welds, bellows, and steel components that are subject to cyclic loading but have no current licensing basis fatigue analysis is corrected by repair or replacement or accepted by engineering evaluation.

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 IWE-3500 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. 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

If moisture has been detected or suspected in the inaccessible area on the exterior of the Mark I containment drywell shell or the source of moisture cannot be determined subsequent to root cause analysis, then:

(a) Include in the scope of license renewal any components that are identified as a source of moisture, if applicable, such as the refueling seal or cracks in the stainless liners of the refueling cavity pools walls, and perform aging management review.

(b) Identify surfaces requiring examination by implementing augmented inspections for the period of extended operation in accordance with Subsection IWE-1240, as identified in Table IWE-2500-1, Examination Category E-C.

(c) Use examination methods that are in accordance with Subsection IWE-2500.

(d) Demonstrate, through use of augmented inspections performed in accordance with Subsection IWE, that corrosion is not occurring or that corrosion is progressing so slowly that the age-related degradation will not jeopardize the intended function of the drywell shell through the period of extended operation.

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 IWE-3500. As discussed in the Appendix for GALL, 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 for GALL, 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 NRC. NRC Information Notice (IN) 86-99, IN 88-82, IN 89-79, IN 2004-09, and NUREG-1522 described occurrences of corrosion in steel containment shells. NRC GL 87-05 addressed the potential for corrosion of BWR Mark I steel drywells in the “sand pocket region.”

NRC IN 97-10 identified specific locations where concrete containments are susceptible to liner plate corrosion; IN 92-20 described an instance of containment bellows cracking, resulting in loss of leak tightness. More recently, IN 2006-01 described a through-wall cracking and its probable cause in the torus of a BWR Mark I containment. The cracking was identified by the licensee in the heat-affected zone at the high pressure cooling injection (HPCI) turbine exhaust pipe torus penetration.

The licensee concluded that the cracking was most likely initiated by cyclic loading due to condensation oscillation during HPCI operation. These condensation oscillations induced on the torus shell may have been excessive due to a lack of an HPCI turbine exhaust pipe sparger that many licensees have installed. Other operating experience indicates that foreign objects embedded in concrete have caused through-wall corrosion of the liner plate at a few plants with reinforced concrete containments.

The program is to consider the liner plate and containment shell corrosion and cracking concerns described in these generic communications. Implementation of the ISI requirements of Subsection IWE, in accordance with 10 CFR 50.55a, augmented to consider operating experience, and as recommended in LR-ISG-2006-01, is a necessary element of aging management for steel components of steel and concrete containments through the period of extended operation.

Degradation of threaded bolting and fasteners in closures for the reactor coolant pressure boundary has occurred from boric acid corrosion, stress corrosion cracking (SCC), and fatigue loading (NRC IE Bulletin 82-02, NRC GL 91-17). SCC has occurred in high strength bolts used for nuclear steam supply system component supports (EPRI NP-5769). The augmented ASME Section XI, Subsection IWE, incorporating recommendations documented in EPRI NP-5769 and TR-104213, is necessary to ensure containment bolting integrity.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

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, 2009.

10 CFR 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2009.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components,* Subsection IWA, *General Requirements,* The ASME Boiler and Pressure Vessel Code, 2004 edition as incorporated by reference in 10 CFR 50.55a, 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*, The ASME Boiler and Pressure Vessel Code, 2004 edition as incorporated by reference in 10 CFR 50.55a, 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,* The ASME Boiler and Pressure Vessel Code, 2004 edition as approved in 10 CFR 50.55a, 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,* The ASME Boiler and Pressure Vessel Code, 2004 edition as incorporated by reference in 10 CFR 50.55a, 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,* The ASME Boiler and Pressure Vessel Code, 2004 edition as incorporated by reference in 10 CFR 50.55a, The 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, April 1988.

EPRI NP-5067, *Good Bolting Practices, A Reference Manual for Nuclear Power Plant Maintenance Personnel*, Volume 1: Large Bolt Manual, 1987; Volume 2: Small Bolts and Threaded Fasteners, Electric Power Research Institute, 1990.

EPRI TR-104213, *Bolted Joint Maintenance & Application Guide,* Electric Power Research Institute, December 1995.

RCSC (Research Council on Structural Connections): *Specification for Structural Joints Using ASTM A325 or A490 Bolts,* 2004.

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.

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 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 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 92-20, *Inadequate Local Leak Rate Testing,* U.S. Nuclear Regulatory Commission, March 3, 1992.

NRC Information Notice 97-10, *Liner Plate Corrosion in Concrete Containment,* U.S. Nuclear Regulatory Commission, March 13, 1997.

NRC Information Notice 2004-09, *Corrosion of Steel Containment and Containment Liner*, U.S. Nuclear Regulatory Commission, April 27, 2004.

NRC Information Notice 2006-01, *Torus Cracking in a BWR Mark I Containment*, U.S. Nuclear Regulatory Commission, January 12, 2006.

NRC Morning Report, *Failure of Safety/Relief Valve Tee-Quencher Support Bolts*, March 14, 2005. (ADAMS Accession Number ML050730347)

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

NUREG-1522, *Assessment of Inservice Conditions of Safety-Related Nuclear Plant Structures*, June 1995.

Staff Position and Rationale for the Final License Renewal Interim Staff Guidance LR-ISG-2006-01, *Plant-Specific Aging Management Program for Inaccessible Areas of Boiling Water Reactor (BWR) Mark I Steel Containments Drywell Shell,* Nuclear Regulatory Commission, November 16, 2006.

XI.S2 ASME SECTION XI, SUBSECTION IWL

Program Description

10 CFR 50.55a imposes the examination requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, Subsection IWL, for reinforced and prestressed concrete containments (Class CC). The scope of IWL includes reinforced concrete and unbonded post-tensioning systems. This evaluation covers the 2004[21] edition of the ASME Code, Section XI, as approved in 10 CFR 50.55a. ASME Code, Section XI, Subsection IWL and the additional requirements specified in 10 CFR 50.55a(b)(2) constitute an existing mandated program applicable to managing aging of containment reinforced concrete and unbonded post-tensioning systems for license renewal.

The primary inspection method specified in IWL-2500 is visual examination, supplemented by testing. For prestressed containments, tendon wires are tested for yield strength, ultimate tensile strength, and elongation. Tendon corrosion protection medium is analyzed for alkalinity, water content, and soluble ion concentrations. The quantity of free water contained in the anchorage end cap and any free water that drains from tendons during the examination is documented. Samples of free water are analyzed for pH. Prestressing forces are measured in selected sample tendons. IWL specifies acceptance criteria, corrective actions, and expansion of the inspection scope when degradation exceeding the acceptance criteria is found.

The 2004 edition of the Code specifies augmented examination requirements following post-tensioning system repair/replacement activities. The post-tensioning system repair/replacement activities are to be in accordance with the requirements of the 2004 edition of the Code.

Evaluation and Technical Basis

1. ***Scope of Program:*** Subsection IWL-1000 specifies the components of concrete containments within its scope. The components within the scope of Subsection IWL are reinforced concrete and unbonded post-tensioning systems of Class CC containments, as defined by CC-1000. The program also includes testing of the tendon corrosion protection medium and the pH of free water. Subsection IWL exempts from examination portions of the concrete containment that are inaccessible (e.g., concrete covered by liner, foundation material, or backfill or obstructed by adjacent structures or other components).

   10 CFR 50.55a(b)(2)(viii) specifies additional requirements for inaccessible areas. It states that the licensee is to evaluate the acceptability of concrete in inaccessible areas when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas. Steel liners for concrete containments and their integral attachments are not within the scope of Subsection IWL but are included within the scope of Subsection IWE. Subsection IWE is evaluated in AMP XI.S1.

2. ***Preventive Action:*** ASME Code Section XI, Subsection IWL is a condition monitoring program. However, the program includes actions to prevent or minimize corrosion of the prestressing tendons by maintaining corrosion protection medium chemistry within acceptable limits specified in IWL.

[21] Refer to the GALL Report, Chapter I, for applicability of other editions of the ASME Code, Section XI.

3. ***Parameters Monitored or Inspected:*** Table IWL-2500-1 specifies two categories for examination of concrete surfaces: Category L-A for all accessible concrete surfaces and Category L-B for concrete surfaces surrounding anchorages of tendons selected for testing in accordance with IWL-2521. Both of these categories rely on visual examination methods. Concrete surfaces are examined for evidence of damage or degradation, such as concrete cracks. IWL-2510 specifies that concrete surfaces are examined for conditions indicative of degradation, such as those defined in ACI 201.1R and ACI 349.3R. Table IWL-2500-1 also specifies Category L-B for test and examination requirements for unbonded post tensioning systems. The number of tendons selected for examination is in accordance with Table IWL-2521-1. Additional augmented examination requirements for post-tensioning system repair/replacement activities are to be in accordance with Table IWL-2521-2. Tendon anchorage and wires or strands are visually examined for cracks, corrosion, and mechanical damage. Tendon wires or strands are also tested for yield strength, ultimate tensile strength, and elongation. The tendon corrosion protection medium is tested by analysis for alkalinity, water content, and soluble ion concentrations. The pH of free water samples is analyzed.

4. ***Detection of Aging Effects***: The frequency and scope of examinations specified in 10 CFR 50.55a and Subsection IWL ensure that aging effects would be detected before they would compromise the design-basis requirements. The frequency of inspection is specified in IWL-2400. Concrete inspections are performed in accordance with Examination Category L-A. Under Subsection IWL, inservice inspections of concrete and unbonded post-tensioning systems are required at 1, 3, and 5 years following the initial structural integrity test. Thereafter, inspections are performed at 5-year intervals. For sites with multiple plants, the schedule for inservice inspection is provided in IWL-2421. In the case of tendons, only a sample of the tendons of each tendon type requires examination during each inspection.

   The tendons to be examined during an inspection are selected on a random basis. Regarding detection methods for aging effects, all accessible concrete surfaces receive General Visual examination (as defined by the ASME Code). Selected areas, such as those that indicate suspect conditions and concrete surface areas surrounding tendon anchorages (Category L-B), receive a more rigorous Detailed Visual examination (as defined by the ASME Code). Prestressing forces in sample tendons are measured. In addition, one sample tendon of each type is detensioned. A single wire or strand is removed from each detensioned tendon for examination and testing. These visual examination methods and testing would identify the aging effects of accessible concrete components and prestressing systems in concrete containments. Examination of corrosion protection medium and free water are tested for each examined tendon as specified in Table IWL-2525-1.

5. ***Monitoring and Trending:*** Except in inaccessible areas, all concrete surfaces are monitored on a regular basis by virtue of the examination requirements. For prestressed containments, trending of prestressing forces in tendons is required in accordance with paragraph (b)(2)(viii) of 10 CFR 50.55a. In addition to the random sampling used for tendon examination, one tendon of each type is selected from the first-year inspection sample and designated as a common tendon. Each common tendon is then examined during each inspection. Corrosion protection medium chemistry and free water pH are monitored for each examined tendon. This procedure provides monitoring and trending information over the life of the plant. 10 CFR 50.55a and Subsection IWL also require that prestressing forces in all inspection sample tendons be measured by lift-off tests and compared with acceptance standards based on the predicted force for that type of tendon over its life.

6. ***Acceptance Criteria:*** IWL-3000 provides acceptance criteria for concrete containments. For concrete surfaces, the acceptance criteria rely on the determination of the "Responsible Engineer" (as defined by the ASME Code) regarding whether there is any evidence of damage or degradation sufficient to warrant further evaluation or repair. The acceptance criteria are qualitative; guidance is provided in IWL-2510, which references ACI 201.1R and ACI 349.3R for identification of concrete degradation. IWL-2320 requires that the Responsible Engineer be a registered professional engineer experienced in evaluating the inservice condition of structural concrete and knowledgeable of the design and construction codes and other criteria used in design and construction of concrete containments. Quantitative acceptance criteria based on the "Evaluation Criteria" provided in Chapter 5 of ACI 349.3R also may be used to augment the qualitative assessment of the Responsible Engineer.

   The acceptance standards for the unbonded post-tensioning system are quantitative in nature. For the post-tensioning system, quantitative acceptance criteria are given for tendon force and elongation, tendon wire or strand samples, and corrosion protection medium. Free water in the tendon anchorage areas is not acceptable, as specified in IWL-3221.3. If free water is found, the recommendations in Table IWL-2525-1 are followed. 10 CFR 50.55a and Subsection IWL do not define the method for calculating predicted tendon prestressing forces for comparison to the measured tendon lift-off forces. The predicted tendon forces are calculated in accordance with Regulatory Guide 1.35.1, which provides an acceptable methodology for use through the period of extended operation.

7. ***Corrective Actions:*** Subsection IWL specifies that items for which examination results do not meet the acceptance standards are to be evaluated in accordance with IWL-3300, "Evaluation," and described in an engineering evaluation report. The report is to include an evaluation of whether the concrete containment is acceptable without repair of the item and, if repair is required, the extent, method, and completion date of the repair or replacement. The report also identifies the cause of the condition and the extent, nature, and frequency of additional examinations. Subsection IWL also provides repair procedures to follow in IWL-4000. This includes requirements for the concrete repair, repair of reinforcing steel, and repair of the post-tensioning system. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

8. ***Confirmation Process:*** As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process.

9. ***Administrative Controls:*** IWA-1400 specifies the preparation of plans, schedules, and inservice inspection summary reports. In addition, written examination instructions and procedures, verification of qualification level of personnel who perform the examinations, and documentation of a quality assurance program are specified. IWA-6000 specifically covers the preparation, submittal, and retention of records and reports. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the administrative controls.

10. ***Operating Experience:*** ASME Section XI, Subsection IWL was incorporated into 10 CFR 50.55a in 1996. Prior to this time, the prestressing tendon inspections were performed in accordance with the guidance provided in Regulatory Guide 1.35. Operating experience pertaining to degradation of reinforced concrete in concrete containments 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). NUREG-1522 described instances of cracked, spalled, and degraded concrete for reinforced and prestressed concrete containments. The NUREG also described cracked anchor heads for the prestressing tendons at three prestressed concrete containments. NRC Information Notice 99-10 described occurrences of degradation in prestressing systems. The program is to consider the degradation concerns described in these generic communications. Implementation of Subsection IWL, in accordance with 10 CFR 50.55a, is a necessary element of aging management for concrete containments through the period of extended operation.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

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, 2009.

10 CFR 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2009.

ACI Standard 201.1R, *Guide for Making a Condition Survey of Concrete in Service,* American Concrete Institute.

ACI Standard 349.3R, *Evaluation of Existing Nuclear Safety-Related Concrete Structures,* American Concrete Institute, 2002.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components,* Subsection IWA, *General Requirements,* The ASME Boiler and Pressure Vessel Code, 2004 edition as approved in 10 CFR 50.55a, 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,* The ASME Boiler and Pressure Vessel Code, 2004 edition as approved in 10 CFR 50.55a, 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,* The ASME Boiler and Pressure Vessel Code, 2004 edition as approved in 10 CFR 50.55a, The American Society of Mechanical Engineers, New York, NY.

NRC Information Notice 99-10, Revision 1, *Degradation of Prestressing Tendon Systems in Prestressed Concrete Containment,* U.S. Nuclear Regulatory Commission, October 7, 1999.

NRC Regulatory Guide 1.35.1, *Determining Prestressing Forces for Inspection of Prestressed Concrete Containments*, U.S. Nuclear Regulatory Commission, July 1990.

NRC Regulatory Guide 1.35, *Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments,* U.S. Nuclear Regulatory Commission, July 1990

NUREG-1522, *Assessment of Inservice Condition of Safety-Related Nuclear Power Plant Structures,* June 1995.

XI.S3 ASME SECTION XI, SUBSECTION IWF

Program Description

10 CFR 50.55a imposes the inservice inspection requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, for Class 1, 2, 3, and metal containment (MC) piping and components and their associated supports. Inservice inspection of supports for ASME piping and components is addressed in Section XI, Subsection IWF. This evaluation covers the 2004 edition[22] of the ASME Code as approved in 10 CFR 50.55a. ASME Code, Section XI, Subsection IWF, constitutes an existing mandated program applicable to managing aging of ASME Class 1, 2, 3, and MC component supports for license renewal.

The IWF scope of inspection for supports is based on sampling of the total support population. The sample size varies depending on the ASME Class. The largest sample size is specified for the most critical supports (ASME Class 1). The sample size decreases for the less critical supports (ASME Class 2 and 3). Discovery of support deficiencies during regularly scheduled inspections triggers an increase of the inspection scope in order to ensure that the full extent of deficiencies is identified. The primary inspection method employed is visual examination. Degradation that potentially compromises support function or load capacity is identified for evaluation. IWF specifies acceptance criteria and corrective actions. Supports requiring corrective actions are re-examined during the next inspection period.

The requirements of subsection IWF are augmented to include monitoring of high-strength structural bolting (actual measured yield strength greater than or equal to 150 ksi or 1,034 MPa) for cracking. The program is augmented to incorporate recommendations delineated in NUREG-1339 and industry recommendations delineated in the Electric Power Research Institute (EPRI) NP-5769, NP-5067, and TR-104213 for high-strength structural bolting, if applicable. These recommendations emphasize proper selection of bolting material, lubricants, and installation torque or tension to prevent or minimize loss of bolting preload and cracking of high-strength bolting.

Evaluation and Technical Basis

1. ***Scope of Program:*** This program addresses supports for ASME Class 1, 2, and 3 piping and components supports that are not exempt from examination in accordance with IWF - 1230 and MC supports. The scope of the program includes support members, structural bolting, high strength structural bolting, support anchorage to the building structure, accessible sliding surfaces, constant and variable load spring hangers, guides, stops, and vibration isolation elements.

2. ***Preventive Action:*** Selection of bolting material and the use of lubricants and sealants is in accordance with the guidelines of EPRI NP-5769, EPRI TR-104213, and the additional recommendations of NUREG-1339 to prevent or mitigate degradation and failure of safety-related bolting. Operating experience and laboratory examinations show that the use of molybdenum disulfide (MoS_2) as a lubricant is a potential contributor to stress corrosion cracking (SCC), especially when applied to high strength bolting. Thus, molybdenum disulfide and other lubricants containing sulfur should not be used. Preventive measures also include using bolting material that has an actual measured yield strength less than

[22] Refer to the GALL Report, Chapter I, for applicability of other editions of the ASME Code, Section XI.

150 ksi or 1,034 MPa. Structural bolting replacement and maintenance activities include appropriate preload and proper tightening (torque or tension) as recommended in EPRI documents, American Society for Testing of Materials (ASTM) standards, American Institute of Steel Construction (AISC) Specifications, as applicable. If the structural bolting consists of ASTM A325, ASTM F1852, and/or ASTM A490 bolts, the preventive actions for storage, lubricants, and stress corrosion cracking potential discussed in Section 2 of RCSC (Research Council for Structural Connections) publication “Specification for Structural Joints Using ASTM A325 or A490 Bolts” need to be used.

3. ***Parameters Monitored or Inspected:*** The parameters monitored or inspected include corrosion; deformation; misalignment of supports; missing, detached, or loosened support items; improper clearances of guides and stops; and improper hot or cold settings of spring supports and constant load supports. Accessible areas of sliding surfaces are monitored for debris, dirt, or indications of excessive loss of material due to wear that could prevent or restrict sliding as intended in the design basis of the support. Elastomeric vibration isolation elements are monitored for cracking, loss of material, and hardening. Structural bolts are monitored for corrosion and loss of integrity of bolted connections due to self loosening and material conditions that can affect structural integrity. High-strength structural bolting (actual measured yield strength greater than or equal to 150 ksi or 1,034 MPa) susceptible to SCC should be monitored for SCC.

4. ***Detection of Aging Effects:*** The program requires that a sample of ASME Class 1, 2, and 3 component supports that are not exempt from examination and 100% of MC component supports be examined as specified in Table IWF-2500-1. The sample size examined for ASME Class 1, 2, and 3 component supports is as specified in Table IWF-2500-1. The extent, frequency, and examination methods are designed to detect, evaluate, or repair age-related degradation before there is a loss of component support intended function. The VT-3 examination method specified by the program can reveal loss of material due to corrosion and wear, verification of clearances, settings, physical displacements, loose or missing parts, debris or dirt in accessible areas of the sliding surfaces, or loss of integrity at bolted connections. The VT-3 examination can also detect loss of material and cracking of elastomeric vibration isolation elements. VT-3 examination of elastomeric vibration isolation elements should be supplemented by feel to detect hardening if the vibration isolation function is suspect. IWF-3200 specifies that visual examinations that detect surface flaws which exceed acceptance criteria may be supplemented by either surface or volumetric examinations to determine the character of the flaw.

   For high strength structural bolting (actual measured yield strength greater than or equal to 150 ksi or 1,034 MPa) in sizes greater than 1 inch nominal diameter, volumetric examination comparable to that of ASME Code Section XI, Table IWB-2500-1, Examination Category B-G-1 should be performed to detect cracking in addition to the VT-3 examination. This volumetric examination may be waived with adequate plant-specific justification. Other structural bolting (ASTM A-325, ASTM F1852, and ASTM A490 bolts) and anchor bolts are monitored for loss of material, loose or missing nuts, and cracking of concrete around the anchor bolts.

5. ***Monitoring and Trending:*** The ASME Class 1, 2, 3, and MC component supports are examined periodically, as specified in Table IWF-2500-1. As required by IWF-2420(a), the sequence of component support examinations established during the first inspection interval is repeated during each successive inspection interval, to the extent practical. Component supports whose examinations do not reveal unacceptable degradations are accepted for

continued service. Verified changes of conditions from prior examination are recorded in accordance with IWA-6230. Component supports whose examinations reveal unacceptable conditions and are accepted for continued service by corrective measures or repair/ replacement activity are reexamined during the next inspection period. When the reexamined component support no longer requires additional corrective measures during the next inspection period, the inspection schedule may revert to its regularly scheduled inspection. Examinations that reveal indications which exceed the acceptance standards and require corrective measures are extended to include additional examinations in accordance with IWF-2430.

6. ***Acceptance Criteria:*** The acceptance standards for visual examination are specified in IWF-3400. IWF-3410(a) identifies the following conditions as unacceptable:

   (a) Deformations or structural degradations of fasteners, springs, clamps, or other support items;

   (b) Missing, detached, or loosened support items, including bolts and nuts;

   (c) Arc strikes, weld spatter, paint, scoring, roughness, or general corrosion on close tolerance machined or sliding surfaces;

   (d) Improper hot or cold positions of spring supports and constant load supports;

   (e) Misalignment of supports; and

   (f) Improper clearances of guides and stops.

   Other unacceptable conditions include:

   (a) Loss of material due to corrosion or wear, which reduces the load bearing capacity of the component support;

   (b) Debris, dirt, or excessive wear that could prevent or restrict sliding of the sliding surfaces as intended in the design basis of the support;

   (c) Cracked or sheared bolts, including high strength bolts, and anchors; and

   (d) Loss of material, cracking, and hardening of elastomeric vibration isolation elements that could reduce the vibration isolation function.

   The above conditions may be accepted provided the technical basis for their acceptance is documented.

7. ***Corrective Actions:*** Identification of unacceptable conditions triggers an expansion of the inspection scope, in accordance with IWF-2430, and reexamination of the supports requiring corrective actions during the next inspection period, in accordance with IWF-2420(b). In accordance with IWF-3122, supports containing unacceptable conditions are evaluated or tested or corrected before returning to service. Corrective actions are delineated in IWF-3122.2. IWF-3122.3 provides an alternative for evaluation or testing to substantiate structural integrity and/or functionality. As discussed in the Appendix for GALL, the staff

finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

8. ***Confirmation Process:*** As discussed in the Appendix for GALL, 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the administrative controls.

10. ***Operating Experience:*** To date, IWF sampling inspections have been effective in managing aging effects for ASME Class 1, 2, 3, and MC supports. There is reasonable assurance that the Subsection IWF inspection program will be effective in managing the aging of the in-scope component supports through the period of extended operation.

    Degradation of threaded bolting and fasteners has occurred from boric acid corrosion, SCC, and fatigue loading (NRC IE Bulletin 82-02, NRC Generic Letter 91-17). SCC has occurred in high strength bolts used for NSSS component supports (EPRI NP-5769).

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

10 CFR 50.55a, *Codes and Standards,* Office of the Federal Register, National Archives and Records Administration, 2009.

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,* The ASME Boiler and Pressure Vessel Code, 2004 edition as approved in 10 CFR 50.55a, 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,* The ASME Boiler and Pressure Vessel Code, 2004 edition as approved in 10 CFR 50.55a, The American Society of Mechanical Engineers, New York, NY.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components,* Subsection IWD, *Requirements for Class 3 Components of Light-Water Cooled Power Plants,* The ASME Boiler and Pressure Vessel Code, 2004 edition as approved in 10 CFR 50.55a, 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,* The ASME Boiler and Pressure Vessel Code, 2004 edition as approved in 10 CFR 50.55a, The American Society of Mechanical Engineers, New York, NY.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components,* Subsection IWF, *Requirements for Class 1, 2, 3, and MC Component Supports of Light-*

Water Cooled Power Plants, The ASME Boiler and Pressure Vessel Code, 2004 edition as approved in 10 CFR 50.55a, The American Society of Mechanical Engineers, New York, NY.

EPRI NP-5067, *Good Bolting Practices, A Reference Manual for Nuclear Power Plant Maintenance Personnel*, Volume 1: Large Bolt Manual, 1987; Volume 2: Small Bolts and Threaded Fasteners, Electric Power Research Institute, 1990.

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

EPRI TR-104213, *Bolted Joint Maintenance & Application Guide*, Electric Power Research Institute, 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 Morning Report, *Failure of Safety/Relief Valve Tee-Quencher Support Bolts*, March 14, 2005. (ADAMS Accession Number ML050730347)

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

RCSC (Research Council on Structural Connections): *Specification for Structural Joints Using ASTM A325 or A490 Bolts,* Chicago, 2004.

XI.S4 10 CFR PART 50, APPENDIX J

Program Description

As described in 10 CFR Part 50, Appendix J, containment leak rate tests are required to "assure that (a) leakage through these containments or systems and components penetrating these containments does not exceed allowable leakage rates specified in the technical specifications and (b) integrity of the containment structure is maintained during its service life."

Appendix J provides two options, Option A and Option B, either of which can be chosen to meet the requirements of a containment leakage rate test (LRT) program. Option A is prescriptive with all testing performed on specified, uniform periodic intervals. Option B is a performance-based approach. Some of the differences between these options are discussed below. More detailed information for Option B is provided in the Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.163[23] and NEI 94-01 as approved by the NRC Final Safety Evaluation for the Nuclear Energy Institute (NEI) Topical Report (TR) 94-01, Revision 2.Three types of tests are performed under either Option A or Option B. Type A tests are performed to determine the overall primary containment integrated leakage rate at the loss of coolant accident peak containment pressure. Type B tests are intended to detect local leaks and to measure leakage across each pressure-containing or leakage-limiting boundary of containment penetrations. Type C tests are intended to detect local leaks and to measure leakage across containment isolation valves installed in containment penetrations or lines penetrating containment. If Type C tests are not performed under this program, they could be included under an ASME Code, Section XI, Inservice Test Program leakage testing for systems containing the isolation valves.

Appendix J requires a general inspection of the accessible interior and exterior surfaces of the containment structure and components be performed prior to any Type A test. General Visual examinations performed in accordance with the ASME Section XI, Subsection IWE (AMP XI.S1) or ASME Section XI, Subsection IWL (AMP XI.S2) program are an acceptable substitute. The purpose of the inspection is to uncover any evidence of structural deterioration that may affect the containment structural integrity or leak-tightness. If there is evidence of structural deterioration, the Type A test is not performed until corrective action is taken in accordance with the repair/replacement procedures.

Evaluation and Technical Basis

1. ***Scope of Program:*** The scope of the containment LRT program includes all containment boundary pressure-retaining components.

2. ***Preventive Action***: The containment LRT program is a performance monitoring program that includes no preventive actions.

3. ***Parameters Monitored or Inspected:*** The parameters to be monitored are leakage rates through containment shells, containment liners, and associated welds, penetrations, fittings, and other access openings.

4. ***Detection of Aging Effects:*** A containment LRT program is effective in detecting leakage rate of the containment pressure boundary components, including seals and gaskets. While the calculation of leakage rates and satisfactory performance of containment leakage rate

[23] RG 1.163 Rev. 0 or the latest Revision.

testing demonstrates the leak-tightness and structural integrity of the containment, it does not by itself provide information that would indicate that aging degradation has initiated or that the capacity of the containment may have been reduced for other types of loads, such as seismic loading. This would be achieved with the additional implementation of an acceptable containment inservice inspection program as described in ASME Section XI, Subsection IWE (AMP XI.S1) and ASME Section XI, Subsection IWL (AMP XI.S2).

5. ***Monitoring and Trending:*** Because the LRT program is repeated throughout the operating license period, the entire pressure boundary is monitored over time. The frequency of these tests depends on which option (A or B) is selected. With Option A, testing is performed on a regular fixed time interval as defined in 10 CFR Part 50, Appendix J. In the case of Option B, the interval for testing may be adjusted on the basis of acceptable performance in meeting leakage limits in prior tests. Additional details for implementing Option B are provided in NRC RG 1.163 and NEI 94-01.

6. ***Acceptance Criteria:*** Acceptance criteria for leakage rates are defined in plant technical specifications. These acceptance criteria meet the requirements in 10 CFR Part 50, Appendix J, and are part of each plant's current licensing basis.

7. ***Corrective Actions:*** Corrective actions are taken in accordance with 10 CFR Part 50, Appendix J, and NEI 94-01. When leakage rates do not meet the acceptance criteria, an evaluation is performed to identify the cause of the unacceptable performance and appropriate corrective actions are taken. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

8. ***Confirmation Process:*** As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process.

9. ***Administrative Controls:*** Results of the LRT program are documented as described in 10 CFR Part 50, Appendix J, to demonstrate that the acceptance criteria for leakage have been satisfied. The test results that exceed the performance criteria are assessed under 10 CFR 50.72 and 10 CFR 50.73. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the administrative controls.

10. ***Operating Experience:*** To date, the 10 CFR Part 50, Appendix J, LRT program, in conjunction with the containment inservice inspection program, has been effective in preventing unacceptable leakage through the containment pressure boundary. Implementation of Option B for testing frequency must be consistent with plant-specific operating experience.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

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, 2009.

10 CFR 50.72, *Immediate Notification Requirements for Operating Nuclear Power Reactors,* Office of the Federal Register, National Archives and Records Administration, 2009.

10 CFR 50.73, *Licensee Event Report System,* Office of the Federal Register, National Archives and Records Administration, 2009.

Final Safety Evaluation for 'Nuclear Energy Institute (NEI) Topical Report (TR) 94-01, Revision 2, *Industry Guideline for Implementing Performance-Based Option of 10 CFR, Part 50, Appendix J,'* and 'Electric Power Research Institute (EPRI) Report No. 1009325, Revision 2, *Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals*, August 2007,' June 25, 2008.

NEI 94-01, Rev. 2-A, *Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50 Appendix J,* Nuclear Energy Institute, August 2007.

NRC Regulatory Guide 1.163, Rev. 0, *Performance-Based Containment Leak-Test Program,* U.S. Nuclear Regulatory Commission, September 1995.

XI.S5 MASONRY WALLS

Program Description

Nuclear Regulatory Commission (NRC) IE Bulletin (IEB) 80-11, "Masonry Wall Design," and NRC Information Notice (IN) 87-67, "Lessons Learned from Regional Inspections of Licensee Actions in Response to IE Bulletin 80-11," constitute an acceptable basis for a masonry wall aging management program (AMP). IEB 80-11 required (a) the identification of masonry walls in close proximity to or having attachments from safety-related systems or components and (b) the evaluation of design adequacy and construction practice. NRC IN 87-67 recommended plant-specific condition monitoring of masonry walls and administrative controls to ensure that the evaluation basis developed in response to NRC IEB 80-11 is not invalidated by (a) deterioration of the masonry walls (e.g., new cracks not considered in the reevaluation), (b) physical plant changes such as installation of new safety-related systems or components in close proximity to masonry walls, or (c) reclassification of systems or components from non-safety-related to safety-related, provided appropriate evaluation is performed to account for such occurrences.

Important elements in the evaluation of many masonry walls during the NRC IEB 80-11 program included (a) installation of steel edge supports to provide a sound technical basis for boundary conditions used in seismic analysis and (b) installation of steel bracing to ensure stability or containment of unreinforced masonry walls during a seismic event. Consequently, in addition to the development of cracks in the masonry walls, loss of function of the structural steel supports and bracing would also invalidate the evaluation basis. The steel edge supports and steel bracings are considered component supports and aging effects are managed by the Structures Monitoring program (AMP XI.S6).

The program requires periodic visual inspection of masonry walls in the scope of license renewal to detect loss of material and cracking of masonry units and mortar. The aging effects that could impact masonry wall intended function or potentially invalidate its evaluation basis are entered in the corrective action process for further analysis, repair, or replacement.

Since the issuance of NRC IEB 80-11 and NRC IN 87-67, the NRC promulgated 10 CFR 50.65, the Maintenance Rule. For license renewal, masonry walls may be inspected as part of the “Structures Monitoring Program” (AMP XI.S6) conducted for the Maintenance Rule, provided the 10 attributes described below are incorporated in AMP XI.S6. The aging effects on masonry walls that are considered fire barriers also are managed by AMP XI.M26, Fire Protection.

Evaluation and Technical Basis

1. ***Scope of Program:*** The scope includes all masonry walls identified as performing intended functions in accordance with 10 CFR 54.4. The aging effects on masonry walls that are considered fire barriers also are managed by AMP XI.M26, Fire Protection, as well as being managed by this program.

2. ***Preventive Action:*** This is a condition monitoring program and no specific preventive actions are required.

3. ***Parameters Monitored or Inspected:*** The primary parameters monitored are potential shrinkage and/or separation and cracking of masonry walls and gaps between the supports and masonry walls that could impact the intended function or potentially invalidate its evaluation basis.

4. ***Detection of Aging Effects:*** Visual examination of the masonry walls by qualified inspection personnel is sufficient. In general, masonry walls should be inspected every 5 years, with provisions for more frequent inspections in areas where significant loss of material or cracking is observed to ensure there is no loss of intended function between inspections. However, masonry walls that are fire barriers are visually inspected in accordance with AMP XI.M26.

5. ***Monitoring and Trending:*** Trending is not required. Condition monitoring for evidence of shrinkage and/or separation and cracking is achieved by periodic examination. Degradation detected from monitoring is evaluated.

6. ***Acceptance Criteria:*** For each masonry wall, the extent of observed shrinkage and/or separation and cracking of masonry may not invalidate the evaluation basis or impact the wall's intended function. However, further evaluation is conducted if the extent of cracking and loss of material is sufficient to impact the intended function of the wall or invalidate its evaluation basis.

7. ***Corrective Actions:*** A corrective action option is to develop a new analysis or evaluation basis that accounts for the degraded condition of the wall (i.e., acceptance by further evaluation). Other alternatives include repair or replacing the degraded wall. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

8. ***Confirmation Process:*** As discussed in the Appendix for GALL, 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the administrative controls.

10. ***Operating Experience:*** Since 1980, masonry walls that perform an intended function have been systematically identified through licensee programs in response to NRC IEB 80-11, NRC Generic Letter 87-02, and 10 CFR 50.48. NRC IN 87-67 documented lessons learned from the NRC IEB 80-11 program and provided recommendations for administrative controls and periodic inspection to ensure that the evaluation basis for each safety-significant masonry wall is maintained. NUREG-1522 documents instances of observed cracks and other deterioration of masonry-wall joints at nuclear power plants. Whether conducted as a stand-alone program or as a part of structures monitoring, a masonry wall AMP that incorporates the recommendations delineated in NRC IN 87-67 should ensure that the intended functions of all masonry walls within the scope of license renewal are maintained for the period of extended operation.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

10 CFR 50.48, *Fire Protection,* Office of the Federal Register, National Archives and Records Administration, 2009.

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, 2009.

10 CFR 54.4, *Scope*, Office of the Federal Register, National Archives and Records Administration, 2009.

NRC Generic Letter 87-02, *Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46,* U.S. Nuclear Regulatory Commission, February 19, 1987.

NRC IE Bulletin 80-11, *Masonry Wall Design,* U.S. Nuclear Regulatory Commission, May 8, 1980.

NRC Information Notice 87-67, *Lessons Learned from Regional Inspections of Licensee Actions in Response to IE Bulletin 80-11*, U.S. Nuclear Regulatory Commission, December 31, 1987.

NRC Regulatory Guide 1.160, Rev. 2*, Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,* U.S. Nuclear Regulatory Commission, March 1997.

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.

NUREG-1522, *Assessment of Inservice Condition of Safety-Related Nuclear Power Plant Structures,* June 1995.

XI.S6 STRUCTURES MONITORING

Program Description

Implementation of structures monitoring under 10 CFR 50.65 (the Maintenance Rule) is addressed in Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.160, Rev. 2, and NUMARC 93-01, Rev. 2. These two documents provide guidance for development of licensee-specific programs to monitor the condition of structures and structural components within the scope of the Maintenance Rule, such that there is no loss of structure or structural component intended function.

The structures monitoring program consists of periodic visual inspections by personnel qualified to monitor structures and components for applicable aging effects, such as those described in the American Concrete Institute Standards (ACI) 349.3R, ACI 201.1R, and American National Standards Institute/American Society of Civil Engineers Standard (ANSI/ASCE) 11. Visual inspections should be supplemented with volumetric or surface examinations to detect stress corrosion cracking (SCC) in high strength (actual measured yield strength greater than or equal to 150 kilo-pound per square inch [ksi] or greater than or equal to 1,034 MPa) structural bolts greater than 1 inch (25 mm) in diameter. Identified aging effects are evaluated by qualified personnel using criteria derived from industry codes and standards contained in the plant current licensing bases, including ACI 349.3R, ACI 318, ANSI/ASCE 11, and the American Institute of Steel Construction (AISC) specifications, as applicable.

The program includes preventive actions delineated in NUREG-1339 and in Electric Power Research Institute (EPRI) NP-5769, NP-5067, and TR-104213 to ensure structural bolting integrity, if applicable.

The program also includes periodic sampling and testing of ground water and the need to assess the impact of any changes in its chemistry on below grade concrete structures.

If protective coatings are relied upon to manage the effects of aging for any structures included in the scope of this aging management program (AMP), the structures monitoring program is to address protective coating monitoring and maintenance.

Evaluation and Technical Basis

1. ***Scope of Program:*** The scope of the program includes all structures, structural components, component supports, and structural commodities in the scope of license renewal that are not covered by other structural AMPs (i.e., "ASME Section XI, Subsection IWE" (AMP XI.S1); "ASME Section XI, Subsection IWL" (AMP XI.S2); "ASME Section XI, Subsection IWF" (AMP XI.S3); "Masonry Walls" (AMP XI.S5); and NRC RG 1.127, "Inspection of Water-Control Structures Associated with Nuclear Power Plants" (AMP XI.S7). Examples of structures, structural components, and commodities in the scope of the program are concrete and steel structures, structural bolting, anchor bolts and embedments, component support members, pipe whip restraints and jet impingement shields, transmission towers, panels and other enclosures, racks, sliding surfaces, sump and pool liners, electrical cable trays and conduits, trash racks associated with water control structures, electrical duct banks, manholes, doors, penetration seals, and tube tracks. The applicant is to specify other structures or components that are in the scope of its structures monitoring program. The scope of this program includes periodic sampling and testing of ground water and may include inspection of masonry walls and water-control structures

provided all the attributes of "Masonry Walls" (AMP XI.S5) and NRC RG 1.127, "Inspection of Water-Control Structures Associated with Nuclear Power Plants" (AMP XI.S7) are incorporated in the attributes of this program.

2. ***Preventive Action:*** The structures monitoring program is a condition monitoring program. The program should include preventive actions delineated in NUREG-1339 and in EPRI NP-5769, NP-5067, and TR-104213 to ensure structural bolting integrity, if applicable. These actions emphasize proper selection of bolting material, lubricants, and installation torque or tension to prevent or minimize loss of bolting preload and cracking of high strength bolting. If the structural bolting consists of ASTM A325, ASTM F1852, and/or ASTM A490 bolts, the preventive actions for storage, lubricants, and stress corrosion cracking potential discussed in Section 2 of RCSC (Research Council for Structural Connections) publication "Specification for Structural Joints Using ASTM A325 or A490 Bolts," need to be used.

3. ***Parameters Monitored or Inspected:*** For each structure/aging effect combination, the specific parameters monitored or inspected depend on the particular structure, structural component, or commodity. Parameters monitored or inspected are commensurate with industry codes, standards, and guidelines and also consider industry and plant-specific operating experience. ACI 349.3R and ANSI/ASCE 11 provide an acceptable basis for selection of parameters to be monitored or inspected for concrete and steel structural elements and for steel liners, joints, coatings, and waterproofing membranes (if applicable).

   For concrete structures, parameters monitored include loss of material, cracking, increase in porosity and permeability, loss of foundation strength, and reduction in concrete anchor capacity due to local concrete degradation. Steel structures and components are monitored for loss of material due to corrosion. Structural bolting is monitored for loose bolts, missing or loose nuts, and other conditions indicative of loss of preload. High strength (actual measured yield strength ≥ 150 ksi or 1,034 MPa) structural bolts greater than 1 inch (25 mm) in diameter are monitored for SCC. Other structural bolting (ASTM A-325, ASTM F1852, and ASTM A490 bolts), and anchor bolts are monitored for loss of material, loose or missing nuts, and cracking of concrete around the anchor bolts. Accessible sliding surfaces are monitored for indication of significant loss of material due to wear or corrosion, debris, or dirt. Elastomeric vibration isolators and structural sealants are monitored for cracking, loss of material, and hardening. These parameters and other monitored parameters are selected to ensure that aging degradation leading to loss of intended functions will be detected and the extent of degradation can be determined. Ground water chemistry (pH, chlorides, and sulfates) are monitored periodically to assess its impact, if any, on below grade concrete structures. If necessary for managing settlement and erosion of porous concrete sub-foundations, the continued functionality of a site de-watering system is monitored. The plant-specific structures monitoring program should contain sufficient detail on parameters monitored or inspected to conclude that this program attribute is satisfied.

4. ***Detection of Aging Effects:*** Structures are monitored under this program using periodic visual inspection of each structure/aging effect combination by a qualified inspector to ensure that aging degradation will be detected and quantified before there is loss of intended function. Visual inspection of high strength (actual measured yield strength ≥ 150 ksi or 1,034 MPa) structural bolting greater than 1 inch (25 mm) in diameter is supplemented with volumetric or surface examinations to detect cracking. Other structural bolting (ASTM A-325, ASTM F1852, and ASTM A490 bolts) and anchor bolts are monitored for loss of material, loose or missing nuts, and cracking of concrete around the anchor bolts. Accessible sliding surfaces are monitored for indication of significant loss of material due to

wear or corrosion, debris, or dirt. Visual inspection of elastomeric vibration isolation elements should be supplemented by feel to detect hardening if the vibration isolation function is suspect. The inspection frequency depends on safety significance and the condition of the structure as specified in NRC RG 1.160, Rev. 2. In general, all structures and ground water quality are monitored on a frequency not to exceed 5 years. Some structures of lower safety significance, and subjected to benign environmental conditions, may be monitored at an interval exceeding five years; however, they should be identified and listed, together with their operating experience. The program includes provisions for more frequent inspections of structures and components categorized as (a)(1) in accordance with 10 CFR 50.65. Inspector qualifications should be consistent with industry guidelines and standards and guidelines for implementing the requirements of 10 CFR 50.65. Qualifications of inspection and evaluation personnel specified in ACI 349.3R are acceptable for license renewal.

The structures monitoring program addresses detection of aging affects for inaccessible, below-grade concrete structural elements. For plants with non-aggressive ground water/soil (pH > 5.5, chlorides < 500 ppm, or sulfates <1500 ppm), the program recommends: (a) evaluating 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 and (b) examining representative samples of the exposed portions of the below grade concrete, when excavated for any reason.

For plants with aggressive ground water/soil (pH < 5.5, chlorides > 500 ppm, or sulfates > 1500 ppm) and/or where the concrete structural elements have experienced degradation, a plant-specific AMP accounting for the extent of the degradation experienced should be implemented to manage the concrete aging during the period of extended operation.

5. ***Monitoring and Trending:*** Regulatory Position 1.5, "Monitoring of Structures," in NRC RG 1.160, Rev. 2, provides an acceptable basis for meeting the attribute. A structure is monitored in accordance with 10 CFR 50.65(a)(2) provided there is no significant degradation of the structure. A structure is monitored in accordance with 10 CFR 50.65(a)(1) if the extent of degradation is such that the structure may not meet its design basis or, if allowed to continue uncorrected until the next normally scheduled assessment, may not meet its design basis.

6. ***Acceptance Criteria:*** The structures monitoring program calls for inspection results to be evaluated by qualified engineering personnel based on acceptance criteria selected for each structure/aging effect to ensure that the need for corrective actions is identified before loss of intended functions. The criteria are derived from design bases codes and standards that include ACI 349.3R, ACI 318, ANSI/ASCE 11, or the relevant AISC specifications, as applicable, and consider industry and plant operating experience. The criteria are directed at the identification and evaluation of degradation that may affect the ability of the structure or component to perform its intended function. Applicants who are not committed to ACI 349.3R and elect to use plant-specific criteria for concrete structures should describe the criteria and provide a technical basis for deviations from those in ACI 349.3R. Loose bolts and nuts and cracked high strength bolts are not acceptable unless accepted by engineering evaluation. Structural sealants are acceptable if the observed loss of material, cracking, and hardening will not result is loss of sealing. Elastomeric vibration isolation elements are acceptable if there is no loss of material, cracking, or hardening that could lead to the reduction or loss of isolation function. Acceptance criteria for sliding surfaces are (a) no indications of excessive loss of material due to corrosion or wear and (b) no debris or dirt

that could restrict or prevent sliding of the surfaces as required by design. The structures monitoring program is to contain sufficient detail on acceptance criteria to conclude that this program attribute is satisfied.

7. ***Corrective Actions:*** Evaluations are performed for any inspection results that do not satisfy established criteria. Corrective actions are initiated in accordance with the corrective action process if the evaluation results indicate there is a need for a repair or replacement. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

8. ***Confirmation Process:*** As discussed in the Appendix for GALL, 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the administrative controls.

10. ***Operating Experience:*** Although in many plants, structures monitoring programs have only recently been implemented, plant maintenance has been ongoing since initial plant operations. NUREG-1522 documents the results of a survey sponsored in 1992 by the Office of Nuclear Regulatory Regulation to obtain information on the types of distress in the concrete and steel structures and components, the type of repairs performed, and the durability of the repairs. Licensees who responded to the survey reported cracking, scaling, and leaching of concrete structures. The degradation was attributed to drying shrinkage, freeze-thaw, and abrasion. The NUREG also describes the results of NRC staff inspections at six plants. The staff observed concrete degradation, corrosion of component support members and anchor bolts, cracks and other deterioration of masonry walls, and ground water leakage and seepage into underground structures. The observed and reported degradations were more severe at coastal plants than those observed in inland plants as a result of brackish and sea water. Previous license renewal applicants reported similar degradation and corrective actions taken through their structures monitoring program. Many license renewal applicants have found it necessary to enhance their structures monitoring program to ensure that the aging effects of structures and components in the scope of 10 CFR Part 54.4 are adequately managed during the period of extended operation. There is reasonable assurance that implementation of the structures monitoring program described above will be effective in managing the aging of the in-scope structures and component supports through the period of extended operation.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

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, 2009.

10 CFR 54.4, *Scope*, Office of the Federal Register, National Archives and Records Administration, 2009.

ACI Standard 201.1R, *Guide for Making a Condition Survey of Concrete in Service,* American Concrete Institute, 1992.

ACI Standard 318, *Building Code Requirements for Reinforced Concrete and Commentary,* American Concrete Institute.

ACI Standard 349.3R, *Evaluation of Existing Nuclear Safety-Related Concrete Structures,* American Concrete Institute, 2002.

AISC, *AISC Specification for Steel Buildings,* American Institute of Steel Construction, Inc., Chicago, IL.

ANSI/ASCE 11-90, 99, *Guideline for Structural Condition Assessment of Existing Buildings,* American Society of Civil Engineers.

EPRI NP-5067, *Good Bolting Practices, A Reference Manual for Nuclear Power Plant Maintenance Personnel*, Volume 1: Large Bolt Manual, 1987; Volume 2: Small Bolts and Threaded Fasteners, Electric Power Research Institute, 1990.

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

EPRI TR-104213, *Bolted Joint Maintenance & Application Guide,* Electric Power Research Institute, December 1995.

RCSC (Research Council on Structural Connections), *Specification for Structural Joints Using ASTM A325 or A490 Bolts,* Chicago, 2004.

NRC Regulatory Guide 1.127, Rev. 1, *Inspection of Water-Control Structures Associated with Nuclear Power Plants,* U.S. Nuclear Regulatory Commission, March 1978.

NRC Regulatory Guide 1.142, Rev. 2, *Safety-related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)*, U.S. Nuclear Regulatory Commission, November 2001.

NRC Regulatory Guide 1.160, Rev. 2, *Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,* U.S. Nuclear Regulatory Commission, March 1997.

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.

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

NUREG-1522, *Assessment of Inservice Condition of Safety-Related Nuclear Power Plant Structures,* June 1995.

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 raw water-control structures associated with emergency cooling water systems or flood protection of nuclear power plants. The NRC 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 NRC 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.

NRC 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. NRC RG 1.127 delineates current NRC practice in evaluating inservice inspection programs for water-control structures.

For plants not committed to NRC RG 1.127, Revision 1, aging management of water-control structures may be included in the “Structures Monitoring” (AMP XI.S6). Even if a plant is committed to NRC RG 1.127, Revision 1, aging management of certain structures and components may be included in the “Structures Monitoring” (AMP XI.S6). However, details pertaining to water-control structures, as described herein, are incorporated in AMP XI.S6 program attributes.

NRC RG 1.127 attributes evaluated below do not include inspection of dams. For dam inspection and maintenance, programs under the regulatory jurisdiction of the Federal Energy Regulatory Commission (FERC) or the U.S. Army Corps of Engineers, continued through the period of extended operation, are 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 evaluates the effectiveness of the aging management program (AMP) based on compatibility to the common practices of the FERC and Corps programs.

Evaluation and Technical Basis

1. ***Scope of Program:*** NRC RG 1.127 applies to raw 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. The scope of the program also includes structural steel and structural bolting associated with water-control structures, steel or wood piles and sheeting required for the stability of embankments and channel slopes, and miscellaneous steel, such as sluice gates and trash racks.

2. ***Preventive Action:*** NRC RG 1.127 is a condition monitoring program. This program is augmented to incorporate preventive measures recommended in NUREG-1339, Electric Power Research Institute (EPRI) TR-104213, EPRI NP-5067, and EPRI NP-5769 to ensure

structural bolting integrity, if applicable. The documents provide guidelines for selection of replacement bolting material, approved thread lubricants, and appropriate torque and preload to be used for installation of bolting. If the structural bolting consists of ASTM A325, ASTM F1852, and/or ASTM A490 bolts, the preventive actions for storage, lubricants, and stress corrosion cracking potential discussed in Section 2 of RCSC (Research Council for Structural Connections) publication “Specification for Structural Joints Using ASTM A325 or A490 Bolts” need to be used.

3. ***Parameters Monitored or Inspected:*** NRC 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 are those described in American Concrete Institute (ACI) 201.1 and ACI-349-3R. These include cracking, movements (e.g., settlement, heaving, deflection), conditions at junctions with abutments and embankments, loss of material, increase in porosity and permeability, 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.

   Steel components are monitored for loss of material due to corrosion.

   Parameters monitored for channels and canals include erosion or degradations that may impose constraints on the function of the cooling system and present a potential hazard to the safety of the plant. Submerged emergency canals (e.g., artificially dredged canals at the river bed or the bottom of the reservoir) should be monitored for sedimentation, debris, or instability of slopes that may impair the function of the canals under extreme low flow conditions.

   Further details of parameters to be monitored and inspected for these and other water-control structures are specified in Section C.2 of NRC RG 1.127. The program is augmented to require monitoring of bolted connections for loss of material and loose bolts and nuts and other conditions indicative of loss of preload. High strength (actual measured yield strength ≥ 150 ksi or 1,034 MPa) structural bolts greater than 1 inch (25 mm) in diameter are monitored for stress corrosion cracking, if applicable. Other structural bolting (ASTM A-325, ASTM F1852, and ASTM A490 bolts) and anchor bolts are monitored for loss of material, loose or missing nuts, and cracking of concrete around the anchor bolts. Accessible sliding surfaces are monitored for indication of significant loss of material due to wear or corrosion, debris, or dirt. The program also is augmented to require monitoring of wooden components for loss of material and change in material properties.

4. ***Detection of Aging Effects:*** NRC RG 1.127 specifies that inspection of water-control structures should be conducted under the direction of qualified engineers experienced in the investigation, design, construction, and operation of these types of facilities. 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. NRC 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. NRC RG 1.127 describes periodic inspections to be performed at least once every 5 years. This interval has been shown to be adequate to detect degradation of water-control structures before a loss of an intended function. The program should include provisions for increased inspection frequency if the extent of the degradation is such that the structure or component may not meet its design basis if allowed to continue uncorrected until the next normally scheduled inspection. NRC 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.

The program should address detection of aging affects for inaccessible, below-grade, and submerged concrete structural elements. For plants with non-aggressive raw water and groundwater/soil (pH > 5.5, chlorides < 500 parts per million [ppm], or sulfates < 1500 ppm), the program should require (a) evaluation of 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 and (b) examination of representative samples of the exposed portions of the below-grade concrete when excavated for any reason. Submerged concrete structures should be inspected during periods of low tide or when dewatered and accessible.

For plants with aggressive environment raw water (pH < 5.5, chlorides > 500 ppm, or sulfates > 1500 ppm) or ground water/soil and/or where the concrete structural elements have experienced degradation, a plant-specific AMP accounting for the extent of the degradation experienced should be implemented to manage the concrete aging during the period of extended operation.

5. ***Monitoring and Trending:*** Water-control structures are monitored by periodic inspection, as described in NRC RG 1.127. Changes of degraded conditions from prior inspection, such as growth of an active crack or extent of corrosion, should be trended until it is evident that the change is no longer occurring or until corrective actions are implemented in accordance with 10 CFR 50.65 and RG 1.160, Rev. 2.

6. ***Acceptance Criteria:*** Quantitative acceptance criteria to evaluate the need for corrective actions are not specified in NRC RG 1.127. However, the "Evaluation Criteria" provided in Chapter 5 of ACI 349.3R provide 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 are acceptable. Acceptance criteria for earthen structures, such as canals and embankments, are consistent with programs falling within the regulatory jurisdiction of the FERC or the U.S. Army Corps of Engineers. Loose bolts and nuts, cracked high strength bolts, and degradation of piles and sheeting are accepted by engineering evaluation or subject to corrective actions. Engineering evaluation should be documented and based on codes, specifications, and standards such as AISC specifications, SEI/ASCE 11, and those referenced in the plant's current licensing basis.

7. ***Corrective Actions:*** NRC RG 1.127 recommends that when inspection 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

8. ***Confirmation Process:*** As discussed in the Appendix for GALL, 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the administrative controls.

10. ***Operating Experience:*** Degradation of water-control structures has been detected, through NRC RG 1.127 programs, at a number of nuclear power plants, and, in some cases, it has required remedial action. NRC NUREG-1522 described instances and corrective actions of severely degraded steel and concrete components at the intake structure and pumphouse of coastal plants. Other degradation described in the NUREG include appreciable leakage from the spillway gates, concrete cracking, corrosion of spillway bridge beam seats of a plant dam and cooling canal, and appreciable differential settlement of the outfall structure of another. 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 NRC RG 1.127 have been successful in detecting significant degradation before loss of intended function occurs.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

ACI Standard 201.1R, *Guide for Making a Condition Survey of Concrete in Service,* American Concrete Institute, 1992.

ACI Standard 349.3R, *Evaluation of Existing Nuclear Safety-Related Concrete Structures,* American Concrete Institute, 2002.

EPRI NP-5067, *Good Bolting Practices, A Reference Manual for Nuclear Power Plant Maintenance Personnel*, Volume 1: Large Bolt Manual, 1987; Volume 2: Small Bolts and Threaded Fasteners, Electric Power Research Institute, 1990.

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

EPRI TR-104213, *Bolted Joint Maintenance & Application Guide,* Electric Power Research Institute, December 1995.

NRC Regulatory Guide 1.127, *Inspection of Water-Control Structures Associated with Nuclear Power Plants,* Revision 1, U.S. Nuclear Regulatory Commission, March 1978.

NRC Regulatory Guide 1.160, Rev. 2, *Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,* U.S. Nuclear Regulatory Commission, March 1997.

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

NUREG-1522, *Assessment of Inservice Conditions of Safety-Related Nuclear Plant Structures*, U.S. Nuclear Regulatory Commission, June 1995.

RCSC (Research Council on Structural Connections), *Specification for Structural Joints Using ASTM A325 or A490 Bolts,* 2004.

XI.S8 PROTECTIVE COATING MONITORING AND MAINTENANCE PROGRAM

Program Description

Proper maintenance of protective coatings inside containment (defined as Service Level I in Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.54, Rev. 1, or latest version) is essential to ensure operability of post-accident safety systems that rely on water recycled through the containment sump/drain system. Degradation of coatings can lead to clogging of Emergency Core Cooling Systems (ECCS) suction strainers, which reduces flow through the system and could cause unacceptable head loss for the pumps.

Maintenance of Service Level I coatings applied to carbon steel and concrete surfaces inside containment (e.g., steel liner, steel containment shell, structural steel, supports, penetrations, and concrete walls and floors) also serve to prevent or minimize loss of material due to corrosion of carbon steel components and aids in decontamination. Regulatory Position C4 in NRC RG 1.54, Rev. 2, describes an acceptable technical basis for a Service Level I coatings monitoring and maintenance program that can be credited for managing the effects of corrosion for carbon steel elements inside containment. American Society for Testing of Materials (ASTM) D 5163-08 and endorsed years of the standard in NRC RG 1.54 are acceptable and considered consistent with NUREG-1801. In addition, Electric Power Research Institute (EPRI) Report 1019157, Guidelines for Inspection and Maintenance of Safety-related Protective Coatings, provides additional information on the ASTM standard guidelines.

A comparable program for monitoring and maintaining protective coatings inside containment, developed in accordance with NRC RG 1.54, Rev. 2, is acceptable as an aging management program for license renewal.

Service Level I coatings credited for preventing corrosion of steel containments and steel liners for concrete containments are subject to requirements specified by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, Subsection IWE (AMP XI.S1). However, this program (AMP XI.S8) reviews Service Level I coatings to ensure that the protective coating monitoring and maintenance program are adequate for license renewal.

Evaluation and Technical Basis

1. ***Scope of Program:*** The minimum scope of the program is Service Level I coatings applied to steel and concrete surfaces inside containment (e.g., steel liner, steel containment shell, structural steel, supports, penetrations, and concrete walls and floors), defined in NRC RG 1.54, Rev. 2, as follows: "Service Level I coatings are used in areas inside the reactor containment where the coating failure could adversely affect the operation of post-accident fluid systems and thereby impair safe shutdown." The scope of the program also should include any Service Level I coatings that are credited by the licensee for preventing loss of material due to corrosion in accordance with AMP XI.S1.

2. ***Preventive Action:*** The program is a condition monitoring program and does not recommend any preventive actions. However, for plants that credit coatings to minimize loss of material, this program is a preventive action.

3. ***Parameters Monitored or Inspected:*** Regulatory Position C4 in NRC RG 1.54, Rev 1, states that "ASTM D 5163-96 provides guidelines that are acceptable to the NRC staff for

establishing an in-service coatings monitoring program for Service Level I coating systems in operating nuclear power plants..." ASTM D 5163-96 has been superseded by ASTM D 5163-08. ASTM D 5163-08 identifies the parameters monitored or inspected to be "any visible defects, such as blistering, cracking, flaking, peeling, rusting, and physical damage."

4. ***Detection of Aging Effects:*** ASTM D 5163-08, paragraph 6, defines the inspection frequency to be each refueling outage or during other major maintenance outages, as needed. ASTM D 5163-08, paragraph 9, discusses the qualifications for inspection personnel, the inspection coordinator, and the inspection results evaluator. ASTM D 5163-08, subparagraph 10.1, discusses development of the inspection plan and the inspection methods to be used. It states that a general visual inspection shall be conducted on all readily accessible coated surfaces during a walk-through. After a walk-through, or during the general visual inspection, thorough visual inspections shall be carried out on previously designated areas and on areas noted as deficient during the walk-through. A thorough visual inspection shall also be carried out on all coatings near sumps or screens associated with the Emergency Core Cooling System (ECCS). This subparagraph also addresses field documentation of inspection results. ASTM D 5163-08, subparagraph 10.5, identifies instruments and equipment needed for inspection.

5. ***Monitoring and Trending:*** ASTM D 5163-08 identifies monitoring and trending activities in subparagraph 7.2, which specifies a pre-inspection review of the previous two monitoring reports, and in subparagraph 11.1.2, which specifies that the inspection report should prioritize repair areas as either needing repair during the same outage or as postponed to future outages, but under surveillance in the interim period.

6. ***Acceptance Criteria:*** ASTM D 5163-08, subparagraphs 10.2.1 through 10.2.6, 10.3, and 10.4, contains one acceptable method for the characterization, documentation, and testing of defective or deficient coating surfaces. Additional ASTM and other recognized test methods are available for use in characterizing the severity of observed defects and deficiencies. The evaluation covers blistering, cracking, flaking, peeling, delamination, and rusting. ASTM D 5163-08, paragraph 11, addresses evaluation. It specifies that the inspection report is to be evaluated by the responsible evaluation personnel, who prepare a summary of findings and recommendations for future surveillance or repair, and prioritization of repairs.

7. ***Corrective Actions:*** A recommended corrective action plan is required for major defective areas so that these areas can be repaired during the same outage, if appropriate. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

8. ***Confirmation Process:*** As discussed in the Appendix for GALL, 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 for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the administrative controls.

10. ***Operating Experience***: NRC Information Notice 88-82, NRC Bulletin 96-03, NRC Generic Letter (GL) 04-02, and NRC GL 98-04 describe industry experience pertaining to coatings degradation inside containment and the consequential clogging of sump strainers. NRC

RG 1.54, Rev. 1, was issued in July 2000. Monitoring and maintenance of Service Level I coatings conducted in accordance with Regulatory Position C4 is expected to be an effective program for managing degradation of Service Level I coatings and, consequently, an effective means to manage loss of material due to corrosion of carbon steel structural elements inside containment.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

ASTM D 5163-05, *Guide for Establishing Procedures to Monitor the Performance of Coating Service Level I Coating Systems in an Operating Nuclear Power Plant,* American Society for Testing and Materials, 2005.

ASTM D 5163-08, *Standard Guide for Establishing a Program for Condition Assessment of Coating Service Level I Coating Systems in Nuclear Power Plants,* American Society for Testing and Materials, 2008.

ASTM D 5163-96, *Standard Guide for Establishing Procedures to Monitor the Performance of Safety Related Coatings in an Operating Nuclear Power Plant,* American Society for Testing and Materials, 1996.

EPRI Report 1003102, *Guideline on Nuclear Safety-Related Coatings*, Revision 1, (Formerly TR-109937), Electric Power Research Institute, November 2001.

EPRI Report 1019157, *Guideline on Nuclear Safety-Related Coatings*, Revision 2, (Formerly TR-109937and 1003102), Electric Power Research Institute, December 2009.

NRC Bulletin 96-03, *Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors*, U.S. Nuclear Regulatory Commission, May 6, 1996.

NRC Generic Letter 98-04, *Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-Of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment,* U.S. Nuclear Regulatory Commission, July 14, 1998.

NRC Generic Letter 04-02, *Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors*, U.S. Nuclear Regulatory Commission, September 13, 2004.

NRC Information Notice 88-82, *Torus Shells with Corrosion and Degraded Coatings in BWR Containments*, U.S. Nuclear Regulatory Commission, November 14, 1988.

NRC Information Notice 97-13, *Deficient Conditions Associated With Protective Coatings at Nuclear Power Plants*, U.S. Nuclear Regulatory Commission, March 24, 1997.

NRC Regulatory Guide 1.54, Rev. 0, *Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants,* U.S. Nuclear Regulatory Commission, June 1973.

NRC Regulatory Guide 1.54, Rev. 1, *Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants,* U.S. Nuclear Regulatory Commission, July 2000.

NRC Regulatory Guide 1.54, Rev. 2, *Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants,* U.S. Nuclear Regulatory Commission, October 2010.

XI.E1 INSULATION MATERIAL FOR ELECTRICAL CABLES AND CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS

Program Description

The purpose of the aging management program (AMP) described herein is to provide reasonable assurance that the intended functions of electrical cables and connections that are not subject to the environmental qualification requirements of 10 CFR 50.49 and are exposed to adverse localized environments caused by temperature, radiation, or moisture are maintained consistent with the current licensing basis through the period of extended operation.

In most areas within a nuclear power plant, the actual ambient environments (e.g., temperature, radiation, or moisture) are less severe than the plant design environment. However, in a limited number of localized areas, the actual environments may be more severe than the plant design environment.

Insulation materials used in electrical cables and connections may degrade more rapidly than expected in these adverse localized environments. An adverse localized environment is a condition in a limited plant area that is significantly more severe than the plant design environment for the cable or connection insulation material that could increase the rate of aging of a component or have an adverse effect on operability. An adverse localized environment exists based on the most limiting condition for temperature, radiation, or moisture for the insulation material of cables or connections. Adverse localized environments can be identified through the use of an integrated approach. This approach may include, but is not limited to, (a) the review of Environmental Qualification (EQ) zone maps that show radiation levels and temperatures for various plant areas, (b) consultations with plant staff who are cognizant of plant conditions, (c) utilization of infrared thermography to identify hot spots on a real-time basis, and (d) the review of relevant plant-specific and industry operating experience.

The program described herein was written specifically to address cables and connections at plants whose configuration is such that most (if not all) cables and connections installed in adverse localized environments are accessible. Cables and connections from accessible areas are inspected and represent, with reasonable assurance, all cables and connections in the adverse localized environments. If an unacceptable condition or situation is identified for a cable or connection in the inspection, a determination is made as to whether the same condition or situation is applicable to inaccessible cables or connections. As such, this program does not apply to plants in which most cables are inaccessible.

As stated in NUREG/CR-5643, “the major concern is that failures of deteriorated cable systems (cables, connections, and penetrations) might be induced during accident conditions.” Since the cables and connections are not subject to the environmental qualification requirements of 10 CFR 50.49, an AMP is required to manage the aging effects. This AMP provides reasonable assurance the insulation material for electrical cables and connections will perform its intended function for the period of extended operation.

Evaluation and Technical Basis

1. ***Scope of Program:*** This AMP applies to accessible electrical cables and connections within the scope of license renewal that are located in adverse localized environments caused by temperature, radiation, or moisture.

2. ***Preventive Actions:*** This is a condition monitoring program and no actions are taken as part of this program to prevent or mitigate aging degradation.

3. ***Parameters Monitored/Inspected:*** Accessible electrical cables and connections installed in adverse localized environments are visually inspected for cable jacket and connection insulation surface anomalies indicating signs of reduced insulation resistance due to thermal/thermoxidative degradation of organics, radiolysis and photolysis (UV sensitive materials only) of organics; radiation-induced oxidation, and moisture intrusion as indicated by signs of embrittlement, discoloration, cracking, melting, swelling or surface contamination. An adverse localized environment is a plant-specific condition; therefore, the applicant should clearly define how this condition is determined. The applicant should determine and inspect the adverse localized conditions for each of the most limiting temperature, radiation, or moisture conditions for the accessible cables and connections that are within the scope of license renewal.

4. ***Detection of Aging Effects:*** Insulation aging degradation from temperature, radiation, or moisture causes cable jacket and connection insulation surface anomalies. Accessible electrical cables and connections installed in adverse localized environments are visually inspected for cable jacket and connection insulation surface anomalies, such as embrittlement, discoloration, cracking, melting, swelling or surface contamination. The inspection of cable jacket and connection insulation surfaces is used to infer the adequacy of the cables and connections. Accessible electrical cables and connections installed in adverse localized environments are visually inspected at least once every 10 years. This is an adequate period to preclude failures of the cables and connection insulation since experience has shown that aging degradation is a slow process. A 10-year inspection interval provides two data points during a 20-year period, which can be used to characterize the degradation rate. The first inspection for license renewal is to be completed prior to the period of extended operation.

5. ***Monitoring and Trending:*** Trending actions are not included as part of this AMP, because the ability to trend visual inspection results is limited. However, inspection results that are trendable provide additional information on the rate of cable or connection degradation.

6. ***Acceptance Criteria:*** The accessible cables and connections are to be free from unacceptable visual indications of surface anomalies that suggest that cable jacket or connection insulation degradation exists. An unacceptable indication is defined as a noted condition or situation that, if left unmanaged, could lead to a loss of the intended function.

7. ***Corrective Actions:*** All unacceptable visual indications of cable jacket and connection insulation surface anomalies are subject to an engineering evaluation. Such an evaluation is to consider the age and operating environment of the component as well as the severity of the anomaly and whether such an anomaly has previously been correlated to degradation of cables or connections. Corrective actions may include, but are not limited to, testing, shielding, or otherwise changing the environment or relocation or replacement of the affected cables or connections. When an unacceptable condition or situation is identified, a determination is made as to whether the same condition or situation is applicable to inaccessible cables or connections. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

8. ***Confirmation Process:*** As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process.

9. ***Administrative Controls:*** The administrative controls for this AMP provide for a formal review and approval process. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the administrative controls.

10. ***Operating Experience:*** Operating experience has shown that adverse localized environments caused by elevated temperature, radiation, or moisture for electrical cables and connections may exist. For example next to or above (within 3 feet of) steam generators, pressurizers, or hot process pipes, such as feedwater lines. These adverse localized environments have been found to cause degradation of the insulating materials on electrical cables and connections that are visually observable, such as color changes or surface cracking. These visual indications can be used as indicators of degradation.

    This AMP considers the technical information and guidance provided in NUREG/CR-5643, IEEE Std. 1205-2000, SAND96-0344, and EPRI TR-109619.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

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

IEEE Std. 1205-2000, *IEEE Guide for Assessing, Monitoring and Mitigating Aging Effects on Class 1E Equipment Used in Nuclear Power Generating Stations.*

NUREG/CR-5643, *Insights Gained From Aging Research,* U. S. Nuclear Regulatory Commission, March 1992.

SAND96-0344, *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.

XI.E2 INSULATION MATERIAL FOR ELECTRICAL CABLES AND CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS USED IN INSTRUMENTATION CIRCUITS

Program Description

The purpose of this aging management program (AMP) is to provide reasonable assurance that the intended functions of electrical cables and connections (that are not subject to the environmental qualification requirements of 10 CFR 50.49 and are used in instrumentation circuits with sensitive, high-voltage, low-level current signals exposed to adverse localized environments caused by temperature, radiation, or moisture) are maintained consistent with the current licensing basis through the period of extended operation.

In most areas within a nuclear power plant, the actual ambient environments (e.g., temperature, radiation, or moisture) are less severe than the plant design environment. However, in a limited number of localized areas, the actual environments may be more severe than the design environment.

Insulation materials used in electrical cables or connections may degrade more rapidly in adverse localized environments. An adverse localized environment is a condition in a limited plant area that is significantly more severe than the plant design environment for the cable or connection insulation material that could increase the rate of aging of a component or have an adverse effect on operability. Exposure of electrical cable and connection insulation material to adverse localized environments caused by temperature, radiation, or moisture can result in reduced insulation resistance (IR). Reduced IR causes an increase in leakage currents between conductors and from individual conductors to ground. A reduction in IR is a concern for all circuits, but especially those with sensitive, high voltage, low-level current signals, such as radiation monitoring and nuclear instrumentation circuits, because a reduced IR may contribute to signal inaccuracies.

In this AMP, either of two methods can be used to identify the existence of aging degradation. In the first method, calibration results or findings of surveillance testing programs are evaluated to identify the existence of cable and connection insulation material aging degradation. In the second method, direct testing of the cable system is performed.

This AMP applies to high-range-radiation and neutron flux monitoring instrumentation cables in addition to other cables used in high voltage, low-level current signal applications that are sensitive to reduction in IR. For these cables, AMP XI.E1 does not apply.

As stated in NUREG/CR-5643, “the major concern is that failures of deteriorated cable systems (cables, connections, and penetrations) might be induced during accident conditions.” Since the instrumentation cables and connections are not subject to the environmental qualification requirements of 10 CFR 50.49, an AMP is required to manage the aging effects. This AMP provides reasonable assurance the insulation material for electrical cables and connections will perform its intended function for the period of extended operation.

Evaluation and Technical Basis

1. ***Scope of Program:*** This AMP applies to electrical cables and connections (cable system) used in circuits with sensitive, high voltage, low-level current signals, such as radiation

monitoring and nuclear instrumentation, that are subject to aging management review and installed in adverse localized environments caused by temperature, radiation, or moisture.

2. ***Preventive Actions:*** This is a performance monitoring program and no actions are taken as part of this program to prevent or mitigate aging degradation.

3. ***Parameters Monitored/Inspected:*** The parameters monitored are determined from the specific calibration, surveillances, or testing performed and are based on the specific instrumentation circuit under surveillance or being calibrated, as documented in plant procedures.

4. ***Detection of Aging Effects:*** Review of calibration results or findings of surveillance programs can provide an indication of the existence of aging effects based on acceptance criteria related to instrumentation circuit performance. By reviewing the results obtained during normal calibration or surveillance, an applicant may detect severe aging degradation prior to the loss of the cable and connection intended function. The first reviews are completed prior to the period of extended operation and at least every 10 years thereafter. All calibration or surveillance results that do not meet acceptance criteria are reviewed for aging effects when the results are available.

   Cable system testing is conducted when the calibration or surveillance program does not include the cabling system in the testing circuit, or as an alternative to the review of calibration results described above. A proven cable system test for detecting deterioration of the insulation system (such as insulation resistance tests, time domain reflectometry tests, or other testing judged to be effective in determining cable system insulation condition as justified in the application) is performed. The test frequency of the cable system is determined by the applicant based on engineering evaluation, but the test frequency is at least once every 10 years. The first test is to be completed prior to the period of extended operation.

5. ***Monitoring and Trending:*** Trending actions are not included as part of this AMP because the ability to trend test results is dependent on the specific type of test chosen. However, test results that are trendable provide additional information on the rate of cable or connection degradation.

6. ***Acceptance Criteria:*** Calibration results or findings of surveillance and cable system testing are to be within the acceptance criteria, as set out in the applicant's procedures.

7. ***Corrective Actions:*** Corrective actions, such as recalibration and circuit trouble-shooting, are implemented when calibration, surveillance, or cable system test results do not meet the acceptance criteria. An engineering evaluation is performed when the acceptance criteria are not met in order to ensure that the intended functions of the electrical cable system can be maintained consistent with the current licensing basis. Such an evaluation is to consider the significance of the calibration, surveillance, or cable system test results; the operability of the component; the reportability of the event; the extent of the concern; the potential root causes for not meeting the acceptance criteria; the corrective actions required; and likelihood of recurrence. When an unacceptable condition or situation is identified, a determination also is made as to whether the review of calibration and surveillance results or the cable system testing frequency needs to be increased. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

8. ***Confirmation Process:*** As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address confirmation process.

9. ***Administrative Controls:*** The administrative controls for this AMP provide for a formal review and approval process. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the administrative controls.

10. ***Operating Experience:*** Operating experience has identified a case where a change in temperature across a high range radiation monitor cable in containment resulted in a substantial change in the reading of the monitor. Changes in instrument calibration can be caused by degradation of the circuit cable and are a possible indication of electrical cable degradation.

    The vast majority of site-specific and industry wide operating experience regarding neutron flux instrumentation circuits is related to cable/connector issues inside containment near the reactor vessel.

    This AMP considers the technical information and guidance provided in NUREG/CR-5643, IEEE Std. 1205-2000, SAND96-0344, EPRI TR-109619, NRC IN 97-45, and NRC IN 97-45, Supplement 1.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

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

IEEE Std. 1205-2000, *IEEE Guide for Assessing, Monitoring and Mitigating Aging Effects on Class 1E Equipment Used in Nuclear Power Generating Stations.*

NRC Information Notice 97-45, *Environmental Qualification Deficiency for Cables and Containment Penetration Pigtails*, U. S, Nuclear Regulatory Commission, July 2, 1997.

NRC Information Notice 97-45, Supplement 1, *Environmental Qualification Deficiency for Cables and Containment Penetration Pigtails*, U. S, Nuclear Regulatory Commission, February 17, 1998.

NUREG/CR-5643, *Insights Gained From Aging Research,* U. S. Nuclear Regulatory Commission, March 1992.

SAND96-0344, *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.

XI.E3 INACCESSIBLE POWER CABLES NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS

Program Description

The purpose of the aging management program (AMP) described herein is to provide reasonable assurance that the intended functions of inaccessible or underground power cables that are not subject to the environmental qualification requirements of 10 CFR 50.49 and are exposed to wetting or submergence are maintained consistent with the current licensing basis through the period of extended operation.

Most electrical cables in nuclear power plants are located in dry environments. However, some cables may be exposed to wetting or submergence, and are inaccessible or underground, such as cables in conduits, cable trenches, cable troughs, duct banks, underground vaults, or directly buried in soil installations. When a power cable (greater than or equal to 400 volts) is exposed to wet, submerged, or other adverse environmental conditions for which it was not designed, an aging effect of reduced insulation resistance may result, causing a decrease in the dielectric strength of the conductor insulation. This insulation degradation can be caused by wetting or submergence. This can potentially lead to failure of the cable's insulation system.

In this AMP, periodic actions are taken to prevent cables from being exposed to significant moisture, defined as periodic exposures to moisture that last more than a few days (e.g., cable wetting or submergence in water. Examples of periodic actions are inspecting for water collection in cable manholes and conduits and draining water, as needed. However, the above actions are not sufficient to ensure that water is not trapped elsewhere in the raceways. For example, (a) if a duct bank conduit has low points in the routing, there could be potential for long-term submergence at these low points; (b) concrete raceways may crack due to soil settling over a long period of time; (c) manhole covers may not be watertight; (d) in certain areas, the water table is high in seasonal cycles, so the raceways may get refilled soon after purging; and (e) potential uncertainties exist with water trees even when duct banks are sloped with the intention to minimize water accumulation.

Experience has shown that insulation degradation may occur if the cables are exposed to 100 percent relative humidity. The above periodic actions are necessary to minimize the potential for insulation degradation. In addition to above periodic actions, in-scope power cables exposed to significant moisture are tested to indicate the condition of the conductor insulation. The specific type of test performed is determined prior to the initial test, and is to be a proven test for detecting deterioration of the insulation system due to wetting or submergence, such as Dielectric Loss (Dissipation Factor/Power Factor), AC Voltage Withstand, Partial Discharge, Step Voltage, Time Domain Reflectometry, Insulation Resistance and Polarization Index, Line Resonance Analysis, or other testing that is state-of-the-art at the time the tests are performed. One or more tests are used to determine the condition of the cables so they will continue to meet their intended function during the period of extended operation.

As stated in NUREG/CR-5643, "the major concern is that failures of deteriorated cable systems (cables, connections, and penetrations) might be induced during accident conditions." Because the cables are not subject to the environmental qualification requirements of 10 CFR 50.49, an AMP is required to manage the aging effects. This AMP provides reasonable assurance the insulation material for electrical cables will perform its intended function for the period of extended operation.

Evaluation and Technical Basis

1. ***Scope of Program:*** This AMP applies to all inaccessible or underground (e.g., in conduit, duct bank, or direct buried) power cables (greater than or equal to 400 volts) within the scope of license renewal exposed to adverse environments, primarily significant moisture. Significant moisture is defined as periodic exposures to moisture that last more than a few days (e.g., cable wetting or submergence in water). Submarine or other cables designed for continuous wetting or submergence are not included in this AMP.

2. ***Preventive Actions:*** This is a condition monitoring program. However, periodic actions are taken to prevent inaccessible cables from being exposed to significant moisture, such as identifying and inspecting in-scope accessible cable conduit ends and cable manholes for water collection, and draining the water, as needed.

   The inspection frequency for water collection is established and performed based on plant-specific operating experience with cable wetting or submergence in manholes (i.e., the inspection is performed periodically based on water accumulation over time and event driven occurrences, such as heavy rain or flooding). The periodic inspection should occur at least annually. The inspection should include direct observation that cables are not wetted or submerged, that cables/splices and cable support structures are intact, and that dewatering/drainage systems (i.e., sump pumps) and associated alarms operate properly. In addition, operation of dewatering devices should be inspected and operation verified prior to any known or predicted heavy rain or flooding events. If water is found during inspection (i.e., cable exposed to significant moisture), corrective actions are taken to keep the cable dry and to assess cable degradation. The first inspection for license renewal is completed prior to the period of extended operation.

3. ***Parameters Monitored/Inspected:*** Inspection for water collection is performed based on plant-specific operating experience with water accumulation in the manhole. Inaccessible or underground power (greater than or equal to 400 volts) cables within the scope of license renewal exposed to significant moisture are tested to provide an indication of the condition of the conductor insulation. The specific type of test to be used should be capable of detecting reduced insulation resistance of the cable's insulation system due to wetting or submergence.

4. ***Detection of Aging Effects:*** For power cables exposed to significant moisture, test frequencies are adjusted based on test results (including trending of degradation where applicable) and operating experience. Cable testing should occur at least once every 6 years. A 6-year interval provides multiple data points during a 20-year period, which can be used to characterize the degradation rate. This is an adequate period to monitor performance of the cable and take appropriate corrective actions since experience has shown that although a slow process, aging degradation could be significant.. The first tests for license renewal are to be completed prior to the period of extended operation with subsequent tests performed at least every 6 years thereafter. The applicant can assess the condition of the cable insulation with reasonable confidence using one or more of the following techniques: Dielectric Loss (Dissipation Factor/Power Factor), AC Voltage Withstand, Partial Discharge, Step Voltage, Time Domain Reflectometry, Insulation Resistance and Polarization Index, Line Resonance Analysis, or other testing that is state-of-the-art at the time the tests are performed. One or more tests are used to determine the condition of the cables so they will continue to meet their intended function during the period of extended operation.

5. ***Monitoring and Trending:*** Trending actions are included as part of this AMP, although the ability to trend results is dependent on the specific type of test(s) or inspection chosen. Results that are trendable provide additional information on the rate of cable insulation degradation.

6. ***Acceptance Criteria:*** The acceptance criteria for each test are defined by the specific type of test performed and the specific cable tested. Acceptance criteria for inspections of manholes are defined by the observation that the cables and support structures are not submerged or immersed in standing water at the time of the inspection.

7. ***Corrective Actions:*** Corrective actions are taken and an engineering evaluation is performed when the test or inspection acceptance criteria are not met. Such an evaluation considers the significance of the test or inspection results, the operability of the component, the reportability of the event, the extent of the concern, the potential root causes for not meeting the test or inspection acceptance criteria, the corrective actions required, and the likelihood of recurrence. When 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, in-scope power cables. Corrective actions may include, but are not limited to, installation of permanent drainage systems, installation of sump pumps and alarms, more frequent cable testing or manhole inspections, or replacement of the affected cable. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

8. ***Confirmation Process:*** As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process.

9. ***Administrative Controls:*** The administrative controls for this AMP provide for a formal review and approval process. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the administrative controls.

10. ***Operating Experience:*** Operating experience has shown that insulation materials are susceptible to water tree formation. The formation and growth of water trees varies directly with operating voltage. Aging effects of reduced insulation resistance due to other mechanisms may also result in a decrease in the dielectric strength of the conductor insulation. Minimizing exposure to moisture mitigates the potential for the development of reduced insulation resistance.

    Recent incidents involving early failures of electric cables and cable failures leading to multiple equipment failures, are cited in NRC IN 2002-12, “Submerged Safety-Related Cables,” and NRC GL 2007-01, “Inaccessible or Underground Power Cable Failures That Disable Accident Mitigation Systems or Cause Plant Transients.”

    The NRC issued GL 2007-001 on inaccessible or underground cables to (a) inform licensees that the failure of certain power cables can affect the functionality of multiple accident mitigation systems or cause plant transients and (b) gather information from licensees on the monitoring of inaccessible or underground power cable failures for all cables that are within the scope of the Maintenance Rule. Based on the review of licensees’ responses, the NRC staff has identified 269 cable failures for 104 reactor units. The data obtained from the GL responses show an increasing trend of cable failures. The NRC staff has noted that the predominant factor contributing to cable failures at nuclear power plants was due to moisture/submergence. The staff also noted that the GL failure data show that the majority of the reported failures occurred at the 4160-volt, 480 volt, and 600-volt service voltage levels for both energized and de-energized cables. These cables are failing within the plants' 40-year licensing period.

The NRC inspectors also have continued to identify safety-related cables which are submerged. The staff noted that licensees had not demonstrated that the subject safety-related cables were designed for wetted or submerged service for the current license period.

This AMP considers the technical information and generic communication guidance provided in NUREG/CR-5643; IEEE Std. 1205-2000; SAND96-0344; EPRI 109619; EPRI 103834-P1-2; NRC IN 2002-12; NRC GL 2007-01; NRC GL 2007-01 Summary Report; NRC Inspection Procedure, Attachment 71111.06, Flood Protection Measures; NRC Inspection Procedure, Attachment 71111.01, Adverse Weather Protection; RG 1.211 Rev 0; DG-1240; and NUREG/CR-7000.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

DG-1240, *Condition Monitoring Program for Electric Cables Used In Nuclear Power Plants*, June 2010.

EPRI TR-103834-P1-2, *Effects of Moisture on the Life of Power Plant Cables,* Electric Power Research Institute, Palo Alto, CA, August 1994.

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

IEEE Std. 1205-2000, *IEEE Guide for Assessing, Monitoring and Mitigating Aging Effects on Class 1E Equipment Used in Nuclear Power Generating Stations.*

NRC Inspection Procedure, Attachment 71111.06, *Flood Protection Measures*, June 25, 2009.

NRC Inspection Procedure, Attachment 71111.01, *Adverse Weather Protection*, April 8, 2009.

NRC Information Notice 2002-12, *Submerged Safety-Related Electrical Cables*, March 21, 2002.

NRC Generic Letter 2007-01, Summary Report, *Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients*, November 12, 2008.

NUREG/CR-5643, *Insights Gained From Aging Research,* U. S. Nuclear Regulatory Commission, March 1992.

SAND96-0344, *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.

RG 1.211 Rev 0, *Qualification of Safety-Related Cables and Field Splices for Nuclear Power Plants*, April 2009.

NUREG/CR-7000, *Essential Elements of an Electric Cable Condition Monitoring Program*, January 2010.

XI.E4 METAL ENCLOSED BUS

Program Description

The purpose of this aging management program (AMP) is to provide an internal and external inspection of Metal Enclosed Buses (MEBs) to identify age-related degradation of insulating material (i.e., porcelain, xenoy, thermoplastic organic polymers), and metallic and elastomer components (e.g., gaskets, boots, and sealants).

MEBs are electrical buses installed on electrically insulated supports that are constructed with each phase conductor enclosed in a separate metal enclosure (isolated phase bus), all conductors enclosed in a common metal enclosure (non-segregated bus), or all phase conductors in a common metal enclosure, but separated by metal barriers between phases (segregated bus). The conductors are adequately separated and insulated from ground by insulating supports or bus insulation. The MEBs are used in power systems to connect various elements in electric power circuits, such as switchgear, transformers, main generators, and diesel generators.

Industry operating experience indicates that failures of MEBs have been caused by cracked insulation and moisture, debris, or excessive dust buildup internal to the bus duct housing. Cracked insulation has resulted from high ambient temperature and contamination from bus bar joint compounds. Cracked insulation in the presence of moisture or debris has provided phase-to-phase or phase-to-ground electrical tracking paths, which has resulted in catastrophic failure of the buses. Bus failure has led to loss of power to electrical loads connected to the buses, causing subsequent reactor trips and initiating unnecessary challenges to plant systems and operators.

MEBs may experience increased resistance of connection due to loosening of bolted bus duct connections caused by repeated thermal cycling of connected loads. This phenomenon can occur in heavily loaded circuits (i.e., those exposed to appreciable ohmic heating). For example, SAND 96-0344 identified instances of termination loosening at several plants due to thermal cycling and NRC IN 2000-14 identified torque relaxation of splice plate connecting bolts as one potential cause of a MEB fault.

This AMP includes the inspection of all bus ducts within the scope of license renewal and a sample of accessible MEB bolted connections for increased resistance of connection. The technical basis for the sample selections should be documented. If an unacceptable condition or situation is identified in the selected sample, a determination is made as to whether the same condition or situation is applicable to other connections not tested.

Evaluation and Technical Basis

1. ***Scope of Program:*** This AMP manages the age-related degradation effects for electrical bus bar bolted connections, bus bar insulation, bus bar insulating supports, bus enclosure assemblies (internal and external), and elastomers. This program does not manage the aging effects on external bus structural supports, which are managed under AMP XI.S6, “Structures Monitoring.” Alternatively, the aging effects on elastomers can be managed under AMP XI.M38, “Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components,” and the external surfaces of MEB enclosure assemblies can be managed under AMP XI.S6, “Structures Monitoring.”

2. ***Preventive Actions:*** This is a condition monitoring program and no actions are taken as part of this program to prevent or mitigate aging degradation.

3. ***Parameters Monitored/Inspected:*** This AMP provides for the inspection of the internal and external portions of the MEB. Internal portions (bus enclosure assemblies) of the MEB are inspected for cracks, corrosion, foreign debris, excessive dust buildup, and evidence of water intrusion. The bus insulation is inspected for signs of reduced insulation resistance due to thermal/thermoxidative degradation of organics/thermoplastics, radiation-induced oxidation, moisture/debris intrusion, or ohmic heating, as indicated by embrittlement, cracking, chipping, melting, discoloration, or swelling, which may indicate overheating or aging degradation. The internal bus insulating supports are inspected for structural integrity and signs of cracks. A sample of accessible bolted connections is inspected for increased resistance of connection. Alternatively, for accessible bolted connections covered with heat shrink tape, sleeving, insulating boots, etc., the sample may be visually inspected for insulation material surface anomalies. The external portions of the MEB, including accessible gaskets, boots, and sealants, are inspected for hardening and loss of strength due to elastomer degradation that could permit water or foreign debris to enter the bus. MEB external surfaces are inspected for loss of material due to general, pitting, and crevice corrosion.

4. ***Detection of Aging Effects:*** MEB internal surfaces are visually inspected for aging degradation including cracks, corrosion, foreign debris, excessive dust buildup, and evidence of moisture intrusion. MEB insulating material is visually inspected for signs of embrittlement, cracking, chipping, melting, discoloration, swelling, or surface contamination. Internal bus insulating supports are visually inspected for structural integrity and signs of cracks. MEB external surfaces are visually inspected for loss of material due to general, pitting, and crevice corrosion. Accessible elastomers (e.g., gaskets, boots, and sealants) are inspected for degradation including surface cracking, crazing, scuffing, dimensional change (e.g. “ballooning” and “necking”), shrinkage, discoloration, hardening and loss of strength.

   A sample of accessible bolted connections is inspected for increased resistance of connection by using thermography or by measuring connection resistance using a micro-ohmmeter. Twenty percent of the population with a maximum sample of 25 constitutes a representative sample size. Otherwise, a technical justification of the methodology and sample size used for selecting components should be included as part of the AMP’s site documentation. If an unacceptable condition or situation is identified in the selected sample, a determination is made as to whether the same condition or situation is applicable to other connections not tested.

   The first inspection using thermography or measuring connection resistance is completed prior to the period of extended operation and every 10 years thereafter provided visual inspection is not used to inspect bolted connections. This is an adequate period to preclude failures of the MEBs since experience has shown that MEB aging degradation is a slow process.

   As an alternative to thermography or measuring connection resistance of bolted connections, for accessible bolted connections that are covered with heat shrink tape, sleeving, insulating boots, etc., the applicant may use visual inspection of insulation material to detect surface anomalies, such as embrittlement, cracking, chipping, melting, discoloration, swelling, or surface contamination. When this alternative visual inspection is used to check the bolted connection sample, the first inspection is completed prior to the period of extended operation and every 5 years thereafter.

5. ***Monitoring and Trending:*** Trending actions are not included as part of this AMP because the ability to trend inspection results is limited. However, results that are trendable provide additional information on the rate of degradation.

6. ***Acceptance Criteria:*** MEB insulation materials are free from regional indications of surface anomalies such as embrittlement, cracking, chipping, melting, discoloration, and swelling, or surface contamination. MEB internal surfaces show no indications of corrosion, cracks, foreign debris, excessive dust buildup, or evidence of moisture intrusion. Accessible elastomers (e.g., gaskets, boots, and sealants) show no indications of surface cracking, crazing, scuffing, dimensional change (e.g. "ballooning" and "necking"), shrinkage, discoloration, hardening, and loss of strength. MEB external surfaces are free from loss of material due to general, pitting, and crevice corrosion.

   Bolted connections need to be below the maximum allowed temperature for the application when thermography is used or a low resistance value appropriate for the application when resistance measurement is used. When the visual inspection alternative for bolted connections is used, the absence of embrittlement, cracking, chipping, melting, discoloration, swelling, or surface contamination of the insulation material provides positive indication that the bolted connections are not loose.

7. ***Corrective Actions:*** Corrective actions are taken and an engineering evaluation is performed when the acceptance criteria are not met. Corrective actions may include, but are not limited, to cleaning, drying, increased inspection frequency, replacement, or repair of the affected MEB 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 MEBs. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

8. ***Confirmation Process:*** As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process.

9. ***Administrative Controls:*** The administrative controls for this AMP provide for a formal review and approval process. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the administrative controls.

10. ***Operating Experience:*** Industry experience has shown that failures have occurred on MEBs caused by cracked insulation and moisture or debris buildup internal to the MEB. Experience also has shown that bus connections in the MEBs exposed to appreciable ohmic heating during operation may experience loosening due to repeated cycling of connected loads.

    This AMP considers the technical information and guidance provided in SAND 96-0344, IEEE Std. 1205-2000, NRC IN 89-64, NRC IN 98-36, NRC IN 2000-14, and NRC IN 2007-01.

References

10 CFR Part 50, Appendix B, *Quality Assurance criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

IEEE Std. 1205-2000, *IEEE Guide for Assessing, Monitoring and Mitigating Aging Effects on Class 1E Equipment Used in Nuclear Power Generating Stations*.

NRC Information Notice 89-64, *Electrical Bus Bar Failures*, September 7, 1989.

NRC Information Notice 98-36, *Inadequate or Poorly Controlled, Non-Safety-Related Maintenance Activities Unnecessary Challenged Safety Systems*, September 18, 1998.

NRC Information Notice 2000-14, *Non-Vital Bus Fault Leads to Fire and Loss of Offsite Power*, September 27, 2000.

NRC Information Notice 2007-01, *Recent Operating Experience Concerning Hydrostatic Barriers*, January 31, 2007.

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.

XI.E5 FUSE HOLDERS

Program Description

The purpose of the aging management program (AMP) described herein is to provide reasonable assurance that the intended functions of the metallic clamps of fuse holders are maintained consistent with the current licensing basis through the period of extended operation.

Fuse holders (fuse blocks) are classified as a specialized type of terminal block because of the similarity in fuse holder design and construction to that of a terminal block. Fuse holders are typically constructed of blocks of rigid insulating material, such as phenolic resins. Metallic clamps (clips) are attached to the blocks to hold each end of the fuse. The clamps, which are typically made of copper, can be spring-loaded clips that allow the fuse ferrules or blades to slip in, or they can be bolt lugs, to which the fuse ends are bolted.

AMP XI.E1, "Insulation for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements," manages the aging of insulating material but not the metallic clamps of the fuse holders. The AMP for fuse holders (metallic clamps) needs to account for the following aging stressors if applicable: increased resistance of connection due to chemical contamination, corrosion, and oxidation or fatigue caused by ohmic heating, thermal cycling, electrical transients, frequent manipulation, or vibration. AMP XI.E1 is based on only a visual inspection of accessible cables and connections. Visual inspection is not sufficient to detect the aging effects from chemical contamination, corrosion, oxidation, fatigue, or vibration on the metallic clamps of the fuse holder.

Fuse holders that are within the scope of license renewal should be tested to provide an indication of the condition of the metallic clamps of the fuse holders. The specific type of test performed is determined prior to the initial test and is to be a proven test for detecting deterioration of metallic clamps of the fuse holders, such as thermography, contact resistance testing, or other appropriate testing justified in the application.

As stated in NUREG-1760, "Aging Assessment of Safety-Related Fuses Used in Low and Medium-Voltage Applications in Nuclear Power Plants," fuse holders experience a number of age-related failures. The major concern is that failures of a deteriorated cable system (cables, connections including fuse holders, and penetrations) might be induced during accident conditions. Since they are not subject to the environmental qualification requirements of 10 CFR 50.49, an AMP is required to manage the aging effects. This AMP ensures that fuse holders will perform their intended function for the period of extended operation.

Evaluation and Technical Basis

1. ***Scope of Program:*** This AMP manages fuse holders (metallic clamps) located outside of active devices that are considered susceptible to the following aging effects: increased resistance of connection due to chemical contamination, corrosion, and oxidation or fatigue caused by ohmic heating, thermal cycling, electrical transients, frequent manipulation, or vibration. Fuse holders inside an active device (e.g., switchgear, power supplies, power inverters, battery chargers, and circuit boards) are not within the scope of this AMP.

2. ***Preventive Actions:*** This is a condition monitoring program and no actions are taken as part of this program to prevent or mitigate aging degradation.

3. ***Parameters Monitored/Inspected:*** The metallic clamp portion of the fuse holder is tested to provide an indication of increased resistance of the connection due to chemical contamination, corrosion, and oxidation or fatigue caused by ohmic heating, thermal cycling, electrical transients, frequent manipulation or vibration.

4. ***Detection of Aging Effects:*** Fuse holders within the scope of license renewal are tested at least once every 10 years to provide an indication of the condition of the metallic clamp of the fuse holder. Testing may include thermography, contact resistance testing, or other appropriate testing methods. This is an adequate period to preclude failures of the fuse holders since experience has shown that aging degradation is a slow process. A 10-year testing interval provides two data points during a 20-year period, which can be used to characterize the degradation rate. The first tests for license renewal are to be completed prior to the period of extended operation.

5. ***Monitoring and Trending:*** Trending actions are not included as part of this AMP because the ability to trend test results is dependent on the specific type of test chosen. However, results that are trendable provide additional information on the rate of degradation.

6. ***Acceptance Criteria:*** The acceptance criteria for each test are defined by the specific type of test performed and the specific type of fuse holder tested. The metallic clamp of the fuse holder needs to be below the maximum allowed temperature for the application when thermography is used; otherwise, a low resistance value appropriate for the application when resistance measurement is used.

7. ***Corrective Action*****:** Corrective actions are taken and an engineering evaluation is performed when the test acceptance criteria are not met in order to ensure that the intended functions of the fuse holders can be maintained consistent with the current licensing basis. Such an evaluation is to consider the significance of the test results, the operability of the component, the reportability of the event, the extent of the concern, the potential root causes for not meeting the test acceptance criteria, the corrective action necessary, and the likelihood of recurrence. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

8. ***Confirmation Process:*** As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process.

9. ***Administrative Controls:*** The administrative controls for this AMP provide for a formal review and approval process. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the administrative controls.

10. **Operating Experience:** Operating experience has shown that loosening of fuse holders and corrosion of fuse clips are aging mechanisms that, if left unmanaged, can lead to a loss of electrical continuity function. Operating experience in NUREG-1760 documented fuse holder failures due to fatigue and recommends maintenance procedures be reviewed to minimize removal and reinsertion of fuses to de-energize components (as this can lead to degradation of the fuse holders).

    This AMP considers the technical information and guidance provided in NUREG-1760, IEEE Std. 1205-2000, NRC IN 86-87, NRC IN 87-42, and NRC IN 91-78.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

IEEE standard 1205-2000, *IEEE Guide for Assessing, Monitoring, and Mitigating Aging Effects on Class 1E Equipment Used in Nuclear Power Generating Stations*.

NRC Information Notice 86-87, *Loss of Offsite Power Upon an Automatic Bus Transfer*, October 10, 1986.

NRC Information Notice 87-42, *Diesel Generator Fuse Contacts*, September 4, 1987.

NRC Information Notice 91-78, *Status Indication of Control Power for Circuit Breakers Used in Safety-Related Application*, November 28, 1991.

NUREG-1760*, Aging Assessment of Safety-Related Fuses Used in Low- and Medium-Voltage Applications in Nuclear Power Plants*, May 31, 2002.

XI.E6 ELECTRICAL CABLE CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS

Program Description

The purpose of the aging management program (AMP) described herein is to provide reasonable assurance that the intended functions of the metallic parts of electrical cable connections that are not subject to the environmental qualification requirements of 10 CFR 50.49 and susceptible to age-related degradation resulting in increased resistance of connection due to thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination, corrosion, or oxidation are maintained consistent with the current licensing basis through the period of extended operation.

Cable connections are used to connect cable conductors to other cable conductors or electrical devices. Connections associated with cables within the scope of license renewal are part of this AMP. The most common types of connections used in nuclear power plants are splices (butt or bolted), crimp-type ring lugs, connectors, and terminal blocks. Most connections involve insulating material and metallic parts. This AMP focuses on the metallic parts of the electrical cable connections. This AMP provides a one-time test, on a sampling basis, to ensure that either aging of metallic cable connections is not occurring and/or that the existing preventive maintenance program is effective such that a periodic inspection program is not required. The one-time test confirms the absence of age-related degradation of cable connections resulting in increased resistance of connection due to thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination, corrosion, or oxidation.

AMP XI.E1, "Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements," manages the aging of insulating material but not the metallic parts of the electrical connections. AMP XI.E1 is based on a visual inspection of accessible cables and connections. Visual inspection may not be sufficient to detect the aging effects from thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination, corrosion, or oxidation on the metallic parts of cable connections.

Electrical cable connections exposed to appreciable ohmic or ambient heating during operation may experience increased resistance of connection caused by repeated cycling of connected loads or of the ambient temperature environment. Different materials used in various cable system components can produce situations where stresses between these components change with repeated thermal cycling. For example, under loaded conditions, ohmic heating may raise the temperature of a compression terminal and cable conductor well above the ambient temperature, thereby causing thermal expansion of both components. Thermal expansion coefficients of different materials may alter mechanical stresses between the components and may adversely impact the termination. When the current is reduced, the affected components cool and contract. Repeated cycling in this fashion can cause loosening of the termination and may lead to increased resistance of connection or eventual separation of compression-type terminations. Threaded connectors may loosen if subjected to significant thermally-induced stress and cycling.

Cable connections within the scope of license renewal should be tested at least once prior to the period of extended operation to provide an indication of the integrity of the cable connections. The specific type of test to be performed is a proven test for detecting increased resistance of connection, such as thermography, contact resistance testing, or another appropriate test. As an alternative to thermography or resistance measurement of cable connections, for the accessible cable connections that are covered with insulation materials such as tape, the applicant may perform visual inspection of insulation material to detect aging effects for covered cable connections. When this alternative visual inspection is used to check cable connections, the applicant must use periodic inspections and cannot use a one-time test to confirm the absence of age-related degradation of cable connections. The basis for performing only a periodic visual inspection is documented.

This AMP, as described, is a sampling program. The following factors are considered for sampling: voltage level (medium and low voltage), circuit loading (high loading), connection type and location (high temperature, high humidity, vibration, etc.). The technical basis for the sample selections should be documented. If an unacceptable condition or situation is identified in the selected sample, a determination is made as to whether the same condition or situation is applicable to other connections not tested. The corrective action program is used to evaluate the condition and determine appropriate corrective action.

SAND96-0344, "Aging Management Guidelines for Electrical Cable and Terminations," indicated that loose terminations were identified by several plants. The major concern is failures of a deteriorated cable system (cables, connections including fuse holders, and penetrations) that could prevent it from performing its intended function. This AMP is not applicable to cable connections in harsh environments since they are already addressed by the requirements of 10 CFR 50.49. Even though cable connections may not be exposed to harsh environments, increased resistance of connection is a concern due to the aging mechanisms discussed above.

Evaluation and Technical Basis

1. ***Scope of Program*****:** Cable connections associated with cables within the scope of license renewal that are external connections terminating at active or passive devices, are in the scope of this AMP. Wiring connections internal to an active assembly are considered part of the active assembly and, therefore, are not within the scope of this AMP. This AMP does not include high-voltage (>35 kilovolts) switchyard connections. The cable connections covered under the Environmental Qualification (EQ) program are not included in the scope of this program.

2. ***Preventive Actions*****:** This is a condition monitoring program, and no actions are taken as part of this program to prevent or mitigate aging degradation.

3. ***Parameters Monitored/Inspected*****:** This AMP focuses on the metallic parts of the connection. The one-time testing verifies that increased resistance of connection due to thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination, corrosion, or oxidation is not an aging effect that requires periodic testing. A representative sample of electrical cable connections is tested. The following factors are considered for sampling: voltage level (medium and low voltage), circuit loading (high load), connection type, and location (high temperature, high humidity, vibration, etc.). The technical basis for the sample selection is documented.

4. ***Detection of Aging Effects*****:** A representative sample of electrical connections within the scope of license renewal is tested at least once prior to the period of extended operation to confirm that there are no aging effects requiring management during the period of extended operation. Testing may include thermography, contact resistance testing, or other appropriate testing methods without removing the connection insulation, such as heat shrink tape, sleeving, insulating boots, etc. The one-time test provides additional confirmation to

support industry operating experience that shows that electrical connections have not experienced a high degree of failures, and that existing installation and maintenance practices are effective. Twenty percent of the population with a maximum sample of 25 constitutes a representative sample size. Otherwise a technical justification of the methodology and sample size used for selecting components for one-time test should be included as part of the AMP's site documentation.

As an alternative to thermography or measuring connection resistance of the cable connection sample, for accessible cable connections that are covered with heat shrink tape, sleeving, insulating boots, etc., the applicant may use visual inspection of insulation materials to detect surface anomalies, such as embrittlement, cracking, chipping, melting, discoloration, swelling or surface contamination. When this alternative visual inspection is used to check cable connections, the first inspection is completed prior to the period of extended operation and every 5 years thereafter. The basis for performing only a periodic visual inspection to monitor age-related degradation of cable connections is documented.

5. ***Monitoring and Trending*:** Trending actions are not included as part of this AMP because it is a one-time testing or, alternatively, a periodic visual inspection program where the ability to trend inspection results is limited. However, results that are trendable provide additional information on the rate of degradation.

6. ***Acceptance Criteria*:** Cable connections should not indicate abnormal temperature for the application when thermography is used; otherwise a low resistance value appropriate for the application when resistance measurement is used. When the visual inspection alternative for covered cable connections is used, the absence of embrittlement, cracking, chipping, melting, discoloration, swelling or surface contamination indicates that the covered cable connection components are not loose.

7. ***Corrective Actions*:** If acceptance criteria are not met, the corrective action program is used to perform an evaluation that considers the extent of the condition, the indications of aging effect, and changes to the one-time testing program or alternative inspection program. Corrective actions may include, but are not limited to, sample expansion, increased inspection frequency, and replacement or repair of the affected cable connection components. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

8. ***Confirmation Process*:** As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process.

9. ***Administrative Controls*:** The administrative controls for this AMP provide for a formal review and approval process. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the administrative controls.

10. ***Operating Experience*:** Electrical cable connections exposed to thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination, corrosion, or oxidation during operation may experience increased resistance of connection. There have been limited numbers of age-related failures of cable connections reported. An applicant's operating experience with detection of aging effects should be adequate to demonstrate that the program is capable of detecting the presence or noting the absence of aging effects for

electrical cable connections where a one-time inspection is used to confirm the effectiveness of another preventive or mitigative AMP.

This AMP considers the technical information and guidance provided in NUREG/CR-5643, SAND96-0344, IEEE Std. 1205-2000, EPRI 109619, EPRI 104213, NEI White Paper on AMP XI.E6, Final License Renewal Interim Staff Guidance LR-ISG-2007-02, Staff Response to the NEI White Paper on AMP XI.E6, Licensee Event Report (LER) 361 2007005, LER 3612007006 and LER 3612008006.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

EPRI 104213, *Bolted Joint Maintenance & Application Guide*, Electric Power Research Institute, Palo Alto, CA, December 1995.

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

Final License Renewal Interim Staff Guidance LR-ISG-2007-02: Changes to Generic Aging Lesson Learned (GALL) Report Aging Management Program (AMP) XI.E6, *Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements*, 74 FR 68287, U.S. Nuclear Regulatory Commission, December 23, 2009.

IEEE Std. 1205-2000, *IEEE Guide for Assessing, Monitoring and Mitigating Aging Effects on Class 1E Equipment Used in Nuclear Power Generating Stations.*

Licensee Event Report 361 2007005, *San Onofre Unit 2, Loose Electrical Connection Results in Inoperable Pump Room Cooler,* U.S. Nuclear Regulatory Commission.

Licensee Event Report 3612007006, *San Onofre Units 2 and 3, Loose Electrical Connection Results in One Train of Emergency Chilled Water (ECW) System Inoperable,* U.S. Nuclear Regulatory Commission.

Licensee Event Report 3612008006, *San Onofre 2, Loose Connection Bolting Results in Inoperable Battery and TS Violation,* U.S. Nuclear Regulatory Commission.

NEI White Paper, *GALL AMP XI.E6 (Electrical Cables)*, Nuclear Energy Institute, September 5, 2006. (ADAMS Accession Number ML062770105)

NUREG/CR-5643, *Insights Gained From Aging Research*, U.S. Nuclear Regulatory Commission, March 1992.

SAND96-0344, *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.

Staff’s Response to the NEI White Paper on Generic Aging Lessons Learned (GALL) Report Aging Management Program (AMP) XI.E6, *Electrical Cable Connections Not Subject to*

10 CFR 50.49 Environmental Qualification Requirements, U.S. Nuclear Regulatory Commission, March 16, 2007. (ADAMS Accession Number ML070400349)

APPENDIX

QUALITY ASSURANCE FOR AGING MANAGEMENT PROGRAMS

QUALITY ASSURANCE FOR AGING MANAGEMENT PROGRAMS

The license renewal applicant must demonstrate that the effects of aging on structures and components (SC) subject to an aging management review (AMR) will be managed in a manner that is consistent with the CLB of the facility for the period of extended operation. Therefore, those aspects of the AMR process that affect the quality of safety-related SCs are subject to the quality assurance (QA) requirements of Appendix B to 10 CFR Part 50. For non-safety-related SCs subject to an AMR, the existing 10 CFR Part 50, Appendix B, QA program may be used to address the elements of corrective actions, confirmation process, and administrative controls on the following bases:

- Criterion XVI of 10 CFR Part 50, Appendix B, requires that measures be established to ensure that conditions adverse to quality, such as failures, malfunctions, deviations, defective material and equipment, and non-conformances, are promptly identified and corrected. In the case of significant conditions adverse to quality, measures must be implemented to ensure that the cause of the condition is determined and that corrective action is taken to preclude repetition. In addition, the cause of the significant condition adverse to quality and the corrective action implemented must be documented and reported to appropriate levels of management.

To preclude repetition of significant conditions adverse to quality, the confirmation process element (Element 8) for license renewal AMPs consists of follow-up actions to verify that the corrective actions implemented are effective in preventing a recurrence. As an example, for the management of internal piping corrosion, the AMP XI.M2, "Water Chemistry," may be used to minimize the piping's susceptibility to corrosion. However, it also may be necessary to institute a condition monitoring program that uses ultrasonic inspection to verify that corrosion is indeed insignificant.

- 10 CFR 50.34(b)(6)(i) requires that the final safety analysis report submitted by a nuclear power plant license applicant includes information on the applicant's organizational structure, allocations of responsibilities and authorities, and personnel qualification requirements. 10 CFR 50.34(b)(6)(ii) also notes that Appendix B to 10 CFR Part 50 sets forth the requirements for managerial and administrative controls used for safe operation. Pursuant to 10 CFR 50.36(c)(5), administrative controls related to organization and management, procedures, record keeping, review and audit, and reporting ensure the safe operation of the facility. Programs that are consistent with the requirements of 10 CFR Part 50, Appendix B, also satisfy the administrative controls element necessary for AMPs for license renewal.

Notwithstanding the suitability of its provisions to address quality-related aspects of the AMR process for license renewal, 10 CFR Part 50, Appendix B, covers only safety-related SCs. Therefore, absent a commitment by the applicant to expand the scope of its 10 CFR Part 50, Appendix B, QA program to include non-safety-related structures and components subject to an AMR for license renewal, the AMPs applicable to non-safety-related SCs include alternative means to address corrective actions, confirmation processes, and administrative controls. Such alternate means are subject to review by the NRC on a case-by-case basis.

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Same as item 8, above.

10. SUPPLEMENTARY NOTES

Revision 2 consolidates NUREG-1801, Volume 1 and Volume 2 into one volume.

11. ABSTRACT *(200 words or less)*

The Generic Aging Lessons Learned (GALL) Report contains the staff's generic ev aluation of the existing plant programs and documents the technical basis for determining where existing programs are adequ ate without modification and where existing programs should be augmented for the period of extended operation. The evaluati on results documented in the GALL Report indicate that many of the existing programs are adequate to manage the aging ef fects for particular structures or components for license renewal without change. The GALL Report also contains recommendatio ns on specific areas for which existing programs should be augmented for license renewal. An Applicant may reference th e GALL Report in a license renewal application to demonstrate that the programs at the applicant's facility corres pond to those reviewed and approved in the GALL Report. However, if an applicant takes credit for a program in the GALL Report, it is incumbent on the applicant to ensure that the conditions and operating experience at the plant are bounded by the conditi ons and operating experience for which the GALL Report program was evaluated. If these bounding conditions are not met, it is incumbent for the applicant to address the additional aging effects and augment the GALL Report programs as appropriate. T he staff will verify that the applicant's programs are consistent with those described in the GALL Report and/or with pla nt conditions and operating experience during the performance of an aging management audit. The focus of the balance of the s taff's review of a license renewal application is on those programs that an applicant has enhanced to be consistent with the G ALL Report, those programs that an applicant has taken an exception to the program described in the GALL Report, and plant-s pecific programs not described in the GALL Report. The information in the GALL Report has been incorporated into the NUREG -1800, "Standard review Plan for Review of License Renewal Applications for Nuclear Power Plants" as directed by the Commi ssion, to improve the efficiency of the license renewal process.

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